Enclosure 3

NORTH ANNA NUCLEAR PLANT INDIVIDUAL PLANT EXAMINATION
TECHNICAL EVALUATION REPORT

(BACK-END)

ERI/NRC 95-105

TECHNICAL EVALUATION REPORT OF THE NORTH ANNA INDIVIDUAL PLANT EXAMINATION BACK-END SUBMITTAL

FINAL REPORT

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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 North Anna nuclear power plant.

E.1 Plant Characterization

Virginia Electric and Power Company operates two nuclear power stations at the North Anna site, each of which has a rated thermal capacity of 2,893 MWt, and is contained within a a sub-atmospheric, large dry containment. The North Anna containment building is a steel-lined concrete shell in the form of a vertical cylinder, with a hemispherical head and a flat base. The base is a 10-foot thick reinforced concrete slab. The mean failure pressure of the containment is 128 psig. In several other aspects, such as RCS water volume, power, and masses of core inventory, the North Anna plant is similar to the Surry and Zion plants.

E.2 Licensee's IPE Process

The IPE back-end analysis was a joint utility/contractor effort, with the contractor, Halliburton NUS, taking the lead role. One Virginia Power engineer and two NUS employees were involved in the back-end analyses. In addition, a number of front-end analysts were also used to perform the tasks related to the front-end/back-end interface. An independent review team composed of station personnel, corporate staff, two senior analysts from Science Applications International Corporation (SAIC) and one containment analyst from Stone and Webster Engineering Corporation (SWEC), was formed to review the results of the IPE.

The methodology employed in the North Anna IPE submittal for the back-end evaluation is clearly described, and the IPE is logical and consistent with GL 88-20. The outcomes of the front-end analyses are grouped into Plant Damage States through the use of an event tree. Probabilistic quantification of severe accident progression involved the development of a large Containment Event Tree (CET). This event tree is qualitatively similar to the tree developed for the NUREG-1150 analyses for the Surry plant. Owing to the complexity of the CET, the large event tree was divided into a number of small Deterministic Event Trees (DETs), that concentrate on specific phenomena, such as in-vessel coolability, early containment failure, etc. Extensive use was made of the information found in NUREG/CR-4551 for the quantification of split fractions for the various CET nodes. 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. To a large extent, the results of the severe accident simulations performed for the Surry IPE submittal were used. However, a few North Anna plant-specific MAAP calculations were also performed.

E.3 Back-End Analysis

The submittal reports an internal events Core Damage Frequency (CDF) of 6.8×10^{-5} per reactor year. LOCAs (30%), loss of offsite power sequences (29%), and transients (27%) contribute

equally to core damage frequency. Steam Generator Tube Rupture (SGTR) sequences and interfacing system LOCAs contribute to approximately 10% and 2% of the CDF, respectively. The CDF for internal flooding is 3.6 x 10^{-6} per reactor year.

The CET analyses were performed using a large event tree methodology. The MAAP code was used to determine the containment response and to calculate the source terms for a selected list of sequences. Results from the containment analyses indicate that, upon core damage, the conditional probability of radiological releases (including both containment failure and bypass) is 26% (see Table E.1). The leading contributors to radiological releases are the steam generator tube rupture sequences and the V sequences, which together contribute to nearly 52% of the release frequency (14% of the CDF). Early containment failure, due to overpressure, contributes to 1.1% of the CDF (4.2% of the release frequency). Late failure due to containment overpressure contributes to 10.2% of the CDF (39% of the release frequency). Basemat meltthrough contributes to 1.1% of the CDF (4.2% of the release frequency). Thus, the conditional probability of late containment failure in the North Anna IPE submittal is nearly twice the value calculated in the NUREG-1150 analysis for the Surry plant. The licensee attributes the difference in results to the consideration of the loss of emergency room cooling, both as an initiator and as a support system, in the front-end analysis. Loss of switchgear room cooling leads to station blackout, since the emergency buses are lost. If AC power is not recovered, or if safety injection systems fail (due to random causes) after AC power recovery, the accident sequence progresses to vessel breach, relocation of core debris to the cavity, and slow pressurization of the containment, leading to late containment failure.

E.4 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 RAIs, the licensee addressed the issue of local and global hydrogen combustion, and associated threats to containment. The licensee stated that hydrogen buildup to sufficient concentrations that can lead to combustion and failure of the containment was unlikely in most accident sequences, since a number of ignition sources are available. However, this situation can occur in a station blackout sequence with late AC power recovery. The licensee stated that a very conservative model for this sequence led to the calculation of a containment failure probability of 0.03. The licensee also stated that buildup of hydrogen concentration to a level that can sustain a detonation was not possible in the North Anna containment.

E.5 Vulnerabilities and Plant Improvements

The submittal attempts to define vulnerability in Section 3.4.2 (page 3-132) of Reference [1] (as pertaining to core damage) as "a failure (component fault or human error) that is significantly

Table E.1 Containment Failure as a Percentage of Total CDF: Comparison with Other PRA Studies

Containment Failure Mode	North Anna IPE	Surry NUREG-1150	Zion NUREG-1150
Early Failure	1.3	0.7	0.5
Late Failure	10.0	5.9	24.0
Very Late Failure	1.1	NA*	NA*
Bypass (V)	2.4	7.6	0.2
Bypass (SGTR)	11.6	4.6	0.3
Isolation Failure	0.2	NA**	1.0
Intact	74.0	81.2	73.0
Core Damage Frequency, yr^{-1}	$6.8x10^{-5}$	$4.1x10^{-5}$	$6.2x10^{-5}$

** Included as a Part of Early Containment Failure
* Included as a Part of Late Containment Failure

greater than others, i.e., contribute more than 10% to the CDF or must be a factor of three greater than the next highest event". No such definition is provided for a vulnerability as pertaining to the back-end analyses. However, in Section 3.4.2.3 of the submittal, the licensee notes that the containment structure and mitigation systems are robust, and the containment capacity is large such that the probability of early containment failure is low. Redundancy and diversity of containment spray systems leads to the fact that the probability of failure of the CHR systems is low. In addition, since North Anna is a sub-atmospheric containment, the probability of having an unisolated containment is also low. Thus, it appears that the licensee perceives that no containment vulnerabilities exist for the North Anna plant.

The licensee has committed to a number of plant improvements. Complete details of the improvements can be found in Section 6.1.2 (page 6-2) of the submittal, and the improvements are summarized in Table 6-1 (page 6.7) of the submittal. To summarize briefly, a number of procedural modifications were deemed necessary, and they include modifications in the testing of auxiliary feedwater flow, procedures for realignment of quench and recirculation spray headers, revision of procedures to verify SI flow, revision of testing procedures for the low head safety injection pump, and improvement and modification of the room chiller repair and testing procedures.

In addition, a number of plant hardware modifications to prevent flooding in the charging pump cubicle, improvement of the fire barrier between the spray pump house and the auxiliary building, and hardware modification in the chiller room, were planned. The modifications (hardware and procedural) are essentially complete. The licensee mentions one back-end related insight, and this insight pertains to SGTR sequences. The SGTR sequences involve failure of the operator to cool down and depressurize the RCS early in the accident. For such sequences, there is a potential for the steam generator PORV to stick open due to steam generator overfill. Thus, the licensee concludes that the most significant insight from the back-end analyses is the importance of regulating the level in the steam generators. The licensee also stated that [11] "the importance of the scenario has been transmitted to the North Anna training department for use in developing training scenarios". In addition, the licensee noted that replacement of steam generators in the North Anna plant is near completion, and that the reliability of the new steam generators is expected to be good.

E.6 Observations

The North Anna submittal calculates an internal events CDF of 6.8×10^{-5} per reactor year, and a CDF of 3.6×10^{-6} per reactor year for internal flooding. LOCAs (30%), loss of offsite power sequences (29%), and transients (27%) all contribute significantly to core damage frequency. The results for the CDF are a factor of two larger than those reported in NUREG-1150 for the Surry plant. The differences are attributed to the identification of an additional initiating event (loss of switchgear room cooling) as an important contributor at North Anna, and differences in assumed success criteria for small break LOCAs and SGTRs. The containment analyses indicate that there is a 26% conditional probability of radiological releases, and 74% conditional probability of intact containment.

The important points of the submittal-only technical evaluation of the North Anna IPE back-end analyses are summarized below:

- The Back-End portion of the IPE supplies a substantial amount of information with regards to the subject areas identified in Generic Letter 88-20. For the most part, the separate models used in the North Anna IPE Back-End analysis are technically sound. Extensive use is made of the quantification of the split fractions used in the NUREG-1150 CET.

- Phenomena that lead to early containment failure (i.e., DCH, steam explosions) have been treated in a manner very similar to NUREG-1150. However, the calculated conditional probability of early containment failure is slightly larger for the North Anna IPE submittal. The differences can be attributed to two reasons. The first reason is that the relatively larger core and RCS in the North Anna plant in a similar containment (as Surry) leads to the calculation of a higher conditional probability of early containment failure. The second reason is the consideration of the loss of emergency room cooling, both as an initiator, and as a support system in the front-end analysis for North Anna. Loss of switchgear room cooling leads to station blackout since the emergency buses are

lost. The increased contribution of station blackout sequences lead to a slightly increased conditional probability of early containment failure.

- The submittal reports a conditional probability of about 10% for late containment overpressure failure, and a probability of about 1% for basemat melt-through. Once again, these values are larger than those obtained from the Surry NUREG-1150 analyses, and they are attributed to the consideration of the loss of emergency room cooling, both as an initiator, and as a support system in the front-end analysis for North Anna.

- The radiological releases for the North Anna plant are dominated by steam generator tube rupture sequences. The release frequency of the SGTR sequennces is 7.9×10^{-6} per reactor year (11.6% of the CDF). It is noted that the release frequency of the SGTR sequences is significant, and the magnitude of the associated releases are large.

- The licensee has addressed the recommendations of the CPI program.

TABLE OF CONTENTS

LIST OF TABLES

NOMENCLATURE

ATWS	Anticipated Transient Without Scram
CDF	Core Damage Frequency
CET	Containment Event Tree
CHR	Containment Heat Rejection
CPI	Containment Performance Improvement
DCH	Direct Containment Heating
DET	Deterministic Event Tree
DF	Decontamination Factor
ECCS	Emergency Core Cooling Systems
EOP	Emergency Operating Procedure
EPA	Electrical Penetration Assembly
EPRI	Electric Power Research Institute
ERI	Energy Research, Inc.
ESF	Engineered Safety Features
EVSE	Ex-Vessel Steam Explosion
GL	Generic Letter
NUS	Halliburton NUS, Inc.
IPE	Individual Plant Examination
ISLOCA	Interfacing Systems Loss of Coolant Accident
IVSE	In-Vessel Steam Explosion
LOCA	Loss of Coolant Accident
LPI	Low Pressure 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
SAIC	Science Applications International Corporation
SGTR	Steam Generator Tube Rupture
SI	Safety Injection Systems
SRV	Safety Relief Valve
STR	Source Term Category
SWEC	Stone and Webster Engineering Corporation
TER	Technical Evaluation Report
VB	Vessel Breach

1. INTRODUCTION

This Technical Evaluation Report (TER) documents the results of our review of the North Anna 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.

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 North Anna 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 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, 4 through 7, and Appendix F 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 are 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 [3] and achieves the four IPE sub-objectives. A draft TER based on the back-end portion of the submittal was submitted to the NRC in August 1994. 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 [11] were reviewed. The final TER is based on the information contained in the IPE submittal [1] and the licensee responses to the NRC Requests for Additional Information (RAIs) [11].

1.2 Plant Characterization

Virginia Electric and Power Company operates two nuclear power stations at the North Anna site, each of which has a rated thermal capacity of 2,893 MWt, and is contained within a sub-atmospheric, large dry containment. A detailed description of the North Anna containment and

plant data are provided in Section 4.4 of the submittal. Figures 4.1.1-1 and 4.1.2-1 (of the submittal) illustrate some of the design features of the cavity and the containment that are important for severe accident progression.

The North Anna containment building is a steel-lined concrete shell in the form of a vertical cylinder with a hemispherical head and a flat base. The base is a 10 ft thick reinforced concrete slab, with a cylinder and dome resting on the top of the reinforced concrete slab. The concrete cylinder is 4.5 ft thick, rises 125 ft above the slab, and is lined with a welded steel liner plate. The dome is a 2.5 ft thick hemisphere with an inside radius of 65 ft. The thickness of the steel liner varies from 0.25 inch on the basemat floor to 0.5 inch on the dome wall. The liner thickness is 0.375 inch on the cylinder wall.

Table 1 provides a summary comparison of the key design features of the North Anna plant and containment systems with the Zion and Surry plants. In addition, the following plant-specific features are important for accident progression in the North Anna plant:

- The cavity in the North Anna plant is isolated from the rest of the lower compartment of the containment. It is composed of a cylindrical region below the RPV (7.75 ft radius) connected to a rectangular tunnel by a normally open keyway. The cavity floor is 10 feet thick, covered with a 0.25 inch liner which is in turn covered with 1.5 ft of concrete. The horizontal, rectangular section of the cavity is 25.2 ft by 10 ft in cross-section and communicates with the lower compartment.

- Water can enter the cavity only from the quench sprays, and only the spray flow that falls onto the refueling pool floor will flow into the cavity. It is estimated by the licensee that it will take approximately 20 minutes for the cavity to be full of water if all the spray systems are functional.

- The maximum water elevation on the containment floor (outside the cavity) when all the contents of the primary system and the RWST are injected into the containment is approximately 6 ft. However, the required depth for overflow from the containment to cavity is 16.5 ft, which means that water will not overflow into the cavity if sprays do not function.

- The outer end of the instrument tunnel is not sloped from the reactor vessel as is the case with the Zion plant. At the outer end of the instrument tunnel, the instrument tubes are directed vertically upwards through a "cofferdam" that is raised above the adjacent tunnel ceiling. The region around the instrument tubes is a likely pathway for the debris, as is a 9 ft^2 ventilation duct on the roof of the instrumentation tunnel. The licensee believes that the design of the instrument tunnel is less conducive to dispersion than the Surry and Zion plants.

- The cavity floor area is 58 m^2, which is somewhat larger than similar Westinghouse designs. If the entire core inventory of the North Anna plant was to relocate on the cavity floor (and spread uniformly), the debris layer thickness will be about 23 cm.

- The concrete used in the North Anna plant is stated to be a basaltic aggregate. This type of concrete will lead to generation of smaller quantities of noncondensable gases due to core-concrete interactions, as compared to limestone concrete.

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

Feature	North Anna	Surry	Zion
Power Level, MW(t)	2,893	2,441	3,236
Volume of RCS Water, m^3	282	283	368
Free Volume of Containment, m^3	51,680	46,440	81,000
Mass of Fuel, kg	82,160	79,652	98,250
Mass of Zircalloy, kg	17,108	16,466	20,230
RCS Water Volume/Power, m^3/MW(t)	0.10	0.12	0.11
Containment Volume/Power, m^3/MW(t)	17.9	19.0	25.0
Zr Mass/Containment Volume, kg/m^3	0.33	0.35	0.25
Fuel Mass/Containment Volume, kg/m^3	1.59	1.72	1.21
Mass of H_2 Generated by Zirconium Oxidation, kg	750	721	886
Maximum H_2 Concentration, 10^{-3} moles/m^3	7	9	7

Table 2 Comparison of Containment Capacities

Feature	North Anna	Surry	Zion
Design Pressure	44 psig (4 bars)	45 psig (4 bars)	47 psig (4.2 bars)
Failure Pressure	128 psig (9.8 bars)	126 psig (9.8 bars)	134 psig (10.2 bars)
Concrete Type	Basaltic aggregate	Basaltic	Limestone

- The North Anna plant has a recirculation spray system that is composed of four independent spray trains. Two trains are located inside the containment, while two trains have spray pumps located outside the containment in the safeguards building. The spray pumps draw water form the sump in the recirculation mode, but the sump is compartmentalized such that each pump has its own sump compartment. The licensee believes that the multiple, redundant spray system design will reduce the probability of failure of the spray system.

In several other aspects such as RCS water volume, power, and masses of core inventory, the North Anna plant is similar to the Surry and Zion plants, as shown in Table 1. Note that there are small differences in the various ratios between the Surry and the North Anna plants.

2. CONTRACTOR REVIEW FINDINGS

The present review compares the North Anna IPE submittal with the intent of the Generic Letter (GL) 88-20, and its supplements, using guidance provided in NUREG-1335. The responses of the licensee were also reviewed. The findings of the present review are reported in this section, and they follow the structure of Task Order Subtask 1.

2.1 Licensee's IPE Process

2.1.1 Completeness and Methodology

The IPE submittal contains a substantial amount of information in accordance with the recommendations of GL 88-20 and NUREG-1335. The submittal appears to be complete and to provide the level of detail requested in NUREG-1335.

The methodology employed in the North Anna IPE submittal for the back-end evaluation is clearly described, and the IPE is logical and consistent with GL 88-20. The outcomes of the front-end analyses are grouped into Plant Damage States through the use of an event tree. Probabilistic quantification of severe accident progression involved the development of a large Containment Event Tree (CET), and this event tree was qualitatively similar to the tree developed for the NUREG-1150 analyses for the Surry plant. Owing to the complexity of the CET, the large event tree was divided into a number of small Deterministic Event Trees (DETs) that concentrated on specific phenomena such as in-vessel coolability, early containment failure, etc. Extensive use was made of the information found in NUREG/CR-4551 in the quantification of split fractions for the various CET nodes. 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. To a large extent, the results of the severe accident simulations performed for the Surry IPE submittal were used. However, a few North Anna plant-specific MAAP calculations were also performed.

The endpoints of the CETs represent the outcomes of possible in-containment accident progression sequences. These endpoints are classified into a manageable number of release categories, characterized by similarities in accident progression and source term characteristics. Associated with each source term category is a release frequency and magnitude of radionuclide release. A "source term logic" diagram is used to classify the CET end states into the appropriate release categories. A total of 24 source term categories were found to result. The magnitude of the releases for each release category were determined by the source term calculations performed for the Surry IPE submittal. For the Surry IPE, MAAP calculations were performed for a selected number of accident sequences representative of the source term bins. For other categories, the release fractions were characterized by results from analyzed sequences that were similar. Calculations were performed for 10 of the 24 release bins.

2.1.2 As-Built/As-Operated Status

It appears that all the North Anna plant and containment systems are modelled by the licensee. The licensee performed a plant walkdown at the beginning of the study to ensure system familiarization.

2.1.3 Licensee Participation and Peer Review of IPE

The IPE back-end analysis was a joint utility/contractor effort, with the contractor, Halliburton NUS, taking the lead role. One Virginia Power engineer and two NUS employees were involved in the back-end analyses. In addition, a number of front-end analysts were also used to perform the tasks related to the front-end/back-end interface. An independent review team composed of station personnel, corporate staff, two senior analysts from Science Applications International Corporation (SAIC) and one containment analyst from Stone and Webster Engineering Corporation (SWEC), was formed to review the results of the IPE. The SWEC employee was involved in the design and modification of the North Anna containment. The review team concluded that the IPE submittal met with or exceeded the requirements of the generic letter. However, with regards to the containment analyses, the review team questioned (1) the applicability of Surry results to the North Anna plant, (2) the definitions and binning of PDSs, (3) the capability of the MAAP code for use in ascertaining the success criteria for SGTR sequences, (4) the application of the PDS rules for different accident sequence types, and (5) the documentation of source term results. The submittal states that the issues raised by the review team were "resolved" by modification of the submittal, and by performing additional North Anna-specific analyses.

2.2 Containment Analysis

2.2.1 Front-End/Back-End Dependencies

The entry points to the containment event trees are the plant damage states. The PDSs are groupings of core melt sequences, based on similarities in accident progression and availability of containment safeguards. The goal of this grouping process is to reduce the number of required containment analyses to a tractable number. In the North Anna IPE submittal, PDSs are defined by a combination of nine different binning characteristics as described below:

1. Status of Containment Building Prior to Core Damage: Considers the success of containment isolation system, and divides PDSs into the two following groups:

   - Containment not isolated
   - Containment isolated

   All sequences with containment isolation failure are grouped into two PDS groups, i.e., with and without the success of quench or recirculation sprays.

2. Transient or LOCA Type: This parameter is used to classify PDSs by accident initiators. This parameter is used to classify sequences by RCS pressure and to distinguish sequences with different key event timings. The following types of initiators are considered:

- o Large break LOCAs
- o Small/Medium break LOCAs
- o Transients

3. Station Blackout Type: This parameter indirectly determines the availability of AC power. Sequences with total loss of AC power are grouped into station blackout PDS, and those with onsite AC power available are grouped into non-station blackout cases.

4. Power Recovery: For station blackout type sequences, this parameter is used to identify the possibility of recovery of offsite power subsequent to core damage. Recovery of offsite power is considered in two time frames, namely, prior to RPV failure, and prior to containment failure. Power recovery after core damage allows for recovery of in-vessel injection which may terminate the accident, and the restoration of sprays and containment heat removal that might prevent containment failure.

5. Recirculation Sprays: Successful operation of sprays in recirculation mode is needed to prevent late containment failure, and mitigate fission product releases to the environment.

6. Containment Heat Removal: In the North Anna plant, operation of one train of recirculation sprays together with a functional heat exchanger is necessary to prevent long term containment overpressure failure. This parameter determines the availability of containment heat removal systems.

7. RCS Pressure During Core Damage: Accident sequences are classified into four types, based on the RCS pressure as follows:

- o Low Pressure (< 200 psig)
- o Low-High Pressure (200 - 2000 psig)
- o High Pressure (2000 - 2335 psig)
- o High-High Pressure (≥ 2335 psig)

The choice of the pressure boundaries was based on a review of NUREG-1150 analyses and assumptions. For RCS pressures below 200 psig, the NUREG-1150 analysts concluded that the energetic dispersal of debris following vessel breach is not believed to be important. The NUREG-1150 in-vessel expert panel determined 2000 psig to be the threshold pressure above which induced hot leg or surge line rupture was likely. The High-High pressure regime corresponds to the PORV setpoint relief pressure.

8. Status of In-Vessel Injection: This indicator determines the availability of the ECCS injection, and consists of four branches:

On	-	ECCS available for injection and operational.
Deadhead	-	ECCS available, but cannot inject because of high pressure.
Recovered	-	Injection recovered after core damage, but before vessel breach.
Failed	-	ECCS never available.

9. Containment Bypass: Steam Generator Tube Rupture (SGTR) and V sequences are binned by this PDS indicator.

The PDS definitions appear to be adequate. The PDS grouping logic is explained by the logic diagram presented in Figure 4.3.2-1 of the submittal. Quantification (i.e., binning of the sequences into proper PDSs) was performed using the HNUS-proprietary NUCAP+ code. Fifty-eight PDSs were initially defined for the IPE. However, all PDSs which have a zero frequency were deleted. Furthermore, PDSs having a frequency of less than 10^{-8} were further binned with other PDSs that were judged to be qualitatively similar. A total of 25 PDSs was found to remain after reclassification.

PDSs 1 and 2 contain sequences that include failure of containment isolation. These sequences contribute to a total of 0.22% of internal events CDF. PDSs 3 through 7 are the Station Blackout (SBO) sequences. SBO sequences contribute to 27.3% of the CDF. PDSs 8 through 11 represent transients other than the station blackout type. They contribute to 2.6% of the internal events CDF. Large LOCA sequences are represented by PDSs 12 and 13, and they contribute to 6.4% of the CDF. Small and medium LOCAs contribute to 51% of the CDF. V sequences contribute to 2.35% of the CDF, and are binned by PDS 24. SGTRs contribute to 10.3% of the CDF, and are binned by PDS 25. High pressure sequences (pressure ~ 2335 psig) contribute to 47% of the CDF, and intermediate pressure (200 < pressure < 2000 psig) sequences contribute to 45% of the CDF. Low pressure sequences contribute to 8% of the CDF. A listing of all the sequences (with frequency approximately above 10^{-8}) in all PDSs is found in Table 4.3.4-2 of the submittal.

2.2.2 Containment Event Tree Development

Probabilistic quantification of severe accident progression is performed using an event tree methodology. To keep the event tree compact and understandable, similar and adjacent top event "headings" are grouped to form "Decomposition Event Trees" (DETs). Each PDS was developed using a different (but similar) event tree, and the general Level-2 event tree is depicted in Figure 4.5.1-1 (page 4-167) of the submittal. The event trees for all the PDSs are provided in Figures 4.5.1-2 through 4.5.1-5 of the submittal. The development of the general event tree is discussed in Section 4.5.1 and the decomposition event trees are discussed in Section 4.5.2 of the submittal. The CET is qualitatively similar to that used in NUREG-1150 [2] and makes extensive use of the results from that reference. The CET is concise and contains the following main top events:

(1) Mode of induced primary system failure,
(2) Debris cooled in-vessel,
(3) No Alpha mode containment failure,
(4) No early containment failure,
(5) No early recirculation spray failure,
(6) Debris cooled ex-vessel,
(7) Mode of late containment failure,
(8) No late recirculation spray failure, and
(9) Long term containment failure.

Mode of induced primary system failure

This DET (i.e., a cluster of adjacent event tree nodes) addresses the potential for failure of the RCS pressure boundary due to natural circulation. Three branch possibilities are considered:

1. No induced RCS failure,

2. Hot leg (or surge line) failure, and

3. Steam generator tube rupture.

The probabilities assigned for the three branches are the same as provided by the NUREG-1150 expert panel. However, the licensee has not credited operator action to depressurize the RCS using the PORVs after core damage.

Debris cooled in-vessel

This DET considers the probability of core coolability after damage due to recovery of injection systems. Two possible branches are considered, namely, debris cooled in-vessel, and debris not cooled in-vessel. However, the negative consequences of core reflood such as reflood-induced hydrogen generation, combustion and containment failure are not treated.

The submittal assigns a probability of 0.7 for successful core coolability for sequences that entail recovery of injection systems. In NUREG-1150, the analysts assigned various values for this event depending on the time of recovery, and the extent of core damage. The level of analyses in the submittal are not as extensive as the NUREG-1150 analyses, and the same probability of 0.7 is assigned for successful core coolability for all PDSs. However, if LPI is available, but not able to function due to high RCS pressure, and if hot leg rupture occurs, the submittal assigns a probability of 0.9 for successful core coolability. Once again the time of hot leg rupture, and the extent of core damage is not considered. For sequences that have induced SGTR, in-vessel core coolability is assigned a zero probability, and this is a conservative assumption.

No Alpha-Mode Containment Failure

In-vessel steam explosions and alpha-mode containment failure are treated by this DET. The values assigned for containment failure due to IVSE are 0.008 for low pressure sequences and 0.0008 for high pressure sequences, and these values are identical to those used in NUREG-1150.

Mode of Early Containment Failure

This node is a risk-dominant question that treats early containment overpressurization failure. The effect of the blowdown of the primary system, DCH, hydrogen combustion, and rapid steam generation in the cavity are considered in determining the containment pressurization loads.

Three possible containment failure modes were considered possible, namely, a leak (leak size of 0.1 ft^2), a rupture (leak size of 1 ft^2), and a catastrophic rupture (leak size of 7 ft^2). Based on NUREG-1150 estimates, containment leakage was assumed to be the most likely failure mode below a failure pressure of 135 psig, and containment ruptures were considered to be the likely failure mode for the pressure range of 135-150 psig.

Early containment failure is assumed to be dependent on the following processes and phenomena:

1. Containment pressure prior to vessel breach,
2. Containment pressurization due to blowdown,
3. Amount of hydrogen produced in-vessel,
4. Fraction of mass participating in DCH, and
5. Extent of hydrogen burn at vessel breach.

The node that is used to estimate containment pressure prior to vessel breach is a summary CET node, and three containment pressures prior to vessel breach were defined, namely, low, intermediate and high. Low containment pressure prior to vessel breach represents all sequences which have functioning sprays and successful heat removal. High containment pressure is assumed to be characteristic of sequences that involve a depressurized RCS and do not have functioning sprays. The intermediate regime is assumed to occur for sequences that are not depressurized prior to containment failure, and do not have sprays.

Containment pressurization due to blowdown is assumed to be dependent only on the RCS pressure prior to vessel breach. "Low" pressure rise is assumed for sequences that involve depressurization prior to vessel breach. "High" pressure rise represents sequences that have RCS pressure greater than 200 psig prior to vessel breach.

In-vessel zirconium oxidation determines the amount of hydrogen generated prior to vessel breach, and two discrete regimes were selected, namely greater than 40% in-vessel oxidation, and less than 40% oxidation. A review of MAAP results indicated that the extent of in-vessel oxidation was more than 40% for all analyzed accident sequences except the large LOCAs.

The fraction of the corium mass participating in DCH is considered in the submittal to be the most important parameter affecting containment pressurization due to DCH. The submittal divides the fraction of mass participating in the DCH into three classes, i.e., nominal (25%), high (50%) and very high (75%). The probability associated with each of these classes is 0.9, 0.09, and 0.01 for nominal, high and very high respectively. These choices were made after a review of the results of the analyses performed for NUREG-1150. A simplified rationale of the choice of the mass of the debris participating in DCH is given in Section 4.5.2.3 of the submittal.

Hydrogen burn associated with DCH events is a significant contributor to containment loads. Given that DCH has occurred, it is possible that a hydrogen burn can occur since the hot debris can act as a source of ignition. The licensee considers that for DCH events, two possible outcomes are possible, namely, burns limited by local flammability conditions in the containment, and unconditional burns. Probabilities varying from 0.1 to 0.7 are assigned for unconditional burns.

It should be noted here that the mass of the debris participating in DCH is not the only parameter that governs DCH-induced containment loads. There are other parameters, the most important being the mass of zircalloy (and steel) in the debris, which govern containment pressurization. The mass of co-dispersed water and the geometry of the cavity and the lower compartment, all play significant roles in determining containment loads. The licensee has acknowledged that parameters such as the mass of codispersed water will play a role in determining the pressure loads due to DCH, but the licensee has not been able to obtain quantitative estimates for pressurization due to these effects. The submittal does not consider the effect of all of these additional parameters, which have significant uncertainties associated with them. However, a recent study sponsored by the NRC [12] extrapolated the DCH calculations performed for the Surry plant to the North Anna plant, and estimated the conditional probability of containment failure due to DCH is approximately 0.1%. Hence, the results of the North Anna IPE submittal (calculated conditional probability of containment overpressure failure of approximately 1% at vessel breach) appears to be conservative.

The expected pressure for each CET branch, and the related probability of containment failure are both provided in Figure 4.5.2-4 (page 4-175) of the submittal. It is not clear how the pressurization loads have been obtained, and it appears that both MAAP results and NUREG/CR-4551 results have been used. In any event, the analyses performed have been extensive, and the licensee seems to be aware of the phenomena that govern early containment failure in PWRs with large, dry containments.

No Early Recirculation Spray Failure

Recirculation sprays can fail during the course of an accident sequence due to several reasons:

- Alpha-mode failure or early containment failure can cause failure of the sprays.
- Environmental conditions in the auxiliary building, due to containment failur and release of steam into the auxiliary building, can cause spray failure.
- Environmental conditions inside the containment can fail the spray pump motors.
- Spray recirculation function can fail due to debris entrainment into the sumps, either by blocking the mesh screens, or by passage into the pump suction and subsequent damage to the pumps.

The DET for this event is very detailed and consists of 10 top event questions. Split fractions varying from 0.01 to 0.1 appear to have been calculated for early recirculation spray failure.

Debris Cooled Ex-Vessel

Ex-vessel cooling of debris by RWST water is treated by this node. Four factors are considered to play a role in ex-vessel coolability of debris, namely, RCS pressure at vessel breach, availability of cooling water on the debris, dispersion of debris out of the cavity, and depth of the debris pool on the cavity floor. Various probabilities are assigned to the coolability of debris after considering combinations of the above-mentioned factors.

If the RCS pressure at vessel breach is greater than 200 psig, then the probability of dispersion of debris out of the cavity is assigned a value of 0.9. If the RCS pressure is less than 200 psig, then all the debris is assumed to accumulate on the cavity floor. For the branch that entails dispersion of debris out of the cavity, there are two possible choices of debris bed depths. Shallow beds refer to debris beds of depth between 10 and 25 cm. Very shallow beds have bed depth of less than 10 cm. For the "dispersed debris" case, the probability of debris bed being very shallow is 0.9, and that for the debris bed being shallow is 0.1. For the no-dispersal (low pressure) scenario, debris beds can be deep or shallow. Deep beds have a depth of greater than 25 cm. It was already noted that, if all the corium inventory was spread on the cavity floor uniformly, the resulting pool will have a depth of 23 cm. Thus, the submittal has included scenarios that involve localized accumulation of debris and considers the possibility of non-coolability of the resulting debris beds.

The final node in the DET considers the coolability of debris for various branches. For deep pools, it was determined that (given there is an overlying pool of water) the coolability of the pools was indeterminate, and a probability of 0.5 was assigned. For shallow beds, the probability of ex-vessel coolability was larger, and a probability of 0.9 was assigned for coolability. For very shallow beds, it is assumed that the debris is always coolable, given an

assured water supply. Even when there is no water supply, it is assumed that radiation and convection can cool very shallow pools, and a probability of 0.9 is assigned for the coolability of very shallow beds without an overlying pool of water.

Late Containment Failure

Late containment failure by overpressure either due to buildup of steam/noncondensable gases, or due to hydrogen burns is questioned by this DET. A key element of this DET is the availability or recovery of AC power and recirculation sprays. The node also depends upon debris bed coolability. Late hydrogen burns due to recovery of sprays are considered, and a simple bounding calculation is used to obtain the peak containment pressure due to a hydrogen burn that will consume all the oxygen in the containment. A peak pressure of 103 psia was calculated, and a resulting probability of 0.03 was assigned to late combustion-induced containment failure. Late containment failure is also probable when the CHR systems are not functional.

No Late Recirculation Spray Failure

This node is qualitatively similar to the early spray failure node, and the details are not repeated here. The corresponding DET is provided in Figure 4.5.2-8 of the submittal.

Long Term Containment Failure

The final node in the CET considers the possibility of basemat melt-through. Although melt-through is highly improbable, it is still considered for sequences that have no sprays or heat removal, and for which the debris is not cooled ex-vessel. Under these circumstances, for deep pools, the probability of basemat melt-through is assigned a value of 0.25. For shallow pools, the corresponding value is 0.05. The source for these values is stated to be NUREG/CR-4551.

In summary, the CET analyses are very detailed and make extensive use of the NUREG-1150 analyses performed for the Surry plant.

2.2.3 Containment Failure Modes and Timing

The North Anna IPE submittal provides a comprehensive discussion of containment failure modes and timing in Section 4.4.1. The IPE submittal utilized the results from the Surry NUREG-1150 analyses for the containment fragilities. Accordingly, the capacities varied from 93 psig (5-th percentile) to 149 psig (95-th percentile), with a mean of 128 psig. A comparison of the Surry and North Anna containments is also provided in Section 4.4 of the submittal, and there is no reason to believe that the containment capacity calculated for the Surry plant is not applicable to the North Anna plant.

The effect of elevated temperatures upon containment penetrations in the North Anna containment is discussed briefly in Reference [11]. O-rings are used in the equipment hatch,

the personnel air lock, and the fuel transfer tube. The licensee stated that "the O-ring material performance in a high temperature environment was not specifically evaluated as a part of the IPE because this issue was not raised in any of the NUREG/CR-4550 documentation".

2.2.4 Containment Isolation Failure

A simplified analysis of the containment isolation system is presented in Section 4.4.1. It should be noted that the North Anna containment is a sub-atmospheric containment maintained at 10 psia, and hence, the probability of containment isolation failure is not anticipated to be large. The NUREG-1150 analyses for Surry (which has a similar, sub-atmospheric containment) did not treat isolation failures in detail. The IPE submittal considers the possibility of a leak from the containment due to isolation valves failing to close in penetrating fluid lines. Four lines to the atmosphere from the containment were identified. It was assumed that the normal isolation arrangement is two valves in series per line. A generic value of 1.1×10^{-2} per valve for failure to close was assumed. A fault tree model for the containment isolation system is presented in page A-313 of the submittal. The total calculated frequency for core damage sequences with containment isolation failure is 1.5×10^{-7} per reactor year, or about 0.2% of the internal events CDF [1,11]. This frequency is similar to that calculated in other IPE submittals.

Containment bypass was also analyzed as part of the IPE. The maximum size of the piping connected to the RCS, and the flow restrictors in the piping, was determined to be six inches. Instead of performing an independent analysis for the North Anna plant, the licensee makes use of the NUREG-1150 results for the Surry plant. The expert panel determined that the frequency of the interfacing systems LOCA was 1.6×10^{-6} per reactor year for Surry, and the same value was used for the North Anna IPE submittal. In addition, unisolated SGTRs were also evaluated, and they are the dominant contributors to containment bypass (7.01×10^{-6} per reactor year, or 10.3% of the internal events CDF).

2.2.5 System/Human Response

No explicit discussion of operator actions in the back-end analyses can be found in the North Anna IPE submittal. In fact, no operator actions were considered in the back-end analyses. The licensee states that this approach was taken because there were no severe accident management procedures in place [11]. Since the recovery actions had not been proceduralized, the licensee decided not to take any credit for recovery actions after core damage [11].

2.2.6 Radionuclide Release Categories and Characterization

The endpoints of the CETs represent the outcomes of possible in-containment accident progression sequences. These endpoints are classified into a manageable number of release categories, characterized by similarities in accident progression and source term characteristics. Associated with each source term category is a release frequency and magnitude of radionuclide release. The main characteristics of the CET end-states considered when developing these release categories in the submittal were:

- Containment bypass (SGTR/Event V)

  CET end-states corresponding to bypass sequences were grouped using this attribute. For V-sequences, the effectiveness of fission product scrubbing by the auxiliary building is also considered.

- Debris cooled in-vessel

  If the debris is cooled in-vessel, and the containment is isolated, fission product release is small.

- Alpha-mode containment failure

  In-vessel steam explosion and missile formation results in a direct path to the environment, and hence the associated end-state was assigned to a separate bin.

- Containment isolation

  This attribute is considered important because pre-existing isolation failures do not permit effective fission product deposition in the containment. Thus, even if containment failure were to occur later in the course of the accident sequence, the magnitude of fission product release is not increased.

- Time of containment failure

  Three time periods were defined in the submittal, namely, early, late and very late (i.e., basemat melt-through).

- Time period of spray operation

  Four possible time periods over which the recirculation sprays are assumed to operate are defined, namely, continuous, early only, late only, and never.

- Mode of containment failure

  Two modes of containment failure were defined, and they are leak and rupture.

- Fission product attenuation effectiveness of the auxiliary building

  This attribute defines the effectiveness of the auxiliary building for scrubbing fission product releases. It is defined for V sequences only.

A "source term logic" diagram is used to classify the CET end states, and is given in Figure 4.7.2-1 (page 4-227) of the submittal. A total of 24 source term categories were found to result.

The magnitude of the releases for each release category were determined by the source term calculations performed for the Surry IPE submittal. For the Surry IPE, MAAP calculations were performed for a selected number of accident sequences representative of the source term bins. For other categories, the release fractions were characterized by results from analyzed sequences that were similar. Calculations were performed for 10 of the 24 release bins.

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 in the IPE submittals. Substantial information is provided in the submittal and the response to the NRC RAIs [11] to obtain the required information.

Only four source term bins appear to have release magnitudes close to the PWR-4 release category of WASH-1400, and they are STCs 22, 23, 24 and 8. STCs 22 and 23 include interfacing systems LOCA sequences (with and without scrubbing by an overlying water pool in the safeguards building, respectively). STC 24 is the source term bin that corresponds to steam generator tube ruptures. The releases are larger than PWR-4 releases, and the frequency of the release category is 7.38 x 10^{-6} per reactor year. The list of sequences that comprise this source term category is not provided in the submittal, however, the breakup of the bin frequency by initiating event is provided in Table 4.7.4-2 of the submittal. From this table, and from the discussion provided in Section 3.4.1.3, it was possible to identify two sequences that meet the GL 88-20 reporting criteria. The sequences in STC 8 have frequencies smaller than 10^{-6} per reactor year. Hence, only the sequences that are binned into STCs 22 and 24 need to be reported. The final list of sequences that meet the reporting criteria is provided in Table 3.

Release category 24 corresponds to SGTRs, and has the largest releases. The SGTR sequences also have a direct pathway to the atmosphere, and the releases are unscrubbed as the steam generator secondary side inventory is boiled dry prior to core damage. There is little mitigation in the secondary system. Large releases are calculated for the interfacing systems LOCA when the release point is not submerged (STC 23). Releases start earlier than for the SGTR sequence. The next largest release is predicted for STC 8, which involves early containment failure with no spray operation. Smaller releases are predicted for STC 7, which is the source term bin corresponding to an early containment leak. All other bins have releases that are orders of magnitudes smaller than the four mentioned above.

A comparison of releases for the steam generator tube rupture sequences between the IPE submittal results and the results available in the literature for other plants based on STCP [6] and MELCOR [7], is shown in Table 4. The results indicate that the iodine and cesium releases reported in the submittal are larger than the STCP results or the MELCOR results. The reported tellurium releases are two orders of magnitude smaller for the North Anna calculations. The submittal also reports strontium, barium, cerium and lanthanum releases, which are one order of magnitude higher than the Surry BCD results. The differences in the releases for the volatile species such as iodine and cesium are probably due to the high degree of revaporization

Table 3 A Listing of Sequences That Meet the Reporting Criteria of GL 88-20

Sequence Title	Frequency (Per RY)	Description	Release Fraction of CsI
T7002	2.99×10^{-6}	SGTR with failure of operator to cooldown and depressurization: Stuck-open relief valve	0.53
VxFm	1.52×10^{-6}	Interfacing Systems LOCA	0.05 - 0.29
T7SGIW	1.1×10^{-6}	SGTR with failure to isolate faulted steam generator	0.53

Table 4 Comparison of Releases for Surry (SGTR Sequence) and North Anna (SGTR Sequence

	W PWR (MELCOR)	Surry (STCP)	North Anna (MAAP)
Group	ERI [7]	BCD [6]	IPE [1]
I	0.28	0.25	0.52
Cs	0.28	0.25	0.52
Te	0.06	0.08	2.6E-3
Sr	3E-3	2E-4	0.034
Ru	1E-4	3E-7	NA+
La	1E-5	2E-8	5.5E-5
Ce	2E-5	0.0	5.2E-3
Ba	NA+	3E-3	0.021

predicted by the MAAP code. The differences in the releases of strontium and barium are probably due to the differences in the treatment of MCCI in the MAAP and CORCON codes.

Realizing that the tellurium releases predicted by the MAAP code are not comparable, the licensee has toggled the in-vessel tellurium release model in the MAAP code, and reports that the releases increased from 0.0026 to 0.55. However, it appears that these releases may also be in error.

Table 5 makes a comparison of releases for the V sequence with the break exit above the water level in the auxiliary building. The releases for the cesium and iodine species are comparable between the NRC codes and the MAAP code. The releases for tellurium are considerably smaller, and the licensee is aware of the fact that the MAAP-predicted releases of tellurium are small. All other reported releases in the submittal are larger than those calculated by STCP. Table 6 makes a comparison of the releases for early containment overpressure failure. In

general, the releases reported for the volatile species are comparable, but the releases reported for tellurium, barium and strontium are considerably smaller.

Table 5 Comparison of Releases for Surry (V Sequence) and North Anna (V Sequence, Release Category 23)

	Surry (STCP)	Surry (STCP)	North Anna (MAAP)
Group	BCD [6]	BNL [8]	IPE
I	0.29	0.40	0.3
Cs	0.26	0.40	0.3
Te	0.05	0.12	1.6E-5
Sr	5E-3	-	0.23
Ru	2E-7	7E-4	NA*
La	4E-4	-	4E-04
Ce	4E-4	-	4E-2
Ba	3E-3	0.01	0.2

Table 6 Comparison of Releases to Other Studies (TMLB' Sequence, Containment Failure At or Around Vessel Breach)

	W PWR (MELCOR)	Surry (STCP)	North Anna (MAAP)
Group	ERI [7]	BCD [6]	IPE[1]
I	0.21	0.07	0.11
Cs	0.15	0.06	0.15
Te	0.34	0.06	0.017
Sr	0.13	0.01	0.02
Ru	3E-4	1E-3	NA
La	3E-3	2E-4	2E-5
Ce	1E-5	-	3E-4

* Not Available

2.3 Quantitative Assessment of Accident Progression and Containment Behavior

2.3.1 Severe Accident Progression

The MAAP 3.0B code was used to evaluate the integrated containment response and the severe accident source terms. The MAAP parameter input file is included in Appendix E of the submittal. However, the majority of calculations appear to have been performed for the Surry plant, and only a limited number of North Anna-specific calculations appear to have been performed. The MAAP analyses performed for severe accident analyses and source term quantification are quite detailed, and a total of 23 sequences were analyzed. The choice of sequences to be analyzed was based on the CET analyses in order to support the quantification of split fractions for the CET, and to enable the proper quantification of source terms for all important end states. A listing of the analyzed accident sequences is provided in Table 4.6.1-1 (page 4-123) of the submittal. Table 4.6.1-2 of the submittal provides a listing of the timing of key events, and Table 4.6.1-3 of the submittal provides results for selected calculated parameters. However, it appears that the results presented in these tables are for the Surry plant.

Only a limited use appears to have been made of the results from the MAAP calculations. It appears that the MAAP results for the following parameters are used in the CET analyses:

1. RCS pressure at vessel breach,

2. Time period for vessel failure and to reach the fifth percentile containment failure pressure (used in the power recovery node of the CET),

3. Containment pressure prior to vessel breach,

4. Containment pressure prior due to blowdown, and

5. In-vessel hydrogen generation.

In addition, the source terms were also evaluated using the MAAP code, but using the Surry plant model. However, it must be noted that only limited sensitivity calculations have been performed with regards to the deterministic severe accident analyses. The performed sensitivity analyses include a study on the effect of the blockage model upon in-vessel hydrogen generation and the effect of altering the tellurium release model upon the source term. Thus, the number of sensitivity analyses performed is very limited, and many of the recommended sensitivity analyses in the EPRI document [3] were not performed. It should be noted that the Surry and the North Anna containments differ slightly in free volume, and the licensee should have scaled the Surry results for containment pressure prior to and at vessel breach properly before using them for the North Anna CET analyses. The licensee noted that [11] the results (for containment pressure) from the Surry analyses were multiplied by a factor of 1.17 to account for the differences in the rated power and the RC volume between the two plants.

2.3.2 Dominant Contributors to Containment Failure

Table 7 of this review shows a comparison of the conditional probabilities of the various containment failure modes of the North Anna IPE submittal with the Surry and Zion/NUREG-1150 results. All comparisons are made for internal initiating events only.

The North Anna core damage frequency for internal events is slightly larger than that calculated by NUREG-1150 for Surry and Zion [4,5]. The conditional probability of early containment failure (due to overpressurization) in the North Anna plant is 1.3%, and is twice as large as the values calculated by NUREG-1150 analyses for Zion or Surry. Transients and station blackout sequences are significant contributors to early containment failure. The licensee attributes the higher probability of early containment failure in the North Anna IPE, as compared to the Surry results, to the larger core inventory, power and RCS water volume in the same containment volume. In any event, it appears from the results that the use of NUREG-1150 analyses have resulted in consistent results for the North Anna plant.

Table 7 Containment Failure as a Percentage of Total CDF: Comparison with Other PRA Studies

Containment Failure Mode	North Anna IPE	Surry NUREG-1150	Zion NUREG-1150
Early Failure	1.3	0.7	0.5
Late Failure	10.0	5.9	24.0
Very Late Failure	1.1	NA^{+}	NA^{+}
Bypass (V)	2.4	7.6	0.2
Bypass (SGTR)	11.6	4.6	0.3
Isolation Failure	0.2	NA^{++}	1.0
Intact	74.0	81.2	73.0
Core Damage Frequency, yr^{-1}	6.8×10^{-5}	4.1×10^{-5}	6.2×10^{-5}

++ Included as a Part of Early Containment Failure
+ Included as a Part of Late Containment Failure

The submittal also calculates a higher probability of late containment failure by overpressure (10.2%) and basemat melt-through (1.1%) than the Surry NUREG-1150 analyses. The licensee attributes the difference in results to the consideration of the loss of emergency room cooling, both as an initiator, and as a support system in the front-end analysis [11]. Loss of switchgear room cooling leads to station blackout since the emergency buses are lost. If AC power is not recovered, or if safety injection systems fail (due to random causes) after AC power recovery, the accident sequence progresses to vessel breach, relocation of core debris to the cavity, and slow pressurization of the containment, leading to late containment failure.

In addition, the calculated conditional probabilities of pre-existing steam generator tube ruptures (10.3%), induced steam generator tube ruptures (1.3%) and interfacing systems LOCA (2.4%) are all larger for the North Anna submittal than for the NUREG-1150 analyses. The differences can be attributed primarily to the differences in the core damage profile between the three analyses. It appears that the methods and assumptions used for containment analyses in the North Anna submittal and the Surry NUREG-1150 report are, for the most part, similar.

The dominance of the SGTR sequences on the release profile of the North Anna plant is to be noted. Another important point to be noted here is that a small, but significant fraction of recovered sequences involve the maintenance of the core in an undamaged state by using the SI systems in the recirculation mode, but without an operable means of CHR. The SI pumps are qualified for pumping water at temperatures in excess of 300°F. If the pumps were not capable of functioning at these temperatures, then core damage and containment failure will occur in rapid succession for these sequences. In addition, the CDF will increase by approximately 10^{-5} per reactor year.

2.3.3 Characterization of Containment Performance

The top ten PDSs with the highest CDF are PDSs 21, 4, 20, 25, 14, 12, 23, 5, 3, and 24. Together the top 10 PDSs contribute to more than 91% of the internal events CDF. The overall results can be understood from a discussion of the results for the top 6 PDSs.

PDS 21, with 28.3% of the internal events CDF, includes small break LOCAs with AC power and recirculation sprays available. Since the RCS pressure is less than the setpoint pressure, the probability of induced hot leg failure is very small. The probability of in-vessel coolability is also negligible, since the sequences involve failure of high and low head injection. Subsequently, the conditional probability of vessel breach is close to 1. However, the probability of early containment failure is low (~ 0.6%), since all sequences in this PDS have functional CHR. The probability of late and very late containment failure is low, and there is a very high probability of intact containment (~ 98%).

PDS 4 (18.6% of the CDF) includes station blackout sequences that involve recovery of sprays after vessel breach. There is a high probability of induced RCS failure (72%) for this PDS. Since AC power is not available early, there is no possibility of in-vessel coolability. The branch of the CET that involves RCS pressure boundary failure, entails no early containment

failure. However, the branch of the CET that does not involve induced failure of the RCS pressure boundary has a considerable probability of early containment failure. The conditional probability of early containment failure for this PDS is 1.6%, while the conditional probability of late containment failure and basemat melt-through is very low for this PDS due to the recovery of CHR.

PDS 20 (12.1% of the CDF) involves small break LOCA sequences with low pressure injection available. The RCS pressure at core damage is below 2000 psig, but above the setpoint of the low pressure injection pumps. Induced failure of RCS is not possible, and the CET results are quite similar to that for PDS 21.

PDS 25 (10.3% of the CDF) involves SGTR sequences, and the end-state of this PDS is the containment bypass bin. No recovery is possible, and containment failure is irrelevant to this PDS.

PDS 14 (4.3% of the CDF) involves sequences with a loss of offsite power or loss of emergency switchgear room cooling. However, the RCS pressure boundary is not intact at core damage due to seal LOCA, and system pressure is below the PORV setpoint pressure. Hence, induced RCS depressurization is assumed not possible. Since alternate sources of power are available, there is a high probability of in-vessel core coolability. The probability of early containment failure for this PDS is 0.2% and the probability of no containment failure is very high (~ 99%).

PDS 12 (4.2% of the CDF) involves large LOCA sequences. The conditional probability of containment failure is very low since recirculation sprays are operating. The only two modes of containment failure are alpha-mode failure (0.04%) and basemat melt-through (0.3%).

In summary, it appears that loss of offsite power sequences and station blackout sequences (induced by loss of offsite power as an initiator, by flooding, and by loss of emergency room cooling as an initiator) are the principal contributors to early containment failure. Transients are the principal contributors to late containment failure. However, the principal mode of releases is due to steam generator tube ruptures (pre-existing or thermally-induced).

2.3.4 Impact on Equipment Behavior

The impact of the accident progression on the performance of some of the equipment after core damage was considered as a part of the CET analyses in the IPE submittal. The submittal has concentrated on the failure of sprays (and the associated recirculation pumps) in the containment analyses. Two DETs determine the performance of the recirculation sprays in the early and the late phase of the accident. The sprays are assumed to fail due to a number of reasons:

1. Containment failure causes spray failure,

2. Excessive debris in the sump plugs the spray headers or fails the spray pumps, and

3. Environmental conditions inside the containment, and outside the containment (due to containment failure and blowdown of steam into the safeguards building), causes the recirculation sprays to fail.

The temperature, pressure and humidity inside the containment and the safeguards building were assessed by a review of MAAP calculations, and compared with the design information available for the spray pumps. The submittal assigns split fractions for the failure of the spray pumps under different severe accident conditions.

However, the submittal does not consider the failure of any other equipment (such as ECCS recirculation function) under severe accident conditions.

2.4 Reducing the Probability of Core Damage and Fission Product Releases

2.4.1 Definition of Vulnerability

The submittal attempts to define vulnerability in Section 3.4.2 (page 3-132) of Reference [1]. In this section, it is said that "a concise definition of vulnerability is not given in the documentation associated with the performance and reporting of the IPE". Hence, the licensee attempts to define vulnerability (as pertaining to core damage) as "a failure (component fault or human error) that is significantly greater than others, i.e., contribute more than 10% to the CDF or must be a factor of three greater than the next highest event". No such definition is provided for a vulnerability as pertaining to the back-end analyses. However, in Section 3.4.2.3 of the submittal, the licensee notes that the containment structure and mitigation systems are robust, and the containment capacity is large such that the probability of early containment failure is low. Redundancy and diversity of containment spray systems leads to the fact that the probability of failure of the CHR systems is low. In addition, since North Anna is a sub-atmospheric containment, the probability of having an unisolated containment is also low. Thus, it appears that the licensee perceives that no containment vulnerabilities exist for the North Anna plant.

2.4.2 Plant Modifications

The licensee has committed to a number of plant improvements based on the review of the results of the submittal. Complete details of the improvements can be found in Section 6.1.2 (page 6-2) of the submittal and the improvements are summarized in Table 6-1 (page 6.7) of the submittal. To summarize briefly, a number of procedural modifications were deemed necessary, and they include modifications in the testing of auxiliary feedwater flow, procedures for realignment of quench and recirculation spray headers, revision of procedures to verify SI flow, revision of testing procedures for the low head safety injection pump, and improvement and modification of the room chiller repair and testing procedures.

In addition, a number of plant hardware modifications to prevent flooding in the charging pump cubicle, improvement of the fire barrier between the spray pump house and the auxiliary

building, and hardware modification in the chiller room, are all planned. All the modifications (hardware and procedural) are essentially complete.

2.5 Responses to CPI Program Recommendations

Generic Letter 88-20, Supplement Number 1 [9] and Number 3 [10] identify specific Containment Performance Improvements (CPIs) to reduce the vulnerability of containments to severe accident challenges. One of the recommendations of the CPI program pertaining to PWRs with large dry containments is 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 RAIs, the licensee addressed the issue of local and global hydrogen combustion, and associated threats to containment. The licensee stated that hydrogen buildup to sufficient concentrations that can lead to combustion and failure of the containment was unlikely in most accident sequences, since a number of ignition sources are available. However, this situation can occur in a station blackout sequence with late AC power recovery. The licensee stated that a very conservative model for this sequence led to a calculation of 128 psig for containment load, which resulted in a containment failure probability of 0.03. The licensee also stated that buildup of hydrogen concentration to a level that can sustain a detonation was not possible in the North Anna containment.

2.6 IPE Insights, Improvements, and Commitments

The licensee has obtained some insights based on the analysis of accident progression, containment response, and radionuclide release [1,11]. The following are the unique safety features (that the licensee has identified) as important to the containment performance:

- The high containment capacity leads to the calculation of lower probabilities of early containment failure (due to hydrogen burn, direct containment heating, etc.)

- The operation of the containment at sub-atmospheric pressure results in the calculation of a low probability of containment isolation failure.

- The redundancy and diversity of recirculation spray system considerably reduces the probability of the failure of the CHR system, and reduces the probability of late overpressure failure of the containment. In addition, the operation of sprays also reduces fission product releases.

- The piping arrangement in the auxiliary building is such that most of the V-sequences will lead to releases under water.

- The design of the cavity and the instrument tunnel is unfavorable to the dispersion of debris to the lower containment. The same design also does not permit water into the cavity unless the sprays function.

The licensee has also concluded from the submittal that a significant fraction of recovered sequences involve the maintenance of the core in an undamaged state by using the SI systems in the recirculation mode, but without an operable means of CHR. The SI pumps are qualified for pumping water at temperatures in excess of 300°F. If the pumps were not capable of functioning at these temperatures, then core damage and containment failure will occur in rapid succession for these sequences. In addition, the CDF will increase by approximately 10^{-5} per reactor year.

As a result of the overall strength of the containment, the dominant pathways for release to the environment are the SGTR and the ISLOCA sequences. The frequency and magnitude of releases associated with the SGTR sequences tend to dominate the overall releases. The SGTR sequences involve failure of the operator to cool down and depressurize the RCS early in the accident. For such sequences, there is a potential for the steam generator PORV to stick open due to steam generator overfill. Thus, the licensee concludes that the most significant insight from the back-end analyses is the importance of regulating the level in the steam generators. The licensee also stated that [11] "the importance of the scenario has been transmitted to the North Anna training department for use in developing training scenarios". The licensee also noted that replacement of steam generators in the North Anna plant is near completion, and that the reliability of the new steam generators is expected to be good [11].

The licensee has committed to a number of plant improvements based on the review of the results of the submittal. Complete details of the improvements can be found in Section 6.1.2 (page 6-2) of the submittal and the improvements are summarized in Table 6-1 (page 6.7) of the submittal. A number of plant and procedural modifications were made, and these are discussed in Section 2.4.2 of this TER.

3. OVERALL EVALUATION AND CONCLUSIONS

The back-end portion of the North Anna IPE submittal provides a substantial amount of information in regard to the subject areas identified in Generic Letter 88-20 and NUREG-1335. The submittal appears to be complete and to provide the level of detail requested in NUREG-1335. The approach and methodology are described clearly in the submittal, and rely extensively on the NUREG-1150 analyses for the Surry plant. The submittal uses a large event tree methodology to perform the probabilistic segment of the back-end analyses, and makes use of the results from the MAAP analyses for the Surry plant to support the CET analyses. Extensive use is made of the NUREG-1150 analyses performed for the Surry plant for the quantification of the split fractions for the CET.

The important points of the technical evaluation of the North Anna IPE back-end analyses are summarized below:

- The results for the CDF are a factor of two larger than those reported in NUREG-1150 for the Surry plant. The differences are attributed to the identification of an additional initiating event (loss of switchgear room cooling) as an important contributor at North Anna, and differences in assumed success criteria for small break LOCAs and SGTRs.

- The Back-End portion of the IPE supplies a substantial amount of information with regards to the subject areas identified in Generic Letter 88-20. The North Anna IPE provides an evaluation of all phenomena of importance to severe accident progression in accordance with Appendix I of the Generic Letter.

- Phenomena that lead to early containment failure (i.e., DCH, steam explosions) have been treated in a manner very similar to NUREG-1150. However, the calculated conditional probability of early containment failure is slightly larger for the North Anna IPE submittal. The differences can be attributed to two reasons. The first reason is that the relatively larger core and RCS in the North Anna plant in a similar containment (as Surry) leads to the calculation of a higher conditional probability of early containment failure. The second reason is the consideration of the loss of emergency room cooling, both as an initiator, and as a support system in the front-end analysis for North Anna. Loss of switchgear room cooling leads to station blackout since the emergency buses are lost. The increased contribution of station blackout sequences lead to a slightly increased conditional probability of early containment failure.

- The submittal reports a conditional probability of about 10% for late containment overpressure failure, and a probability of about 1% for basemat melt-through. Once again, these values are larger than those obtained from the Surry NUREG-1150 analyses, and they are attributed to the consideration of the loss of emergency room cooling, both as an initiator, and as a support system in the front-end analysis for North Anna.

- The radiological releases for the North Anna plant are dominated by steam generator tube rupture sequences. The release frequency of the SGTR sequennces is 7.9×10^{-6} per reactor year (11.6% of the CDF). It is noted that the release frequency of the SGTR sequences is significant, and the magnitude of the associated releases are large.

- The licensee has addressed the recommendations of the CPI program.

5. REFERENCES

1. "Probabilistic Risk Assessment for the Individual Plant Examination Final Report North Anna Power Station Units 1 and 2," prepared by Virginia Electric and Power Company, (December 1992).

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

3. Kenton, M. K., and Gabor, J. R., "Recommended Sensitivity Analyses for an Individual Plant Examination Using MAAP 3.0B," EPRI report (1988).

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

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

6. R. S. Denning et al., "Radionuclide Release Calculations for Selected Severe Accident Scenarios, Supplemental Calculations," U. S. Nuclear Regulatory Commission, NUREG-4551, Vol. 6 (August 1990).

7. M. Khatib-Rahbar, R. Vijaykumar, I. K. Madni, E. G. Cazzoli, H. P. Isaac, and U. Schmocker, "Simulation of Severe Reactor Accidents: A Comparison of MELCOR and MAAP Computer Codes", Paper Presented at the ANS Probabilistic Safety Assessment International Topical Meeting, Clearwater Beach, Florida, January 26-29 (1993).

8. E. G. Cazzoli et al., "Independent Verification of Radionuclide Release Calculations for Selected Accident Scenarios," U. S. Nuclear Regulatory Commission, NUREG-4629 (July 1986).

9. 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.

10. 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.

11. "Responses to NRC Questions," prepared by Virginia Electric and Power Company, Letter from M. L. Bowling, Virginia Electric and Power Company, dated April 27, 1995.

12. 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

North Anna

Containment Type

Large, dry containment operating at sub-atmospheric pressure

Unique Containment Features

Operation of containment at sub-atmospheric pressure

Multiple, redundant sprays

Unique Vessel Features

None found

Number of Plant Damage States

25

Containment Failure Pressure

Dry 128 psig (mean)

Additional Radionuclide Transport and Retention Structures

Auxiliary building structures have not been credited.

Conditional Probability That The Containment Is Not Isolated

0.0022

Important Insights Including Unique Safety Features

Operation of containment at sub-atmospheric pressure

Multiple, redundant recirculation sprays

Auxiliary building design that leads to radionuclide release under water for interfacing systems LOCAs

Cavity design that is unfavorable to the dispersion of debris form the cavity to the containment

Implemented Plant improvements

No containment-related improvements were found necessary

C-Matrix

Definition of 25 PDSs and 24 STCs makes the display of the results in the form of the C-matrix very cumbersome. The results are provided in Appendix F, Figures F.4-1 through F.4-24 of the submittal.