Docket Files

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 2, 1994

Docket No. 50-219

Mr. John J. Barton
Vice President and Director
GPU Nuclear Corporation
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Barton:

SUBJECT: REVIEW OF OYSTER CREEK NUCLEAR GENERATING STATION INDIVIDUAL PLANT EXAMINATION (IPE) SUBMITTAL (TAC NO. M74443)

Enclosed is the staff's evaluation of GPU Nuclear Corporation's (GPUN) Oyster Creek IPE for internal events and internal flood. The evaluation package consists of: a Staff Evaluation Report (SER) (Enclosure 1); and contractor Technical Evaluation Reports (TERs) for the front-end, back-end, and human reliability analysis reviews (Enclosures 2, 3, and 4).

Based on our review, we conclude that GPUN has met the intent of Generic Letter 88-20, and we do not recommend that a further review be conducted. However, our review identified a deficiency (lack of treatment of pre-initiators) in the human reliability analysis portion of the IPE which may limit the IPE's usefulness in other applications. In addition, GPUN plans to address a number of potential operator mitigation actions during its accident management development phase, specifically the need for the interconnection between the fire protection water system and the drywell spray system.

We would also like to mention that GPUN did not explicitly state that they plan to maintain their Probabilistic Risk Assessment (PRA) "living." The staff notes that a "living" PRA could enhance plant safety and provide additional assurance that any potentially unrecognized vulnerabilities would be identified and evaluated during the life of the plant.

By this letter we are closing TAC No. M74443.

Sincerely,

Original signed by

Alexander W. Dromerick, Sr. Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Staff Evaluation Report
2. TER (Front-End)
3. TER (Back-End)
4. TER Human Reliability
   Analysis

cc w/enclosure 1:
See next page

DISTRIBUTION: w/enclosures 1, 2, 3, 4:
Docket File NRC & Local PDRs
PDI-4 Plant ADromerick
RHernan ELois, RES
JRidgely, RES
JChung, RES

DISTRIBUTION: w/enclosure 1:
SVarga
JCalvo
SNorris
OGC
ACRS (10)
JRogge, RGI

OFFICE	LA:PDI-4	PM:PDI-4	D:PDI-4		
NAME	SNorris	ADromerick:bp	JStolz		
DATE	8/1/94	8/1/94	8/2/94		

OFFICIAL RECORD COPY
Document Name: G:\DROMERIC\M74443.CLO

By this letter we are closing TAC No. M74443.

Sincerely,

Alexander W. Dromerick

Alexander W. Dromerick, Sr. Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Staff Evaluation Report
2. TER (Front-End)
3. TER (Back-End)
4. TER Human Reliability Analysis

cc w/enclosure 1:
See next page

Mr. John J. Barton
Vice President and Director

Oyster Creek Nuclear
Generating Station

cc:

Ernest L. Blake, Jr., Esquire
Shaw, Pittman, Potts & Trowbridge
2300 N Street, NW.
Washington, DC 20037

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

BWR Licensing Manager
GPU Nuclear Corporation
1 Upper Pond Road
Parsippany, New Jersey 07054

Mayor
Lacey Township
818 West Lacey Road
Forked River, New Jersey 08731

Licensing Manager
Oyster Creek Nuclear Generating Station
Mail Stop: Site Emergency Bldg.
Post Office Box 388
Forked River, New Jersey 08731

Resident Inspector
c/o U.S. Nuclear Regulatory Commission
Post Office Box 445
Forked River, New Jersey 08731

Kent Tosch, Chief
New Jersey Department of
Environmental Protection
Bureau of Nuclear Engineering
CN 415
Trenton, New Jersey 08625

Jack S. Wetmore
TMI Licensing Manager
GPU Nuclear Corporation
Post Office Box 480
Middletown, Pennsylvania 17057

ENCLOSURE 1

STAFF EVALUATION OF THE OYSTER CREEK INDIVIDUAL PLANT EXAMINATION

(IPE)

(INTERNAL EVENTS ONLY)

TABLE OF CONTENTS

EXECUTIVE SUMMARY

The NRC staff completed its review of the internal event portion of the Oyster Creek IPE submittal and associated documentation which includes GPU Nuclear Corporation's (GPUN/licensee) responses to staff generated questions and request for additional information.

The licensee's IPE is based on a Level 1 and 2 Probabilistic Risk Assessment (PRA) consistent with Generic Letter 88-20, Appendix 1. The PRA was performed by PLG Inc., with the support from other consultants. GPUN personnel familiar with detail design, controls, procedures, and systems maintained involvement in the development, analysis, and technical reviews of the Oyster Creek PRA models.

The Oyster Creek IPE did not identify any severe accident vulnerabilities which the licensee defined as any core damage sequence that exceeds 1E-4 per reactor year, or containment bypass that exceeds 1E-6 per reactor year. The IPE did, however, take credit for a number of modifications that were installed during the 14R refueling outage. These include the interconnection to the combustion turbine generators at the adjacent Forked River Site; hard piped containment vent system; and operator training for manual initiation of the containment spray system.

The IPE estimated the total mean core damage frequency (CDF) from internal events including internal flood as 3.96E-6/yr. Dominant initiating events and their percent contribution (%) to CDF include loss of offsite power (32.8%), turbine trip (13.1%), and reactor trip (7.7%). IPE importance measures identified failure of electromatic relief valves (EMRV) to close as the largest component contributor to total CDF (48%). The significance of this contributor stems from the success criteria which requires (for many accident initiators) opening and subsequent closing of up to 4 of 5 EMRVs. Essential AC power bus failures had also been found to be an important contributor (37%) to core damage. In addition, a number of the OC IPE dominant sequences involve loss of DC power. DC power is required to activate the isolation condenser, and open the EMRVs to allow for vessel injection with the low pressure firewater system.

The OC IPE found a relatively low station blackout induced core damage frequency of 7.7E-7/yr. The IPE basis for this low frequency primarily stems from utilization of isolation condensers and the firewater system as a source of makeup. System activation does not require AC power nor long-term DC power for extended operation. The staff review noted, however, that the IPE analysis did not specifically model recirculation pump seal loss-of-coolant accident (LOCA). This assumption substantially reduces the significance of SBO as a contributor to core damage. This finding is not consistent with NUREG-1032 "Evaluation of Station Blackout Accidents at Nuclear Power Plants." Unlike other boiling water reactors (BWRs), Oyster Creek does not have steam driven makeup capability during station blackout and, therefore, must rely entirely on natural circulation for core cooling (analogous to pressurized water reactors (PWRs) with steam generators). A pump seal LOCA under these conditions could disrupt natural circulation and compromise decay heat

removal. Although the licensee provided references to support its position on seal LOCA, the issue remains open and under staff consideration (as a possible generic issue separate from Generic Issue 23). Because the issue is being addressed separately for BWRs, the staff (IPE) review team did not pursue this aspect further.

All modelled operator actions were found to contribute 21% to core damage. The IPE, however, did not perform a pre-initiator human event analysis. Generic Letter 88-20 requested that licensees examine maintenance and surveillance practices as part of their effort to identify potential vulnerabilities. These areas are plant-specific and require an examination of routine personnel activities to uncover potential maintenance errors. The staff finds the lack of pre-initiator event analysis a weakness in the licensee's IPE, which may limit the usefulness of the IPE for future regulatory applications.

The Oyster Creek IPE takes substantial credit (50%) for in-vessel recovery following core damage. For low pressure sequences, vessel breach is prevented by injection through condensate, control rod drive system, or through the use of core spray supplied by the fire protection system. The IPE also assumes that vessel failure will result in a guaranteed containment failure. For many sequences involving extensive core damage, the IPE did not credit any operator actions. The licensee indicated its concern that the potential for adverse effects (which could result from operator mitigation action), could exceed the perceived benefit. For example, in response to staff questions on sequences involving recovery of electrical power, the licensee stated that prompt action to vent containment without proper "Accident Management Guidelines" could result in an earlier source term release than if no action was taken. Other issues associated with accident progression also remain open, e.g., the consequence of activation of drywell sprays with corium in the drywell. The licensee stated that it plans to postpone further evaluation of potential operator mitigation action to the accident management development phase "when better tools will be available (MAAP4)."

In response to containment performance improvement (CPI) program recommendations, the licensee considered a plant modification to provide water from the fire protection system to the drywell sprays and has concluded that this modification is not cost beneficial. The licensee has taken the position that the containment will always fail when the reactor vessel fails. This position may have masked the true potential benefit from enhanced drywell sprays. Other licensees have concluded that having the drywell sprays will significantly reduce the probability of drywell liner melt-through. The licensee has stated that it is unclear how operator actions will affect the accident progression, and they intend to evaluate the effects of potential operator actions when appropriate tools (MAAP4) become available. The staff recommends that the licensee continue to evaluate the need for drywell sprays as part of its accident management program evaluation.

Based on the review of the Oyster Creek IPE submittal and associated documentation, the staff concludes that the licensee met the intent of Generic Letter 88-20. This conclusion is based on the following findings: (1) the IPE is complete with respect to the information requested in Generic Letter

88-20 and associated guidance document NUREG-1335; (2) the front-end systems analysis, the back-end containment performance analysis, and the portion of human reliability analysis performed (post-initiator events) are technically sound and capable of identifying plant-specific vulnerabilities to severe accidents; (3) the licensee employed a viable means (documentation reviews and walkdowns) to verify that the IPE reflected the current plant design and operation; (4) the PRA which formed the basis of the IPE had been peer reviewed; (5) the licensee participated fully in the IPE process consistent with the intent of Generic Letter 88-20; (6) the licensee appropriately evaluated Oyster Creek's decay heat removal (DHR) function for vulnerabilities consistent with the intent of the USI A-45 resolution; and (7) the licensee responded appropriately to recommendations stemming from the CPI program.

It should be noted that the staff's review primarily focused on the licensee's ability to examine Oyster Creek for severe accident vulnerabilities. Although certain aspects of the IPE were explored in more detail than others, the review is not intended to validate the accuracy of the licensee's detailed findings (or quantification estimates) which stemmed from the study.

I. BACKGROUND

On November 23, 1988, the NRC issued Generic Letter 88-20 which requires licensees to conduct an Individual Plant Examination (IPE) in order to identify potential severe accident vulnerabilities at their plant and to report the results to the Commission. Through the examination process, a licensee is expected to: (1) develop an overall appreciation of severe accident behavior; (2) understand the most likely severe accident sequences that could occur at its plant; (3) gain a more quantitative understanding of the overall probabilities of core damage and fission product releases; and (4) if necessary, reduce the overall probability of core damage and radioactive material releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

As stated in Appendix D of NUREG-1335, the IPE submittal guidance document, all IPEs are to be reviewed by NRC teams to determine the extent to which each licensee's IPE process met the intent of Generic Letter 88-20. The IPE review itself is a two-step process; the first step, or "Step 1" review, focuses on completeness and the quality of the submittal. Only selected IPEs are investigated in more detail under a second step or "Step 2" review. The decision to go to a "Step 2" review is primarily based on the ability of the licensee's methodology to identify vulnerabilities, and the consistency of the licensee's IPE findings and conclusions with previous PRA experience. A unique design may also warrant a "Step 2" to better understand the implication of certain IPE findings and conclusions. As part of this process, the Oyster Creek IPE only required a "Step 1" review.

On August 14, 1992, GPU Nuclear Corporation (GPUN) submitted the Oyster Creek IPE in response to Generic Letter 88-20 and associated supplements. (Oyster Creek is a General Electric BWR-2 Mark I single-unit plant with isolation condensers.) The IPE submittal was based on a Level 1 PRA, and a Level 2 PRA consistent with Generic Letter 88-20, Appendix 1. The IPE submittal contains the results of an evaluation of internal events, including internal flooding. The licensee plans to provide a separate submittal on findings stemming from the IPE for external events (IPEEE). The staff will review the IPEEE separately, within the framework prescribed in Generic Letter 88-20, Supplement 4.

As part of its review, the NRC contracted with Science & Engineering Associates, Inc. (SEA), Scientech Inc./ Energy Research Inc., and Concord Associates to review the front-end analysis, the back-end analysis, and the human reliability analysis, respectively. SEA's review is documented in NRC-04-91-066 Task 8 report, "Oyster Creek Nuclear Power Plant IPE: Front-End Review." Scientech's review is documented in SCIE-NRC-212-92, "Oyster Creek Individual Plant Examination Back-End Technical Evaluation Report." Concord's review is documented in CA/TR 92-019-08, "Technical Evaluation Report: Oyster Creek Nuclear Generating Station Individual Plant Examination Assessment of Human Reliability Analysis, Document-Only."

On July 27, 1993, the staff sent a request for additional information to the licensee. The licensee responded to the staff's request in a letter dated October 1, 1993. In addition, the licensee, in a letter dated July 3, 1993,

provided to the staff a feasibility study for implementation of a portable DC generator.

This report documents findings and conclusions which stemmed from the NRC review. Specific numeric results and other insights taken from the licensee's IPE submittal are listed in the Appendix to this Staff Evaluation Report.

II. STAFF'S REVIEW

1. Licensee's IPE Process

The Oyster Creek IPE submittal of August 14, 1992, describes the approach taken by the licensee to confirm that the IPE represents the as-built and as-found plant. In addition to detailed document reviews, plant walk-throughs were performed by members of the licensee's PRA team (consultants and plant personnel) for familiarization with plant/system operations, equipment layout for origin and susceptibility to floods, and containment walk-throughs for information to be used for the back-end analysis. On the basis of review of the information submitted with the IPE, the staff concludes that the licensee's walkdowns and documentation reviews constitute a viable process for confirming that the IPE represents the as-built and as-found plant.

The IPE submittal contains a summary description of the licensee's IPE process, the plant personnel participation in the process, and the subsequent in-house peer review of the final product. The staff reviewed the licensee's description of the IPE program organization, composition of the peer review teams, and peer findings and conclusions. The staff notes the considerable participation of the GPUN personnel in virtually all aspects of the IPE through technology transfer, model development, reviews, data collection, and requantification of the models with plant-specific data. In addition to the IPE team, other GPUN and plant organizations were involved to insure that the models accurately portrayed the plant. Although GPUN did not indicate its intentions of maintaining a "living PRA," the submittal stated that, GPUN recognizes the potential benefit of the PRA and its potential use in future evaluation.

As part of the IPE process, GPUN established an independent review team which consisted of personnel from all appropriate organizations including engineering, operations, training, and an independent safety engineering group. This review was in addition to internal reviews performed by the GPUN consultants. Based on the review of the IPE submittal and associated documentation, the staff concluded that the licensee's peer review process provided reasonable assurance that the IPE analytic techniques had been correctly applied and documentation was accurate.

The submittal defined "vulnerability" as "any core damage sequence that exceeds 1E-4 per reactor year or containment bypass that exceeds 1E-6 per reactor year." The fundamental contributors to risk and risk-important accident scenarios were determined by delineating the sequence characteristics and evaluating their importance on the basis of their respective contribution

to core damage frequency and release category frequency. No plant vulnerabilities were identified and, therefore, no potential enhancements were identified to specifically address vulnerabilities.

The licensee probed the quantitative results by performing: (a) an uncertainty analysis; (b) a sensitivity study on several key variables; and (c) an importance analysis to identify the most important systems to plant safety. The sensitivity analysis concluded that changes to data or assumptions do not have a significant effect on the overall results. The results of the importance analysis provided a basis for the identification of possible low-cost improvements.

The staff finds the licensee's IPE process capable of identifying severe accident risk contributors (or vulnerabilities) and that such capability is consistent with the objective of Generic Letter 88-20.

2. Front-End Analysis

The staff examined the IPE front-end analysis for completeness and consistency with acceptable PRA practices. The licensee capitalized on insights stemming from the Oyster Creek PRA Level 1 study, NUREG-1150, and several other PRAs of plants with similar designs.

The Level 1 IPE involves a "plant model" which integrates the system and human action analysis (and associated data), and delineates accident progression from the initiating events (IEs) to plant damage states. Event tree sequence diagrams (ESDs) were used to identify available success paths needed to mitigate accident initiators, and to identify subsequent system failures, translating them into rules. Plant-specific analysis and transient assessment reports, in combination with the plant procedures, served as the basis for the ESD. Top events in the ESDs were sequenced by initiating events and intersystem dependencies. Event sequences explicitly represent support systems, front-line systems, human responses, and dependencies. Functional success criteria and specific system success criteria for each major plant safety function with respect to each IE category are clearly and appropriately described. The dominant accident sequence groups and their contributions to the core damage frequency are identified along with the contributing important systems.

The front-end IPE analysis used the large event tree/small fault tree methodology which treats dependencies on the event tree as split fractions rather than through the logical linking of fault trees. The licensee used the latest modification of this method in which the event trees are replaced by logic diagrams, i.e., tables of rules. Thus, no event trees were explicitly presented in the Oyster Creek submittal.

The licensee's IPE submittal identified 28 initiating event groups for Oyster Creek. These groups were further categorized into three broad groups: (1) general transients (15 initiating events); (2) loss of coolant accidents, small LOCAs (6 initiating events); and (3) large LOCAs (7 initiating events). Initiating events were determined by using a master logic diagram which identifies the various plant functions that could fail and lead to a plant

trip. These groups were reviewed against previous PRAs and industry studies, plant operational experience, and the Final Safety Analysis Report (FSAR).

The IPE identified and analyzed plant-specific initiators. These included: interfacing system LOCA, loss of intake channel flow to the intake structure, loss of Turbine Building Component Cooling Water (TBCCW), unisolated steamline breaks and large pipe breaks inside containment, and internal flooding. In response to staff questions on success criteria, the licensee stated that only RELAP5/RETRAN computer codes had been used in the development of thermal hydraulic analysis in support of the Level 1 analysis. Further, core damage is defined as water at the top of active fuel and decreasing. MAAP had not been used to develop Level I success criteria.

The IPE analyzed front-line systems and major support systems including but not limited to AC/DC vital power, service/circulating water, and instrument air The IPE provides a clear description of the top events considered; the success criteria; the support systems required; the systems' configuration, operation, testing, maintenance, and technical specifications assumptions; and the systems' boundaries. The system analysis task utilized the fault tree approach to logically combine the basic events and failure probability in order to derive the split fraction values used in the plant model. A comprehensive analysis of system dependencies was performed and included support to support, support to front-line, and front-line to front-line system.

In order to develop plant-specific IE frequencies, a Bayesian update of generic (BWR) IE data was performed utilizing plant-specific information. The data sources used were clearly identified in the IPE. The staff notes that the licensee made an effective use of both generic and plant-specific IE data. Further, the IPE submittal provides a detailed discussion of the dependencies between IEs and mitigating systems (including front-line and support systems), and clearly presents how each IE group affects the split fractions used in the model.

A Bayesian update process was also used to develop the IPE's systems' database. A generic database encompassing the cumulative experience from a large population of nuclear power plants was combined with a comprehensive plant-specific database containing more than 10 years of Oyster Creek experience. The update was performed using the data analysis module of the RISKMAN program. Plant-specific features were considered in selecting the appropriate generic distributions in order to obtain a "coherent" integration and updating of the database. As recommended in NUREG-1335, the IPE made extensive use of plant-specific data. Systems and components such as emergency core cooling pumps, batteries, diesel generators, electric buswork and breakers, service water pumps, instrument air, primary containment isolation, Automatic Depressurization System (ADS) valves, and other components were quantified using plant-specific (mainly post-1982) data.

The common cause failures (CCFs) were analyzed in two categories. The first category includes sharing of common components, effects of floods, and human errors during test and maintenance. The second category includes design errors, construction errors, procedural deficiencies, and unforeseen

environmental variations. Common cause events were incorporated into the system analysis in order to identify the CCF mechanism. The quantification of the CCF factors was accomplished by a multiple greek letter (MLG) methodology, consistent with NUREG/CR-4780. Responding to the staff's request for additional information, the licensee listed 50 common cause failure events and their associated contribution to core damage.

The submittal contains the top 100 most probable core damage sequences in Appendix C, Table C.5-1, of the Level 1 report in accordance with the reporting guidelines in NUREG-1335. These 100 highest frequency sequences account for 82% of total core damage frequency. The IPE derived a point estimate mean of 3.96 E-6/year for a total CDF. An uncertainty analysis identified the 5th and 95th percentile as 1.31E-6/year and 9.82E-6/year, respectively.

Among the dominant accident sequences, about 20.8% (7.69E-7/year) of the total CDF was contributed by the loss of all AC power (station blackout) with failure of an EMRV to reclose. Turbine trip with loss of all DC power contributed 7% of the total CDF (2.59E-7/yr), and reactor trip with the loss of all DC power contributed 5.7% of the total CDF (2.1E-7/yr). Other dominant sequences included: inadvertent MSIV closure with loss of all DC power (3.3%); loss of offsite power events (LOSP) with EMRV closure and core spray failures (3.2%); loss of TBCCW with EMRV closure and core spray failures (2.8%); and large below core LOCA with spray failure (2.6%).

The dominant IEs include: loss of offsite power (32.8% of total CDF); turbine trip (13.1%); reactor trip (7.7%); MSIV closure (7.7%); and total loss of feedwater (5.7%). The IPE did not find anticipated transients without scram (ATWS) as a significant contributor to the total CDF, based on credit taken for plant modifications for ATWS prevention and mitigation and the incorporation of operator recovery actions in the emergency operating procedures (EOPs).

The IPE performed an importance analysis that showed that EMRV failure to close contributes most to total CDF (48%). Essential AC power bus failures contribute 37% and DC power failures about 33%. This importance measure percent CDF is that percentage resulting from the summation of the frequency of all sequences involving the top events, and it represents the percentage decrease in the CDF that would result if the top event or system failure could be made zero. A sensitivity study was performed on several key variables in the study: LOSP events recovery; EMRV failures to close; and recovery of containment heat removal (including recovery of DC power and containment spray). The analysis concluded that changes to data or assumptions do not have a significant effect on the overall results.

A number of the Oyster Creek IPE dominant sequences involve loss of DC power. (Oyster Creek has only 3-hour battery capacity). These sequences and associated contribution to core damage include: turbine trip with loss of all DC power (7%); reactor trip with loss of DC power (5.7%); and inadvertent MSIV closure with loss of DC power (3.3%). DC power is required to activate the isolation condenser and open the EMRVs to allow for vessel injection with the low pressure firewater system.

The licensee's IPE station blackout analysis did not address recirculation pump seal LOCA, although the staff identified gross seal failure as a potentially dominant core damage sequence in station blackout accidents at nuclear power plants (NUREG-1032). Oyster Creek, for example, does not have a steam driven makeup system available during station blackout (unlike other BWRs). A LOCA during station blackout would compromise decay heat removal by degrading natural circulation between the reactor core and isolation condenser.

In response to staff questions, the licensee stated that loss of coolant through the recirculation pump seals would be "insignificant" on loss of pump seal cooling, a condition which would exist during station blackout. Although the licensee provided references to support its position, the issue remains open and under independent staff consideration (as a possible generic issue separate from Generic Issue 23). Because the issue is being considered separately for BWRs, the staff (IPE) review team did not pursue this aspect further. The staff notes, however, that the Oyster Creek IPE analysis is sensitive to assumptions associated with recirculation pump seal failures (i.e., impact the estimated core damage frequency by more than an order of magnitude).

The IPE's flooding analysis was divided into two parts. In the first part, effects were addressed in the rules and modules of the mitigating systems analyses. In the second part, flood source and equipment location data were compiled and catalogued and only components that were deemed significant to plant risk were analyzed. The flooding analysis considered the effects on components (including electrical) of being submerged, sprayed, or exposed to condensing steam. The calculated flood-induced CDF is 2.08 E-7. Approximately 78% of the flood-induced CDF is due to floods in the turbine building, with the remaining due to floods in the reactor building.

Based on the IPE description and licensee responses to questions, the staff finds the licensee's IPE methodology clearly described and justified in its submittal. Based on the staff's review of the front-end analysis and the staff's finding that the analytical techniques used are capable of identifying potential core damage vulnerabilities, the staff concludes that the IPE front-end analysis meets the intent of Generic Letter 88-20.

3. Back-End Analysis

The staff examined the licensee's back-end analysis for completeness and consistency with the guidance specified in Generic Letter 88-20, Appendix 1. The Oyster Creek consultant, PLG Incorporated, used the RISKMAN methodology to quantify the event trees and version 7.03 of MAAP-3.0B. The analyses conformed to Electric Power Research Institutes (EPRI's) recommendations related to selected model parameter values. MAAP was not used to investigate in-vessel recovery under damaged core conditions.

The licensee, through PLG, had EQE Engineering Consultants perform a plant-specific containment structural analysis to develop containment failure pressure, temperature, and location insights. The mean ultimate containment failure pressure was determined to be 134 psig. The staff found the approach

consistent with Generic Letter 88-20, Appendix 1 (Guidance on the Examination of Containment System Performance).

Three issues are unique at Oyster Creek. First, the torus was strengthened in the 1980's. This resulted in about a 25% increase in pressure capacity to a best estimate limit of 153 psig. Second, the sand, normally between the drywell shell and the concrete wall at the drywell floor elevation, has been removed. Corrosion has occurred at this location which has reduced the structural integrity to about 8 psi below the drywell head flange leakage pressure. Finally, Oyster Creek has a 1-foot thick, 6-inch high curb at the liner—drywell floor interface. The volume of the sump and within the curb is sufficient to contain all of the estimated corium volume. This reduces the liner melt-through probability by approximately 50%. Thus, there are no wetwell failures, and drywell failures are at the drywell floor location due to over pressure with a small contribution from liner melt-through.

The translation of the Level 1 accident sequences into Level 2 Containment Event Tree (CET) and accident release characteristics was performed by mapping each of the accident sequences into Plant Damage States (PDS). The PDS were defined by the condition of the plant at the end of the Level 1 analysis. The PDS considers the reactor pressure (high or low), drywell floor conditions (wet or dry), containment integrity (intact, bypassed, failure within a few hours of event initiation, or fails later), status of active systems (containment vent, suppression pool cooling, drywell sprays, and water to cool debris), and status of reactor building (isolated, firewater system in the reactor building, and standby gas treatment system (SGTS) operability). However, the reactor building and SGTS effectiveness was assumed to be zero based on the dominant containment failure mode being a catastrophic breach of containment. The licensee reduced the suggested screening criteria identified by an order of magnitude to ensure consideration of sequences which could be important to containment integrity and risk. The licensee has listed all of the Level 2 sequences with a frequency equal to or greater than 1 E-10 (49 sequences), exceeding the NUREG—1335 screening guidelines.

The licensee identified 19 PDS which were mapped into seven key plant damage states (KPDS). The KPDSs were used as the entry states to the CET. The CET models the core degradation, vessel failure, containment behavior, and reactor building behavior.

The CET was developed to resemble the Peach Bottom NUREG/CR-4551 accident progression event trees. The quantification of the CET for each KPDS was carried through a number of split fractions defined for each top event. The results were used to define CET end-states bins which were subsequently used to develop source term categories. The source term was evaluated using a source term event tree (STET). The STET considered six questions: drywell spray availability; reactor pressure at time of vessel failure; condition of containment (intact, vented, early or late failure); containment failure mode (leak or gross); availability of pool scrubbing; and availability of reactor building mitigation. The results of the STET were grouped into six key release categories (KRC) based on similarities of containment failure, timing, and mitigative features. The source terms for the KRCs were calculated by selecting representative sequences and using MAAP to model the behavior and

release of 12 radionuclide groups. The timing of the release was based on the estimated containment failure time from the initiation of the accident, as follows:

- Early (E) - 3 hours or less after vessel failure,
- Late (L) - More than 3 hours after vessel failure.

Sensitivity studies concerning accident phenomenology were not performed. Instead, the licensee stated in response to the staff's request for additional information, a combination of parameters were chosen from those recommended by EPRI, to give a conservative response in source term released.

Substantial credit (50%) is taken for in-vessel recovery following core damage. This is partially a result of the licensee's definition of core damage. (Core damage is defined as water at the top of active fuel and decreasing.) For low pressure sequences, vessel breach is prevented by using the condensate system, control rod drive system, or fire protection system through the core spray system. The assumption was made that vessel failure will result in a guaranteed containment failure. For many sequences involving extensive core damage, no credit was given to operator actions. The licensee indicated its concern that the potential for adverse effects (which could result from operator mitigation action) could exceed the perceived benefit. For example, in response to staff questions on sequences involving recovery of electrical power, the licensee stated that prompt action to vent containment without proper "accident management guidelines" could result in an earlier source term release than if no action was taken. Other issues associated with accident progression also remain open, e.g., the consequence of activation of drywell sprays with corium in the drywell. The licensee stated that it plans to postpone further evaluation of potential operator mitigation action to the accident management development phase "when better tools will be available (MAAP4)." The accident management program is a key element in closure of severe accident concerns, and the staff recommends that the licensee address these issues within that framework.

The licensee considered the effects of containment temperature and pressure on the elastomer seals. These seals are used for the drywell head flange and equipment and manway hatches. For all of the potential accident sequences considered, the temperature and pressure profiles are expected to result in no or little leakage. This result is based on their consultant's analysis (EQE) and agrees with the results of analysis discussed in NUREG/CR-5565, NUREG/CR-4944, NUREG/CR-5096, and NUREG/CR-4064.

The licensee also examined the failure of containment isolation. The modeling of containment isolation failure is based on a fault tree model. The fault tree incorporates modeling of automatic containment isolation valves that penetrate containment and are open to the containment atmosphere (e.g., vent and purge lines) as well as potential containment bypass lines whose system pressure is less than 90 psig, larger than 1-inch in diameter, and contains non-manual isolation valves. The fault tree considers automatic and manual isolation signal failures and component and common cause failures.

The licensee employed a process to understand and quantify severe accident progression. The process lead to a determination of conditional containment failure probabilities and containment failure modes consistent with the intent of Generic Letter 88-20, Appendix 1.

The following tables show the conditional containment failure probability as a function of failure location and timing, respectively:

Containment Failure Locations

• Drywell	42.3%
(Liner Melt-through	17%)
• Wetwell	0.0%
• Bypass	7.3%
• Intact	50.4%

Containment Failure Timings

• Early	15.9%
• Late	26.4%
• Bypass	7.3%
• Intact (following vessel breach)	0.0%
• No Vessel Breach	50.4%

Of particular interest is that the probability of containment failure is zero if reactor vessel failure is prevented and one if reactor vessel fails. This is due to the fact that the recovery of electric power was not considered once core damage commenced. Therefore, there was no recovery of containment heat removal or drywell sprays.

The process of determination of conditional containment failure probabilities and containment failure modes was consistent with the intent of Generic Letter 88-20, Appendix 1. The dominant contributors to containment failure were found to be consistent with insights from other analysis of similar designs. The licensee characterized containment performance for each of the CET end-states. The licensee considered the failure of containment seals and containment isolation failures. The staff's review did not identify any significant problems or errors in the back-end analysis. The overall assessment of the back-end analysis is that the licensee has made reasonable use of probabilisitic techniques in performing the back-end analysis, and that the techniques employed are capable of identifying plant vulnerabilities. Based on these findings, the staff concludes that the licensee's back-end IPE process is consistent with the intent of Generic Letter 88-20.

4. Human Factor Considerations

The licensee acknowledged three types of human errors, *pre-initiator human events* associated with errors during routine activities (such as valve misalignment) leaving equipment disabled ("Group A"), *initiating human events* associated with errors, causing a plant abnormal condition ("Group B") and

post-initiator human events associated with errors during operator response to an abnormal condition, i.e., an initiator ("Group C").

The IPE did not perform a pre-initiator human event analysis. The rationale provided by the IPE and in the licensee's responses for this approach is that: (a) usually few pre-initiator events are identified during a Human Reliability Analysis (HRA); (b) typically they are not significant contributors to core damage frequency; and (c) the frequency of pre-initiator events is captured in the basic equipment failure rates and, hence, there could be double counting of failures if a separate analysis was performed.

These types of errors have been shown to be dominant contributors in other studies and are not necessarily part of the basic equipment failure rate (e.g., NUREG 1150 analyses). In addition, Generic Letter 88-20 requested the licensees to examine maintenance and surveillance practices as part of their effort to identify potential vulnerabilities. Generally most plants have administrative controls for preventing system unavailability due to test and restoration activities. The process by which these controls are implemented, however, determines whether there are practices creating the potential for leaving a system in an undetecte' disabled state (resulting in equipment unavailability on demand). While the staff agrees that a portion of pre-initiator events can be captured when performing a Bayesian update (provided ample operational data is available), unless routine personnel activities are examined as part of the IPE HRA, such instances of potential errors may not be uncovered. The staff finds the lack of pre-initiator event analysis a weakness of the licensee's HRA, which may limit the usefulness of the IPE for future regulatory applications.

Initiator human events were analyzed as part of the IE analysis consistent with acceptable PRA practices.

Post-initiator human events were extensively analyzed. They were further distinguished to human events associated with response-type actions and to human events associated with recovery-type actions. Response-type actions include those human actions performed in response to the first level directive of the EOPs, such as reading instrumentation to determine reactor water level status or maintaining reactor water level with different systems. Recovery-type actions include actions performed to recover from a specific failure or fault, i.e., cross connecting electrical busses following loss of offsite power (proceduralized action) or gaging a failed instrument air relief valve (non-proceduralized action).

In order to identify post-initiator human events, the licensee examined the EOPs, system instructions, and off-normal event procedures associated with the accident sequences delineated and the systems modeled. Further, discussions were held with plant operators on the interpretation and implementation of plant procedures to identify and understand the specific actions and the specific components manipulated when responding to the accident sequences modeled.

88-20The licensee employed the Success Likelihood Index Methodology (SLIM) to quantify post-initiator events. The licensee's evaluation was based on

eliciting the control room operators' judgement of each action analyzed. Important factors influencing human performance (for example, the type and location of plant procedures, operator communication, location of required actions, effect of annunciators and alarms, and the time available versus the time required to perform the needed human action) were considered in the analysis. Plant-specific performance shaping factors were used in the calculation of the human error probabilities (HEPs).

The HRA dealt extensively with the issue of accounting for the effects of multiple operator actions and the dependencies among human actions. A "confusion" performance shaping factor was included in the quantification of each human error to account for dependencies among steps of an individual task. Further, the IPE performed a thorough "sensitivity to multiple operator actions" analysis that included a quantitative and a qualitative portion. The quantitative sensitivity re-estimated the CDF by increasing the HEPs to determine their relative importance. The qualitative sensitivity reviewed the time available versus the time required for an action and crew changes, for all actions in a scenario. This sensitivity did not identify any dependent actions that were treated as independent. The staff also notes that the licensee used a sound approach to address multiple operator actions, and that the study had been used for planning plant modifications in a way that allowed a better understanding of various operator interactions.

The post-initiator quantitative results and the insights derived from the analysis are also discussed in a clear and concise manner in the IPE. A total of 34 functionally different operator actions (and a total of 66 individual actions) were modeled in the IPE. The total human error contribution to CDF is 21%. The IPE lists (Tables 2.1-6 and 2.1.4) and discusses the most important human actions in the context of their contribution to the total core damage frequency. It should be noted that no individual (or combination of) human action(s) were dominant in the Oyster Creek IPE. The most important human action to CDF is initiation of containment cooling (2.76%), followed by failure of manual core spray function core spray (2.70%) and recovery of DC power (2.50%).

An improvement regarding operator training for initiating the containment spray system was identified and implemented. In addition, one procedural and several operator training improvements were identified that are under review and consideration by the licensee.

In summary, the staff finds the HRA methodology described in the licensee's submittal supports the quantitative understanding of the overall probability of core damage during plant operations, as well as an understanding of the contribution of human actions to that probability. Therefore, the staff finds the licensee's assessment of human reliability capable of discovering severe accident vulnerabilities from human errors and consistent with the intent of Generic Letter 88-20. The staff notes that the licensee used a thorough, systematic, and traceable post-initiator event human analysis. However, the staff finds the lack of pre-initiator human event analysis a weakness of the licensee's HRA which may have an impact to the IPE's usefulness in other applications. The staff encourages the licensee to consider pre-initiators explicitly in its HRA in any future revisions of its PRA.

5. Containment Performance Improvements (CPI)

As a result of the Containment Performance Improvement Program, recommendations for improvements were made for licensees to consider as part of the IPE process. These recommendations were identified in Generic Letter 88-20, Supplement 1. Each of these proposed improvements is discussed separately below.

(1) A hardened vent: The licensee has proposed installation of an 8-inch hardened vent from the torus air space to the stack. Venting is initiated directly before containment pressure reaches 3.0 psig (corresponding to the containment spray start signal and ADS actuation logic setpoints) and again before torus pressure reaches the primary containment pressure limit, as directed by Oyster Creek's EOP.

The licensee has suggested an alternate strategy to protect the emergency core cooling system (ECCS) pumps from a loss of net positive suction head (NPSH). In lieu of venting containment, the licensee suggests using suppression pool cooling with the residual heat removal (RHR) and using the wetwell sprays to reduce the suppression pool water temperature. After a sufficient reduction in pool water temperature, the drywell sprays would be used to reduce the drywell temperature. This would prevent a potential loss of NPSH by venting containment. The licensee proposed considering this procedure during the preparation of its accident management guidelines, and should be considered as part of its Accident Management Program.

(2) An alternate water supply for vessel injection or drywell sprays: Provisions for using the fire protection system pumps aligned to supply both divisions of the core spray system have been provided. The fire protection system consists of two diesel driven 2000 gpm pumps. The diesel driven pump has its own DC power supply, and it can be started locally. Connection of the fire protection system to the Core Spray (CS) system is by means of a 4-inch (which reduces to 3-inches) line from the 12-inch fire main ring header. The fire protection system can also be used to provide make-up to the isolation condenser. Each isolation condenser is provided fire protection system water through a 6-inch line from the 12-inch fire main ring header.

No provision exists for using the fire protection system with the drywell sprays. The licensee has concluded that this capability is not cost beneficial for the following reasons. First, those sequences where drywell sprays could be beneficial represent only 8.75% of the total core damage frequency. Second, the flow rate at the nozzle would not develop a full spray pattern, but would "run out of the spray nozzles." Without a full spray pattern, the fission product scrubbing would be greatly reduced. And finally, "without a fully developed spray, the capability to cool the containment shell is greatly reduced." Furthermore, "it is highly likely that fire protection water exiting the hole in the vessel left by the exiting corium would provide a comparable degree of containment shell cooling." This last argument is only true if water exits the hole sufficiently before the corium reaches the

drywell liner and that the drywell is flooded at the liner. This depends on the melt progression and vessel failure assumptions. Given that the water will exit the reactor vessel through the failure location and that a pool of water will overlay the corium, drywell sprays could still be important. As discussed in NUREG/CR-5978, the drywell sprays could be important if: (a) the water pool could not be kept subcooled; (b) there is excessive, late, release of coarse aerosols from residual fuel in the reactor vessel directly to the drywell atmosphere; or (c) there is extensive revaporization of deposited fission products from the reactor coolant system after reactor vessel failure. Furthermore, NUREG/CR-5869 states that flooding containment prior to core relocation onto the bottom head can significantly delay or prevent vessel failure.

The licensee has taken the position that the containment will always fail when the reactor vessel fails. This position may have masked the true potential benefit from enhanced drywell sprays. Other licensees have concluded that having the drywell sprays will significantly reduce the probability of drywell liner melt-through. The licensee has stated that it is unclear how operator actions will affect the accident progression, and they intend to evaluate the effects of potential operator actions when appropriate tools (MAAP4) become available. The staff recommends that the licensee continue to evaluate the need for drywell sprays as part of its accident management program evaluation.

(3) <u>An enhanced reactor pressure vessel (RPV) depressurization system reliability</u>: The IPE submittal stated that the licensee would consider procurement of a portable generator, based on its cost effectiveness. The station batteries will provide DC power for a minimum of 3 hours. However, in a letter dated July 2, 1993, the licensee stated that: (1) portable DC generators were not readily available; (2) for extended station blackout conditions, portable AC generators, to be used for battery charging, could readily be obtained through an outside supplier; and (3) providing a portable DC power supply was not cost-effective. Based on their analysis, the licensee stated that portable generators were not procured at this time but will be reconsidered during preparation of the accident management guidelines. This need for alternate power supply at the site will be reviewed as part of the Accident Management Program.

(4) <u>Incorporation of the BWROG Emergency Procedure Guidelines (EPGs), Revision 4, into the plant procedures</u>: The licensee has incorporated Revision 4 of the BWROG EPGs.

Based on this review, the staff concludes that the licensee has responded to the CPI Program recommendations, has searched for vulnerabilities associated with containment performance during severe accidents, and its evaluation is consistent with the intent of Generic Letter 88-20 and associated Supplement 1. However, certain aspects of the analysis are to receive further consideration as part of the licensee's Accident Management Program.

6. DHR Evaluation

In accordance with the resolution of Unresolved Safety Issue (USI) A-45, the licensee performed an examination of Oyster Creek to identify decay heat removal (DHR) vulnerabilities. The evaluation considered various combinations of reactor vessel inventory makeup and decay heat removal rejection pathways. The analysis took minimal credit for human recovery actions.

The plant features listed below were considered in the licensee's evaluation of the Oyster Creek decay heat removal function:

(1) The normal path for decay heat removal involves the feedwater system and main condenser. The success criteria for this path require that main steam isolation valves (MSIVs) are open and that the main condenser and the support systems are available. The required support systems include instrument air system for control of the feedwater regulating valves, 4160 VAC system for the feedwater and the condensate pumps, 120 VAC feedwater control power, and 125 VDC for the instrument and logic. The turbine building closed cooling water (TBCCW) is also required for pump and lube oil coolers.

(2) The decay removal path through the isolation condenser can be utilized following reactor isolation transients where either the main condenser is unavailable or MSIVs are closed. The success criteria for this path require initiation of one of two isolation condensers, followed by the successful long term shell side makeup water. The emergency makeup water for the long term operation, due to the boil-off of the shell side inventory, can be provided by either the condensate transfer system or the fire protection water system. The high pressure makeup on the eventual loss of the reactor coolant system inventory can be provided via the control rod drive hydraulic system.

(3) Decay heat may be transferred through coolant discharged into the containment. The discharge may involve a pipe break (in the event of a LOCA), or through the operation of relief or safety valves. The decay heat is removed from the containment via the spray/emergency service water system and transferred to the intake canal.

(4) Upon failure of the containment spray/emergency service water system, decay heat may be transferred to the containment and outside atmosphere through the hardened vent system.

The recovery of containment heat removal is well documented in the submittal. The overall contribution of loss of decay heat removal to CDF had been found to be 3.96%.

Based on the process that the licensee used to search for DHR vulnerabilities, and review of plant-specific features, the staff finds the licensee's DHR evaluation to be consistent with the intent of Generic Letter 88-20 and resolution of USI A-45.

7. Generic Safety Issues

As part of the IPE submittal, the licensee proposed resolution of several generic issues including USI A-17, "System Interaction in Nuclear Power Plants;" USI A-47, "Safety Implications of Control Systems;" Generic Issue (GI)-101, "BWR Water Level Redundancy;" and GI-105, "Interfacing System LOCA at BWRs." However, USI A-17, GI-101, and GI-105 were resolved by staff with no new requirements. Accordingly, the licensee's proposed resolution of these issues was not reviewed in detail. The review of the licensee's response to Generic Letter 89-19, "Request for Action Related to Resolution of Unresolved Safety Issue A-47," addresses USI A-47 resolution.

8. Licensee Actions and Commitments From the IPE

The licensee used the IPE process to identify plant and/or procedural modifications. The IPE took credit for several modifications that the licensee installed during the 14R refueling outage. These include installation of a hard piped containment vent system; operator training for manual initiation of the containment spray system; and installation of interconnection to the combustion turbine generators at the adjacent Forked River Site. The combustion turbine interconnection will make it possible to supply power from the combustion turbines directly to non-essential 4160 V bus 1A and emergency loads of essential 4160 V buses 1C and 1D via cross-tie.

Purchasing a portable power generator and developing procedures for recovering offsite or onsite power were identified as additional improvements for coping with station blackout. While the procedure development is underway, the licensee plans to evaluate the purchasing of an additional AC generator before the 15R refueling outage. The staff recognizes the licensee's intent to address station blackout events by the interconnection to the two combustion turbines and recovery of AC power procedure development.

IPE findings indicate that there are a number of additional "low-cost" improvements which could enhance overall reactor safety. These planned actions include:

- Development of an emergency procedure for Loss of Offsite Power.
- Development of an emergency procedure for Loss of DC Power.
- Increased training on the importance of the core spray system.
- Changes to maintenance scheduling for the core spray system to improve downtime.
- Programs instituted to reduce blockage and fouling of the isolation condensers.
- Modifications to implement the Reactor Overfill Protection System.
- Consider the development of specific guidance, training, and procedures for reactor overfill transients.
- Increased emphasis in training on key operator actions as defined by the IPE.
- Consideration of alternate containment heat removal capability to maintain minimal NPSH as part of Accident Management.
- Alternate water supply for drywell sprays (Accident Management).

Although the NRC review did not examine the merits of the above recommendations in detail, the staff notes that the licensee is applying PRA/IPE findings to enhance plant safety. The staff, therefore, finds the licensee's actions reasonable.

III. CONCLUSION

The staff finds the licensee's IPE submittal for internal events including internal flooding is consistent with the information requested in NUREG-1335. Based on the review of the submittal, the licensee's response to questions and associated information, the staff finds the licensee's IPE conclusion that no fundamental weakness or severe accident vulnerabilities exist at Oyster Creek to be reasonable. The staff notes that:

(1) GPUN personnel participated in virtually all aspects of the IPE through technology transfer, model development, reviews, data collection, and requantification of the models with plant-specific data. In addition to the IPE team, other GPUN and plant organizations were involveo to insure that the models accurately reflect the as-built, as-operated plant.

(2) The licensee established an independent review team which consisted of personnel from all appropriate organizations including engineering, operations, training, and an independent safety engineering group. This review was in addition to internal reviews performed by the GPUN consultants and provides assurance that the IPE analytic techniques had been correctly applied and documentation was accurate.

(3) The front-end IPE analysis is complete with respect to the level of detail requested in NUREG-1335. In addition, the analytical techniques were found to be consistent with other NRC reviewed and accepted Probabilistic Safety Analyses (PSAs).

(4) The back-end analysis addressed the most important severe accident phenomena associated with Mark I containments. No obvious or significant problems or errors were identified.

(5) The HRA allowed the licensee to develop an understanding of the contribution of human errors to CDF and containment failure probabilities. However, lack of analysis of pre-initiator events is a limitation of the licensee's IPE.

(6) The employed analytical techniques in the front-end analysis, the back-end analysis, and the HRA are capable of identifying potential plant-specific vulnerabilities.

(7) The licensee's IPE process searched for DHR vulnerabilities consistent with the USI A-45 (Decay Heat Removal Reliability) resolution.

(8) The licensee responded to CPI Program recommendations which include searching for vulnerabilities associated with containment performance

during severe accidents. However, the licensee plans to address a number of issues in its follow-on accident management program.

Based on the above findings, the staff concludes that the licensee demonstrated an overall appreciation of severe accidents, has an understanding of the most likely severe accident sequences that could occur at the Oyster Creek facility, has gained a quantitative understanding of core damage and fission product release, and responded appropriately to safety improvement opportunities identified during the process. The staff, therefore, finds the Oyster Creek IPE process acceptable in meeting the intent of Generic Letter 88-20.

The staff, however, finds the lack of analysis of pre-initiator human events a weakness of the licensee's IPE that may limit its usefulness in other applications. The staff encourages the licensee to improve its HRA by including pre-initiators in any future revisions of its PRA. The staff also notes that GPUN did not explicitly state that they plan to maintain their PRA "living." The staff notes that a "living" PRA could enhance plant safety and provide additional assurance that any potentially unrecognized vulnerabilities would be identified and evaluated during the life of the plant.

Principal Contributors: Erasmia Lois
John Ridgely
Jin Chung

Date:

APPENDIX
Oyster Creek Unit 1 DATA SUMMARY SHEET*
(INTERNAL EVENTS)

- Total core damage frequency (CDF) point estimate: 3.69 E-6/Year

- Initiating event importance to total CDF:

  - Loss of offsite power 32.8%
  - Turbine trip 13.1%
  - Reactor trip 7.7%
  - MSIV closure 6.9%
  - Total loss of feedwater 5.7%
  - Loss of condenser vacuum 4.0%
  - Loss of TBCCW 4.0%
  - Loss of intake structure 3.3%
  - Electric pressure regulator fail 3.2%
  - Large below core inside cont. LOCA 2.9%

- Dominant core damage sequences and contribution to CDF:

  - Station blackout with failure of an EMRV to reclose 20.8%
  - Turbine trip with loss of all DC power 7.0%
  - Reactor trip with loss of all DC power 5.7%
  - Inadvertent MSIV closure with loss of all DC power 3.3%
  - LOSP with EMRV failure to close and core spray failure 3.2%
  - Loss of TBCCW with failures of EMRV close and core spray 2.8%
  - Large below core LOCA with core spray failure 2.6%
  - RWCU overpressurization with core spray failure 2.0%
  - Loss of intake flow with EMRV failure and core spray failures 2.0%
  - Loss of condenser vacuum with loss of all DC power 1.8%

- Operator action importance to total CDF:

  - Initiation of Containment Cooling 2.76%
  - Core spray (Manual initiate or injection with fire protection) 2.70%
  - Recover of DC Power 2.50%
  - Recover Offsite Power 2.20%
  - Initiation of IC makeup 1.51%
  - Containment Venting 1.47%
  - Manual initiation of ADS 1.23%
  - Initiation of Boron injection (ATWS) 1.22%

- Level and Power Control Following ATWS 1.08%
- Control Post Trip RPV Level 1.03%

• System importance to total CDF:

o EMRV closure	48%
o 4160 VAC essential bus 1D	37%
o 4160 VAC essential bus 1C	37%
o 125 VDC bus C	33%
o 125 VDC bus B	31%
o Recovery from LOSP	26%
o Core spray	21%
o Reactor scram	6%
o 4160 VAC bus 1A	5%
o 4160 VAC bus 1B	4%

• Conditional containment failure probability given core damage:

o Drywell	42.3% (Liner Melt-through 17%)
o Wetwell	0.0%
o Bypass	7.3%
o Intact (Vessel Breach Prevented)	50.4%

• Important plant hardware and plant characteristics regarding containment performance:

- 8-inch hardened torus vent.
- 6-inch high, 1-foot thick drywell floor curb at the drywell liner.
- Two isolation condensers, operate with only opening one DC powered valve.
- Torus structural strength increased by 25% due to installation of straps.
- Liner corrosion at the liner-sand (which has been removed) - concrete interface (reduces strength by about 8 psi).
- Alternate water supply to reactor vessel and isolation condenser.

• Modifications the IPE took credit:

- Interconnection to the combustion turbine generators at the adjacent Forked River Site.
- Hard piped containment vent system,
- Operator training for manual initiation of the containment spray system.

• Significant PRA findings:

- IPE importance measures identified failure of electromatic relief (EMRV) to close as the largest component contributor to total CDF (48%). The significance of this contributor stems from mitigation success criteria which requires (for many accident initiators) opening and subsequent closing of up to 4 of 5 EMRVs.

- Losses of offsite power are significant contributions to CDF; the planned modification to use independent offsite power source will help mitigate the effects of loss of offsite power event.
- The plant is highly dependent on DC power; battery monitoring and maintenance will continue to be important.
- The licensee installed an 8-inch hard vent to reduce containment pressure. The analysis, however, showed that containment venting could result in inadequate NPSH for the RHR pumps, an effect that can be alleviated by reducing the suppression pool temperature with sprays before venting.

▪ Potential improvements under evaluation:

- Integrated loss of offsite power and station blackout procedure which includes cross-tieing buses and alignment of the alternate AC capability.
- Loss of all DC power procedure and a portable power generator for the essential loads.
- Training in the containment spray system and changes in the preventive maintenance on the containment spray and emergency service water.
- Post trip reactor feedwater control (Reactor Overfill Protection System (ROPS)).
- Alternate containment heat removal capability to maintain minimal NPSH (as part of Accident Management).
- Alternate water supply for drywell sprays (as part of Accident Management).

* Information has been taken from the Oyster Creek IPE and has not been validated by the NRC staff.

ENCLOSURE 2

OYSTER CREEK INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(FRONT-END)

SEA 92-553-008-A:2
February 17, 1994

Oyster Creek Nuclear Power Plant IPE: Front-End Review

Contractor Technical Evaluation Report
NRC-04-91-066, Task 8

Bernard Weber, DNV Technica, Inc.
Andrew Wolford, DNV Technica, Inc.
John Darby, Science & Engineering Associates, Inc.

Performed for:
Science and Engineering Associates, Inc.

By:
DNV Technica, Inc.

Prepared for the
Nuclear Regulatory Commission

1 INTRODUCTION

1.1 SEA Review Process

This report summarizes the review of the IPE Submittal for Oyster Creek. The issues raised in this report are based on a review of the Submittal only. Also, a visit to the OCNGS site is outside the scope of this review. The purpose of this review is to identify issues related to the IPE front-end analyses for OCNGS and to supply these findings to the NRC. The Review Process is provided in Figure 1, and subsequently described below.

The review was performed by reviewers from DNV Technica Inc. (DNV), under contract to Science and Engineering Associates Inc. (SEA). The reviewers followed the process used by SEA to perform previous IPE front end reviews.

This report does not include an evaluation of licensee responses to NRC questions that were generated based on our review.

1.1.1 Review of FSAR and Tech Specs

The NRC provided the submittal to SEA in September. The submittal was subsequently transmitted to the reviewers at DNV. DNV began work on October 1, 1992. Between October 1 and October 18, the review focused on a detailed review of the submittal to develop an understanding of the front-line and support systems, and to identify apparent deficiencies, if any, in the information assembly process of the IPE. The purpose of the preliminary review was to identify specific areas in the FSAR that should be consulted for confirmation, clarification, and additional discussion of information in the IPE submittal.

On October 14 and 15, 1992, the latest Updated Final Safety Analysis Report (FSAR) and Technical Specifications (Tech Specs) for OCNGS were reviewed. This review was performed at NRC NRR using up-to date documentation provided by the NRR project manager. The focus of this review was to gain a better understanding of various plant systems, plant design, and accident analyses.

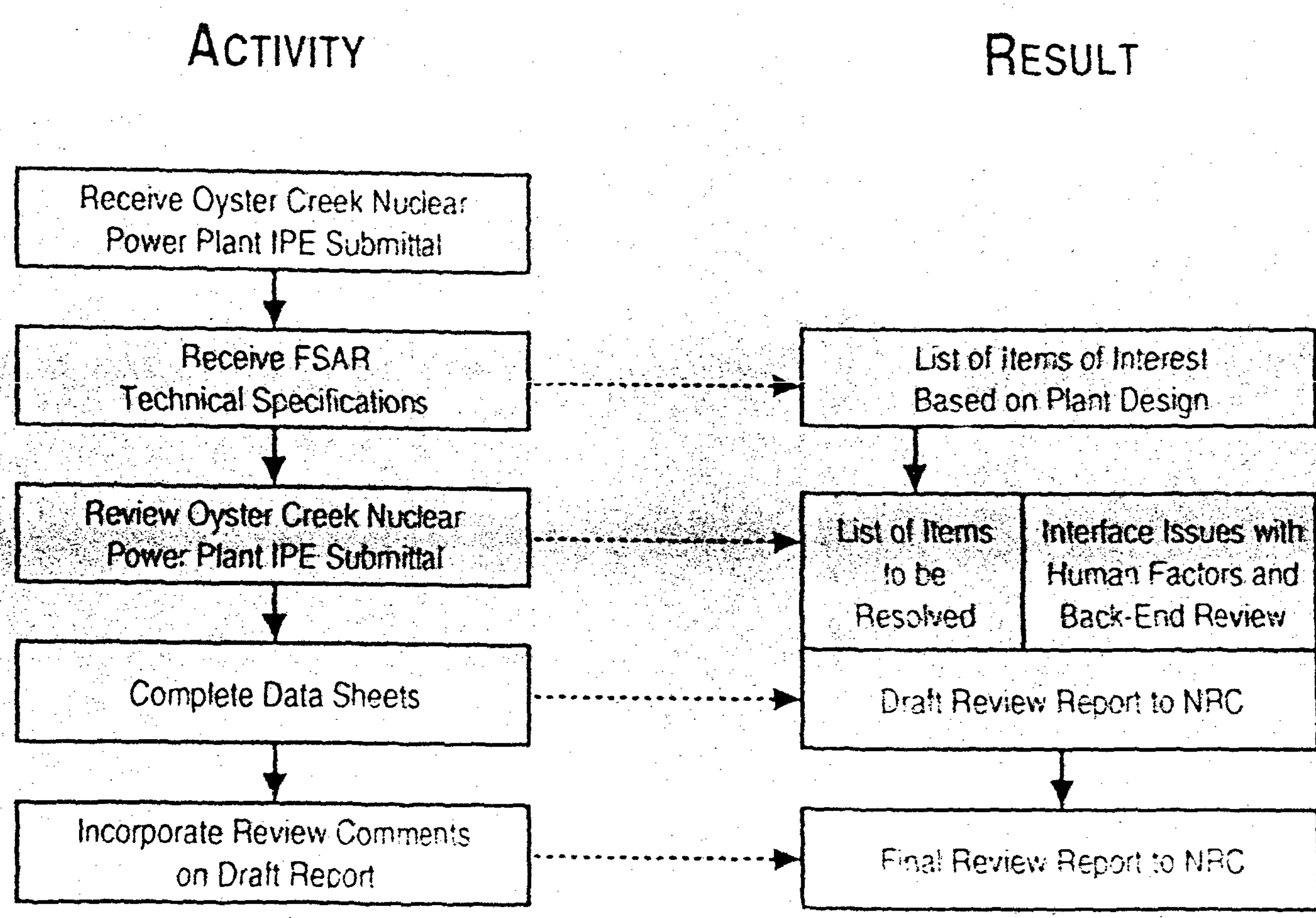

Figure 1. SEA Step 1 Review for Oyster Creek Nuclear Power Plant Unit 1 Front-End IPE

1.1.2 Review of IPE submittal

Between October [illegible] and November 13 a detailed review of the IPE submittal for OCNGS was performed. A Level 2 PRA was submitted by Oyster Creek to satisfy IPE requirements. A smaller report was submitted to address issues not covered in the PRA. The OCNGS Probabilistic Risk Assessment (PRA) was completed in December, 1991 and revised in June, 1992. A 'roadmap' was provided with the submittal to guide the reader to specific sections of the PRA which satisfied IPE requirements. The OCNGS PRA itself has not been subjected to a separate review by NRC.

The review effort incorporated a horizontal review of all aspects of 'front-end' issues as well as vertical reviews of selected key issues. The findings of this review are documented in Section II of this report. The review procedure focused on each item listed in the 'Step 1' Review Guidance Document.

1.2 OCNGS IPE Methodology

The PRA was performed using a new version of the PLG Inc. method. The PLG method has been referred to as the "large event tree, small fault tree approach," in that the dependencies are considered by split fractions on the event trees rather than the logical linking of fault trees. The latest modification to this technique is the rules based approach in which the event trees are represented by tables of rules; successful outcomes are not typically shown in the rules tables, only core damage outcomes. The tables of rules effectively replace the event trees, and in the Oyster Creek submittal, no event trees are provided for the front-end analysis.

Detailed fault trees were developed down to the component level for each front-line and support system in the plant. In all, 25 separate systems were mode[illegible] [illegible]rator actions were incorporated into the system fault trees. Containment heat recovery [illegible] [illegible]ssed as a separate module, using specialized event trees. Common cause failures were [illegible] [illegible]nly at the system level. Both generic and plant-specific data were incorporated into the fault trees. The quantification of core damage frequency via the large event trees was carried out using the 'rules methodology,' as implemented by the *RISKMAN* software package. An uncertainty analysis was performed on the major contributors to the CDF.

The methodology, although complete, was difficult to follow and at times a bit confusing.

The methodology used in the IPE front-end analysis of OCNGS meets the criteria stated in NUREG-1335.

1.3 OCNGS Plant

The OCNGS is a single unit facility located in Ocean County, New Jersey. The prime contractor and NSSS supplier was General Electric. Burns and Roe, Inc. provided engineering support and construction management. The unit achieved initial criticality in May, 1969 and was placed in commercial operation in December, 1969.

The unit is a Boiling Water Reactor (BWR-2) with a Mark I type containment. The plant is a non-jet pump BWR with five external loops for forced circulation of primary coolant. Shutdown cooling is provided by the main feedwater/condenser system. Backup cooling is provided by two isolation condensers, an Automatic Depressurization System (ADS), or a two-train low pressure core spray system.

1.3.1 Similar Plants and PSAs

The reviewer did not find a listing of similar plants or a list of PSAs of similar plants. The only other BWR-2 vintage plant is Nine Mile Point Unit 1 (NMP-1). The FSAR states that certain original design features of OCNGS, particularly in the areas of reactor, pressure vessel, are similar to BWRs of the same vintage; however, because modifications have taken place at OCNGS and those facilities over the course of time, a detailed comparison would not be meaningful.

The major differences between OCNGS and NMP-1 are that OCNGS has two isolation condensers, while NMP-1 has four.

1.3.2 Unique Features

Oyster Creek is a BWR-2 reactor, one of only two in existence. A Mark I containment is used. It is similar in design to the other BWR-2, Nine Mile Point Unit 1. Unique features of the Oyster Creek design include:

- Combination safety/relief valves are not used at OCNGS; such valves are typically seen in later vintage BWRs. Separate safety and relief valves are used at Oyster Creek. The relief valves are not air operated but do require DC power to open.
- There is no high pressure ECCS system, and no low pressure coolant injection system (LPCI). Later BWR designs have steam-driven or motor driven high pressure core spray and LPCI in addition to a low pressure core spray system.
- The station batteries have a three-hour capacity. *Some plants have longer battery capacities; this is important since DC power is required to open the relief valves and depressurize the system.*
- There is no reactor core isolation cooling system (RCICS). Cooling is provided by isolation condensers as a backup to the main condenser.
- There is an offsite combustion turbine which is interconnected to the unit to provide backup power during a station blackout.
- There are no (internal) jet pumps at Oyster Creek; recirculation flow is provided by five external pumps.
- Dedicated containment venting is available.

Based on these unique features, it was identified that the key areas for review are reactor depressurization sequences, and station blackout sequences lasting more than three hours.

II. CONTRACTOR REVIEW FINDINGS

II.1 Review and Identification of IPE Insights

This section presents our findings, including a summary of IPE strengths and weaknesses. The following sections address each work area explicitly in the order they appear in the SOW.

II.1.1 General Overview of Front-End Analysis

II.1.1.1 Completeness Check

A detailed review of the OCNGS IPE submittal was performed between October 16 and November 13. The initial review effort focused on the documentation to verify that all required information was presented in detail, according to the guidelines of NUREG-1335. Since the submittal consisted of a PRA with supplemental documentation and a cross-reference table, this review was time-consuming. The cross-reference table was helpful, but not completely adequate. For instance, the description of methodology for some major subtasks were found among the study details, not in the summary report. The Review Guidance as provided by the NRC was used extensively in this review of the PRA/IPE submittal.

Except for minor discrepancies cited throughout this review, the documentation provided is considered to be complete.

II.1.1.2 Methodology Check

The submittal employs the "rules modules" approach to the quantification of plant risks. The methodology is described briefly in Section 2 of the OCPRA. Detailed fault trees were constructed for front-line and support systems. System interdependencies were handled via specific rules which governed the quantification of the modules. Common cause failures were incorporated directly into the fault tree models. Intrinsic component dependencies, such as common environments, testing or maintenance, were treated within each system, but not across systems. Recovery actions were applied as separate modules in the quantification process. An uncertainty analysis was performed. **In conclusion, the methodology used in the OCNGS IPE submittal is consistent with the methods identified in Generic Letter 88-20 and NUREG-1335.**

Very little description is provided regarding the actual quantification of the model. In fact, the generation and evaluation of results does not appear in Figure 2-1 or in the text of Section 2.

II.1.1.3 Process to Confirm Representation of As-Built As-Operated Plant

Sections 1.3 and 1.4 of the submittal provide the sources of documentation and methods used to familiarize the utility staff with the as-built, as-operated plant. The information used to generate the IPE front-end analysis came from a variety of sources: Updated FSAR, Technical Specifications, Operations Plant Manual, Emergency Operating Procedures, P&IDs and Electrical Diagrams, and other related documents. To familiarize the PRA staff and verify the accuracy of the models, the following procedure was used:

- general walkdowns to familiarize the team with the arrangement of the site and plant systems
- systems analysis walkdowns, often with a cognizant plant engineer, STA, or operator, to review pertinent information
- plant model walkdowns to verify the impacts of initiating events, systems interactions and system interdependencies
- internal flood analysis walkdowns to verify component locations, collect source information, determine propagation paths and determine flood impacts
- repetitive reviews of the Event Sequence Diagrams (ESDs) were performed, in meetings with various utility personnel - including operations, safety analysis and training departments - to verify the validity of the plant models.

This procedure is thorough and would ensure that the plant models represent the as-built, as-operated plant. The ESDs are very helpful in documenting plant response to accident initiating events.

II.1.1.4 Internal Flooding Methodology

The OCNGS flood analysis was divided in to two parts: one completed as part of the Level 1 PRA in which LOCA initiating events are propagated through the base plant model, the other performed as a separate 'screening' analysis of specific flooding events.

For the flood events which appear explicitly in the Level 1 PRA, flood effects are addressed in the rules modules for the mitigating systems analyses. For the screening analyses, flood source

and equipment location data were compiled and catalogued. Only components which were deemed to be significant to plant risk were included. Using component and source information, potential flooding events were identified. Conservative isolation and recovery actions were then applied to these floods.

[illegible] critical internal flood areas were identified. The screening analysis showed a total CDF of [illegible], which represents about 5% of the core damage frequency. Approximately 78% of the flood induced CDF is due to floods in the Turbine Building (TB), the remaining occurred in the Reactor Building (RB). Significant contributors to CDF due to flood were:

- feedwater line failure in TB
- circulating water line failure in TB
- service water line failure in RB.

Flood frequencies were quantified using the March 1990 *Database for Probabilistic Risk Assessment of Light Water Reactor Power Plants, Volume 9, Flood Data*. The frequencies for each building were partitioned according to the number of systems in the applicable OCNGS buildings.

The reviewers were unable to find a <u>separate</u> summary of results for the internal flood effects which were included in the LOCA analysis, for LOCAs outside of containment that are not isolated.

In our judgement, the analysis was thorough, although difficult to follow. Without the benefit of a plant tour, it is difficult to gain a clear understanding of the spatial aspects of the flood analysis. We are not convinced that all flooding sources or water propagation effects have been considered. Based on the details provided, the central conclusion - namely that there is no significant threat from internal flooding - seems reasonable. We do, however, have one comment:

On page 7.1-19 of Section 7.1.2.2 of the PRA, flooding and subsequent loss of core spray was disregarded following a small above core LOCA outside containment in the Reactor Building that is not isolated. The reason for disregarding this is that sufficient flow will be diverted to the torus via the ADS, and there will not be enough steam and water in the Reactor Building to cause a failure of the core spray pumps. A manual operator action is most likely needed to open the EMRVs in this case, yet the event was screened out. No reference is provided to substantiate this conclusion.

II.1.1.5 Utility Peer Review

Details of the independent peer review are provided in Appendix D of the Level 1 PRA. The PRA was reviewed using two parallel efforts: one by an independent in-house review group (IHREG), the other by an external consultant. Both reviews took place early in 1991.

The IHREG was comprised of a multi-disciplinary and multi-organizational group of management personnel not directly involved in the Level 1 PRA. The group met on nine occasions from to February to May, 1991. Based on the comments listed in Appendix D, the group performed an in-depth review of the PRA.

Review by plant personnel who were not familiar with PRAs occurred mostly at the Event Sequence Diagram (ESD) level. **As stated in the PRA, "*It is sometimes difficult to directly link the ESD with the plant models rules. This provides further justification for the reader to ensure that they fully understand the plant rules files rather than focusing solely on the ESD diagrams and discussions.*"**

The external review was performed by Dr. David H. Johnson, a consultant from PLG. The nature of the comments indicate that his review was detailed. **It is worth noting that the independent reviewer was also the project manager for PLG's contribution of the Level 1 PRA.**

II.1.2 Review of Accident Sequence Delineation and Systems Analysis

II.1.2.1 Initiating Event Review

The review of initiating events was carried out as recommended by Section 3.1.1 of the Draft Review Guidance provided by the NRC. The findings of the review are as follows.

The identification of initiating events (IEs) was performed using a "master logic diagram" based on all plant safety functions. The master logic diagram was constructed to guide the effort of searching for ways in which the hazard of radioactive material release may become unacceptable by a loss of control of the essential safety functions. The diagram is similar to a fault tree in construction in that it depicts various conditions which might lead to a release of radioactive material. The approach represents a thorough method for identifying possible IEs. **The master logic diagram appears to omit containment bypass sequences.** The diagrams produced a list of initiating events at the "functional" level.

The list of final IEs was produced from the logic diagram by reviewing plant and industry operating experience, other PRAs, and feedback from other parts of the risk model (i.e. systems analyses). The completeness of the list was verified by a review of operating procedure manuals (OPMs).

The OCNGS PRA reports initiating events as groups, rather than distinct events. At this level, it is difficult to determine if any events were screened out during quantification. Based on the information presented in Section 4.6, some events were screened.

The OCNGS IPE appears to not justify screening out of several initiating events, namely:

1. **Feedwater line breaks outside containment**
2. **Core flow blockage initiating events**
3. **Leakage at CRD or instrumentation penetrations or RWCU bottom head piping from the bottom of the vessel.**

No HVAC-related IEs were analyzed. The Updated FSAR at OCNGS concludes that safety related equipment is capable of operating without room cooling, including the control room. Some rooms require ventilation fans, such as electrical switchgear rooms, but these failures are not modeled explicitly as initiating events.

The OCNGS PRA provides a detailed discussion of the dependencies between IEs and mitigating systems, including front-line and support systems. Table 7.3-8 is the "Initiating Event Impact Table" which provides a summary of how each initiating event group affects the split fractions used in the model.

The quantification of initiating events is described in Section 4.6 of the Level 1 PRA. This section provides a detailed discussion of the methodology and considerations used in the IE quantification process.

The methodology used to develop plant-specific IE frequencies was the same as that used to quantify component failure probabilities for this study. The method is based on the Bayesian interpretation of probability, and involves the development of a prior distribution for "generic" information regarding an event probability. The event probability is then modified using plant-specific evidence. Some initiating events, such as loss of TBCCW, were quantified using a systems analysis approach.

The generic plant frequency distribution is taken from operating experience reported for 29 BWRs. Not all BWR data were used for all events because of differences in plant designs. Plant-specific data were taken from Oyster Creek Scram Data and transient event reports. Error factors, or similar information regarding the distribution of generic data, are not provided in the IPE. This is important information, since the error factors can heavily influence the final point estimate used in the CDF quantification.

The quantification of IE frequencies is reasonable. The submittal makes effective use of both generic and plant specific data. The LOCA data is not referenced, however. Also, a spot-check of a key IE - Loss of Offsite Power - showed that an error factor of approximately

12 was used for this event (assuming a lognormal prior distribution). Typically, this data comes from public utilities commissions and is of very high quality. One would expect an error factor of less than 3. The high error factor tends to reduce the mean value. Assuming an error factor of 3, instead of 12, produces a mean value of 6.2E-2 rather than 3.2E-2: almost twice the value used in the IPE. Furthermore, if the plant evidence were 1, instead of 0, with an error factor of 3, the LOSP frequency jumps to 8.0E-2. Since the LOSP initiator dominates this IPE (and many others) an examination of the prior distribution and plant-specific evidence may be warranted.

Several of the more likely initiating events, such as reactor trip and turbine trip have fairly low frequencies at Oyster Creek - between 0.7 and 0.9 per year. It is not unusual to see 3-4 turbine trips per year. The low values for these more likely events may be due to the maturity of the plant.

In summary, the initiating event data have been derived from a combination of generic and plant-specific data. The sources have been identified, in most cases, although some of the LOCA data is not referenceable. The discussion of initiating event quantification is thorough. The impact of initiating events on front-line and support systems has been modeled.

A better description of the process for screening initiating events from consideration should be provided.

II.1.2.2 Review of Front-line and Support Systems Analysis

The following front-line and support systems were analyzed in detail:

Front-line Systems

- Isolation Condenser
- Turbine Trip and Bypass
- Reactor Protection
- Main Steam Isolation

Core Spray

Containment Spray

Recirculation Pump Trip

Condensate/Feedwater

Automatic Depressurization System (ADS)

Standby Liquid Control

Containment Isolation

Standby Gas Treatment

Firewater

Condensate Transfer

Control Rod Drive Hydraulics

Reactor Building Isolation

Main Steam Relief

Torus Vent

Support Systems

AC Electrical Power

DC Electrical Power

Service Water

Turbine Building Closed Cooling Water (TBCCW)

Engineered Safety Features Actuation System (ESFAS)

Circulating Water

Instrument Air

For each of the above systems, the IPE presented a brief description of the system as modeled, a description of the top events considered, and success criteria. References are provided for the bases of most success criteria. Also provided are support systems required, systems supported, configuration and operation, periodic testing requirements, maintenance, operator actions, the potential to cause an IE, applicable Technical Specifications, modeling assumptions, split fraction definitions, common cause analysis, and results. The discussions are complete and to the level of detail adequate for review. Fault trees are enclosed, but that is not a requirement.

The final list of systems analyzed was developed using a screening process. A preliminary list was developed using a previous PRA for OC and by review of other plants' PRAs. The FSAR was also used to identify key safety systems. The final list was developed following the completion of the plant logic model. The plant model then provided the criteria for including a system in the final list.

No detailed discussion of HVAC systems is provided in the Submittal. The Updated FSAR at OCNGS concludes that safety related equipment is capable of operating without room cooling, including the control room, but that some rooms require ventilation fans, such as electrical switchgear rooms. The PRA does model ventilation for some systems, for example, for the 480 V switchgear, but it is not clear if all required ventilation is considered. For example, for the containment spray system, the model assumes that room coolers are not required, but no justification for this assumption is provided. No discussion of the need for cooling (or justification for the lack of need) for the 4160 V switchgear is provided. (Perhaps the 4160V switchgear is located in the 480v switchgear rooms, for which ventilation is modeled. The PRA is not clear on this issue.) We could find no discussion of HVAC for the control room in the PRA.

No modeling of the recirculation system was performed aside from the recirc pump trip system for ATWS sequences, and the consideration of a pump seal LOCA as an initiating event. As discussed later in section II.1.2.5, the submittal is deficient in not addressing seal cooling during mitigation of transient accident initiating events.

Based on the review, it is concluded that all important front-line and support systems required for prevention of core damage are modeled in the OCNGS IPE, except for HVAC and recirc pump seal cooling. Further justification for not modeling these systems should be provided.

II.1.2.3 System Dependencies and Support Systems

The OCNGS IPE performed a comprehensive analysis of system dependencies and support systems. Inter-system dependencies were treated in three groups: support to support, support

to front line, and front line to front-line. Dependencies are examined at different levels depending on the nature and configuration of the system. Some dependencies are listed at a train level or subsystems, or complete systems. The information in these tables was then used to generate impact tables which provide the split fractions to be used when top events, or combinations of top events are failed.

As a point of clarification, in the discussion of top event DB, page 7.4-3 of the PRA states that "Failure of this top event is assumed to result in a failure of all equipment that requires battery A and B DC control and start power." If this is true, failure of DB should cause a total loss of MDS and other systems dependent on DC power. This condition is not reflected in the model. If this is not the case, the statement in the PRA should be corrected.

From the review, it is concluded that the IPE treated dependencies between plant systems in a reasonable and consistent manner. No deficiencies were identified, other than the issues associated with HVAC and recirc pump seal cooling as previously discussed.

II.1.2.4 Treatment of Common Cause Failures

Common cause failures are treated, in the OCNGS IPE, by means of explicitly incorporating dependencies in the systems modeling, and implicitly, by quantifying the likelihood of certain faults due to common causes.

Definitions of common cause, common mode and dependent failure are only given by example, rather than by strictly defining the scope and extent of these types of faults. This lack of definition leaves the reviewer in a questioning state regarding the common use of these terms.

The approach used to quantify dependent failures is the well-known multiple Greek letter (MGL) method developed and refined by PLG. The OC IPE relies heavily upon referenced materials to document the method, procedures and database used in the CCF treatment. A succinct development of the Oyster Creek-specific MGL database is given in the OCPRA section 4.4.

An abbreviated vertical review of the contribution of Diesel Generator CCF to core damage frequency revealed extreme difficulty in simply tracing the database Z_DGS_ through its accident sequence. The numerical value for diesel generator CCF probability appeared quite low in magnitude.

One potential bias that may have been introduced into the quantification of CCF, and which could not be evaluated is the effect of censoring the event data, without corresponding censoring of the population data. Simply put, if the generic database of common cause events is reviewed specifically for potential to occur at Oyster Creek, and some of the events are deemed inapplicable, then a similar exercise must be conducted on all other plants to determine the population exposed to the same CCF event. This bias may need to be investigated further.

Because of the nature of the rules-modules approach, the contribution of common cause events to total CDF cannot be easily derived. As a result, significance of common cause events could not be determined for OCNGS.

II.1.2.5 Review of Event Trees

The OCNGS does not explicitly use event trees in the quantification of core damage frequencies. Rather, it uses the rules methodology -- a modification to the event tree approach which is logically equivalent but computationally more efficient. The rules simply represent the conditions which must be present for the assignment of the status of the top event. No pictorial event trees appear in the Level 1 analysis.

The development of scenarios was accomplished by the construction of event sequence diagrams(ESDs). ESDs were used to document success paths available to mitigate the consequences of initiating events and subsequent system failures. The ESDs are reviewed and then translated into rules. It is important to note that the ESDs provide only guidance to the modeler and a framework for non-PRA analyst reviews. The plant model rules files are the only representation of the plant as modeled and quantified. In general, the rule sets are more difficult to peruse than event trees.

Functional success criteria is stated in Section 8 of the PRA (Plant Model Endstates). Specific criteria for system success is provided in the system notebooks (Appendix E). In general, the IPE rules development process provides the bases for all success criteria. However, there is no overall summary of the criteria, and it is not obvious where the analysis diverged from the FSAR analyses.

The linkage of transfers between modules is simplified by the use of interim variables. This complicated the review of the linkage between modules. A spot check of some important links, such as transients to LOCAs showed no discrepancies. While it does not affect our conclusions, we do have one comment:

The ESDs were very difficult to follow with respect to linkages between modules. For example the reader is transferred from Loss of Feedwater Control module to ESD LL3, which does not exist. Attempts to find LL3 among module LT3 also prove futile. Logistic errors such as these do not affect the outcome of the study, but they do make the auditing process very frustrating.

The following paragraphs provide our specific comments related to each individual event response modules.

General Transient Module

The general transient module is used to evaluate plant response to a broad range of initiating events. Some initiating events enter this module directly, while others, particularly those support system failures that lead to plant trips, cascade into the general transient module. This module is directly linked to the long term transient model. Initiating events explicitly modeled are reactor trips and turbine trips. Top events questioned are appropriate for the IEs identified. They include: reactor trip, turbine trip, CST inventory, condenser vacuum, turbine bypass, control of feedwater (level setdown), mode switch to shutdown, main steam isolation, condensate and feedwater systems.

The event sequence diagrams show transfers from the general transient module to long term LOCA modules. A spot-check of the rules modules verified that this link occurred. This module is complete and no deficiencies were found.

Loss of Feedwater Control Module

This module was developed for a specific initiating event - the loss of feedwater control. This event is separated from the other general transients because the only automatic action expected is a turbine trip on high reactor water level. The event may lead to steamline flooding and subsequent failure. Events appearing in this module are turbine trip, reactor scram, CST inventory, condenser vacuum, turbine bypass, main steam isolation, IC isolation, condensate, feedwater, and MSIV closure. This module appears accurate and complete.

Long Term General Response Module

The long term general response module is used to evaluate the long term plant response to the general transient and loss of feedwater control modules. This module is quantified in conjunction with the support module and either of the aforementioned transient modules. The output of this module is directed to the recovery module and eventually to the plant damage state module. Functions modeled in this module include recirculation pump trip, isolation condenser actuation, condensate transfer, relief valve operation, core spray, containment spray, ADS, and others.

It should be noted that the success criteria for core spray in the IPE differs from that assumed in the FSAR. While the FSAR Chapter 15 analysis requires two main pumps and one booster pump be operable, the IPE assumes that only one main and one booster pump be operable. This assumption is based on recent LOCA analyses for OCNGS.

The general transient module does not address the integrity of the recirculation pump seals following a general transient. That is, there is no consideration of recirculation pump seal LOCAs occurring during the mitigation portion of a general transient. If the recirculation pumps at OCNGS are typical of BWRs, the seals are cooled by injection (probably from the CRD system) or by controlled leakage with cooling by the pump lube oil/bearing cooling water system; however if both these modes of cooling are lost, the seals will fail even if the

pumps have tripped. The loss of seal cooling affects five pumps at Oyster Creek. Also, Oyster Creek has isolation condensers instead of a RCIC system, and contrary to the case with RCIC, core cooling does not directly involve injection to the vessel, and therefore losses through the seals are not directly compensated for. Sections 8.1.1 and 8.2.1.2 of the PRA indicate that long term makeup to the vessel is not required when cooling with the isolation condensers is provided.

Issue 1.9.37 of the FSAR for Oyster Creek states that the leakage resulting from loss of seal cooling is acceptable for two hours; however, this does not justify ignoring the long term requirements for seal cooling or the consequences of the loss of seal cooling. For example, if a seal leak rate of 100 gpm is assumed, and a vessel inventory of 200 gal per inch is assumed with 300 inches of water normally available above the top of the core, then without makeup to the vessel the top of the core will uncover in about 10 hours. The seal LOCA can be isolated by closing the isolation valves in the recirculation piping, but this requires recognition of the event and electrical power.

This module is considered deficient because it does not address the possibility of a post-transient LOCA via the recirculation pump seals.

Small LOCA Module

The small LOCA spectrum of breaks is assumed to include those leak sizes below which depressurization due to inventory loss is not expected to reduce RPV pressure below core spray or condensate pump shutoff head before the onset of core damage. An equivalent hole size for small LOCAs is not stated in the IPE submittal. PRAs for BWRs typically consider several small LOCA break sizes for steam/water breaks, since steam flashing provides more efficient depressurization.

The following functions are modeled in the small LOCA analysis: reactor trip, condensate, feedwater, ADS, core spray, fire water system (possible backup to core spray), and MSIV closure.

Because plant response may vary depending on the location of the break, six small LOCA groups are defined. The interfacing systems LOCA (ISLOCA) is included as a small LOCA group.

A review of the rules modules for small LOCAs shows that the rules are functionally correct and account for short term plant response to small LOCAs at varying locations.

Large LOCA Module

This module addresses short term plant response to large LOCAs. Due to the design of the Oyster Creek reactor core, there are a number of possible locations for a large LOCA that will not allow RPV reflood above the top of active fuel (TAF). For this reason, the IPE modeled two basic categories of large LOCA: above core and below core, depending on the break location with respect to TAF. **A feedwater line LOCA outside containment is not addressed, as mentioned in Section II.1.2.1 of this review.**

This module considers the following front-line systems: reactor scram, condensate storage tank, condensate, core spray, and fire water injection (backup to core spray). An interfacing systems large LOCA (overpressurization of the RWCU) is included in the large leaks below TAF.

A review of the rules modules for large LOCAs shows no deficiencies or inconsistencies for the IEs identified. The feedwater line LOCA was not addressed.

Long Term LOCA Response Module

This module models the long term plant response to both small and large LOCAs. All LOCA initiating events utilize this module. The output of the long term LOCA response module is input to the recovery module which is in turn analyzed by the PDS module. The long term LOCA module addresses the performance of the following systems: primary containment isolation, containment cooling, containment venting, reactor building isolation, and standby gas treatment. These systems primarily provide containment cooling and isolation functions. **The rules module is functionally correct, no deficiencies were found.**

Containment Heat Removal Recovery Module

This module is used in the Oyster Creek IPE to address the recovery of long term containment heat removal. This form of recovery applies to scenarios in which long term vessel injection is available from sources outside containment, with continuing discharge of decay heat to containment, eventually challenging the strength of the primary containment.

The functions which are modeled in this module include: long term recovery of dc power bus C, recovery of core spray, recovery of instrument air, recovery of torus vent, and long term RPV cooldown. Although it is shown as a single module, separate sub-modules are created for general transient and LOCA response. The output from this module is then directed into the PDS module.

The module is functionally correct. **No deficiencies were found.**

II.1.2.6 Dominant Sequences

The mean core damage frequency for OCNGS is 3.7E-6/yr. The IPE reports CDF in the following ways:

- initiator contributions to CDF
- system contributions to CDF
- operator action contributions to CDF
- plant damage state contributors to CDF.

A general narrative of the sequences is given in the rationale for specific initiators, systems, or operator actions appearing as dominant contributors. Detailed narratives for each of the top 20 scenarios are provided in Appendix C. These contribute to about 62% of the total CDF.

Dominant Initiating Events

The dominant IEs are: Loss of Offsite Power (Station Blackout), Turbine Trip, Reactor Trip, MSIV Closure, Total Loss of Feedwater, Loss of Condenser Vacuum, and Loss of TBCCW. Together, they account for about 76% of the total CDF. The largest other contributors are Loss

of Intake Structure, Electric Pressure Sensing Regulator Fails Low, and a Large LOCA Below Core, each contributing about 3% to the total CDF.

Dominant System Failures

The system failure which contributes most to the CDF is Electro-Matic Relief Valve (EMRV) failure to close (importance of 48%). Essential AC power buses contribute to 37% of the CDF. DC power buses contributed to about 33% of the CDF. The IPE takes credit for an offsite emergency power source for its LOSP recovery model, however the amount of credit is not relatively large. The emergency offsite source - an offsite combustion turbine - is not yet in place, but is scheduled to be in operation after the 14R outage.

Dominant Human Actions

Human actions contribute approximately 21% to the total CDF. The operator actions as modeled in the PRA range from post-trip control to emergency actions, to recovery of systemic or functional failures. The important operator actions with regards to CDF are initiation of containment cooling, initiation of core spray, recovery of DC power, recovery of offsite power, initiation of IC makeup, and containment venting. No single operator action is disproportionately larger than the others, in fact, the top ten operator actions contribute from 2.76% to 1.03% to the total CDF.

Dominant Equipment Failures

Failure of an EMRV to close is significant because it allows reactor coolant to discharge to the torus (equivalent to a small LOCA) which requires ADS to allow for low pressure coolant injection. The continued heat rejection to the torus presents demands on the containment cooling systems. The absence of containment cooling can lead to loss of NPSH for the core spray pumps.

It is noted that a common cause failure of the DGs does not appear explicitly in the top sequences. It appears in the top sequence but is effectively masked. The modeling of the common cause failure of the DGs is rather confusing, due to the change in variable names

for the CCF event from EE4 to EDD. Furthermore, the narrative describes the events as if they were independent failures.

Dominant Sequences

The following paragraphs summarize the dominant sequences in the IPE. The reviewers performed a spot check of the CDF calculation, based on the list of sequences provided in Appendix C of the PRA (Detailed Results). In some cases, we could not reproduce the calculational value provided in the text. The IPE results in Section 3 (and Appendix C of the PRA) represent point estimate values and should correspond to our spot-check calculations. In the sequences below, our hand-calculated value is shown in parentheses. The difference in our estimate and the value quoted in the IPE is because the IPE included some success events with probabilities slightly less than 1.0 - we assumed 1.0 for these events.

7.60E-7 ([illegible]) Loss of offsite power, followed by independent failure of both emergency diesel generators (station blackout) with successful lift of EMRVs and actuation of the isolation condenser. At least one of the EMRVs fails to close. Attempts to recover offsite power fail, leaving no source for reactor level makeup. Reactor level drops below the top of active fuel, regardless of IC operation. Fuel failure is assumed to occur shortly thereafter. Scenario timing is 37 minutes from IE to core damage. The recovery of offsite power event (mean probability of failure .258) includes credit for the future emergency power source at the Forked River Site, adjacent to OCNGS.

2.59E-7 (2.7E-7) Turbine trip IE, followed by independent failures of both divisions of DC power for 3 hours. Loss of DC power disables all 4160 VAC switchgear. DGs may start and run, but cannot be loaded onto buses, resulting in a station blackout. Reactor makeup is not possible; however, MSIVs will close on the loss of DC power. Safety valves will cycle, eventually depleting reactor inventory until fuel is uncovered. Scenario timing, from IE to core damage is only about 20-30 minutes.

2.10E-7 (2.2E-7) This scenario is identical to the above scenario, except that the IE is a reactor trip.

1.29E-7 (1.3E-7) This scenario is identical to the above two scenarios, except that the IE is a inadvertent closure of an MSIV.

1.0E-7 (1.0E-7) Loss of offsite power with EMRV failure to close and failure of core spray system. DGs successfully start and load, and core injection is aligned through the CRDs. Initial post-trip pressure is reduce by the EMRVs, one of which fails to close. Core spray experiences an independent failure. CRD injection is unable to match flow out of the EMRV and core uncovery results.

Similar narratives are provided for 15 additional sequences.

The submittal provides an excellent presentation of results with respect to the front-end analysis. The level of detail is sufficient to identify dominant contributors.

II.1.2.7 Front-End and Back-End Interfaces

The Level 1/Level 2 interface was accomplished through a set of Plant Damage States (PDS). Rules modules were used in place of event trees to delineate all possible PDSs. These modules examined all possible Level 1/Level 2 interfaces such as containment cooling and torus venting. The PDS quantification was performed using the *RISKMAN* code, similar to the process followed to quantify the core damage frequency. The PDS screening criteria used a truncation value of 5E-10, although, if sequences were combined for simplification, the truncation limit became zero.

Extensive combining of PDSs was performed to keep the number of distinct states below the code limit of 150. This was performed by consolidating zero or low frequency endstates into combined endstates. **No further information is provided on how the consolidation was performed. For example, the cutoff limit for sequence quantification was found as a footnote in a large printout of sequences.**

Based on our review, the following conclusions have been drawn:

- Important sequences were not screened out. It is possible that consolidation may have caused some sequences to be incorrectly categorized, although this does not appear to be the case. The screening criteria is consistent with NUREG-1335 guidelines
- The bases for grouping logic is not provided in the IPE. This should be provided in some form.
- Plant Damage States explicitly considered all important reactor and containment systems. The Level 1 sequences contain all necessary information for the PDS analysis, therefore no systems were added in the PDS modules.
- The IPE does not address the timing of the containment failure with respect to the failure of core cooling equipment by high temperature. In other words, the IPE does not address, upon loss of cooling to the containment and torus, which failure occurs first: containment due to overpressure, or core cooling equipment due to high temperature or loss of adequate NPSHA. **This important physical insight should be addressed.**

II.1.2.8 Multi-Unit Considerations

Oyster Creek Nuclear Generating Station is single unit facility. Multi-unit considerations are not applicable.

II.1.3 Review of the IPE Quantitative Process

The OCNGS IPE used the *RISKMAN* code package for sequence quantification and plant damage state analysis. A description of how the code produced qualitative and quantitative results is not provided. Details are provided in Section 5.4 of the PRA, with broad references to the integrated quantification process. **Informative details pertaining to the quantification process should be provided.** A truncation limit of 1E-13 is reported in the table of top 100 sequences, Appendix C of the Level 1 PRA. No truncation limits were reported in the systems analysis. A truncation limit of 5E-10 was used in the plant damage state screening analysis.

To evaluate the uncertainty in the results, a dominant scenario model was developed consisting of sequences which represented 95% of the core damage frequency. Probability distributions for initiating events, component failure data, and human errors were then input to the model. Monte Carlo mathematical methods were then used to propagate uncertainties through the dominant scenario model. **The overall approach to uncertainty is widely accepted and no deficiencies were found.**

Frequency distributions were plotted for total core damage frequency and the top six plant damage states' frequencies. The point estimate for the CDF is 3.69E-6/yr. The CDF distribution based on the uncertainty analysis is:

Mean	3.69E-6
95th percentile	9.82E-6
Median	3.2E-6 (taken from Figure 9.1-1 of PRA)
5th percentile	1.31E-6

The top 100 most dominant sequences are provided in Appendix C of the PRA.

The quantification process is valid, although details of the procedure are scattered throughout the PRA. It would be convenient if these details were concentrated into a single location.

II.1.3.1 Quantification of the Impact of Integrated Systems and Component Failures

Split fractions were quantified separately in the CDF calculation. Because component-level information is lost in the process, the quantification of CDF is fully integrated only by use of the dependency matrices.

A sensitivity study was performed on several key variables in the study: loss of offsite power recovery, EMRV failures to close, and recovery of containment heat removal (including recovery of DC power and containment spray). An analysis was made regarding the sensitivity to data and some key assumptions. The analyses conclude that changes to data or assumptions regarding

the LOSP and containment heat removal recoveries do not have a significant effect on overall results. The same is not true for the case of the EMRVs. The study concludes that relaxing assumptions for the EMRVs would reduce total CDF by 18%, but those same conclusions fail to point out that a 10% increase in failure probabilities produces a 5% increase in total CDF.

It would have been beneficial to examine the sensitivity to changes in data or assumptions pertaining to events which have traditionally dominated other PRAs, such as common cause failure of both emergency diesel generators.

II.1.3.2 Fault Tree Component Failure Data

In general, the OCNGS PRA database was developed using a Bayesian update process to combine the cumulative experience from a large population of nuclear plants with a comprehensive plant-specific database that represents a collection of over 10 years of operational and maintenance experience at Oyster Creek. The following sections discuss the submittal's treatment of fault tree component failure data. Separate discussions are presented for plant-specific, generic, and common cause component failure data.

Plant-Specific Data

The IPE for OCNGS made extensive use of plant-specific records for failure data, success data, and maintenance/testing unavailabilities. Components recommended by NUREG-1335 for analysis with plant-specific data include: emergency core cooling pumps, batteries, diesel generators, electric buswork, and breakers. All of these components, except electrical buswork, were analyzed. In addition, service water pumps, instrument air, primary containment isolation, ADS valves, and other components were quantified using plant-specific data.

Generic Data

All generic data is listed in Table 4.3-8 of the Level 1 PRA. No description is provided regarding the generic data used. Error factors are not presented, either. The only information provided is a reference to PLG's database for nuclear reactor PRAs. Since

only 50 component failure rates utilized plant-specific data, this leaves a vast amount of failure data without a description of the data sources.

Common Cause Data

The approach used to quantify dependent failures is the well-known multiple Greek letter (MGL) method developed and refined by PLG. The OC IPE relies heavily upon referenced materials to document the method, procedures and database used in the CCF treatment.

Essentially, the development of the plant specific database begins with an industry-wide database of dependent failure events, and by way of engineering review, determines the vulnerability of the Oyster Creek plant systems to each individual failure event. Modifications to the individual events are made to correct for dissimilarities in equipment physical parameters and levels of system redundancy (i.e. 3 train system versus 2 train system). Of vital importance, it should be noted that the data are censored at this point in the data analysis, if individual events are deemed inapplicable (not possible in Oyster Creek). A comment was made on 4.4-1, "the primary source of generic common cause data ... was the PLG generic common cause database", yet we could find no other event data source documented. Specifically, we would expect an Oyster Creek event reporting system to have been utilized, or mentioned.

II.1.4 Review of IPE Approach to Reducing the CDF

II.1.4.1 Methodology for Identification of Plant Vulnerabilities

This section presents our comments regarding our review of the IPE's methodology to identify plant vulnerabilities. The OCNGS IPE presents a clear definition of a vulnerability as "any core damage sequence that exceeds 1E-4 per reactor year, or any containment bypass sequence that exceeds 1E-6 per year." **No vulnerabilities were found, according to the IPE submittal's definition. The reviewers accept the conclusion that no vulnerabilities exist at Oyster Creek. It is possible, using the OCNGS criteria, to have a component contribute to 99% of the CDF, and yet no vulnerabilities would be identified by their numerical criteria.**

II.1.4.2 **Plant Improvements and Planned Modifications**

Despite having no identified vulnerabilities, a number of potential areas for low-cost improvements were developed that could, according to the licensee, enhance overall reactor safety.

1. Development of an emergency procedure for Loss of Offsite Power.
2. Development of an emergency procedure for Loss of DC Power
3. Evaluate the purchase of a portable DC power generator.
4. Increased training on the importance of core spray system.
5. Changes to maintenance scheduling for the core spray system to improve downtime.
6. Programs instituted to reduce blockage and fouling of the isolation condensers.
7. Modifications (after 15R) to implement the Reactor Overfill Protection System.
8. Consider the development of specific guidance, training, and procedures for reactor overfill transients.
9. Increased emphasis in training on key operator actions as defined by the IPE.

These low cost modifications show a good application of IPE insights. The impacts of these planned modifications were not quantified however. In light of the strong dependencies on DC electric power, the purchase of a portable DC generator (item #3 above) should be evaluated.

The IPE already takes credit for the following planned modifications (implemented after 14R):

1. Use of the Forked River site for alternate AC power.
2. A hard-piped containment vent system.
3. Provisions for an all-manually initiated containment spray system.

II.1.5 Review of the Licensee's Evaluation of the DHR Function

II.1.5.1 IPE Focus on Reliability of DHR

The OCNGS IPE modeled the plant's dependency on the DHR function. Failure to remove decay heat is reflected in the Level 1 PRA as plant damage sequences with LH for the second and third characters. These damage states contribute to about 4% of the total calculated core damage frequency. The licensee considers this value low enough to declare the issue closed. The reviewers contend that the criteria for identifying DHR dependency is not consistent with the IPE definition of DHR. By focusing strictly on containment heat removal, many other facets of DHR are omitted.

II.1.5.2 IPE Considered Diverse Means of DHR

The IPE considered several diverse means of DHR. The various means of DHR, in order of preference, are:

1. Feedwater/Condenser
2. Two Isolation Condensers with makeup from firewater
3. Containment Spray/Emergency Service Water, if EMRVs are available
4. Hard piped containment vent system (to be installed in outage 14R)
5. Firewater makeup to vessel using EMRVs.

The systems themselves were modeled in the IPE along with the support required for operation.

The issue of recirculation pump seal failures after a general transient has not been adequately addressed and this affects the resolution of DHR issues.

II.1.5.3 Unique Features

The Oyster Creek facility has several unique features with respect to DHR:

- There is no high pressure ECCS system, and no low pressure coolant injection system (LPCI). Later BWR designs have steam-driven high pressure core spray and LPCI in addition to a low pressure core spray system.
- The station batteries have a three-hour capacity. *Some plants have longer battery capacities; this is important since DC power is required to open the relief valves and depressurize the system.*
- There is no RCIC system. Cooling is provided by a passive convection cooling system (isolation condensers).
- Dedicated containment venting is available.

III. OVERALL EVALUATION AND CONCLUSION

The OCNGS IPE submittal is based on a Level 2 PRA for internal events. The overall methodology is consistent with Generic Letter 88-20 and NUREG-1335. The information provided supported a horizontal review.

The submittal includes a full set of system analyses for both front-line and support systems. The initiating events list includes events typically appearing in PRAs for BWR plants; however, a few have been omitted, such as leaks in the RWCU line or feedwater line breaks outside containment. The IE frequencies were derived using a reasonable approach, although the errors factors were not provided.

The systems analysis portion is complete. Fault trees were drawn for most system failures. All major front-line and support systems were modeled, with the possible exception of recirculation pump seal cooling. Loss of HVAC room cooling is not included, based on FSAR analyses. A combination of plant specific and generic data were used to quantify basic event probal .ities. Common cause failures were evaluated using the Multiple Greek Letter Method and are propagated correctly through the model. Some data censoring occurred which may . quire further investigation.

The overall quantification of CDF was performed using the rules methodology - an evolution of the large event tree / small fault tree method. This method is credible, however, it is not amenable to review. Risk results were presented in an excellent manner.

In conclusion, it is our opinion that the OCNGS IPE is a good Level 1 PRA, with minor weaknesses. Responses to the issues raised in this review will help to correct [illegible] weaknesses in the submittal. The approach follows industry practice and it supports OCNGS's conclusion that no vulnerabilities exist at the facility. Further information should substantiate this conclusion.

ENCLOSURE 3

OYSTER CREEK INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(BACK-END)

SCIE-NRC-212-92

OYSTER CREEK
INDIVIDUAL PLANT EXAMINATION
BACK-END
TECHNICAL EVALUATION REPORT

Revision 1

W. H. Amarasooriya
J. F. Meyer

Prepared for the U.S. Nuclear Regulatory Commission
Under Contract NRC-05-91-068-03
March 9, 1994

SCIENTECH, Inc.
11821 Parklawn Drive
Rockville, Maryland 20852

OYSTER CREEK
INDIVIDUAL PLANT EXMINATION
BACK-END
TECHNICAL EVALUATION REPORT

Preparer

W.H. Amarasooriya
Safety Analysis Group

Technical Review

Mohammad Modarres
Manager, Safety Analysis Group

Approval

James F. Meyer
Manager, Safety Analysis Group

SCIENTECH, INC

March 9, 1994

TABLE OF CONTENTS

1. INTRODUCTION

This technical evaluation report (TER) documents the results of the SCIENTECH Review of the Oyster Creek Individual Plant Examination (IPE) Back-End submittal [1]. This technical evaluation report complies with the requirements of the U.S. Nuclear Regulatory Commission contractor task order for Step 1 reviews, and adopts the NRC Step 1 Review objectives, which include the following

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

- To assess the strengths and the weaknesses of the IPE submittal
- To pose a preliminary list of questions about the IPE submittal, based on this limited Step 1 review
- To complete the IPE Evaluation Data Summary Sheet

In Section 2 of the TER, we summarize our findings, and briefly describe the Oyster Creek IPE submittal as it pertains to the work requirements outlined in the contractor task order. Each portion of Subsection 2.1 corresponds to a specific work requirement. In Subsection 2.2, we set out our assessment of the Oyster Creek submittal strengths and weaknesses. In Section 3, we present our evaluation of the Oyster Creek IPE overall, as well based on the Step 1 review. Appended to this report is the IPE Evaluation Summary Sheet, which we completed on the Oyster Creek IPE.

2. CONTRACTOR REVIEW FINDINGS

2.1 Review and Identification of IPE Insights

This section is structured in accordance with Task Order Subtask 1.

2.1.1 General Review of IPE Back-End Analytical Process

2.1.1.1 Completeness

The Oyster Creek Individual Plant Examination (IPE) Back-End submittal is essentially complete with respect to the level of detail requested in NUREG-1335. The IPE submittal meets the NRC sequence selection screening criteria described in Generic Letter 88-20, and summarizes how this was done (see Table 8-1, page 8-5, of the Level 2 PRA).

2.1.1.2 Description, Justification, and Consistency

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

2.1.1.3 Process Used for IPE

As noted in Subsection 1.3, page 1-1, of the IPE submittal report, "The analysis of Oyster Creek containment performance was accomplished in the context of extending the Level 1 study [2] to Level 2, as defined in NUREG/CR-2300."[3] The Level 1 model quantification led to identification of 19 plant damage states (PDSs) with a frequency of 1×10^{-8} per year or greater. For the Level 2 analysis, these PDSs were condensed into a set of seven key plant damage states (KPDSs) for which containment event trees (CETs) were developed. Representative sequences were selected for each KPDS. MAAP3.0B, Rev. 7.03, was used to calculate severe accident event timing and containment loads for each of the representative sequences.

2.1.1.4 Peer Review of IPE

Independent peer review of the Oyster Creek IPE by the six-member Independent In House Review Group (IIHRG) is discussed in Subsection D.3.2, Appendix D, of the Level 2 PRA. The IIHRG review comments and their disposition are described in Subsection D3.3. Of the nine IIHRG comments listed, only one led to a textual revision. It was concluded that the other IIHRG comments could be adequately addressed by responding only to the reviewers who made them. It appears that the Oyster Creek IPE did receive adequate and appropriate peer review.

2.1.2 Containment Analysis/Characterization

2.1.2.1 Front-end, Back-end Dependencies

The interface between the Level 1 system analysis and the Level 2 containment analysis consists of a set of plant damage states (PDSs), as discussed in Section 5 of the Level 2 PRA. A PDS is the result of one or more of a number of physical conditions, which were analyzed in Level 1, including the following: pressure inside the reactor at the time of vessel breach, presence of water on the drywell floor, containment pressure boundary integrity status, availability of water to cool the core debris, suppression pool cooling, and containment venting. Other secondary conditions were also considered. The Level 1 analysis of Oyster Creek ended, in principle, at the onset of significant core damage, which was defined as the time when "... the top of the active fuel is uncovered, and vessel water level is continuing to drop." The Level 1 event trees identified 19 plant damage states with a frequency of 1.0E-8 per year or greater. For the purpose of Level 2 analysis, these were reduced to a set of seven key plant damage states, selected on the basis of the Generic Letter reporting criteria. Each of the seven KPDSs received consideration during the subsequent Level 2 analyses. It appears that proper account was taken of front-end to back-end dependencies, and, overall, the analysis of front-end, back-end dependency is logically and clearly presented.

2.1.2.2 Sequences with Significant Probability

Accident sequences with a significant probability of occurrence were evaluated, as noted in the Level 2 PRA, Sections 5 and 8. A number of KPDSs with a higher annual frequency rate than 1.0E-8/year received further consideration, using the containment event tree (CET). These KPDSs were "... a factor of 10 lower that the NRC sequence frequency criteria," and are listed in Table 8-1 (on page 8-5 of the Level 2 PRA). The CET events used to further analyze the KPDSs were selected to address in-vessel core degradation, the potential for in-vessel recovery, the phenomena associated with ex-vessel progression, containment integrity challenges, containment failure, its timing, and the effectiveness of other safeguard systems to mitigate offsite releases. The CET end-states were binned together into a number of release categories. Because only one CET was actually developed, it had to be quantified for each KPDS, and therefore the CET branching probabilities (split fractions) varied, in most cases, for each KPDS. The CET used in the Level 2 PRA was developed to resemble the NUREG-1150 (and NUREG/CR-4551) accident progression event trees (APET) developed for the Peach Bottom examination. However, fewer events were analyzed during the Oyster Creek IPE (The examiners asked 145 questions at Peach Bottom, as compared with 16 at Oyster Creek). The Oyster Creek IPE submittal explains that answers to many of the questions asked at Peach Bottom were "implicit in the definitions of the Oyster Creek plant damage states" and therefore not included in the Oyster Creek CET. (In actuality, however, some of the questions whose answers were considered implicit were asked as part of the Oyster Creek CET.)

The Oyster Creek CET, shown in Figure 7-1, page 7-11, of the Level 2 PRA. See Subsection 2.1.3.3 for a further description of CET top events.

The quantification of the CET for each KPDS was carried through a number of split fractions defined for each applicable CET top event. The process appears reasonable, but it is difficult to follow the nature of each split fraction, the terminology employed, and the split fraction logic used (see Table 10-1 of the Level 2 PRA). The final results of the quantification of the CET sequences are not given. The results were used to define the release categories and to calculate their frequencies directly. The leading release fractions are discussed in Subsection 2.1.2.6.

2.1.2.3 Failure Modes and Timing

The Oyster Creek containment failure characterization is described in Section 6 of the Level 2 PRA, and it is detailed in an EQE Engineering calculation report, which is appended to the Level 2 PRA. Page 15 of the EQE Engineering report notes the following:

> [T]he potential failure modes of the containment shell are investigated. The loads considered include temperature, pressure, and dead load. . . . Failures of both the drywell and the suppression chamber (torus) are considered. The failure modes examined include:
>
> 1. Membrane failures of the drywell shell (sphere, cylinder, head shell)
> 2. Failure of the drywell head flange seal
> 3. Failure of the vent line from the drywell to the suppression chamber
> 4. Failure of the suppression chamber shell
> 5. Failure at penetrations

For each of the failure modes examined, the probability of failure was calculated as a function of internal pressure within containment metal temperatures ranging from 300°F to 1,200°F. A log-normal failure model was used to perform these calculations. An overall uncertainty value was estimated, using material-strength and modeling uncertainties, whose rationale for selection as the basis of the overall estimate was not given.

The containment capacities of each failure mode at temperatures of 300°F and 700°F are shown in Tables 6-1 and 6-2 of the Level 2 PRA. The failure mode with the least containment pressure was that of leakage through the bolted drywell head flange connection: 121 psig at 300°F after 70 hours. However, as 70 hours is a relatively long time for accident progression, the subsequent failure mode—the membrane failure of the drywell shell (134 psig also at 300°F)—was more significant (from the source term viewpoint).

The liner melt-through (a consequence of direct contact of the containment shell with fuel debris) was also analyzed. The results appear reasonable when compared with NUREG/CR-5423 (Table 10-10 of the Level 2 PRA). A difference factor of 2 cited in the Oyster Creek results is attributed to a 6-inch-high and 1- foot-wide curb on the drywell floor. The analysis of this difference from NUREG/CR-5423 the appears to have been adequate.

2.1.2.4 Containment Isolation Failure

Containment isolation failure is considered part of a plant damage state. One of the primary conditions taken into account before binning plant damage states is the "Containment Pressure

Boundary Integrity Status." This condition addresses the containment isolation failures and potential containment bypass, as well as early or late containment failures.

As stated on page 10-6 of the IPE submittal report, "Pre-existing leaks were not specifically addressed in this study. Because the Oyster Creek containment is continuously monitored for oxygen content to insure inerting, plant operating staff would be alerted to such leaks and would respond accordingly." This is reasonable and consistent with other back-end assessments of facilities with inerted containments.

2.1.2.5 System/Human Response

The CET considers possible methods for arresting the core under the top event VB of CET. Each KPDS is screened to identify the ones with reasonable recovery potential.

Top Event Number 11 in the CET (Emergency Crew vents Containment in Core Damage Scenarios) modeled the intentional venting of the suppression pool air space. Success of this event means that the vent flow capacity is adequate and that long-term containment failure would be precluded. Depending on the KPDS, the probability of this top event occurring (probability of failure to vent) ranges from 0.017 to 1.0.

It is assumed that the probability that the crew vents the containment does not change from "pre-core damage" venting in the Level 1 analysis to the "post-core damage" venting in the Level 2 analysis. In addition to wetwell (torus) venting, venting via the drywell is available and is "procedure-activated" for the representative sequence in the KPDS OJAU (note Section 10.11 on page 10-12).

About half of the CDF is allocated to "No Vessel Breach," resulting in no radionuclide releases. Vessel breach (after core damage) is prevented by either introducing fire protection water when the vessel is under low pressure or providing sufficient "control rod drive hydraulic system" flow when the vessel is under high pressure. For both vessel injection modes, operator action is required. (Note page 10-2 of the Level 2 PRA.) Thus it appears that this operator-controlled cooling function has a key bearing on the radiological release profile for Oyster Creek.

2.1.2.6 Radionuclide Release Characterization

The radionuclide release characterization is described in Section 11 of the Level 2 portion of the submittal. Release categories (qualitative descriptions of the containment event tree end- state bins) and associated source terms (quantitative descriptions of the CET end-state bins, including release timing and release fractions) are generated. As an aid in defining the release categories, a source term event tree (STET) is used, as stated in the submittal: "The purpose of such a tree is to define the different release categories for which the source term characteristics could be sufficiently different to warrant a separate source term definition."

The STET is shown in Figure 11-1. The characterization of the radionuclide releases from the containment are a function of the seven top events in the STET. A seven-character end-state

identifier was developed, as described on pages 11-2 and 11-3. The seven STET top events were keyed to either a plant damage state group or to the status of certain CET top events. The rules for binning CET sequences to release categories are listed in Tables 11-1 and 11-2. The results of the release category binning are shown in Table 11-3. Each release category was assigned to an enveloping release category "according to the rules of conservative condensation." From this, six key release categories (KRCs) were generated, as a simplified and conservative characterization of the releases to the environment. The source terms for these KRCs were calculated by selecting representative sequences and using MAAP to model the behavior and release of 12 radionuclide groups, as listed on page 11-5 of the Level 2 PRA. The analysis process and assumptions are described for each KRC starting on page 11-6. In Table 11-4, page 11-16, the frequency of each KRC is listed, while in Table 11-5, page 11-17, source term information (release fractions and release times) for each KRC is provided. In addition, Table 11-6, page 11-18 summarizes the distribution of CsI within the vessel and containment at the end of the MAAP runs.

The discriminator interrogatories used to define the release categories were the following:

- Reactor coolant system pressure at vessel breach?
- Drywell sprays available?
- Core damage arrested in-vessel?
- Time of containment failure?
- Size of containment failure?
- Containment bypassed?
- Suppression pool scrubbing prior to containment failure?
- Accident-mitigation in reactor building?

Generic Letter 88-20 states that the following should be reported: "any functional sequence that has a core damage frequency greater than 1×10^{-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." The Oyster Creek IPE submittal meets this reporting requirement. See Table 8-1, page 8-5, for a summary.

The radionuclide characterization is well developed and portrayed in the submittal. It appears to be reasonable assessment of radionuclide transport and release.

2.1.3 Quantitative Core Damage Estimate

2.1.3.1 Severe Accident Progression

The accident progression analysis, performed with MAAP computer code, is discussed in Section 9 of the Level 2 PRA. (Note the limitations and assumptions used in the MAAP analysis, as described in Section 4.3 of the Level 2 PRA.) MAAP results are discussed and presented in figures and tables for the following KPDSs:

- Low Pressure Station Blackout with Stuck-Open Relief Valve (PIFW)
- High Pressure Station Blackout (NIFW)

- Large DBA LOCA with No Core Spray (OIAU)
- Turbine Trip ATWS with SLC Failure (MKCU)
- Reactor Water Cleanup (RWCU) System Failure in Pressure Reducing Station (OJAU) - Bypass Sequence
- Loss of Feedwater with Failure of Scram Discharge Volume (SDV) to Isolate (MJAU) - Bypass Sequence
- Station Blackout with SDV Failure to Isolate (NJHW) - Bypass Sequence

A discussion of the phenomenological uncertainties of severe accident progression could not be located in the submittal.

2.1.3.2 Dominant Contributors: Consistency with IPE Insights

Table 1-5, page 1-10, of the Level 2 PRA compares results at Oyster Creek and NUREG-1150 Peach Bottom for the dominant contributors to containment failure. These results and those of the Fitzpatrick IPE are given in Table 1, below, where it can be seen that the early containment failures at Oyster Creek are less than one-third of those at Fitzpatrick or Peach Bottom. Correspondingly, Oyster Creek has a significantly higher fraction of no vessel breach than does Fitzpatrick or Peach Bottom. However, the Oyster Creek containment is assumed to always fail, in part because no AC recovery is assumed.

Table 1. Containment Failure as a Percentage of CDF: Comparison to Fitzpatrick IPE and Peach Bottom NUREG-1150 Results

Containment Failure	Fitzpatrick IPE	Peach Bottom/ NUREG-1150	Oyster Creek
CDF (per year)	1.9×10^{-6}	4.5×10^{-6}	3.2×10^{-6}
Early Failure	60.4	55.7	15.9
Bypass	na	na	7.3
Late Failure	26.0	16.0	26.4
Intact	2.5	18.0	0
No Vessel Breach	11.1	10.0	50.4

2.1.3.3 Characterization of Containment Performance

The containment performance observed during the Oyster Creek IPE was characterized using containment event trees. The top events of these event trees are discussed in Subsection 7.2 and listed in Table 7-1 of the Level 2 PRA. The CET chronologically models core degradation, vessel failure, containment behavior, and reactor building behavior. The first top event analyzed was one that would occur in the entry state from the front-end (i.e., a KPDS). The next top five events consisted of phenomena that could occur from the time core damage began until vessel

failure seemed imminent (these events were: vessel breach, safety valve failure, containment intact prior to vessel breach, containment leakage, and suppression pool bypass).

The next four events analyzed were phenomena that could occur during and shortly after vessel breach. These four events could affect transient loading conditions significantly and could lead to early containment failure (blowdown, direct containment heating, ex-vessel fuel coolant interaction, drywell liner melt-through). The next three events were analyzed with attempts to formulate a long-term containment response, to prevent containment failure by establishing adequate debris bed cooling, and to either remove or vent containment heat. These events were occurrence of containment venting, incidence of containment remaining intact late, and occurrence of containment leak areas. The last two top events set out in the Oyster Creek CET were simulated to study the phenomena that could affect the reactor building integrity and the ability of the phenomena to reduce an offsite source term if the containment failed.

As shown in the summary in Table 2, below, many of the top events addressed containment behavior. The containment loading was calculated using the MAAP computer code.

2.1.3.4 Impact on Equipment Behavior

A discussion of the impact of severe accidents on equipment behavior could not be located in the submittal.

2.1.4 Reducing Probability of Core Damage or Fission Product Release

2.1.4.1 Definition of Vulnerability

As noted in Subsection 3.2, page 3-2, of IPE Submittal Report, "A vulnerability is defined as any core damage sequence that exceeds $1x10^{-4}$ per reactor year, or any containment bypass sequence or large early containment failure sequence that exceeds $1x10^{-6}$ per reactor year." GPU found no vulnerabilities for the Oyster Creek nuclear power plant.

2.1.4.2 Plant Improvements

With respect to plant improvements, Subsection 8.2, page 8-5, of the submittal report explained that "Because of the relatively low frequencies associated with the various containment failure modes, no specific hardware modifications or changes to existing procedures beyond those identified in the level 1 analysis are planned at this time. The level 2 PRA will be used as a major input to the development of accident management guidelines."

Although no back-end improvements are planned as such, the submittal does address front-end improvements, which might affect mitigation of the consequences of back-end events. Note Section 2.1.5, below.

A plant improvement, planned as part of the 14R modifications, is the hard-piped containment vent system. This system is assumed available in the IPE and becomes important for back-end

Table 2. Oyster Creek CET Top Event Descriptions

	Top Event Designator	Top Event Description
CET Entry State		
	VB	Vessel Breach Prevented
	ES	EMRV(s) or Safety Valve(s) Sticks Open Prior to Vessel Breach in High Pressure Melt Scenarios
	I1	Containment Intact Prior to Vessel Breach
	L1	Small Leak Area if Containment Fails to Top Event I1
	S1	Suppression Pool Not Bypassed Prior to Vessel Breach
Events during or Shortly after Vessel Breach		
	LE	Debris Not Entrained
	I2	Containment Intact after Vessel Breach
	L2	Small Leak Area if Containment Fails in Top Event I2
	LM	No Significant Release of Fission Products into the Reactor Building due to Drywell Liner Melt-Through
Long-Term Containment Events		
	S3	Suppression Pool Not Bypassed Late
	DV	Emergency Crew Vents Containment in Core Damage Scenarios
	I3	Containment Intact Late
	L3	Small Leak Area if Containment Fails in Top Event I3
Events Pertaining to Reactor Building Effectiveness		
	HB	No Hydrogen Burn in Reactor Building
	BE	Reactor Building Effective

assessment, during "dirty venting," that is, venting after core damage. This is addressed under CET Top Event I1, discussed on page 7-7 and page 10-12 of the Level 2 PRA. For the Key Plant Damage State (KPDS) OIAU (note the containment matrix in the TER Appendix), the venting is judged to be effective 98.3% of the time, the same value used for the Level 1 "clean" venting.

2.1.5 Responses to CPI Program Recommendations

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

- Alternate water supply for drywell spray/vessel injection
- Reactor pressure vessel depressurization system reliability enhancement
- Emergency procedures and training

In Section 4 of the submittal report, these recommendations are addressed

- An alternate water supply for vessel injection is in place and credit is taken for it in the IPE. Specifically, it is GPU's position that injection is timely enough and of sufficient quantity to prevent vessel breach. This is addressed in top event VB in the CET. An alternate water supply for drywell spray was considered to not be cost-effective, although there are situations where, if water was provided to the dry, non-coolable core debris, it would mitigate the consequences of the accident

- Reactor pressure vessel depressurization system reliability enhancement is accomplished by providing an alternate AC source connection scheduled for the 14R refueling outage. This will reduce the likelihood of an extended station blackout, thereby improving depressurization reliability

- GPU has implemented the BWR Revision 4 EPGs and they are reflected in the IPE submittal

2.2 IPE Strengths and Weaknesses

2.2.1 IPE Strengths

1. The IPE submittal appears to have addressed most of the important and relevant phenomena in sufficient detail. Most of the leading severe accident hazards, such as liner melt-through, are systematically addressed

2. The results of the IPE at Oyster Creek are compared in sufficient detail with the NUREG-1150 results at Peach Bottom, and the differences are well documented

3. The back-end analysis is robust, (i.e., it was performed in a way that protects the results from the impact of changes that may occur later as the result of a front-end analysis). A front-end analysis should not significantly affect the conditional calculations of the back-end results

2.2.2 IPE Weaknesses

1. A discussion of the impact of severe accidents on equipment behavior could not be located in the submittal.

2. Although the probabilities of sequence occurrence seem reasonable throughout the submittal, the sources of such estimates are often not cited. For example, CET split fractions probabilities are not well defined.

3. The Oyster Creek IPE seems to focus on the events and containment characteristics that have early, detrimental effects on health, and on large, early releases to the environment. Large, early releases are important, but constitute only a small fraction of the probable accident events that should be considered and the subsequent containment responses that make up a back-end assessment. Attention to long-term effects and consequences is wanting.

3. OVERALL EVALUATION

As discussed in Section 2, this IPE submittal contains a large amount of back-end information, which contributes to the resolution of severe accident vulnerability issues at Oyster Creek. A large segment of the back-end portions of the IPE submittal is well written and directed to addressing Generic Letter 88-20 issues. The issue of liner melt-through is addressed well. The concrete curb is an attractive design feature to protect the liner.

However, there appear to be some weaknesses in the submittal, as set out in Section 2 of this report. In summary, our concerns about the submittal follow:

- The methodology appears to be geared to providing consequence assessment and comparison to safety goals, as contrasted to understanding the vulnerabilities of the containment and the impacts that equipment, EOPs, and phenomenology have on containment performance.

- There appears to be no discussion of back-end uncertainties. This coupled with a lack of sensitivity analysis weakens the overall conclusions and could give the wrong impression of the state of knowledge of containment vulnerabilities.

- A large fraction of the core damage frequency results in no vessel breach, that is the accident is arrested in-vessel. There appears to be no discussion of the uncertainties in this conclusion or of the sensitivity of this result to major assumptions.

- The quantification of the CETs could not be traced.

4. REFERENCES

1 General Public Utilities Corporation, "Oyster Creek Individual Plant Examination Report," August 1992

2 GPU Nuclear Corporation and PLG, Inc., "Oyster Creek Probabilistic Risk Assessment (Level 1)," Vols. 1 through 6, November 1992.

3 American Nuclear Society and Institute of Electrical and Electronics Engineers "PRA Procedures Guide - A Guide to the Performance of Probabilistic Rick Assessments for Nuclear Power Plants," prepared for the U.S. Nuclear Regulatory Commission. NUREG/CR-2300, Vols. 1 and 2, January 1983.

APPENDIX

IPE EVALUATION AND DATA SUMMARY SHEET

BWR Back-end Facts

Plant Name

Oyster Creek

Containment Type

Mark I

Unique Containment Features

Presence of a drywell floor concrete curb; a thinning of the liner in the sandbed region; 25 percent increased structural capability of torus as a result of a backfit performed, and an increased containment cooling capability as a result of improving the NPSH limits with a rise in drywell pressure

Unique Vessel Features

None found

Number of Plant Damage States

19

Ultimate Containment Failure Pressure

134 psig at 300°F

Additional Radionuclide Transport And Retention Structures

Suppression Pool scrubbing is assumed. However, the containment failure modes appear to preclude the possibility of post-containment-failure scrubbing. Reactor Building mitigation does not appear to be credited.

Conditional Probability That The Containment Is Not Isolated

A value could not be found in the IPE submittal, but it is assumed to be very low because of the characteristics of the Mark I inerted containments.

APPENDIX (continued)

IPE EVALUATION AND DATA SUMMARY SHEET

Important Insights, Including Unique Safety Features

Presence of a drywell floor concrete curb, a thinning of the liner in the sandbed region, the earliest release occurs 2 hours after the accident, and the worst release is caused by a bypass scenario, 25 percent increased structural capability of torus as a result of a backfit performed, and an increased containment cooling capability as a result of improving the NPSH limits with a rise in drywell pressure.

Implemented Plant Improvements

None implemented, but the containment cooling capability increased as a result of improving the NPSH limits with a rise in drywell pressure.

C-Matrix

Simplified Oyster Creek C-Matrix

Key Plant Damage State	Frequency per year	Early Failure	Bypass	Late Failure	No Vessel Breach
PIEW	1.13E-6	0	0	0	1.0
NIEW	1.06E-6	0.31	0	0.69	0
OIAU	5.74E-7	0.01	0	0.18	0.81
MKCU	1.70E-7	1.0	0	0	0
OJAU	1.64E-7	0	1.0	0	0
MJAU	5.26E-8	0	1.0	0	0
NJHW	1.54E-8	0	1.0	0	0

ENCLOSURE 4

OYSTER CREEK INDIVIDUAL PLANT EXAMINATION

TECHNICAL EVALUATION REPORT

(HUMAN RELIABILITY ANALYSIS)

CA/TR-92-019-08

TECHNICAL EVALUATION REPORT

OYSTER CREEK NUCLEAR GENERATING STATION INDIVIDUAL PLANT EXAMINATION

ASSESSMENT OF HUMAN RELIABILITY ANALYSIS DOCUMENT ONLY

M. G. Beck
P. M. Haas

Prepared for

U.S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Division of Safety Issue Resolution

Draft, November 1992
Final, February, 1994

CONCORD ASSOCIATES, INC.
Systems Performance Engineers

725 Pellissippi Parkway
Knoxville, TN 37932

Contract No. NRC-04-91-069
Task Order No. 8

9406210226-XA

TABLE OF CONTENTS

1. INTRODUCTION

This technical evaluation report (TER) is a summary of the documentation-only review of the Human Reliability Analysis portion of the Oyster Creek Nuclear Generating Station Individual Plant Examination (IPE) submittal to the U. S. Nuclear Regulatory Commission (NRC). The body of the report consists of four sections, per the instructions of the Task Order: (1) this Introduction, which provides a brief summary of the approach to this Step 1 review and of the Oyster Creek IPE HRA approach; (2) Contractor Review Findings, a detailed documentation of findings for each work requirement specified in the Task Order; (3) overall Evaluation and Conclusions, which summarizes the important findings and results from the review, and (4) the NRC summary data sheets.

1.1 Step 1 HRA Review Approach

The documentation-only review approach for Oyster Creek IPE HRA involves the following six steps, illustrated in Figure 1. These steps, especially steps 2 through 4, are interactive and iterative, but follow this general progression:

(1) Scoping Review - an overview of the entire IPE submittal. Read summary sections, plant descriptions, the major HRA-pertinent section(s), and result sections. Skim/scan the entire submittal, including appendices and detailed front-end and back-end analyses. Identify the basic approach used for the HRA and the organization of the HRA documentation, including any obvious major omissions. Identify notable features of the plant, the overall IPE approach, or the HRA approach that deserve special attention. Identify and obtain references that may need to be reviewed or checked; obvious points of interface with front-end and back-end analysis. Review descriptions of IPE/HRA team qualifications.

(2) Detailed Review of HRA Sections - a detailed review and assessment of the primary HRA section(s) of the submittal. This involves first a thorough (re)reading of descriptions of methodology, noting assumptions, data sources, and other important aspects of the analysis, and annotating any questions, potential problem areas, missing information, or issues for further investigation. Second, it involves a comparison of information and documentation found in the submittal about the overall HRA methodology/approach to the information/ documentation "requirements" identified in accepted HRA approaches used in other PSAs. For example, since the Oyster Creek IPE used a Success Likelihood Index based methodology (SLIM), this comparison involved reviewing the information contained in the submittal regarding the major steps in the SLIM approach as described in NUREG/CR-3518 and 4016 (Refs. 1 and 2). Finally, the detailed review involves an attempt to "track" the complete assessment of a few key operator actions through the HRA process described in the submittal. By tracking, we mean simply identifying that the submittal contains sufficient information to clearly delineate

methodology, major assumptions, important parameters such as performance shaping factors, data sources, and references for both the qualitative and quantitative assessment of human actions. There is no attempt to reproduce quantitative analysis.

(3) **Response to Work Requirements** - assessment of specific issues identified in the Task Order work requirements. This is an item-by-item assessment responding to each work requirement. The focus is identification of strengths and weaknesses of the HRA portions of the submittal and insights regarding important results or potential areas of improvement. Any questions that require additional input from the licensee are identified. This step includes completion of the NRC Data Sheets, which is Work Requirement 2 in the Task Order.

(4) **Interface with Front-End and Back-End Reviewers** - two-way exchange of information and discussion of issues. The focus is on HRA aspects of front-end or back-end analysis, but includes a general exchange of information and findings. The interaction takes place informally throughout the review, but primarily after completion of the overview in Step 1 above, and again after completion of Steps 2 and 3 as writing of the TER begins. More formal interaction occurs during the closing meeting of NRC staff and IPE review contractors in Step 6.

(5) **Prepare the TER** - develop and write this technical evaluation report. This involves: preparation of a draft report documenting all work accomplished, findings, and conclusions; internal technical review verifying findings and conclusions and compliance with Task Order Requirements: editorial review and printing.

(6) **NRC Staff and Contractor Meeting** - held after submittal of the TERs from contractors to review findings and conclusions and finalize questions for the licensee (if any).

1.2 Oyster Creek IPE HRA Approach

The Oyster Creek IPE consists of Level 1 and 2 Probabilistic Risk Assessment (PRA) without evaluation of external events. The PRA's methodology employs the "large event tree - small fault tree" approach. The PRA is innovative in that the logic of the plant model is entered as logic statements or "modules" that can be directly linked, eliminating the need for support states. Specific operator actions are identified by the analysts based on review of operating procedures, system analysis and development of plant model, and incorporated into the system analysis for system split fractions and plant model.

The HRA approach described in the submittal, essentially directed at quantifying human error probability (HEP) estimates, was performed using a Success Likelihood Index (SLIM) based methodology. This method relies heavily on the use of operator input in evaluating human actions. The submittal provides details of performance shaping factors used, the structured operator survey format and process for determination of PSF values, and the process for evaluation of HEP.

2. CONTRACTOR REVIEW FINDINGS

The subsections below address explicitly, item by item, each of the work requirements specified in the Task Order. For each item, there is an attempt to identify notable points about the submittal, both strengths and weaknesses, and insights as to how the submittal might be improved with regard to the specific work requirement and the overall intent of Generic Letter 88-20. Information obtained from the license in response to NRC questions has been factored into this final report.

2.1 Work Requirement 1.1 Perform a general review of the human reliability analysis.

2.1.1 WR 1.1.1 The IPE submittal is essentially complete with respect to the type of information and level of detail requested in the IPE Submittal Guidance Document NUREG-13435. List any obvious omissions.

Table 2-1 lists the major items identified in NUREG-1335 pertinent to HRA that were checked. The following are the findings for this work item:

(1) General Methodology. The plant model is developed by combining the response of plant systems with operator functions as provided in plant procedures (EOPs and abnormal response procedures) to represent the integrated plant response. These operator functions are included as top level events. Models for most of these functions require operation of systems or components. System models required to support the top event also include important operator actions (including many of those described above) which affect system operability. They are documented in the system notebooks. Specific operator actions are identified by the analysts were evaluated, and the results were incorporated into the respective system fault trees or sequence event trees.

The overall description of the HRA effort in Section 6 of the PRA (Level 1) report provides a clear understanding of the general methodology and approach to addressing human actions within the IPE. The model of human interactions used for the evaluation of HEPs splits the response into three phases: identification, diagnosis and response. The actions of operators were classified as skill, rule and knowledge based actions and evaluated accordingly.

The SLIM-based method was used to evaluate the operator actions in the IPE. Input pertinent to performance shaping factors (PSFs) was obtained from operators. The submittal provided reasonably detailed descriptions of the structured questionnaire used to obtain operator input. PSFs used were described and justified. The calculation of HEPs based on input was outlined in the submittal. Only post-event human errors were evaluated for the IPE.

(2) Information Assembly. A listing of reference PRAs of similar plants, including Peach Bottom (Ref. 3), that were reviewed for the Oyster Creek PRA was provided in Section 1.6 of

Table 2-1 NUREG-1335 HRA Items Checked - WR 1.1.1

NUREG-1335 REFERENCE	INFORMATION PERTINENT TO HRA
2.1.1 General Methodology	Concise description of HRA effort and how it is integrated with the IPE tasks/analysis.
2.1.2 Information Assembly	2.1.2.2 List of reference PRAs, insights regarding HRA, human performance. 2.1.2.3 Concise description of plant documentation used for HRA information; concise discussion of the process used to confirm that the HRA represents conditions in the as-built, as-operated plant. 2.1.2.4 Description of the walkthrough activity, including HRA specialist participation.
2.1.3 Accident Sequence Delineation	Description of process for assuring human actions considered in initiating events and accident sequence delineation; HRA specialist involvement.
2.1.4 System Analysis	Description of process for assuring that the impacts of human actions are included in systems analysis; process for integrating HRA.
2.1.5 Quantification Process	2.1.5.1 HRA in common cause analysis. 2.1.5.3 Types of human failures considered in the IPE; a categorization and concise description exist. 2.1.5.4 List of human reliability data and time available for recovery actions; data sources clearly identified; if screened, a list of errors considered, criteria for screening, and results of screening. 2.1.5.5 List of HRA data obtained from plant experience and method/process for obtaining data, list of generic data. 2.1.5.6 Concise description of method by which HEPs are quantified, including break down such as task analysis, and techniques for combining probabilities, assessing dependencies, etc.

Table 2-1 NUREG-1335 HRA Items Checked - WR 1.1.1

NUREG-1335 REFERENCE	INFORMATION PERTINENT TO HRA
2.1.6 Front-End Results and Screening Process	Human contributions to important sequences are clearly identified. A concise definition of vulnerabilities is provided, along with a discussion of criteria used to identify vulnerabilities. A listing of vulnerabilities is provided, with clear definition of those related to human performance. Underlying causes of human related vulnerabilities are identified. 2.1.6 Sequences that, were it not for low human error rates in recovery actions, would have been above the applicable core damage frequency screening criteria are identified and discussed. 2.1.7 Any human performance issues pertinent to USIs or GSIs are identified and discussed as appropriate.
2.2 Back-End Submittal	Impacts of operator action on containment response are identified. Actions assumed to be accomplished by operators can reasonably expected to be accomplished under the severe accident conditions expected; equipment accessibility, survivability, information availability, etc. have been considered. Critical human actions have been identified and included in the event trees and quantitative HRA assessments.
2.3 Specific Safety Features and Potential Improvements	Any human performance related aspects of unique and/or important safety features are discussed, including any that resulted in significantly lowering typically high frequency core melt sequences. Human related potential improvements - procedures, training, etc.- in response to vulnerabilities are clearly identified and discussed.
2.4 IPE Utility Team and Internal Review	The submittal describes the utility staff participation and involvement in the HRA. An independent in-house review of the HRA was conducted.

the submittal. The methodology used for the HRA analysis is based on the methods used in the TMI 1 PRA (Ref. 4) and is a refinement of that analysis. Plant documentation to acquire HRA information was identified. It included: plant operating procedures, emergency operating procedures (EOPs), and surveillance and maintenance procedures. A detailed description was provided for each action to be analyzed by plant operators, including plant conditions and other constraints. The plant operators evaluate the PSFs by completing the "PRA Human Action Survey Form." The survey process is a structured method to evaluate the performance shaping factors. The survey form used was provided in the submittal, as well as detailed information on the PSF breakdown and linkage to the survey form rating system.

"Human Action Walkdowns" were performed by team members responsible for evaluating operator actions with experienced operator personnel. They were conducted to familiarize themselves with the operator actions modeled as well as to verify operator action survey forms. The SLIM-based evaluation process used plant operator input from the survey form to evaluate PSFs which were converted to the success likelihood index values. The survey process and information collection appear to be well structured.

(3) Accident Sequence Delineation. Technical information on the plant design and supporting calculations are combined with abnormal response and EOP procedures to form the basis of the Event Sequence Diagrams. Specific operator actions required to prevent degradation of plant conditions are identified by the analysts during development and evaluation of Event Sequence Diagrams. A "Detailed Human Action Description" is provided by HRA analysts and verified/ modified by HRA walkdown. Details of each operator action were provided in Appendix E of the submittal. Incorporation of operator actions into the PRA is discussed in Section 2.1.3 of this TER.

(4) System Analysis. The System analysis is described in section 5 of the Oyster Creek PRA (OCPRA) Level 1 report. System descriptions are appropriately detailed and comprehensive. System notebooks were developed for each system analyzed. A summary of the contents was included in the submittal, and notebooks are provided in Appendix F to the submittal. Included in each notebook are the important operator actions for the system operation. In addition to routine information on major components and instrumentation, the notebooks include information on system dependencies and interfaces, testing and maintenance, technical specifications, system operation, modeling assumptions, and success criteria. Operator actions are incorporated into the PRA in appropriate system fault trees. Documentation of system fault trees are provided in the system notebooks. Documentation appears to be sufficient to support a detailed evaluation, if one were necessary. The incorporation of EOP steps in system models was addressed above.

(5) Quantification Process. Human Interactions (HIs) were grouped into three major classifications for quantification, depending on the time at which the action occurs in the accident scenario. "Group A" HIs occur prior to the initiator event, and are the result of human errors during maintenance, testing, or calibration activities. "Group B" HIs are those that result in initiating events. These are captured in the initiating event frequencies obtained from plant operating experience. Therefore, Group B HIs are not included in the IPE HRA analysis.

"Group C" are broken down into two sub-classes: (CP) operator actions performed in response to procedures, particularly Emergency Operating Procedure (EOP), and (CR) recovery actions in response to unavailability of a safety function, which may or may not be proceduralized. CP events appear as headings in the event trees or as basic events in system or functional fault trees. Type CR events are separately added to the model following initial quantification and are addressed at the accident sequence cutset level.

Group A HIs error frequencies were considered to be captured in the basic equipment failure rates for misalignment or failure to restore systems. The submittal states that this failure mode is not a large contributor to system failure. The submittal stated that certain Group A errors were included in system models, but no details were provided. This is the subject of a request for additional information in section 2.2.1

The quantification process used for Group C HIs in the Oyster Creek IPE is described in considerable detail. For each operator action, a fairly detailed description of plant conditions and other constraints was provided to the operator. The SLIM-based evaluation process uses plant operator input for evaluating Performance Shaping Factors (PSFs). Selection of PSFs is justified in the submittal. Conversion of these PSFs to the success likelihood index value is accomplished by use of weighting factors based on the class of action (rule, knowledge or skill based) for each "phase" of identification, diagnosis and response. The likelihood index value is converted to error probability for each behavior model phase using reference actions to "calibrate" the Success Likelihood Index for each action phase.

The survey sheets completed by the operators are structured to a level of detail and with questions intended to reduce the variability of the subjective responses. All inputs were analyzed to provide a data spread for statistical analysis for estimating the uncertainty of the values obtained.

There is a concise summary of the common cause analysis provided in Section 5.3.3.3 of the PRA Level 1 report in the submittal. The submittal states pre-initiator human errors are not considered because they are captured in the component failure data analyzed. Pre-initiator (Group A) human errors are discussed in section 2.2.1 of this report, and a request for additional information on their treatment is provided in that section. Common cause events are a subset of these pre-initiator errors.

(6) <u>Front-End Results and Screening Process</u>: The IPE submittal defines vulnerability as any core damage sequence that exceeds 1.0 E-4 per reactor year, or any containment bypass sequence or large early containment failure that exceeds 1.0 E-6 per reactor year. No vulnerabilities were identified. A structured review was performed to identify potential low cost improvements. The results of level 1 and 2 PRAs, as well as contributors to system unavailability and operator action error rates were reviewed.

No listing was provided of sequences that were it not for low human error rates in recovery actions, would have been above the applicable core damage frequency criteria; nor was any clear statement that no such sequences exist.

As required by NUREG-1335, GSE and other safety issues, such as internal flooding, Loss of Feedwater Control, and alternate water supply for drywell spray/vessel injection, were analyzed by Oyster Creek, and the results are reported in the IPE submittal. No vulnerabilities were identified. Several analysis of these safety issues involved human actions which were considered important enough to have potential improvements identified:

1. The alternate drywell spray source considered cross-tie of fire protection diesel water with manual operated valves. Because of high radiation from core damage, the required shielding to allow access would make the modification cost prohibitive for the minimal affect on cooling core debris.

2. Procedure changes to improve operator response to internal flooding were recommended.

3. A new reactor overfill prevention system is to be installed for loss of feedwater system control because of concerns about operator responses to isolate MSIVs within the allowed time.

(7) Back-End Submittal. The Containment Event Trees (CETs) consider the influence of the physical and chemical processes on changing the containment pressure and (in the case of containment failure or bypass) on affecting the release of fission products from the containment. The end state of the front-end analysis is binned according to plant damage states and used as input to CETs. The plant damage state information includes the following categories: physical condition in the reactor coolant system and containment at time of vessel breach; integrity of primary containment and status of associated active systems; integrity of secondary containment and status of associated active systems .

Containment models include "dirty venting." These are the only human actions directly modeled in the analysis. The containment analysis used the results of the level 1 system status as input for the back-end plant damage state. Therefore, many human actions were indirectly incorporated into the back-end analysis. The results of the front-end containment venting HEPs were used in the back-end analysis.

(8) Specific Safety Features and Potential Improvements. A number of specific safety features of the Oyster Creek plant were discussed in Section 8 of the IPE submittal. Specific procedure changes and modifications were identified as cost effective and are being implemented. These include:

- Containment vent modifications and associated procedure revisions.

- Station blackout technical basis document and integrated loss of offsite power procedure to provide: recovery of offsite or onsite power; for alignment and cross-tieing buses to critical equipment; and for startup and alignment of alternate AC capability.

- Loss of all DC power procedure to be coordinated with the integrated loss of offsite power procedure

- Reactor overfill prevention system is to be installed for reactor overfill transients because of concern for operator response to isolate MSIVs within the required time.

Improvements or enhancements under consideration include:

- Development of a specific procedure and training on reactor overfill transients.

- Operator training should emphasize important actions (listed in Section 8.1.5) which were identified by the PRA as important in reducing core damage risk.

(9) IPE Utility Team and Internal Review. While the IPE development was supported by a consultant (PLG, Inc.), the submittal states that one of the objectives of the study was to build on in-house PRA expertise and develop tools for ongoing risk management activities after the completion of the PRA. GPU provided system analysts, engineers and plant operations personnel as a part of the PRA team. HRA specialists from the contractor organization as well as GPU were included on the IPE team.

The internal review process described in the submittal appears to be extensive. Multiple engineers and operations personnel with expertise in Oyster Creek design and operation were involved in the reviews. A review of the comments suggests that the team provided a thorough review. An outside consultant with expertise in PRA methodology reviewed the IPE for technical methods. With regard to the personnel on the team, however, no individual was identified as the HRA reviewer or as having previous HRA experience. The submittal would be strengthened if a thorough review of the HRA portion of the IPE were included in the review process.

2.1.2 WR 1.1.2 The employed HRA methodology is clearly described and justified for selection.

Section 6 of the Level 1 report included in the submittal clearly describes the steps performed in the HRA portion of the IPE. The SLIM methodology is a well established and documented HRA approach. The SLIM-based evaluation process used at Oyster Creek uses plant operator input as the basis for PSFs which are converted to the success likelihood index value using weighting factors. The success likelihood index value is converted to error probability using calibration values from "known" HEPs. There are requests for additional information on the implementation of the SLIM-based methodology which are detailed in the sections which follow.

2.1.3 WR 1.1.3 The methodology (including the human action taxonomy) employed is capable of identifying important human actions, and contains a discussion of the most important human actions and errors.

The human action taxonomy used in the HRA was clearly identified in the submittal. The model of human interactions used for the evaluation divides the response into three phases: identification, diagnosis and response. The actions of operators were classified as skill, rule or knowledge based actions and were evaluated accordingly. Details on the human actions and the quantification were provided in Section 6 and Appendix E of the submittal.

The submittal stated that procedures were reviewed to identify operator actions to be included in the plant model. One important operator action that was not included in the plant model, and which is in the EOPs, is containment flooding. This was identified by review of independent review comments for March 27 IIHRG meeting in Appendix D to the submittal. The response to the comment was that the operator action was not required "to establish or maintain stable shutdown conditions." Because the steps are in the EOPs the containment flooding would likely be carried out by the operators. IPEs for BWRs with Suppression Pool type containments have identified containment flooding as a source of containment failure when core damage and vessel melt through occur after the torus is flooded (loss of pressure suppression capability). The Licensee indicated in response to an NRC question on this point, that this potential "down-side" of containment flooding had been evaluated and was not included because of its low likelihood of occurrence.

2.1.4 WR 1.1.4 The IPE submittal employed a viable process to confirm that the IPE represents the as-built, as operated plant.

Technical information on the plant design and supporting calculations are combined with abnormal response and EOP procedures to form the basis of the Event Sequence Diagrams (ESD). The ESDs were presented to various GPUN organizations including plant operations, safety analysis, and training departments for review. The resulting final ESDs were used as the primary input in the development of the plant model.

In addition, walkdowns were held to verify information was correct. A structured program was provided to prepare detailed descriptions of all Human Actions to be analyzed. Plant walkdowns by risk assessment personnel, a human factors specialist, and plant operators over a 3 day period confirmed the accuracy of the detailed descriptions.

The final check on as-built and as-operated was provided by the Independent Review Group. Members were chosen for their expertise in plant design and operation. The Independent Review Group reviewed the entire submittal including system notebooks and operator action sections.

This process appears to be a reasonable and systematic approach to assuring that the IPE represents the as-built, as-operated plant.

2.1.5 WR 1.1.5 The HRA had been peer-reviewed to help assure the analytic techniques were correctly applied

The internal review process described in the submittal and discussed in Section 2.1.1(9) above appears to be comprehensive, with exception of the HRA analysis. No individual was identified as the HRA reviewer or as having previous HRA experience. No other peer-review was identified in the submittal for the HRA analysis. Peer-review by qualified HRA personnel helps provide additional confidence that the HRA methods were appropriately applied and results are correct. The submittal would be strengthened by additional information concerning any HRA review and qualifications of the HRA reviewer(s).

2.2 Work Requirement 1.2 Review the most likely sequences that could occur at the plant.

2.2.1 WR 1.2.1 The accident sequences appropriately considered human actions consistent with other NUREG-1150 and other NRC accepted PSAs (see table NUREG-1335 Appendix B).

The human actions of Grand Gulf (Ref. 5) were compared to the OCPRA human actions. The review shows that equivalent actions were considered in the OCPRA sequences. Additional human actions were included in the OCPRA because of the additional operator instructions provided by the new (Rev. 4) EOPs. As was noted in Section 2.1.3 earlier, a potential discrepancy in the incorporation of EOP steps was identified and additional information on the process for identifying and including proceduralized operator actions into the PRA.

Pre-initiator (Group A) human errors such as calibration error or misalignment of systems or instrumentation are not modeled in the PRA. The submittal states that "misalignment of systems are not modeled in the OCPRA since these causes of unavailability are captured in the component failure data." Pre-initiator human errors are normally considered in PRAs; (e.g., see Grand Gulf (Ref. 5) and Surry (Ref. 6) PRAs). While it is true that, in general, pre-initiators typically have less impact on estimated CDF than do post-initiators, significant contributions from pre-initiators have been identified in some PRAs. A systematic analysis of pre-initiator human errors and contributing factors would provide much greater confidence that no important errors have been missed. And, the information gained on "generic" factors influencing human performance, e.g. procedures on administrative controls, may indicate relatively low-cost means for significant improvement.

2.2.2 WR 1.2.2 The accident sequences screened out because of low human error (see NUREG-1335, Section 2.1.6.6) appears appropriate, based on HRA techniques employed.

The submittal addressed the importance of human actions by examining the contribution to core damage for three groupings: (1) all operator actions, (2) operator actions grouped into nine general categories, and (3) top 10 individual operator actions . All modeled operator actions

were found to contribute 21% of total core damage. The most important groups of operator action were those associated with establishing RPV injection and removal of containment heat. The individual operator actions were from the most important groups and ATWS sequences, and their contribution to total core damage ranged from 1% to 2.76%. Detailed information about each operator action is available in Appendix E of the Level 1 Report.

The analysis of operator action contribution to core damage provides insight into which operator actions are the most important, but the specific information required by NUREG-1335, Section 2.1.6.6 was not found.

2.3 Work Requirement 1.3 Review the quantitative nature of the IPE submittal.

2.3.1 WR 1.3.1 The employed human error probability (HEP) screening values appear capable of screening in significant human errors.

Screening or "conservative" values were used for only a few selected operator actions including circulating water system flooding and loss of off-site power recovery in this IPE. In these cases values are provided without referencing any source. The values appear to be appropriate, but the submittal would be strengthened if the source of the values is referenced or additional information on the technical basis for these estimates were provided.

While there were few actions for which numerical screening was performed, it should be noted that potentially significant qualitative screening is performed in the process of selecting those human actions to be evaluated. Operator actions modeled, including recovery actions appear to be appropriate based on review of similar PRAs. However, the submittal does not provide much information on the process by which the specific ones selected for Oyster Creek. In general, the basis was said to be "required" operator actions, EOPs, and abnormal procedures. The submittal would be strengthened by a discussion of the specific rationale, assumptions and criteria for selection of actions.

2.3.2 WR 1.3.2 The IPE developed human error probabilities (HEPs) for significant human actions, or provided rationale for using screening values.

With the exception of the screening values cited above, no numerical screening of HEPs typical in many PRAs was identified from the submittal review. Actions selected for analysis were analyzed directly, and HEPs were developed. The method used to quantify HEPs are discussed in Section 2.3.3.

2.3.3 WR 1.3.3 Sources of generic human reliability data used in the IPE were identified and rationale for their use provided. Generic human error probabilities (HEP) data were modified using plant-specific Performance Shaping Factors (PSFs) as appropriate, and rationale provided for selection of employed PSFs.

The SLIM-based evaluation process uses plant operator input to evaluate operator actions for the HRA. Thus the data is neither "generic" nor "plant-specific" in the usual sense of those words. There is some merit to the assertion that since the operators are from this particular plant, their judgments probably reflect some degree of plant-specific experience. On the other hand, the operator judgment primarily specifies the *relative* importance of PSFs. The *absolute* values are determined by the selected anchor points; and the submittal does not discuss the selection of those anchor values in much depth. The process for selection of PSFs and justification of the ones selected is reasonably well described in the submittal. The PSFs chosen for use along with the process attempt to account for dependencies and effects of multiple and successive operator actions. As noted earlier, the process for elicitation of expert judgment from the operators appears to be well structured and systematically applied.

Conversion of these PSFs to the success likelihood index value is accomplished by use of weighting factors based on the class of action (rule, knowledge or skill based) for each model for action phase of identification, diagnosis and response. An additional factor to account for the significance of the class of action for the diagnosis phase was used to increase the value for knowledge based activities. The submittal provides a general overview of the basis for the weighting factors used. Because of the importance of the weighting factors in calculation of Success Likelihood Index and the HEPs, it is felt that the submittal would be strengthened by inclusion of a more detailed description of the basis and structured process used in developing the weighting factors used.

The Success Likelihood Index value is converted to error probability using reference actions to "calibrate" the Success Likelihood Index value for each action identification, diagnosis and response phases. As indicated above, there is little information provided in the submittal on the selection of reference actions/values to calibrate the SLIM methodology.

2.3.4 WR 1.3.4 The recovery method is clearly described and credit for recovery actions appear justified.

Three types of recoveries are addressed in the submittal: system recoveries incorporated into system logic models, procedurally directed recoveries, and non-procedurally directed recoveries. The later two types of recoveries were added to the plant model following initial quantification and refinement. With exception of "dirty venting" discussed in Section 2.1.1.(7), no credit was included for post vessel breach recoveries in the back-end PRA.

Methods, data, and assumptions used to quantify recovery actions are clearly and concisely summarized. Information provided includes a description of each recovery action, amount of time available for the action, manual actions required, procedure availability, how the need for action is perceived, cognition class for activities, and success criteria for recovery. A concise description of PSFs and their use in the SLIM method was provided. While we did not perform detailed checks to validate numerical estimates, the HEP values overall appear to be reasonable and consistent with other PSAs. Values for selected operator actions were compared with previous PRAs in section 6.3.6 of the Level 1 report and found to be consistent.

2.4 Work Requirement 1.4 Review the IPE approach to reducing the probability of core damage or fission product release.

2.4.1 WR 1.4.1 The IPE analysis appears to support the licensee's definition of vulnerability, and that the definition provides a means by which the identification of potential vulnerabilities (as so defined) and plant modifications (safety enhancements) is made possible.

The IPE submittal defines vulnerability as any core damage sequence that exceeds 1.0 E-4 per reactor year, or any containment bypass sequence or large early containment failure that exceeds 1.0 E-6 per reactor year. No vulnerabilities were identified. A structured review was performed to identify potential low cost improvements. The results of level 1 and 2 PRAs were reviewed, well as major contributors to system unavailability and operator action error rates. Results of this review are discussed below. The overall process employed in the IPE for identifying vulnerabilities and cost effective safety enhancements appears to be comprehensive and able to systematically identify cost effective safety enhancements.

2.4.2 WR 1.4.2 The identification of plant improvements include human-related plant modifications (e.g., procedures and training,), and proposed modifications are reasonably expected to enhance human reliability and plant safety.

Cost effective plant improvements identified during the IPE process and being incorporated are discussed in Section 8 of the IPE report. The results of level 1 and 2 PRAs, contributors to system unavailability, and operator action error rates were reviewed to identify potential enhancements. No information was provided on any evaluation of the improvement in the IPE results, but it appears that the additional guidance in procedures should enhance the operator performance. Specific cost effective enhancements identified are being implemented including:

- Containment Vent modifications and associated procedure revisions.

- Station Blackout technical basis document and integrated loss of offsite power procedure to provide: recovery of offsite or onsite power; alignment and cross-tieing buses to critical equipment; and startup and alignment of alternate AC capability.

- Loss of all DC power procedure to be coordinated with the integrated loss of offsite power procedure

- A new Reactor overfill Prevention system is to be installed for reactor overfill transients because of concern for operator responses to isolate MSIVs within the required time.

Improvements or enhancements under consideration include:

- Development of specific procedure and training on reactor overfill transients.
- Operator Training should emphasize important actions listed in Section 8.1.5 were identified by the PRA as important in reducing core damage risk.

While no discussion of the evaluation for improvements was provided in the submittal, the procedure changes, training emphasis and modifications should help address problems identified by the PRA as contributors to operator error or system unavailability.

2.5 Work Requirement 2.0 Complete data sheets.

Completed data sheets are included in Section 4 of this TER.

3. OVERALL EVALUATION AND CONCLUSIONS

On the basis of our review, we concluded that with regard to the HRA, the submittal demonstrates that the licensee used a reasonable process to meet the intent of Generic Letter 88-20. Overall, the HRA methodology used for identification of important actions, analysis of factors influencing human performance, quantification of human error, assessing the impact of human error on system response (and therefore CDF and releases) appears reasonable and consistent with practice in other PSAs. A reasonable process was in place to identify potential human-related improvements.

Notable weaknesses of the submittal are the failure to treat pre-initiator errors explicitly and the description of basis for weighting factors and choice of reference Human Error events to calibrate the SLIM-based HRA evaluation. It is typical practice in PRAs to test pre-initiators such as maintenance, test and calibration errors explicitly. The submittal should include a clear and concise justification for the assertion that such errors are negligible and /or are incorporated in component failure data. The conversion of PSFs to Success Likelihood Index is accomplished by use of weighting factors for different types of human interaction. A more detailed description of the derivation of these weighting factors would have strengthened the submittal. The SLIM-based methodology must be "calibrated" using known or accepted HEPs. The submittal would have been strengthened if the discussion and justification was expanded for the HEPs used for calibration of the SLIM-based HRA.

4. IPE EVALUATION AND DATA SUMMARY SHEETS

IPE DATA SUMMARY SHEETS
(HUMAN RELIABILITY)

Plant Name: Oyster Creek Nuclear Generating Station

Information Assembly

- List of plants, PSAs or other analysis known to have employed similar methodology.

  TMI 1 (PLG)

- Ex-Control Room actions treated? List.

  Yes, Multiple actions as required for recovery. Table 6.4.1.a-c provides operator actions for sequences and plant location for actions

Human Failure Data (Generic and Plant Specific)

- Analytical method used, e.g., Expert Judgment, THERP, SLIM-MAUD, HCR. TRC.

  SLIM-based

- Were the following human errors considered:

  (1) Pre-initiator, e.g., maintenance error including testing, equipment calibration, and restoration.

  Assumed to be included in component failure data.

  (2) Post-initiator procedural?

  Yes

(3) Post-initiator recovery

- Control Room

Yes

- Ex-Control Room

Yes

- Types of human errors considered, e.g. omission, commission

  Errors of omission only

- Source of human reliability data,

  Generic Data?

  No

  Simulator Data?

  No

  Expert Judgment?

  Used SLIM method: Operator input based on detailed descriptions of operator actions using structured survey form to provide input for PSFs.

- Most significant operator actions,

  The most important groups of operator action were those associated with establishing RPV injection, removal of containment heat and ATWS sequences.

- Human Error contribution to core damage frequency (if known).

  21 %

- Vulnerabilities associated with human error.

  None identified

PLANT IMPROVEMENTS AND UNIQUE SAFETY FEATURES

- Improvement insights stemming from HRA.

Appendix B of submittal reviews contributors to operator errors and provides following recommendations (Section 8.1.5):

- Consider specific procedures and training for Reactor overfill transients.

- Consider training emphasis that consistently successful performance of following actions can reduce Core Damage risk:

  Operator injects fire water through Core Spray system during loss of AC power and unisolated LOCA outside containment events.

  Operator inhibits ADS and controls level near TAF during ATWS with FW available and condenser failed and EMERV/SV closure.

  Operator inhibits ADS during ATWS with FW failed and EMERV/SV closure.

  Operator manually re-energizes bus 1A1/1B and restarts at least one TBCCW pump following a loss of offsite power.

  Operator trips reactor after TT failure (high level)

  Operator secures or isolates condensate transfer header to reactor building within 1 or 2 hours after condensate transfer supply line break in the reactor building

  Operator trips plant and isolates feedwater following line break in the trunion room.

- Implemented human factor improvements

- Containment Vent modifications and associated procedure revisions.

- Station Blackout technical basis document and integrated loss of offsite power procedure to provide: recovery of offsite or onsite power; for alignment and cross-tieing buses to critical equipment; and for startup and alignment of alternate AC capability.

- Loss of all DC power procedure to be coordinated with the integrated loss of offsite power procedure

- A new Reactor overfill prevention system is to be installed for reactor overfill transients because of concern for operator responses to isolate MSIVs within the required time.

- Enhancements under consideration.

- Development of specific procedure and training on reactor overfill transients.

- Operator Training should emphasize important actions listed in Section 8.1.5 were identified by the PRA as important in reducing core damage risk (listed under improvement insights above).

- The alternate drywell spray source considered cross-tie of fire protection diesel water with manual operated valves. Because of high radiation from core damage, the required shielding to allow access would make the modification cost prohibitive for the minimal affect on cooling core debris.

- Procedure changes to improve operator response to internal flooding were recommended.

- Portable DC generator and equipment necessary to supply essential DC loads

REFERENCES

1. Embrey, D.E.,"SLIM-MAUD: An Approach to Assessing Human Error Probabilities Using Structured Expert Judgment," NUREG/CR-3518, USNRC, March 1984.

2. Rose, E.A. et al., "Application of SLIM-MAUD: A Test of an Interactive Computer-Based Methods for Organizing Expert Assessment of Human Performance and Reliability," NUREG/CR-4016, USNRC, September 1985.

3. USNRC, "Analysis of Core Damage Frequencies from Internal Events: Peach Bottom Unit 2," NUREG/CR-4550/Vol. 4, October 1986.

4. Pickard, Lowe and Garrick, Inc, "Three Mile Island Unit 1 Probabilistic Risk Assessment," prepared for GPU Nuclear Corp., PLG-0525, December 1986.

5. USNRC, "Analysis of Core Damage Frequencies from Internal Events: Grand Gulf-1," NUREG/CR-4550/Vol. 6, Rev. 1.

6. USNRC, "Analysis of Core Damage Frequencies from Internal Events: Surry, Unit-1," NUREG/CR-4550/Vol. 3, Rev 1.