



UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555
February 28, 1992

Docket 50-443

Mr. Ted C. Feigenbaum
President and Chief Executive Officer
New Hampshire Yankee Division
Public Service Company of New Hampshire
Post Office Box 300
Seabrook, New Hampshire 03874

Dear Mr. Feigenbaum:

SUBJECT: STAFF EVALUATION OF SEABROOK INDIVIDUAL PLANT
EXAMINATION (IPE) - INTERNAL EVENTS, GL 88-20
(TAC NO. M74466)

The purpose of this letter is to transmit our evaluation of your Independent Plant Examination (IPE) which you submitted March 1, 1991, in response to Generic Letter 88-20.

New Hampshire Yankee (NHY) responded to Generic Letter 88-20 and its supplements in letters dated November 1, 1989, March 1, 1991, and December 9, 1991.

The NRC staff completed its review of the internal events portion of the IPE submittal, its associated documentation which included the Seabrook Station Probabilistic Safety Assessment (SSPSA), response to Unresolved Safety Issue (USI) A-45 "Decay Heat Removal" resolution, an internal flood assessment, and response to staff generated questions which focused on the Seabrook IPE process and consideration of Containment Performance improvement (CPI) program recommendations. No additional unresolved safety issues (USIs) or generic safety issues (GSIs) were proposed for resolution as part of the Seabrook IPE.

The SSPSA which formed the basis of your IPE, is a full-scope Level 3 PSA completed in 1983. Subsequently, three substantial updates were performed and completed in 1986, 1989, and 1990. For each update, the applicable plant documents, including design documents and change requests, were reviewed and models changed as necessary. This update approach has been proceduralized as part of the risk management process at Seabrook. The latest PSA update is current through July 1990. Each update involved increasing levels of participation by utility staff, with the final update being conducted completely by utility personnel. We understand that you plan to keep the SSPSA as a living document.

Walkdowns discussed in the IPE submittal included systems walkdowns for system familiarity, spatial interactions walkdowns (which included consideration of

Dfoi

fire, flood and seismic effects), containment walkdowns, and containment bypass walkdowns. The IPE submittal states that during each walkdown, utility personnel from engineering and operations participated. The walkdowns constituted the process used to confirm that the IPE represented the as-built, as-operated plant.

The Seabrook IPE did not identify any severe accident vulnerabilities associated with either core damage or "unusually poor" containment performance. However, the IPE did identify potential improvements which you plan to evaluate following completion of the IPE for External Events (IPEEE) and accident management evaluations. These potential improvements focus on reducing both core damage frequency and offsite release of radioactivity. Although the NRC staff did not examine the merits of these improvements in detail, the improvements do not appear to be of sufficient safety significance to require regulatory action.

Our review found the IPE submittal weak with regard to the level of documentation provided on the process used to develop the conditional Human Error Probabilities (HEPs) as part of the human reliability analysis. NHY indicated in the IPE submittal, and in discussions with the IPE review team, that the next SSPSA update would include a revised human reliability analysis. The staff, therefore, recommends that you further document the basis for the HEPs and that they be checked for consistency with plant procedures as part of the accident management program.

Based on the Step 1 review of the Seabrook IPE submittal, and previous staff reviews of the SSPSA which included reviews by both Brookhaven National Laboratory (BNL) and Lawrence Livermore National Laboratories (LLNL), the staff concludes that the Seabrook IPE meets the intent of Generic Letter 88-20. This conclusion is based on the following findings: (1) the IPE is complete with regard to the information requested in Generic Letter 88-20; (2) the IPE front-end and back-end analysis is technically sound and capable of identifying plant-specific vulnerabilities to severe accidents; (3) although the IPE had weak documentation on the process employed for developing explicit human error probabilities, the human reliability analysis is capable of identifying severe accident vulnerabilities which could result from pre-initiating (test and maintenance) through post initiating (operator recovery) human interactions; (4) the licensee has performed plant walkdowns to verify that the IPE reflects the current plant design and operation; (5) the SSPSA which formed the basis of the IPE had an extensive independent peer review; (6) the licensee participated fully in the IPE process; (7) the licensee is actively using the IPE as a living document to enhance plant safety; (8) decay heat removal capability was appropriately evaluated (responding to USI A-45); (9) the licensee responded appropriately to the recommendations stemming from the CPI program.

The staff notes that the IPE review is not intended to validate the accuracy of your IPE, nor the bottom-line numbers so generated. Although certain aspects of the IPE were explored in more detail than others, the review primarily focused on your IPE process and its ability to identify vulnerabilities. The Seabrook numerical results and safety insights are summarized in the appendix.

Mr. Ted C. Feigenbaum

3 February 28, 1992

It is our understanding that you expect to submit an IPE for external events (IPEEE) by October 2, 1992. By this letter we are closing IAC Number M74466.

Sincerely,

Original signed by
Gordon E. Edison, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects I/II
Office of Nuclear Regulatory Research

Enclosure: Staff Evaluation

cc w/enclosure:
See next page

* SEE PREVIOUS CONCURRENCE

OFFICE:	LA/PDI-3	: PM:PDI-2	: PM:PDI-3	: (A)PDI-3	:
NAME :	MRusbrook	* RHernan	GEEdison/jb	VN...	:
DATE :	2/27/92	: 2/27/92	: 2/27/92	: 2/28/92	:

OFFICIAL RECORD COPY
Document Name: SEABROOK.SER

Mr. Ted C. Feigenbaum

Seabrook

cc:

Thomas Dignan, Esq.
John A. Ritscher, Esq.
Ropes and Gray
One International Place
Boston, Massachusetts 02110-2624

Dr. Mauray Tye, President
Sun Valley Association
209 Summer Street
Haverhill, Massachusetts 01831

Barbara J. Saint Andros, Esquire
Kopelman & Paige, P.C.
Counsel for Amesbury, Newburyport &
Salisbury
101 Arch Street
Boston, Massachusetts 02110

Robert Backus, Esq.
Backus, Meyer and Solomon
116 Lowell Street
Manchester, New Hampshire 03106

Diane Curran, Esq.
Harmon and Weiss
2001 S Street, N.W.
Suite 430
Washington, DC 20009

Mr. Peter Brann
Assistant Attorney General
State House, Station #6
Augusta, Maine 04333

Seacoast Anti-Pollution League
5 Market Street
Portsmouth, New Hampshire 03801

Resident Inspector
U.S. Nuclear Regulatory Commission
Seabrook Nuclear Power Station
Post Office Box 1149
Seabrook, New Hampshire 03874

Mr. T. L. Harpster
Public Service Company of New Hampshire
Post Office Box 300
Seabrook, New Hampshire 03874

Regional Administrator, Region 1
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, Pennsylvania 19406

Ashod N. Amirian, Esq.
145 South Main Street,
P.O. Box 38
Bradford, Massachusetts 01830

Mr. Alfred Sargent
Chairman
Board of Selectmen
Salisbury, Massachusetts 01950

Office of the Attorney General
One Ashburton Place
20th Floor
Boston, Massachusetts 02108

Durham Board of Selectmen
Town of Durham
Durham, New Hampshire 03824

Chairman, Board of Selectmen
RFD 2
South Hampton, New Hampshire 03827

R. Scott Hill-Whilton, Esquire
Lagoulis, Clark, Hill-Whilton
& Rotondi
79 State Street
Newburyport, Massachusetts 01950

Board of selectmen
Town of Amesbury
Town Hall
Amesbury, Massachusetts 01913

Mr. Ted C. Feigenbaum

Seabrook

cc:

Town of Exeter
10 Front Street
Exeter, New Hampshire 03823

Gerald Garfield, Esq.
Day, Berry and Howard
City Place
Hartford, Connecticut 06103-3499

Mr. J. F. Opeka
Northeast Utilities
P.O. Box 270
Hartford, Connecticut 06141-0270

Mr. F. M. Kacich
Northeast Utilities Service Company
P. O. Box 270
Hartford, Connecticut 06141-0270

Gene Spector
Federal Energy Regulatory Commission
825 North Capital Street, N.E.
Room 8105
Washington, D.C. 20426

Mr. Rob Sweeney
Three Metro Center
Suite 610
Bethesda, Maryland 20814

Mr. George L. Iverson, Director
New Hampshire Office of Emergency
Management
State Office Park South
107 Pleasant Street
Concord, New Hampshire 03301

Adjudicatory File (2)
Atomic Safety and Licensing Board
Panel Docket
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. Leon Maglathlin
Public Service Company of New Hampshire
1000 Elm Street
Manchester, New Hampshire 03105

Mr. Jack Dolan
Federal Emergency Management Agency
Region I
J.W. McCormack Post Office &
Courthouse Building, Room 442
Boston, Massachusetts 02109

Mr. J. M. Feschel
Public Service Company of New Hampshire
P.O. Box 300
Seabrook, New Hampshire 03874

Stephen G. Burns, Director
Office of Commission Appellate
Adjudication
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. David Rodham, Director
Massachusetts Civil Defense Agency
400 Worcester Road
P.O. Box 1496
Framingham, Massachusetts 01701-0317
ATTN: James Muckerheide

Atomic Safety and Licensing Board
Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

John P. Arnold, Attorney General
G. Dana Bisbee, Associate Attorney
General
Attorney General's Office
25 Capitol Street
Concord, New Hampshire 03301

Congressman Nicholas Mavroules
U. S. House of Representatives
70 Washington Street
Salem, Massachusetts 01970

Mr. Ted C. Feigenbaum

DISTRIBUTION w/enclosure:

Docket File 50-443 ←
NRC & Local PDRs
PDI-3 Reading
J. Calvo
W. Beckner
S. Varga
OGC 15B18
R. Hernan
ACRS (10)
E. Jordan, NMBB 3701
J. Linville, RI
G. Edison (2)
F. Congel
E. Chow (RES)
W. Minners (RES)
C. Ader (RES)
J. Schiffgens
J. Flack (RES)
K. Brockman
S. Crockett
J. Gray
E. McKenna
J. Scarborough
J. Linville, Region I

w/o enclosure

M. Rushbrook

ENCLOSURE

STAFF EVALUATION OF SEABROOK INDIVIDUAL PLANT EXAMINATION (IPE)
(INTERNAL EVENTS ONLY)

TABLE OF CONTENTS

PAGE

	EXECUTIVE SUMMARY	1
I.	BACKGROUND	2
II.	STAFF'S REVIEW	3
	1. Licensee's IPE Process	3
	2. Front-End Analysis	5
	3. Back-End Analysis	7
	4. Human Factor Considerations	9
	5. Containment Performance Improvements (CPI)	10
	6. Decay Heat Removal (DHR) Evaluation	10
	7. Licensee Actions and Commitments from the IPE	11
III.	CONCLUSION	11
IV.	REFERENCES	13
APPENDIX	SEABROOK DATA SUMMARY SHEET	15

EXECUTIVE SUMMARY

The NRC staff completed its review of the internal events portion of the IPE submittal, its associated documentation which included the Seabrook Station Probabilistic Safety Assessment (SSPSA), response to USI A-45 "Decay Heat Removal" resolution, an internal flood assessment, and response to staff generated questions which focused on the Seabrook IPE process and consideration of Containment Performance Improvement (CPI) program recommendations. No additional unresolved safety issues (USIs) or generic safety issue (GSIs) were proposed for resolution as part of the Seabrook IPE. Although the SSPSA contains a full range of internal and external event models, the Individual Plant Examination (IPE) submittal addresses only internal events. Hence, the staff has not reviewed external events. A separate updated Individual Plant Examination for external events (IPEEE) will be submitted at a later date.

The SSPSA which formed the basis of the licensee's IPE, is a full-scope Level 3 PSA completed in 1983. Subsequently, three substantial updates were performed and completed in 1986, 1989, and 1990. For each update, the applicable plant documents, including design documents and change requests, were reviewed and models changed as necessary. This update approach has been proceduralized as part of the risk management process at Seabrook. The latest PSA update is current through July 1990. Each update involved increasing levels of participation by utility staff, with the final update being conducted completely by utility personnel. The licensee plans to keep the SSPSA as a living document.

Walkdowns discussed in the IPE submittal included systems walkdowns for system familiarity, spatial interactions walkdowns (which included consideration of fire, flood and seismic effects), containment walkdowns, and containment bypass walkdowns. The IPE submittal states that during each walkdown, utility personnel from engineering and operations participated. The walkdowns constituted the process the licensee used to confirm that the IPE represented the as-built, as-operated plant.

The Seabrook IPE did not identify any severe accident vulnerabilities associated with either core damage or "unusually poor" containment performance. However, the IPE did identify potential improvements which the licensee plans to evaluate following completion of the IPE for External Events (IPEEE) and accident management evaluations. These potential improvements focus on reducing both core damage frequency and offsite release of radioactivity. Although the staff did not examine the merits of these improvements in detail, the improvements do not appear to be of sufficient safety significance such that they require regulatory action.

The review found the IPE submittal weak with regard to the level of documentation provided on the process used to develop the conditional Human Error Probabilities (HEPs) as part of the human reliability analysis. The licensee indicated in the IPE submittal, and in discussions with the IPE review team, that the next SSPSA update would include a revised human

review team, that the next SSPSA update would include a revised human reliability analysis. The staff, therefore, recommends that the licensee further document the basis for the HEPs and that they be checked for consistency with plant procedures as part of the accident management program.

Based on the Step 1 review of the Seabrook IPE submittal, and previous staff reviews of the SSPSA which included reviews by both Brookhaven National Laboratory (BNL) and Lawrence Livermore National Laboratories (LLNL), the staff concludes that the licensee meets the intent of Generic Letter 88-20. This conclusion is based on the following findings: (1) the IPE is complete with regard to the information requested in Generic Letter 88-20; (2) the IPE front-end and back-end analysis is technically sound and capable of identifying plant-specific vulnerabilities to severe accidents; (3) although the IPE had weak documentation on the process employed for developing explicit human error probabilities, the human reliability analysis is capable of identifying severe accident vulnerabilities which could result from pre-initiating (test and maintenance) through post initiating (operator recovery) human interactions; (4) the licensee has performed plant walkdowns to verify that the IPE reflects the current plant design and operation; (5) the SSPSA which formed the basis of the IPE had an extensive independent peer review; (6) the licensee participated fully in the IPE process; (7) the licensee is actively using the IPE as a living document to enhance plant safety; (8) decay heat removal capability was appropriately evaluated (responding to USI A-45), and (9) the licensee responded appropriately to the recommendations stemming from the CPI program.

In conclusion, the staff notes that the IPE review is not intended to validate the accuracy of the licensee's IPE, nor the bottom-line numbers so generated. Although certain aspects of the IPE were explored in more detail than others, the review primarily focused on the licensee's IPE process and its ability to identify vulnerabilities. The licensee's numerical results and safety insights are summarized in the appendix.

I. BACKGROUND

On August 8, 1985, the NRC issued a policy statement on severe accidents and concluded that there is a need for a systematic examination of all nuclear power plants for plant-specific severe accident vulnerabilities. In response to the policy statement, the NRC staff presented a plan for closure of severe accident issues (SECY-88-147) which contained six major elements, one requiring examination of existing plants for severe accident vulnerabilities. The NRC issued Generic Letter 88-20 on November 23, 1988 (Ref. 1) which promulgated the Individual Plant Examination (IPE) requirement.

On January 31, 1989, the NRC issued draft NUREG-1335 (Ref. 2) which established format and content guidelines for submitting the IPE results. After a public workshop to discuss these guidelines, the NRC issued Generic Letter 88-20 Supplement 1 (Ref. 3) on August 29, 1989 with the final NUREG-1335 (Ref. 4). Issuance of Supplement 1 to Generic Letter 88-20 initiated the internal event IPE process.

On March 1, 1991, the licensee formally submitted the Seabrook IPE (Ref. 6). The IPE submittal contained the results of an evaluation of both internal and external events; however, an updated Individual Plant Examination for External Events (IPEEE) is expected to be submitted in the future in response to Generic Letter 88-20, Supplement 4. (The staff will review the external events portion of the Seabrook IPE separately following receipt of the licensee's response to Supplement 4.) The licensee met with the NRC IPE review team to present IPE findings and conclusions. Following the team review of the Seabrook IPE submittal, the Seabrook Station Probabilistic Safety Assessment (SSPSA), and associated NRC contractor review reports, the IPE team generated a list of questions (Ref. 7) which were sent formally to the licensee. The licensee's response (Ref. 8) provided the review team with additional information necessary to complete the internal events review.

The NRC team review of the Seabrook IPE submittal involved an examination of the submittal for completeness, formulation of questions for additional information, meeting and discussions with the licensee to better understand the licensee's IPE process, and consolidation of IPE insights and findings for data base storage. This review is limited in scope as it is designed to look for significant omissions, or inconsistencies with commonly accepted Probabilistic Safety Assessment (PSA) practices. The review process is not intended to validate the accuracy of the licensee's IPE, nor the numerical results generated as part of the analytic process.

What follows is a summary of the Step 1 review team findings of the internal events portion of the licensee IPE and supporting information. Specific numerical results and other insights stemming from the licensee's IPE can be found in the appendix.

The information examined during the IPE review included:

1. Seabrook supplemental response to Generic Letter 88-20 (Ref. 6)
2. Seabrook response to NRC request for additional information (Ref. 8)
3. Seabrook Station Probabilistic Safety Assessment (SSPSA) (Ref. 9)
4. Contractor report by Lawrence Livermore National Laboratory (Ref. 10)
5. Contractor report by Brookhaven National Laboratory, NUREG/CR-4552 (Ref. 11)

Discussions were also held between team members and the licensee in order to gain additional insight into the Seabrook IPE analysis, and understanding of licensee participation in the IPE process.

II. STAFF'S REVIEW

1. Licensee's IPE Process

In 1983, Pickard, Lowe, and Garrick (PLG), Inc., Seabrook Station staff and Yankee Atomic Electric Company (YAEC) staff together performed a full-scope Level 3 risk assessment of the Seabrook Station. The analysis, or "Seabrook Station Probabilistic Safety Assessment" (SSPSA) was submitted to the NRC for review in 1984 (Ref. 9). NRC contracted Lawrence Livermore National Laboratory (LLNL) to review the SSPSA core damage models (Ref. 10) and Brookhaven National Laboratory (BNL) to review the containment analysis (Ref. 11). The SSPSA contains a full range of internal and external event models. Results

from these studies have been included in the current IPE submittal, although a separate updated Individual Plant Examination for External Events (IPEEE) will be submitted at a later date.

Since the completion of the SSPSA, three substantial updates referred to as the Seabrook Station Probabilistic Safety Studies (SSPSS) have been performed [1986 (Ref. 12), 1989 (Ref. 13), and 1990 (Ref. 14)]. The NRC also contracted the review of the SSPSA-1986 (BNL contract, Ref. 15) for specific issues relevant to emergency planning.

The Seabrook IPE submittal described the approach taken by the licensee to confirm that the IPE represents the as-built, as currently operated plant. In addition to plant walk-throughs, the original SSPSA received reviews by both in-house personnel and independent experts. For each subsequent SSPSA update (latest update current through July 1990), applicable plant design and change requests were reviewed, and models updated accordingly. This process has been proceduralized as part of the risk management process at Seabrook. Successive SSPSA updates also involved increasing levels of participation by utility staff with the latest update being conducted completely in-house. The licensee intends to maintain the IPE which is based on the SSPSA and subsequent SSPSS updates as a living document.

The staff examined the information associated with the walkdown activities of the licensee's IPE team including scope and team makeup. The IPE submittal documented licensee walkdowns performed for system familiarity, and spatial interactions which included consideration of fire, flood and seismic effects, containment walkdowns, and walkdowns which focused on containment bypass paths. The IPE submittal noted that utility personnel from engineering, operations, or both, participated in the walkdown activities. Based on this review, the staff concludes that the walkdowns constituted a viable process by which the licensee could confirm that the IPE represented the as-built, as-operated plant.

The IPE submittal contained a summary description of the licensee's staff participation in the IPE 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 that utility personnel have participated fully in the IPE process, and that an extensive peer review had been performed on the original SSPSA which forms the basis of the IPE submittal. The submittal documented comments and resolutions which resulted from the noted "six person-years" of peer-review effort. The effort involved two separate independent review boards, one to assure product quality, the other to assure technical credibility. According to the submittal, independence meant that "no reviewers on the board was [sic] allowed to contribute to a document or deliverable other than reviewing it."

Quantitative contributions to Core Damage Frequency (CDF) by functional sequences, initiating events, individual systems, and individual operator actions formed the licensee's basis for evaluating potential vulnerabilities to core damage. The functional sequences examined included 97.6% of the total ($1.1E-4/\text{yr}$) CDF for both internal and external events. The results indicate that 69.4% of the CDF stems from reactor coolant pump (RCP) seal loss-of-coolant accidents (LOCAs) that occur due to loss of offsite power (LOSP) or

transients involving loss of component cooling water (See section II).

The licensee defined vulnerabilities as those components, systems, operator actions, and/or plant design configurations that contribute significantly to an unacceptably high severe accident risk. The term, "contribute significantly" is defined as a contribution of more than 50% of the total frequency for a given risk measure. Two risk measures were identified: (1) the mean frequency of core damage (unacceptable: exceeding 2×10^{-4} /year) and (2) the mean frequency of large release (unacceptable: exceeding 2×10^{-6} /year). The Seabrook IPE analysis did not find any single initiating event, system, or human action that would have resulted in a risk measure that exceeded the above criteria.

Based on the employment of the above criteria, the staff notes the reasonableness of the licensee's conclusion that the study indicates that no fundamental weakness or severe accident vulnerabilities exist at the Seabrook Station. The staff finds the Seabrook IPE process capable of identifying "unacceptably high" severe accident risk contributors (or vulnerabilities) and that such capability is consistent with the objective of Generic Letter 88-20. Furthermore, the licensee's plan to update the Seabrook IPE periodically will provide additional assurance that any unforeseen vulnerability would be identified during the lifetime of the plant.

2. Front-End Analysis

The staff examined the front-end analysis (accident sequence delineation, system analysis, quantification, and sequence screening) for completeness and consistency with other PSAs. The overall review findings are that (1) the licensee's IPE is essentially complete, with the level of detail consistent with the information requested in NUREG-1335, and (2) the IPE techniques, and findings which stem from the analysis, are consistent with other NRC reviewed and accepted PSAs.

The SSPSA and associated updates (which forms the basis of the Seabrook IPE) contained directly or by reference, all of the plant information used in the IPE. The bulk of the plant layout information is contained in the Final Safety Analysis Report (FSAR), with additional containment design information in SSPSA-1990 Section 7 and its references. Appendix E of the IPE submittal contained summaries of various system analyses, including a brief description of system function, configuration, dependencies and operation. Also included were brief descriptions of system models, top events, success criteria and analysis conditions, and the results of the quantification of system unavailabilities and event tree split fractions.

In addition to referencing previous safety analyses conducted on the Seabrook Station, the IPE submittal referenced insights from the Zion PRA (Ref. 17) and Indian Point PRA (Ref. 18). These insights principally focused on dependencies, common mode failures, support system failures, and external hazards. The RCP seal LOCA is the dominant contributor to core damage frequency at Seabrook, intrinsic to two functional accident sequences, (1) station blackout stemming from LOSP, and (2) loss of component cooling. Taken together, these two sequences total almost 70% of the core damage frequency. The licensee utilized the NUREG-1150 (Ref. 19) RCP seal LOCA assumptions in the SSPSA-1990.

The SSPSA and subsequent updates employed the "large event tree - small fault tree" modeling technique, sometimes called the "event tree linking approach." In the latest update, the plant (front-end) model and the containment response (back-end) model were linked by computerized logic rules resulting in direct production of accident sequences in terms of release categories. This eliminated the intermediate step of manually constructing plant damage states by binning the front-end core damage sequences. Although plant damage states were not explicitly determined, the staff finds the large event-small fault tree approach used by the licensee technically sound, and acceptable for meeting the intent of Generic Letter 88-20.

The initiating events appeared to have been appropriately reflected in the plant design dependency models and success criteria. The submittal contained 72 initiating events, of which 36 were identified as internal events. The staff compared the list of initiators to similar lists generated in other PSAs and NUREG/CR-2300 (Ref. 16), and found the list complete with the exception of the loss of instrument air. This particular initiator was modeled as an addition to the frequency of the total loss of main feedwater initiator, and was not included explicitly in the dependency matrices. In response to staff's questions (numbers 1 through 4 in Ref. 7), the licensee explained that given a loss of instrument air, components would fail safe or resort to backup air accumulators. Following the review of this additional information the staff concluded that the loss of instrument air initiator would not significantly increase the total CDF or release of radioactive material. The licensee, however, agreed to include instrument air in the dependency matrix during the next SSPSA update.

The IPE submittal contained the event trees, system dependency matrices, top event descriptions, top event interdependencies, success criteria, and system descriptions necessary to understand the sequences. In most cases, the bases for the top event success criteria were not provided explicitly in the IPE submittal, but were found to be available in the various referenced documents. The success criteria presented by the submittal were reviewed on an audit basis, and found to be reasonable when compared to criteria used in other PSAs.

The PLG computer code RISKMAN (Ref. 20) was used to evaluate the model's event trees. Event tree split fractions were evaluated using fault trees and/or reliability diagrams and the IRRAS (Ref. 21) computer code. Dependent failure mechanisms were treated by a combination of explicit and parametric approaches. Master logic diagrams, heat balance fault tree methodology and specialized failure modes and effects analyses were used to identify common cause initiating events. Functional and shared equipment dependencies were modeled explicitly in the event tree logic. The Seabrook IPE employed the "multiple Greek letter" method to model common cause failures among like components in parallel applications.

The SSPSA and updates incorporate plant-specific logic models of systems, system dependencies, spatially related interactions, success criteria and operator actions. Because Seabrook Station has only recently begun commercial operation, the IPE utilized generic initiating event frequency and component failure rate data from the PLG database PLG-0500.

The IPE submittal referenced the original SSPSA internal flood risk analysis,

which had been recently updated to reflect the as-built plant configuration and more recent industry experience. The analysis included identification of critical flooding areas, calculation of flood frequency distributions, and flood severity and mitigation possibilities. Critical locations were identified by combining a plant systems location matrix with a plant level fault tree to identify minimal cutsets for core damage and radioactive release. All components in a flooded area were considered disabled; fragilities were not considered. Critical locations identified from this analysis were examined in detail to develop flood scenarios and estimate frequencies. The only significant internal flooding scenarios resulted from floods originating in the turbine building and affecting the adjacent switchgear rooms. The sequences lead to loss of offsite power with concurrent loss of one or both vital buses, however, the overall contribution to core damage frequency had been estimated to be less than 1%.

The submittal contained the top 100 most probable core damage sequences accounting for 97.6% of the total mean core damage frequency ($1.1E-4$ /year). Internal events contributed 55% of this total with the remainder associated with external events. The submittal identified the dominant sequences and contributors by initiator, system and operator action. The top twenty sequences were described with respect to their accident progression, and a list of potential improvements were identified to be analyzed for their cost-benefit for reducing the CDF (Table 6.2 in Ref. 6). These potential improvements or safety enhancements are to be evaluated by the licensee after completion of the IPEEE and accident management review. Evaluation of these improvements is not in response to any identified or perceived vulnerability, and it is therefore reasonable to perform these evaluations after completion of the related IPEEE and accident management reviews.

The staff did not identify any obvious or significant weaknesses with the front-end analysis. The licensee employed techniques consistent with acceptable PSA practices, and these techniques were capable of identifying potential severe accident vulnerabilities defined earlier in Section II.1. The staff, therefore, finds the IPE front-end analysis consistent with the intent of Generic Letter 88-20.

3. Back-End Analysis

The IPE review examined the back-end analysis which included the containment feature description, containment failure characterization, Containment Event Tree (CET) representation, and radionuclide release. The Seabrook containment structure is noted to be large ($2.6E+6$ cu ft) and relatively stronger than most plants in its class, i.e., 210 psia for wet containment sequences. [Dry sequences can lead to a superheated containment atmosphere and higher containment temperatures (above 700 degrees F), tending to fail the containment at a lower pressure.] Bunkers house the RHR pumps which allow for scrubbing of releases that could result from interfacing system LOCAs occurring in the RHR system.

The staff examined the documentation of referenced codes, analytical models and data input. As discussed earlier in Section II.1, the Seabrook IPE back-end analysis does not join to the front-end analysis through plant damage state binning. The front-end sequences are linked directly to the back-end sequences which address the 19 CET top events contained in the RISKMAN

software package. Logic rules that determine split fractions for top events in the CETs implicitly perform the binning process. (The logic rules intrinsically include availability of safety equipment such as the emergency feedwater system.) The CET end points are subsequently binned into nine distinct release categories.

The present CETs evolved from CETs which originally consisted of only 12 top events and 154 sequences. New additions include direct containment heating (DCH) and induced steam generator tube rupture (ISGTR). Plant specific containment response analyses were performed using MARCH (Ref. 22), COCOCLASS9 (Ref. 23), MODMESH, and CORCON-MOD1 (Ref. 24) computer codes. For additional insight, results were compared to those obtained in the Zion and Indian Point PRA studies.

The IPE determined that the dominant contributors to containment isolation failure sequences were primarily due to purge valve isolation signal failures. The dominant contributors to containment by-pass stemmed from ISGTR sequences (Fig 1.6 of Ref. 6).

The licensee defined "unusually poor" containment performance as all events resulting in an early large failure of containment (i.e. failures sufficiently large to prevent containment pressurization.) The licensee estimates the conditional and absolute probabilities of all events resulting in early large failure of containment to be small (0.002 and $2.1E-7$ /year respectively). The most likely mode of containment failure is a Type B, defined by the licensee as sufficiently large to be self-limiting, i.e., the failure is of sufficient size to limit or prevent further containment pressure increase. Type B failures are principally associated with failure of the containment piping penetrations seals.

Failure of elastomer material primarily used to seal personnel and equipment hatches and electrical penetration assemblies, were considered in the evaluation of containment integrity. The mechanical and thermal properties of the elastomer seals enabled seal failure pressures to be in excess of the failure pressures predicted by the structural analyses. Heat transfer, mass transport analyses, and evaluation of maximum leakage areas afforded by clearances between metal to metal contacting surfaces were utilized to support the above conclusion.

The licensee did not find any vulnerabilities that would lead to previously defined "unusually poor" containment performance (Section 1.4.2 of Ref. 6). However, the licensee did identify a list of potential improvements which are to be analyzed for their cost-benefit for reducing the offsite release. The list of candidates includes limiting the use time of containment purge valves and procedures to direct RCS depressurization in order to preclude DCH. The licensee indicated that these potential procedural and administrative improvements will be evaluated following completion of the IPEEE and accident management evaluations. According to the IPE, implementation of these improvements are not in response to any identified or perceived vulnerability and, therefore, do not impact the IPE conclusions regarding containment performance. Based on the IPE review team findings, the staff finds the licensee's evaluation approach acceptable, i.e., the decision and time frame for integrating these improvements into the plant would best be determined by the licensee in order to minimize interference with other ongoing safety

activities.

In summary, the 1984 SSPSA has been amended and augmented to incorporate new methodology, current plant configuration (as of July 1990), new phenomenological insights, and current equipment performance characteristics. Specifically, the licensee's IPE addressed the most important severe accident phenomena normally associated with large dry containments, i.e., DCH, Induced Steam Generator Tube Rupture (ISGTR), and hydrogen combustion. The IPE review did not identify any obvious or significant problems or errors in the back-end analysis. The IPE team's overall assessment of the back-end analysis is that the licensee has made reasonable use of PSA techniques in performing the back-end analysis, and that the techniques employed were capable of identifying potential severe accident 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's IPE treated human reliability explicitly. Three types of human interactions were included in the analysis: (1) pre-initiating event interactions or those operator or technician actions that can inadvertently disable safety equipment during test or maintenance, (2) initiating event related interactions which can cause reactor transients, and (3) post-initiating event related interactions which include operator response to reactor transients and recovery actions. The submittal contained a list of human reliability data, a list of the data sources, and a list of important human errors and recovery actions.

The Seabrook IPE process employed the Human Reliability Analysis (HRA) contained in the original SSPSA, dated 1984. The HRA used operator action trees, a qualitative operator-plant status confusion matrix, and results from Seabrook simulator trials. One simulator trial was used directly in the quantification of human error by identifying an anchor point to validate the HEPs derived from other sources. A method that systematically incorporated expert opinion (SLIM-MAUD-like technique see Ref. 25), was also used to develop Human Error Probability (HEP) estimates by incorporating plant-specific information and performance shaping factors (i.e., time, potential for misdiagnosis, and level of stress). However, the NRC IPE review team found the Seabrook submittal weak with regard to the level of documentation provided on the process used for developing the conditional HEPs. The licensee indicated in the IPE submittal that the next SSPSA update will include a revised HRA.

Human errors previously identified in the LLNL review (Ref 10) were added to the plant logic models for the Seabrook IPE. These errors included: (1) operator failure to provide makeup to the refueling water storage tank (RWST) during a small LOCA, (2) operator failure to recover engineered safety features actuation system (ESFAS) with long response time (60 minutes), and (3) operator failure to recover ESFAS during LOCA conditions. The licensee also updated the electric power recovery model as part of the IPE analysis.

The Seabrook HRA performed in 1983 used then state-of-the-art methods which included qualitative and quantitative techniques, and simulator trials, both of which are still viable methods today. Insights from the original HRA and

other follow-on studies have been incorporated into plant procedures. For example, the licensee used the simulator to evaluate operator response to plant changes and develop operator training programs. The licensee also indicated that it had reviewed the Emergency Operating Procedures (EOPs) to see if any recent changes would impact the analysis. None were identified.

The IPE submittal did not identify sequences that, except for low human error rates in recovery actions, would have been above the licensee's screening criteria which follow the guidance from NUREG-1335. The submittal did, however, provide a table of the risk achievement worth (RAW) importance measures for the important operator actions in order to evaluate their sensitivity. Although Generic Letter 88-20 did not require RAW importance measures, they are an important means by which significant insights can be gleaned from probabilistic studies.

Based on the information contained in the IPE submittal, the SSPSA, responses to staff questions, and discussions with the licensee, and contractor reviews, the staff judges that the HRA process used by the licensee is capable of uncovering severe accident vulnerabilities from human error, and that the process employed is consistent with the intent of Generic Letter 88-20.

5. Containment Performance Improvements (CPI)

Generic letter 88-20 Supplement 3 (Ref. 26) contained CPI recommendations which focused on the vulnerability of containments to severe accident challenges. For large dry containments, such as the Seabrook design, the CPI program results recommended that licensees in their IPE focus on hydrogen production and control during severe accidents, particularly on the potential for local hydrogen detonation.

With regard to hydrogen combustion and detonation, the licensee has estimated that the conditional containment failure probability resulting from global adiabatic burn of all the hydrogen produced by oxidation of 100% of the zirconium in the reactor, is less than $1E-4$ with a maximum predicted containment pressure of 110 psia. Also, as a result of a review and analysis of the Seabrook containment design, a site walkdown, and comparisons to the Indian Point 3 containment design, the licensee concluded that there is negligible probability of containment failure or severe damage that could result from local detonations due to hydrogen "pocketing" inside the containment. The licensee based this conclusion upon the open containment features, minimal enclosed spaces and the liberal use of open floor gratings. The licensee's conclusions are consistent with those for Indian Point 3 which has a containment design that closely resembles that of Seabrook. The staff, therefore, concludes that the licensee's response to CPI Program recommendations, which includes searching for vulnerabilities associated with containment performance during severe accidents, is reasonable and consistent with the intent of Generic Letter 88-20 and associated Supplement 3.

6. Decay Heat Removal (DHR) Evaluation

In accordance with the resolution of USI A-45 "Shutdown Decay Heat Removal Requirements," the licensee performed an examination of the Seabrook DHR system to identify vulnerabilities. The examination method employed in the IPE has been found to be consistent with approaches identified in Generic

Letter 88-20. The licensee's examiner did not identify any DHR vulnerabilities using the criteria defined earlier in Section II.1.

The Seabrook IPE used a 24-hour mission time for DHR following reactor trip which is consistent with IPE requirements. In addition, the IPE conservatively modeled feed and bleed cooling success criteria by requiring operation of both pressurizer PORVs. The IPE also noted the reliability of the PORVs because of their independence from all support systems except DC power. Recent analyses were cited in the submittal which indicate that, with the available combinations of high head pumps, only one PORV is needed to provide sufficient cooling. This would further reduce the contribution of DHR function to core damage frequency. Procedures and training are in place at Seabrook to justify credit for feed and bleed cooling.

Based on these findings, the staff concludes that the licensee's DHR evaluation is consistent with the intent of Generic Letter 88-20 to resolve USI A-45 as part of IPE. Therefore, USI A-45 is considered resolved for the Seabrook Station.

7. Licensee Actions and Commitments from the IPE

The IPE submittal provides a discussion of potential improvements which are to be analyzed for their cost-benefit. Table 6.1 in the IPE submittal (Ref. 6) lists the top 24 core damage sequences with potential improvements for each. Table 6.2 in the IPE submittal summarizes the benefits for each potential plant design improvement. The improvements are associated primarily with the reduction in the frequency of RCP seal LOCA. An additional, independent, automatically initiated charging pump is shown to provide a 61% reduction in CDF. The addition of an independent, automatic seal injection pump indicates a 59% reduction. Manually actuated versions of either option would result in only a 28% reduction in CDF. Automatic initiation was found to be important because many of the initiators leading to core damage through RCP seal LOCAs were external events such as control room fires and earthquakes, and these negatively affect operator actions.

The licensee also identified a list of potential improvements which are to be analyzed for their cost-benefit for reducing the offsite release (Table 6.3 in Ref. 6). The list of candidates included limiting the use of containment purge valves and procedures to direct reactor cooling system depressurization in order to preclude DCH. The licensee indicated that potential procedural and administrative improvements will be evaluated following completion of the IPEEE and the accident management evaluations. Although the team did not examine the merits of these improvements in detail, the improvements do not appear to be of sufficient safety significance such that they require regulatory action. Therefore the licensee's decision to evaluate plant improvements following completion of the IPEEE and accident management studies is acceptable.

III. CONCLUSION

Based on the team review of the internal events portion of the licensee's IPE submittal, the staff finds that:

- (1) The Seabrook IPE is complete with regard to the information and

subject areas identified in Generic Letter 88-20 and associated NUREG-1335 document;

- (2) The PSA methodology used by the licensee for both the front-end and back-end analysis is technically sound and is capable of identifying plant-specific vulnerabilities to severe accidents
- (3) Although the IPE had weak documentation on the process employed for explicitly developing the HEPs as part of the HRA, further discussions with the licensee have led to the conclusion that the HRA process is capable of identifying severe accident vulnerabilities which could result from pre-initiating (test and maintenance) through post-initiating (operator recovery) human interactions;
- (4) The licensee performed walkdowns to verify that the IPE models reflect the current plant design and operation, consistent with NUREG-1335 guidance document;
- (5) The peer review of the SSPSA IPE, is consistent with the guidance provided in NUREG-1335;
- (6) The licensee participated fully in the IPE process with minimal reliance on contractors;
- (7) The licensee is actively using the IPE as a living document to enhance plant safety;
- (8) The IPE is capable of identifying vulnerabilities associated with the decay heat removal system, therefore, USI A-45 is considered resolved for the Seabrook Station, and
- (9) The licensee's response to the recommendations stemming from the CPI program appeared reasonable.

Based on the above, 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 Seabrook facility, and has gained a quantitative understanding of core damage and fission product release. The staff, therefore, finds the Seabrook IPE process acceptable, and meets the intent of Generic Letter 88-20.

IV. REFERENCES

1. NRC letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR §50.54(f)," Generic Letter No. 88-20, dated November 23, 1988.
2. USNRC, "Individual Plant Examination: Submittal Guidance," NUREG-1335, Draft Report for Comment, January 1989.
3. NRC letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities, "Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR §50.54(f)," - Generic Letter No. 88-20, Supplement No. 1, dated August 29, 1989.
4. USNRC, "Individual Plant Examination: Submittal Guidance," NUREG-1335, Final Report, August 1989.
5. T. Feigenbaum, New Hampshire Yankee to USNRC, "Response to Generic Letter 88-20," NHY Letter NYN-89126, November 1, 1989.
6. B. Drawbridge, New Hampshire Yankee to USNRC, "Supplementary Response to Generic Letter 88-20," NHY Letter NYN-91034, March 1, 1991.
7. G. Edison, USNRC to T. Feigenbaum, New Hampshire Yankee, "Seabrook - Individual Plant Examination (IPE) Review - Request for Additional Information (TAC No. 74466)," July 5, 1991.
8. T. Feigenbaum, New Hampshire Yankee to USNRC, "Response to Request for Additional Information Regarding the Seabrook Station IPE Report," NHY Letter NYN-91116, July 23, 1991.
9. New Hampshire Yankee to USNRC, "Seabrook Station Probabilistic Safety Assessment," NHY Letter SBN-617, January 30, 1984.
10. Garcia, A., et al., "A Review of the Seabrook Station Probabilistic Safety Assessment," Lawrence Livermore National Laboratory, December 1984.
11. NUREG/CR-4552, "A Review of the Seabrook Station Probabilistic Safety Assessment - Containment Failure Modes and Radiological Source Terms," Brookhaven National Laboratory, March 1987.
12. New Hampshire Yankee to USNRC, "Seabrook Station Probabilistic Safety Study - 1986 Update," July 1987.
13. New Hampshire Yankee to USNRC, "Seabrook Station Probabilistic Safety Study - 1989 Update," December 1989.
14. New Hampshire Yankee to USNRC, "Seabrook Station Probabilistic Safety Study - 1990 Update," December 1990.

15. USNRC Letter, "Transmission of Brookhaven National Laboratory Technical Evaluation of the Emergency Planning Sensitivity Study for Seabrook," transmitting Technical Report A-3852, March 9, 1987.
16. NUREG/CR-2300, "PRA Procedures Guide," January 1983.
17. Pickard, Lowe, and Garrick, Inc., Westinghouse Electric Corporation, and Fauske & Associates, Inc., "Zion Probabilistic Safety Study," September 1981.
18. Pickard, Lowe, and Garrick, Inc., Westinghouse Electric Corporation, and Fauske & Associates, Inc., "Indian Point Probabilistic Safety Study," March 1982.
19. NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Plants," (Second Draft for Peer Review), June 1989.
20. Pickard, Lowe, and Garrick, Inc., "RISKMAN PRA Workstation Software User Manual" (Proprietary), Release 2.0.
21. NUREG/CR-4844, "Integrated Reliability and Risk Analysis System (IRRAS) User's Guide - Version 1.0 (Draft)," June 1987.
22. NUREG/CR-1711, "MARCH (Meltdown Accident Response Characteristics) Code Description and User's Manual," BMI-2064, Battelle Columbus Laboratories, October 1980.
23. Bordelon, F. M. and Murphy, E. T., "Containment Pressure Analysis Code," WCAP-8327, COCOCLASS9 Code, Westinghouse Electric Company, July 1974.
24. MODMESH & CORCON-MOD 1 Codes, Seabrook Station Probabilistic Safety assessment, Volume 6, December 1983.
25. NUREG/CR-3518, "SLIM-MAUD: An Approach to Assessing Human Error Probabilities Using Structured Expert Judgment," 1984.
26. NRC letter to All Licensees Holding Operating Licenses and Construction Permits for Nuclear Power Reactor Facilities Except Licensees for Boiling Water Reactors with MARK I Containments, "Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the Individual Plant Examination for Severe Accident Vulnerabilities - Generic Letter No. 88-20, Supplement No. 3, dated July 6, 1990.

APPENDIX
SEABROOK DATA SUMMARY SHEET*
(INTERNAL AND EXTERNAL EVENTS)

o Total Core Damage Frequency:

1.1E-4/year (mean value)
55% resulting from internal events
45% resulting from external events

o Major Initiating Events and contribution to core melt frequency (internal and external events):

	Internal	External
Transients:	(42%)	(41%)
- LOSP	(16%)	(24%)
- Loss of Support Systems	(7%)	(17%)
- General Transient	(19%)	(0%)
LOCAs	(7%)	(1%)
ATWS	(6%)	(3%)
Total		
	(97%)	(96%)

o Major systems and contribution to core melt frequency:

Diesel Generator	(27.5%)
Primary Component Cooling	(17.5%)
Service Water	(15.7%)
Emergency Feedwater	(14.8%)
Residual Heat Removal	(3.8%)

o Major operator action failures (in decreasing risk importance):

- Failure to recover electric power
- Failure to recover signal
- Failure to recover EFW
- Failure to perform manual reactor shutdown
- Failure to control SGTR break flow and depressurize
- Failure to feed and bleed
- Failure to provide makeup to the RWST
- Failure to switchover to high pressure recirculation
- Failure to depressurize during SBO
- Failure to control EFW

o Conditional containment failure probability given core damage

Late Containment Failure	(65.4%)
Intact Containment	(20.2%)
Early Small Containment Failure/Bypass	(14.2%)
Early Large Containment Failure/Bypass	(0.2%)

- o Conditional containment failure mode contributions to early large containment failure/bypass (Unusually poor containment performance)

Containment Isolation Failure	(58.7%)
Induced Steam Generator Tube Rupture	(26.8%)
Direct Containment Heating	(11.1%)

- o Proposed modifications under consideration to reduce core damage frequency:

1. Independent, automatic seal injection pump
2. Independent, manual seal injection pump
3. Independent, manual charging pump
4. Alternate emergency AC power source (e.g., swing diesel)
5. Alternate offsite power source that bypasses switchyard
6. Alternate scram button to remove power from MG sets to control rod drives
7. DC power enhancement:
 - independent AC source for battery chargers
 - credit operator action to cross-tie batteries within each train
 - additional batteries

- o Proposed modifications under consideration to reduce offsite release:

1. Administrative control to reduce time the purge valves are open
2. Procedure to direct depressurization of reactor coolant system
3. Alternate, independent emergency feedwater pump
4. Containment leakage monitoring
5. Residual heat removal isolation valve leakage monitoring system

- o Future Activities: - Periodic update of SSPSA
(* All information is taken from the Seabrook IPE and has not been validated by the NRC staff.)