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

OCT 31 1991

MEMORANDUM FOR: Frank J. Congel, Director
Division of Radiation Protection and
Emergency Preparedness, NRR

FROM: Warren Minners, Director
Division of Safety Issue Resolution, RES

SUBJECT: REVIEW OF SEABROOK INDIVIDUAL PLANT EXAMINATION (IPE) SUBMITTAL - INTERNAL EVENTS

We have completed a Step 1 review of the Seabrook IPE. The review is based on the efforts of a review team which consists of the team leader Ed Chow, William Milstead, Dale Rasmuson, all from RES and Steve Long from NRR.

Our review has covered the internal event analysis in the IPE submittal, its associated documentation which includes a probabilistic safety assessment (PSA), licensee's response to USI A-45 "Decay Heat Removal," internal flood assessment, and licensee response to staff's questions which focused on the Seabrook IPE process and consideration of Containment Performance Improvement (CPI) program recommendations. No other generic safety issues were proposed for resolution in the IPE submittal.

Based on the Step 1 review, we conclude that the licensee has met the intent of Generic Letter 88-20 and do not recommend a Step 2 review be conducted. This conclusion is based on the staff's review of the IPE submittal, the licensee's involvement in PRA activities, implementation of safety enhancements and continued employment of their PRA to enhance safety at Seabrook. A discussion of the team review is provided in the enclosed draft Safety Evaluation Report (SER) which we recommend be issued to document the staff's findings. A separate draft SER will be provided to document the staff's review of the external event portion of the Seabrook IPE submittal.

The licensee has not found any vulnerabilities associated with core damage or "unusually poor" containment performance. However, the licensee has identified a number of potential procedural and administrative improvements which will be evaluated following completion of the IPEEE and the accident management evaluations. These potential improvements are to reduce core damage frequency or offsite release (Table 6.2 and Table 6.3 of the IPE submittal- Ref. 6). The list of improvements includes limiting the time containment purge valves are allowed open and procedures to direct RCS depressurization. We suggest NRR keep track of these actions to ensure that the implementation will be carried out properly.

Finally, it is important to note that as discussed in the SER, the review process is not intended to validate the accuracy of the licensee's IPE, including the bottom-line numbers generated in the performance of the IPE.

If you have any questions regarding the enclosed SER, please contact Ed Chow on x23984.

Warren Minners

Warren Minners, Director
Division of Safety Issue Resolution, RES

Enclosure: As stated

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ENCLOSURE 1

EVALUATION OF SEABROOK INDIVIDUAL PLANT EXAMINATION (IPE)

(INTERNAL EVENTS ONLY)

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I. INTRODUCTION

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. Subsequently, the NRC staff presented a plan for closure of severe accident issues, and on November 23, 1988 issued Generic Letter 88-20 (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 IPE internal event submittals. A public workshop was held in Fort Worth, Texas on February 28 and March 1-2, 1989 to discuss these guidelines. Finally, on August 29, 1989, the NRC issued Generic Letter 88-20 Supplement 1 (Ref. 3) with a final NUREG-1335 (Ref. 4) which initiated the internal event IPE process.

On November 1, 1989, New Hampshire Yankee (NHY) submitted to the NRC a schedule for completing its IPE (Ref. 5). On March 1, 1991, the licensee submitted its IPE to the NRC (Ref. 6). The IPE submittal contained the results of an evaluation of internal and external events; however, an updated external events portion is to submitted in the future. On April 24, 1991, the licensee met with the staff and presented its IPE findings. Based on the review of the IPE submittal, the Seabrook probabilistic safety assessment (PSA), and associated contractor reports, the staff generated a list of questions (Ref. 7) which were sent to the licensee. The subsequent licensee's responses (Ref. 8) to the questions provided the staff with additional information necessary to complete the internal event review. The staff's review of the external events portion of the submital will be documented separately.

The staff's review of IPE submittals is a two step process. The first step, or Step 1 review involves a check of the submittal for completeness, formulating questions for additional information, discussions with the licensee to better understand the licensee's IPE process, and extracting IPE insights and findings. Since the Step 1 review is a limited scope review, it is an attempt to look for any obvious or significant omissions, or inconsistencies with known probabilistic risk assessment (PRA) practices. Step 2 provides for a more detailed review of the submittal, but is only conducted where the Step 1 review indicates a more detailed review is warranted. In either case, the review process is not intended to validate the accuracy of the licensee's IPE, including the bottom-line numbers generated in the performance of the IPE. The results of the Step 1 review for Seabrook do not indicate the need for a Step 2 review.

What follows is a summary of the staff's findings based on the Step 1 review of the internal events portion of the licensee IPE and supporting information. This information 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)

In addition, meetings held between the staff and licensee on April 24, 1991 also provided additional insight into the Seabrook IPE and participation of the licensee in the IPE process.

II. STAFF'S REVIEW

1. Licensee's IPE Process

The Seabrook Station Probabilistic Safety Assessment (SSPSA), is a full-scope, Level 3 PRA, completed in 1983 and initially submitted to NRC for review in 1984 (Ref. 9). The original SSPSA was performed by Pickard, Lowe, and Garrick (PLG), Inc. with participation by Seabrook Station staff and Yankee Atomic Electric Company (YAEC) staff. Since then, three substantial updates, referenced as Seabrook Station Probabilistic Safety Studies (SSPSSs) have been performed, with completion dates in 1986 (Ref. 12), 1989 (Ref. 13), and 1990 (Ref. 14). NRC reviews of the original SSPSA were conducted by Lawrence Livermore National Laboratory (LLNL) for the core damage models (Ref. 10) and by Brookhaven National Laboratory (BNL) for the containment performance models (Ref. 11). NRC review of the SSPSS-1986 for specific issues relevant to emergency planning was also conducted by BNL (Ref. 15).

The IPE submittal describes the utility's approach to confirming that the IPE represents the as-built, as operated plant. The original SSPSA was based on then-current plant design documents, procedures and plant walk-throughs. The original SSPSA was performed by contractors with substantial participation by Seabrook engineering staff and YAEC staff. The original SSPSA was reviewed by both in-house and independent experts. For each update of SSPSA (SSPSS), the applicable plant documents, including design documents and change requests, were reviewed and the models were changed as necessary. This process has been proceduralized as part of the risk management process. The IPE submittal states that the SSPSS-1990 is current through July, 1990. Successive updates (SSPSSs) involved increasing levels of participation by utility and YAEC staff. The IPE submittal states that the 1990 update, which forms the basis for the submittal was conducted completely by NHY and YAEC personnel. The licensee intends to keep the SSPSS as a living document.

The SSPSA contains a full range of internal and external event models. These results have been included in the current IPE submittal, although a separate Individual Plant EXamination for External Events (IPEEE) will be submitted at a later date.

The staff examined the information associated with the walkdown activities of the licensee's IPE team including scope and team makeup. The IPE states that the walkdowns performed included systems walkdowns for system familiarity, spatial interactions walkdowns - including 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/or Operations participated. The walkdowns constituted the process the licensee used to confirm that 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 in-house licensee peer review. The staff examined the licensee's description of the IPE program organization, composition of independent review team, areas of licensee review, licensee findings and conclusions, and response to NRC questions and comments. The staff notes that utility personnel have participated fully in the IPE process and have responded to all the staff's questions and comments.

The core damage vulnerability screening was based on evaluation of the contributions made to core damage frequency (CDF) by functional sequences, initiating events, individual systems, and individual operator actions. The functional sequences examined included 97.6% of total CDF which is 1.1E-4/year for internal and external events. They indicate that 69.4% of the total results from reactor coolant pump (RCP) seal loss-of-coolant accidents

(LOCAs) that occur due to loss of ac power or transients involving loss of component cooling water.

The licensee defines 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" means a contribution of more than 50% of the total frequency for a given risk measure. In addition, the term "unacceptably high severe accident risk" refers to two risk measures, namely, mean frequency of core damage exceeding 2E-4/year and mean frequency of large, early release exceeding 2E-6/year. The IPE submittal concludes that the IPE revealed no fundamental weakness or design vulnerability for Seabrook Station. This conclusion is based on the reasoning that no single initiating event, system, or human action is involved with a majority of the risk (e.g., >1.0E-4 in CDF).

Based on the team review of the licensee's IPE process, the staff finds that the licensee's IPE process has met the intent of Generic Letter 88-20.

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 PRA methods. The staff's review determined that the licensee submitted the appropriate information with sufficient level of detail consistent with NUREG-1335, and that the appropriate information sources were identified. Based on the staff's review, the initiating events appeared to have appropriately reflected the plant design and dependencies, and sequences and the associated bases for front-line system success appeared to be appropriate.

The "large event tree - small fault tree" modeling technique, sometimes called the "event tree linking approach," was used for the SSPSA and its updates. In the latest update, the plant (front-end) model and the containment response (back-end) model are linked by computerized logic rules, resulting in direct production of accident sequences in terms of release categories. This eliminates the intermediate step of manually constructing plant damage states by binning the front-end core damage sequences. Therefore, plant damage states have not been provided in the submittal. The IPE submittal provided information consistent with NUREG-1335 on event tree modeling, system analysis, dependency treatment, and the quantification process. The IPE submittal contains 72 initiating events, 36 of which were identified as internal events. Initiators specific to the Seabrook plant were identified, where appropriate. The staff compared the list of initiators to similar lists from other PRAs and NUREG/CR-2300 (Ref. 16). The list appears to be comprehensive with the exception of the loss of instrument air initiator. This event was modeled as an addition to the frequency of the total loss of main feedwater initiator. However, as discussed later, New Hampshire Yankee, in response to the staff's questions (numbers 1 through 4 - Ref. 7), provided additional information that has allowed the staff to conclude that the modeling of the loss of instrument air initiator will not significantly increase the total CDF or release of radioactive material.

The New Hampshire Yankee SSPSS contained, directly or by reference, all of the plant information used for the IPE. The bulk of the plant layout information is contained in the Final Safety Analysis Report (FSAR), with additional containment design information in SSPSS-1990, Section 7 and its references. Appendix E of the IPE submittal contains summaries of the various system analyses, including a brief description of system function, configuration, dependencies and operation. Also included are 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 were principally in the areas of the importance of dependencies and common mode failures, support system failures and external hazards. The RCP seal LOCA is the dominant contributor to core damage frequency at Seabrook, but, due to design differences, the dominant initiating events leading to seal LOCA at Seabrook differ from the other plants referenced. The licensee utilized in the conduct of the original SSPSA and the NUREG-1150 (Ref. 19) RCP seal LOCA assumptions used in the SSPSS-1990.

Emergency operating procedures (EOPs) have been reviewed and modeled by the licensee. In particular, insights from the performance of a Seabrook operations crew during simulation of key accident sequences on the Seabrook simulator were used to help model operator actions such as cognitive errors, errors of omission and commission, and recovery actions.

The 100 most probable core damage sequences are provided in the IPE submittal. The total CDF is reported as 1.1E-4/year (mean value). Internal events contributed 55% of this total. External events contributed the remainder.

The IPE submittal contains the event trees, system dependency matrices, top event descriptions, top event interdependencies, success criteria, and system descriptions necessary to understand the sequences. However, the instrument air system was not included explicitly in the dependency matrices. In response to staff's question number 4 (Ref. 7), NHY has explained that in many cases loss of instrument air results in components failing safe and for other applications, air accumulators have been added to reduce the impact of loss of instrument air. New Hampshire Yankee has stated that it will be including instrument air in the dependency matrices for the next SSPSS update. In most cases, the bases for the top event success criteria were not provided either explicitly or in the IPE submittal. They are available in the various referenced documents, however. The success criteria presented by the submittal were reviewed on an audit basis, and none were found to be unreasonable when compared to criteria used in other PRAs.

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 "multiple greek letter" method was used for parametric modeling of 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, generic initiating event frequency and component failure rate data from the PLG database was utilized. However, in future updates of the SSPSA the use of plant specific data is encouraged as such data becomes available.

The IPE submittal states that analysis of internal flooding risk was conducted as part of the spatial interaction study for the original SSPSA in 1982-83, and has been recently updated to represent the as-built plant configuration and incorporate recent industry experience. The analysis included identification of critical flooding areas, calculation of frequency distributions for floods in these areas, and consideration of 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 which resulted from this analysis were examined in detail to develop flood scenarios and estimate frequencies. The only significant internal flooding scenarios result from floods originating in the turbine building and affecting the adjacent switchgear rooms. The results can include loss of offsite power with concurrent loss of one or both vital buses.

The submittal identified the dominant sequences and contributors by initiator, system and operator action. The top twenty sequences are explained briefly with respect to accident progression. The submittal also identified a list of potential improvements which are to be analyzed for their cost-benefit for reducing the CDF (Table 6.2, "Potential Plant Enhancements to Reduce Core Damage Frequency" in Ref. 6). These potential improvements are to be evaluated by the licensee after completion of the IPEEE and accident management program.

The staff did not identify any obvious and significant problems or errors in the front-end analysis. The staff's overall assessment of the front-end analysis is that the licensee has made reasonable use of the PRA techniques in performing the front-end analysis, and that these techniques were capable of identifying potential severe accident vulnerabilities.

3.0 Back-End Analysis

The staff examined the back-end analysis which includes containment feature description, containment failure characterization, containment event tree (CET) representation, and radionuclide release. The staff examined the documentation of referenced codes, analytical models and data inputs used.

The staff notes that the back-end analysis is not performed as a separate analysis joined to the front-end analysis through plant damage state binning. Instead the front-end accident sequences are linked to the 989 back-end sequences through the 19 Containment Event Trees (CET) top events using the Riskman software package. Plant damage state binning is accomplished in the form of logic rules that determine split fractions for top events in the CETs. The logic rules also include availability of safety equipment such as the emergency feedwater system. The present CET has evolved from the original CET which consisted of 12 top events and 154 sequences. Additional issues added since the 1983 SSPSA have resulted in a CET that has 19 top events and 989 sequences. New issues incorporated into the CET include direct containment heating (DCH) and induced steam generator tube rupture (ISGTR). Binning is used at the end points of the CETs to define nine distinct release categories. Plant specific containment response analysis was performed using MARCH (Ref. 22), COCOCLASS9 (Ref. 23), MODMESH, and CORCON-MOD1 (Ref. 24) computer codes. 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 valves - isolation signal failure. The dominant contributors to containment by-pass sequences stemmed from Induced Steam Generator Tube Rupture (ISGTR).

The licensee defined "unusually poor" containment performance as all events resulting in early large containment failure (i.e. events in which containment releases are sufficiently large to prevent containment pressurization or result in depressurization of the containment building). The licensee's estimates of the conditional and absolute probabilites of "unusually poor" containment performance are both low (0.002 and 2.1E-7/year respectively). This is mainly the result of an unusually robust containment design. Median containment failure pressures were determined by structural analysis to be 210 psia and 187 psia for dry and wet sequences respectivly. Thus a relativly large conditional probability (0.202) of no containment failure (i.e. intact containment) is estimated. The most likely mode of containment failure is a Type B failure, 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, but the containment does not blow down catastrophically). Type B failure principally involves containment piping penetrations. Failures of elastomer material used as valve seats and seals, personnel and equipment hatch seals, and electrical penetration assembly seals were considered in the evaluation of containment integrity and were found to provide adequate assurance of failure pressures in excess of the failure pressures predicted by the structural analyses, through heat transfer and mass transport analyses and evaluation of maximum leakage areas afforded by clearances between metal to metal contacting surfaces.

The licensee did not find any vulnerabilities to "unusually poor" containment performance. However, the licensee has identified a number of potential procedural and administrative improvements which will be evaluated following completion of the individual plant examination of external events (IPEEE) and the accident management evaluations. 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 1984 SSPSSA has been ammended and augmented to incorporate new methodology, reflect current plant configuration (as of July 1990) and incorporate new phenomonological insights and equipment performance characteristics. Specifically the licensees IPE has adequately incorporated Direct Containment Heating (DCH), Induced Steam Generator Tube Rupture (ISGTR), and hydrogen combustion phenomonolgy.

The staff did not identify any obvious or significant problems or errors in the back-end analysis. The staff's overall assessment of the back-end analysis is that the licensee has made reasonable use of the PRA techniques in performing the back-end analysis, and that these techniques were capable of identifying potential severe accident vulnerabilities.

4. Human Factor Considerations

The licensee's IPE treated human reliability explicitly. 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 used the Human Reliability Analysis (HRA) contained in the original SSPSA. In the original SSPSA, HRA techniques used included operator action trees, qualitative operator-plant status confusion matrix, and results from Seabrook simulator trials. A SLIM-MAUD-like technique (Ref. 25), a method using expert opinion, was used to develop plant-specific human error probability (HEP) estimates based on plant-specific information and performance shaping factors (i.e., time, potential for misdiagnosis, and level of stress).

Three additional human errors were identified and added to the plant logic models for the IPE analysis. These errors are (1) operator provides makeup to the refueling water storage tank (RWST) during a small LOCA, (2) operator recovers engineered safety features actuation system (ESFAS) with long response time (60 minutes), and (3) operator recovers ESAFS during LOCAs. Screening values were assigned to these events for their HEPs. The recovery of electric power model was also updated and used in the IPE. The licensee indicated in the IPE submittal and in discussions with the staff that the next update of the PRA would include a revised HRA.

One simulator trial was used directly in the quantification of human error. It was then used as an anchor point to validate the HEPs derived from other sources. The licensee discussed this anchoring process in its response to the staff's question 22 (Ref. 7). In its response to the staff's question 20 (Ref. 7), the licensee indicated that it had reviewed the EOPs to see if any significant changes had been made to the EOPs which would impact upon the analysis. None were found.

The staff notes that the Seabrook HRA performed in 1983 used state-of-the-art methods, which include qualitative and quantitative techniques, and simulator trials to provide a reasonable analysis. Insights from the HRA of the original SSPSA and follow-on studies have been incorporated into plant procedures since 1983. The licensee uses the simulator to evaluate operator response to plant changes and develop operator training programs. Based on the information contained in the IPE submittal, responses to staff questions, and discussions with the licensee, the staff judges that the HRA process used by the licensee is capable of uncovering severe accident vulnerabilities from human error.

The IPE submittal did not identify sequences that, except for low human error rates in recovery actions, would have been above the screening criteria which follow the guidance from NUREG-1335. The submittal however, does provide a table of the Risk Achievement Worth (RAW) importance measures for identifying important operator actions.

The staff found that the HRA write-up in the SSPSA and the IPE submittal did not provide detailed information on the process used for developing HEPs. The staff recommends that as the licensee develops an accident management program, the basis for these low conditional HEPs be developed and be consistent with plant procedures.

5.0 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 indicated that licensees should focus on hydrogen control during their IPE, particularly on the potential for local hydrogen detonation.

With regard to the combustion and detonation, the licensee has determined that the containment failure probability, as a result of 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 has concluded that there is a negligible probability of containment failure or severe damage as a result of hydrogen "pocketing" and local detonation inside the containment. The licensee bases this conclusion upon the open containment features, i.e. for structural barriers, minimal enclosed spaces and the liberal use of open grating for flooring throughout the containment, and the fact that local hydrogen deflagrations or detonations require conditions of nearly stagnant or quiescent atmospheres which are not considered very likely during accident conditions. The Seabrook licensee's conclusions are similar to those for Indian Point 3 whose containment design closely resembles the Seabrook containment design. Based on the staff's findings, the staff concludes that the licensee's CPI analysis is acceptable.

6.0 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 staff noted that the licensee's examination did not identify any DHR vulnerabilities. The IPE submittal bases its discussion on a 24-hour mission time for DHR following reactor trip. Feed and bleed cooling is conservatively modelled in the IPE as requiring operation of both pressurizer PORVs. Under this assumption, DHR is not a functional vulnerability at Seabrook. Independence of the PORVs from all support systems except DC power provides reliability. Recent analyses are cited in the submittal to indicate that, with some combinations of high head pumps available, only one PORV is needed to provide sufficient cooling. This would further reduce the contribution of DHR function to CDF.

7.0 Licensee Actions and Commitments from IPE

The IPE submittal provides a discussion of potential improvements in risk analysis, operating procedures and plant design. 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 sums 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, and an independent, automatic seal injection pump is shown to provide 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, which were assumed to negatively affect operator actions.

The potential improvements listed in Table 6.1 will be evaluated by the licensee; however, NHY states that it will first update the external events analysis before making a decision to proceed with any plant hardware improvements.

The submittal also provides a discussion of the unique safety features of the Seabrook Station. These are generally the same features associated with modern Westinghouse pressurized water reactor designs. Of some special note are the containment structure, which is large and relatively stronger than most plants in its class, and the bunkers housing the RHR pumps, which provide for probable scrubbing of releases resulting from an interfacing system LOCA in the RHR system. Although presented as a strength by the submittal, the staff notes that the Seabrook emergency feedwater system has only one safety grade motor-driven pump and one safety grade turbine-driven pump. An additional motor-driven startup pump is credited by the analysis as a commercial grade subsystem.

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 includes limiting the use time of containment purge valves and procedures to direct RCS depressurization in order to preclude DCH. The licensee has indicated that these potential procedural and administrative improvements will be evaluated following completion of the IPEEE and the accident management evaluations.

The licensee also indicated to the staff that its SSPSA is a living document and will be updated; however, the licensee did not commit to an update period.

III. CONCLUSION

The licensee's IPE is mainly based on the 1983 SSPSA which had been partially reviewed by BNL. A summary of key data from the IPE submittal is provided in the Appendix to this report. The SSPSA has been updated previously and is to be maintained as a living document with future updates.

Based on the staff's review of the HRA analysis, the staff notes that the HRA write-up in the SSPSA and the IPE submittal did not provide adequate information about the process for developing HEPs. The staff recommends that in the development of an accident program the licensee should develop the bases for the HEPs and ensure they are consistent with plant procedures.

The staff recognizes that the licensee has been actively involved in using the results of PRA analyses to make plant improvements over the years.

Based on the team review of the internal events portion of the licensee's IPE submittal, the staff finds that the licensee has demonstrated an overall appreciation of severe accidents, has an understanding of the most likely severe accident sequences that could occur at the Seabrook facility, has gained a quantitative understanding of core damage and fission product release. The implementation of any safety enhancements will be determined following submittal of an update of the external events portion of the IPE and accident management evaluations. The staff, therefore, finds that the internal events portion of the Seabrook IPE 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-89136, 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.

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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.

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APPENDIX

SEABROOK DATA SUMMARY SHEET*

(INTERNAL AND EXTERNAL EVENTS)

o Total Core Damage Frequency: 1.1E-4/year (mean value)

55% of total is due to internal events

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

	Total	Internal	External
Transients	(83%)	(42%)	(41%)
- LOSP	(40%)	(16%)	(24%)
- Loss of Support Systems	(24%)	(7%)	(17%)
- General Transient	(19%)	(19%)	(0%)
LOCAs	(8%)	(7%)	(1%)
ATWS	(9%)	(6%)	(3%)

o Major systems and contribution to core melt frequency:

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

- 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 control 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

- 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%)

- 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%)

- 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

- 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.)