
Safety Evaluation Report

related to the operation of
**Seabrook Station,
Units 1 and 2**

Docket Nos. 50-443 and 50-444

Public Service Company of New Hampshire, et al.

**U.S. Nuclear Regulatory
Commission**

Office of Nuclear Reactor Regulation

July 1985



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ABSTRACT

This report is Supplement 3 to the Safety Evaluation Report (NUREG-0896, March 1983) for the application filed by the Public Service Company of New Hampshire, et al., for licenses to operate Seabrook Station, Units 1 and 2 (Docket Nos. STN 50-443 and STN 50-444). It has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission and provides recent information on open items identified in the SER. The facility is located in Seabrook, New Hampshire. Subject to favorable resolution of the items discussed in this report, the staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public.

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1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

On March 7, 1983, the Nuclear Regulatory Commission staff (NRC or staff) issued a Safety Evaluation Report (SER), NUREG-0896, on the application of Public Service Company of New Hampshire (PSNH, hereinafter referred to as the applicant) for licenses to operate Seabrook Station, Units 1 and 2. In April 1983, the NRC issued the first supplement (SSER 1) and in June 1983, the second supplement (SSER 2) was issued to that document. This is the third supplement to that SER (SSER 3) that provides information to update the status of the NRC review.

Each of the sections and appendices of this supplement is designated the same as the related portion of the SER. The contents of this document are supplementary to the initial SER, SSER 1, and SSER 2, and not in lieu of those documents unless otherwise noted. The NRC Project Manager for the Seabrook operating license review is Mr. Victor Nerses. He may be reached by telephone at (301) 492-7238 or by mail at the following address:

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1.7 Outstanding Issues

Section 1.7 of the SER noted that certain outstanding issues in the staff's review had not been resolved by the time the report was issued. This supplement closes six of those items. These items, and the sections of this supplement that present results of the staff's evaluation, are

- (3) Reassessment of assumptions for safe shutdown earthquake design response spectra (2.5.2.6)
- (5) Loading combinations, design transients, and stress limits (3.9.3.1)
- (7) Flow measurement uncertainties (4.4.5.2)
- (8) TMI Action Plan items (12.3, in Section 1.7 of the SER this is listed as 12.2 but this was a typographical error and noted in Appendix H of SSER 1; 15.9.5)
- (15) Service water monitoring system (11.5.2)
- (18) Outstanding QA issues (17.5, in Section 1.7 of the SER, this is listed as Q-List (17))

As of this supplement, the remaining outstanding issues are:

- (2) Emergency preparedness (2.3.3, 13.3)
- (4) Preservice and inservice inspection and testing programs (2.5.5.3, 3.9.6, 5.2.4, 5.4.12, 6.6.1)
- (6) Stresses/dynamic and environmental qualification of equipment (3.10, 3.11)
- (8) TMI Action Plan items (3.9.3.2, 4.4.5.4, 6.2.8, 9.3.4.2, 13.3, 13.5.1, 14, 15.3.5, 15.9.7, 15.9.9)
- (9) Fracture toughness of secondary system materials (10.3.6)
- (10) Level measurement error (7.3.2.8)
- (11) Instrumentation and control for safe shutdown (7.4.2.1, 7.4.2.4)
- (12) Radiation data management system (7.5.2.2)
- (14) Solid radwaste management system (11.4)
- (16) Shift technical advisor (I.A.1.1) (13.1.2.2, 13.2.2.1)
- (17) Steam generator tube rupture (15.6.3)
- (19) Control room design review (18)

1.8 Confirmatory Issues

Section 1.8 of the SER noted that there are some items that have been resolved essentially to the staff's satisfaction but for which certain confirmatory information has not yet been provided by the applicant. This supplement closes eight of the confirmatory items. These items, and the sections of this supplement that present results of the staff's evaluation, follow:

- (2) Scour protection of plant grade (2.4.10)
- (3) Cooling tower performance (2.4.11.2)
- (4) Evaluation of data from recent earthquake (2.5.2.6)
- (8) ASME Code case for Section III, Class 1, reactor coolant system components (5.2.1.2)
- (34) Ventilation of radioactive gaseous waste system (RGWS) cubicles (11.3)
- (35) Conformance of process and effluent monitors with Appendix 11.5A and Regulatory Guide (RG) 4.15 (11.5)
- (36) Airborne radioactivity monitoring (12.3.4.2)
- (39) Emergency operating procedures (13.5.2.3)

As of this supplement, the remaining confirmatory items are:

- (1) Underground transmission line easement (2.1.2)
- (5) Staff review of applicant response to IE Bulletin 79-02 (3.9.3.3)
- (6) Loose parts monitoring system (4.4.5.3)
- (7) Conformance of reactor internals and control rod drive mechanism materials aging and tempering temperatures to staff guidelines (4.5.2)
- (9) Staff review of LOFTRAN computer code (5.2.2.1)
- (10) Conformance with RG 1.36 for compatibility of thermal insulation with RCPB and ESF materials (5.2.3.1 and 6.1)
- (11) Confirmation of maximum yield strength of RCPB materials (5.2.3)
- (12) Analysis of the containment purge and vent system (6.2.4, 6.2.8)
- (13) Containment subcompartment analyses (6.2.1.2)
- (14) Formal documentation of previously provided information related to several instrumentation and control systems (7.2.2, 7.3.2, 7.4.2.5, 7.5.2.1, 7.6.7, 7.7.2)
- (15) Test of engineered safeguards P-4 interlock (7.3.2.3)
- (16) Main steam atmospheric relief valves (7.4.2.2)
- (17) Pressurizer auxiliary spray (7.4.2.5)
- (18) RCS pressure control during low temperature operations (7.6.7.2)
- (19) RHR system (7.6.7.5, 7.6.7.7)
- (20) Tower actuation signal (7.6.7.8)
- (21) Routing of offsite power circuits (8.2.2.3)
- (22) Compliance with BTP PSB-1 (8.3.1.1.1, 8.3.1.1.3, 8.3.1.1.4)
- (23) Compliance with RG 1.9 (8.3.1.2)
- (24) Nonsafety loads powered from the Class 1E ac system (8.3.1.4)
- (25) Automatic load transfers between redundant divisions (8.3.1.8)
- (26) Battery supports for onsite dc systems (8.3.2.2)
- (27) Compliance of non-Class 1E circuits to Class 1E requirements (8.3.3.3.1)

- (28) DC nonsafety loads (8.3.2.4)
- (29) Compliance with RG 1.63 (8.3.3.6.3)
- (30) Diesel generator control panel mounts (9.5.4.1)
- (31) Diesel generator exhaust inspection and protection (9.5.8)
- (32) Conformance to GDC 35 for fracture toughness of main steam and feedwater materials (10.3.6)
- (33) Sampling capability for vacuum pumps during startup (10.4.2)
- (37) Health physics organization (12.5.1)
- (38) Experience level for the ISEG (13.4.1)
- (40) Inadvertent boron dilution (15.4.6)
- (41) Systems outside containment containing radioactive material (III.D.1.1)
(15.9.2)

2 SITE CHARACTERISTICS

2.4 Hydrologic Engineering

2.4.10 Flood Protection Requirements

In the SER, the staff concluded that the shore protection features for the Seabrook Station would survive the design-basis flood and that the plant's design flood protection met the guidelines of Regulatory Guide (RG) 1.102 and the requirements of General Design Criterion (GDC) 2, except as noted regarding scour protection behind the vertical seawall and adjacent to electrical manholes #13/14 and #15/16. The scour protection remained as a confirmatory item (Confirmatory Item 2).

Subsequent to the SER, the applicant has revised the shore protection features on the basis of changes in site conditions that occurred during the construction stage. During construction the land area on the east side of revetment B was graded to about elevation (el) 13 ft mean sea level (msl). The original design for revetment B was based on the revetment with a 1 on 2 slope extending to the original marsh elevation. On the basis of these changed conditions, the applicant has determined that the higher ground on the harbor side of revetment B provides significant protection. The changed conditions result in a reduction in the design waves that can impact revetment B during the design-basis hydrologic event (probably maximum hurricane). As a result, the applicant has redesigned stone revetment B from a 1 on 2 slope to a structure with both a 1 on 3 slope and a nominal 1 on 2.5 slope and reduced the weight of armor stone required for the revetment consistent with the reduced design wave height (see revised Figure 2.7).

In addition, the applicant has addressed the confirmatory issue involving the requirement for scour protection to the plant grade area behind the vertical concrete seawall adjacent to electrical manholes #13/14 and #15/16. This area would be subjected to significant wave overtopping and wave scour of plant grade if not protected during the design-basis flooding event. The applicant has stated that asphalt paving, 6 in. thick, will be provided for scour protection (see revised Figure 2.7).

On the basis of its analysis of the changed site conditions, the redesign of revetment B, and the asphalt scour protection, the staff concludes that these shore protection features would survive the design-basis flood. On the basis of its analysis using Sections 2.4.2, 2.4.5, and 2.4.10 of the Standard Review Plan (SRP, NUREG-0800), the staff concludes that the applicant has shown that the plant design-basis flood protection meets the guidelines of RG 1.102 and thus the requirements of GDC 2.

2.4.11 Cooling Water Supply

2.4.11.2 Adequacy of Cooling Water Supply

The adequacy of the backup emergency cooling tower to provide water at or below the 90°F design-basis temperature was left as Confirmatory Item 3 in the SER. The staff and its consultant, Argonne National Laboratory (ANL), have subsequently performed analyses that determined that the design basis will be essentially met.

The staff's consultant reviewed the applicant's detailed analysis of cooling tower performance, which included proprietary information on the mechanical draft cooling tower and tower fill design. The consultant then performed an independent analysis of several possible scenarios using its own mathematical model of mechanical draft cooling tower performance which incorporated data from the tower manufacturer, projected design-basis heat loads, and adverse meteorology of the severity suggested in RG 1.27. The consultant's model agreed well with the manufacturer's performance data. The model predicted a maximum instantaneous return service water temperature of 90.2°F, which essentially agrees with the applicant's analysis and meets the design basis of 90°F within the limits of modeling accuracy.

Thus, on the basis of its review of the Final Safety Analysis Report (FSAR) according to SRP Section 2.4.11, additional data obtained from the applicant, and an independent analysis of mechanical draft tower performance by ANL, the staff concludes that the mechanical draft cooling tower can adequately provide return service water temperatures at or below design limits. Therefore, the staff concludes that the ultimate heat sink design meets the suggested criteria of RG 1.27 and the hydrologic aspects of GDC 44. Confirmatory Item 3 has been resolved.

2.5 Geology and Seismology

2.5.2 Vibratory Ground Motion

2.5.2.6 Safe Shutdown Earthquake

2.5.2.6.4 High-Frequency Ground Motion

The SER indicated one outstanding seismology issue (Outstanding Issue 3): The applicant should reassess the assumptions for safe shutdown earthquake (SSE) design response spectra in light of new ground motion information from earthquakes in New Brunswick and Gaza, New Hampshire.

As has been reported by Chang (NUREG/CR-3327), Cranswick et al. (1982), and Weichert et al. (1982), strong ground motion from the Gaza, New Hampshire, earthquake of January 18, 1982 (body wave magnitude (m_b) = 4.8) and aftershocks of the New Brunswick earthquake of January 9, 1982 (m_b up to 4.8) appear to indicate that the recorded motion was enriched in high frequencies (above about 10 Hz) when compared with typical recordings from California. This observation raises the question of whether earthquake sources in the eastern United States differ from those in the western United States. Since the relative lack of data from eastern U.S. earthquakes has resulted in reliance on western strong motion

records (such as those used to develop RG 1.60 design spectra), the adequacy of ground motion estimates at high frequencies should be examined.

The applicant addressed this issue in discussions with the staff, responses to Requests for Additional Information (RAIs) 230.7 and 230.8, and a report by Weston Geophysical Corporation (1983). The applicant argues that, for the New Hampshire event, high accelerations (0.15 to 0.5g) recorded at the Franklin Falls Dam sites were recorded in the very near field, possibly very close to the rupture. Furthermore, none of these instrument sites were founded on bedrock; thus, the recording conditions are not comparable to the Seabrook rock site. Similarly, all but one of the recording sites of the 1982 New Brunswick aftershocks were an alluvium. Furthermore, the applicant questioned the validity of instrument corrections at the high frequencies where the peak accelerations were recorded (18 to 47 Hz). In fact, response spectra of strong motion accelerograph (SMA)-1 records are routinely cut off at 25 Hz by the U.S. Geological Survey (USGS) to avoid presenting unreliably corrected data.

Considerable effort has been and is being expended, including specific NRC Office of Nuclear Regulatory Research contracts, in an attempt to determine the cause of the high-frequency energy. It is apparent that this issue is extremely complex, and it is likely to be some time before the cause of the recorded high frequencies is fully understood. The following examples illustrate some of the parameters that complicate the issue. All but one of the strong motion instruments in New Brunswick which recorded aftershock data in 1982 were located in shallow soil sites, and thus, site effects from resonance in the soil column may be one reason some of the high frequencies were recorded. Different types of strong motion instrumentation have also been used at different times, each having different frequency-response and high-frequency limitations. This makes it extremely difficult to draw conclusions regarding high frequencies in general, and in particular magnitudes above about 5.0, the size of the events which are of concern in the eastern United States.

As reported by Mueller and Cranswick (1982), Boatwright and Astrue (1983), and Haar et al. (1984) for the New Brunswick aftershock data recorded in 1982, there do not appear to be systematic source differences between the New Brunswick data and data recorded in California. Preliminary results of work undertaken in 1983 (Cranswick, 1984) appear to show that stress drops for very small magnitude (less than about 2.0) events decrease as the magnitude decreases and that the site response has an effect on the high frequency content of the recorded ground motion. These examples demonstrate the complexity of this issue. It is the staff's current judgment that little evidence exists to support systematic source differences (although this issue is still under investigation) and that the high-frequency issue may have generic implications. The short-duration, high-frequency ground motion associated with the highest peak accelerations are similar to those recorded from small nearby earthquakes at other locations that also have produced little or no damage. The NRC staff will continue to monitor and support generic work on this topic and to monitor any impact this work has on the design basis of power plant sites. Regarding the Seabrook site, it is the staff's judgment that the most appropriate and applicable data on strong motion are the accelerograph data used to develop the rock site-specific spectra discussed in SER Section 2.5.2.6.2.1.

The SER indicated one confirmatory seismology issue (Confirmatory Issue 4), evaluation of data from recent earthquakes. First, the applicant was asked to

assess and describe any implications that the occurrence of the New Brunswick earthquake may have for the choice of the maximum historical (1755 Cape Ann) earthquake as the controlling earthquake for seismic design. Second, the applicant was asked to evaluate data resulting from the New Hampshire and New Brunswick earthquakes with respect to the assumed relationship between vertical and horizontal ground motion at frequencies greater than 33 Hz.

Maximum Earthquake

In reviews performed since about 1976 for nuclear power plants located in New England and the northernmost Piedmont, the NRC staff has recognized the New England-Piedmont Tectonic Province, in which the maximum historical earthquake has been characterized by a maximum Modified Mercalli intensity (MMI) of VII or a magnitude of about 5.3. On January 9, 1982, a magnitude 5.7 earthquake occurred in central New Brunswick, Canada, in geologic terrain that is similar to that which composes the New England-Piedmont Tectonic Province. In accordance with Appendix A to 10 CFR 100, this may require that an earthquake similar to the 1982 New Brunswick earthquake be assumed to occur elsewhere in the New England-Piedmont Tectonic Province.

Analysis of the aftershock sequence of the January 1982 main event suggests the presence of a specific structure. Wetmiller et al. (1984) describe the aftershock activity following the New Brunswick earthquake of January 9, 1982. The aftershock activity was arranged in a north-trending conjugate "V" pattern. The earthquakes were predominantly thrust earthquakes occurring on north-trending faults, indicating an east-west compressive regional stress field. The New Brunswick earthquakes did not produce any significant surface faulting and are not associated with any known preexisting fault. The applicant argues that the 1982 New Brunswick earthquakes are associated with a seismogenic structure and that focal mechanisms are consistent with this interpretation. Also, the applicant argues that the 1982 events are not out of character with the region's moderate level of seismicity.

The staff has reviewed seismicity in central New Brunswick before the January 1982 earthquake. The earthquakes in central New Brunswick are seen as part of a broad scattered pattern that extends throughout New Brunswick, Maine, and southern New Hampshire. Before the January 1982 event, earthquakes in 1855 and 1922 that were not located instrumentally, with maximum MMI V, may have occurred near the epicenter of the 1982 event, with magnitudes of at least 5.0 (Stevens, 1982). The historical and instrumental record results in earthquake location uncertainty and a relatively high detection threshold because of sparse population and poor instrumental coverage before 1975. An event of magnitude at least 5 is not unexpected however.

The Seabrook site lies in the New Hampshire-Cape Ann seismic zone, which is part of the New England-Piedmont Tectonic Province. In past reviews of New England sites, the staff concluded that there was a spatial relationship between the zone defined by the White Mountain Plutonic series, which represent the youngest significant deformation features in New England, and historic seismicity. The largest New England earthquakes have occurred within this zone (the 1727 Newbury event (intensity VII), the 1755 Cape Ann event (intensity VIII), the 1817 Woburn event (intensity VI), and two intensity VII events in 1940 at Lake Ossipee, New Hampshire). The 1755 Cape Ann event is the maximum historical earthquake within the New Hampshire-Cape Ann seismic zone. This event, which

caused intensity VIII shaking, had a large body wave magnitude (m_{blg}) of approximately 6.0 (Street and Lacroix, 1979).

The conjugate faults defined by the New Brunswick aftershocks might reflect the existence of faults that could localize seismicity in central New Brunswick. However, similar faults could exist in other areas of eastern Canada or the eastern United States.

The seismicity varies throughout the New England-Piedmont Tectonic Province, with regions of relatively higher and lower rates of earthquake activity. Large earthquakes have usually occurred in areas of high seismicity. Thus, it is reasonable to assume that future large earthquakes are more likely to happen in areas of past and present high seismicity (NUREG-1048). The Seabrook site has already been defined as one of these areas of relatively higher seismicity. The staff still considers a magnitude of 6.0 event, similar to the 1755 Cape Ann event, to be the appropriate controlling earthquake in determining the SSE. The staff find the occurrence of the magnitude 5.7 New Brunswick event does not impact the controlling earthquake for determining the SSE at Seabrook.

Vertical Ground Motion

The vertical-component design response spectra for Seabrook deviate from the recommendations of RG 1.60 for frequencies greater than 33 Hz. The vertical-component spectral levels are reduced from the horizontal-component levels by a factor of 1-to-2/3 for frequencies between 33 and 50 Hz and by a factor of 2/3 for frequencies greater than 50 Hz. The staff asked the applicant to analyze strong-motion data from the New Hampshire and New Brunswick earthquakes to confirm the appropriateness of the assumed relationship between vertical and horizontal ground motion at frequencies greater than 33 Hz. The applicant, in responding to RAI 230.8, concludes that the New Hampshire and New Brunswick data sets do not offer a sufficient and reliable basis to study the appropriateness of the vertical to horizontal ratio beyond 33 Hz. The applicant questioned the validity of instrument corrections at high frequencies.

In examining the New Hampshire and New Brunswick data at frequencies up to 33 Hz, no consistent trend in the vertical to horizontal ratio is evident and the scatter in results at a given frequency is at least an order of magnitude. Although Newmark and Hall (NUREG/CR-0098) recommended the use of 2/3 for all frequencies, they recognized that large scatter exists in this ratio. Studies of western U.S. earthquakes (e.g., NUREG/CR-1175; Agbabian, 1983) have shown that the assumption of the vertical taken as 2/3 of the horizontal is generally conservative. The staff has also examined peak acceleration data included in site-specific spectra and found that the 2/3 assumption is a good average, although individual values show wide scatter.

Considering the above, the staff concludes that the vertical design spectrum at Seabrook is adequate. Since the current eastern U.S. strong motion data currently are undergoing analysis, the staff will continue to monitor ongoing generic work and any new data that become available and will assess these results with respect to the ratio of vertical to horizontal ground motion.

Conclusion

The staff concludes that there are no remaining outstanding or confirmatory seismologic issues. Outstanding Issue 3 and Confirmatory Issue 4 are resolved.

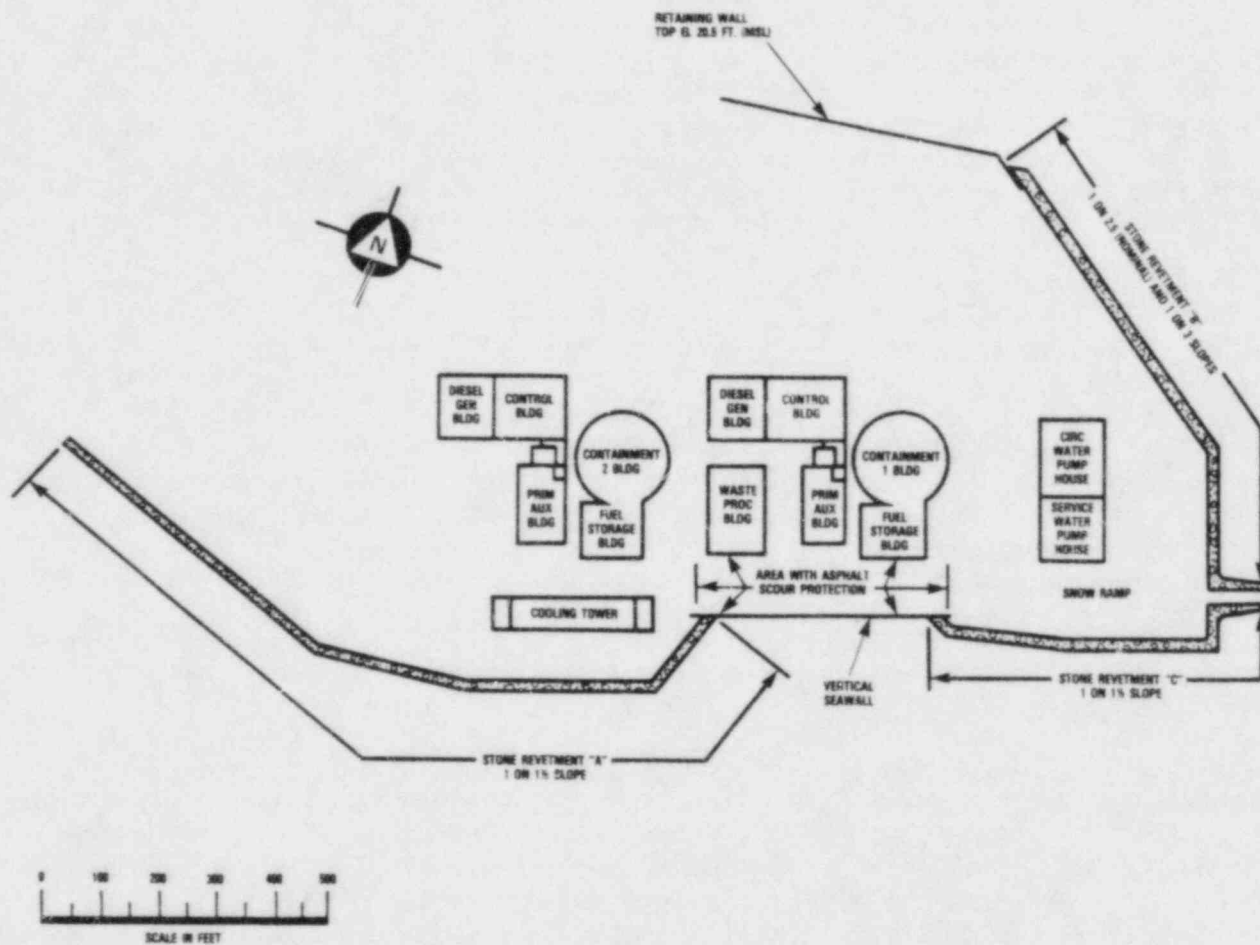


Figure 2.7 (revised) Wave protection stone revetment and seawall
Source: FSAR Figure 2.4-20

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.5 Missile Protection

3.5.2 Structures, Systems and Components To Be Protected From Externally Generated Missiles

Nuclear power plants must be designed to withstand the effects of tornado- and high-wind-generated missiles so that unacceptable damage cannot occur and thereby prevent impact on the health and safety of the public in accordance with the requirements of GDC 2 and 4. The current licensing criteria governing tornado missile protection are contained in SRP Sections 3.5.1.4 and 3.5.2. These criteria specify that safety-related systems are to be provided positive tornado missile protection (i.e., barriers) from the maximum credible tornado threat. However, SRP Section 3.5.1.4 includes guidance on the use of probabilistic risk assessment (PRA) methodology in lieu of a deterministic approach for assessing tornado missile protection. The acceptance criterion in this regard is similar to that identified in SRP Section 2.2.3, which deals with the identification of design-basis events using probabilistic methods. The tornado missile acceptance criterion is as follows: "The probability of significant damage to structure, systems and components required to prevent a release of radioactivity in excess of 10 CFR Part 100 following a missile strike, assuming loss of offsite power, shall be less than or equal to a median value of 10^{-7} or a mean value of 10^{-6} per year."

On November 10, 1982, the applicant for the Seabrook Station reported a potential 10 CFR 50.55(e) deficiency regarding the adequacy of tornado missile shield protection on the north and west walls of the fuel storage building above elevation 64 ft 0 in. where ductwork penetrates the walls. Ductwork barriers are not installed, and a clear line of sight is available through openings in the walls. Under the present configurations, a tornado missile could enter the fuel storage building through the north wall opening and enter the spent fuel pool area through an opening in the floor slabs at elevation 64 ft 0 in. The missile could damage spent fuel or the pool cooling system. FSAR Section 3.5.2 currently states that the spent fuel cooling/cleanup system and storage pool is protected against tornado missiles.

The applicant's evaluation of the safety implications of this concern resulted in the initiation of a detailed review of all exterior openings in seismic Category I (safety-related) buildings. A total of 30 openings were identified as potential tornado missile targets. These targets include vents, louvers, heating, ventilation, and air conditioning exhaust and intake openings, and doorways that have a potential for missile entrance into the seismic Category I buildings. These openings are categorized by the applicant into three groups:

- (1) openings protected by missile shields, but the adequacy of those shields was questionable because of the area of coverage

- (2) openings vulnerable to sudden depressurization caused by damage by a missile hit
- (3) openings identified during design review that do not have missile shields

The applicant elected to demonstrate compliance with the tornado missile protection criterion by PRA methodology rather than provide positive protection for the openings in the seismic Category I buildings. The applicant provided a detailed probability analysis report entitled "Seabrook Nuclear Power Plant, Tornado Missile Analysis, Final Report C569" prepared by Applied Research Associates, Inc., dated September 1983. Additional information to support the analysis was provided in a submittal dated April 18, 1985.

Because of the specialized nature of the study, the staff contracted with the National Bureau of Standards (NBS) to assist in the review of the applicant's PRA analysis. NBS provided a technical evaluation report (TER) regarding the probability of a tornado missile strike for the identified openings in seismic Category I buildings. Concerns that were identified during the staff's review were satisfactorily resolved by the applicant in the April 18, 1985, submittal. The consultant's TER forms a part of the staff's supplemental safety evaluation report (Appendix J).

The staff has reviewed the consultant's TER and agrees with its contents. The consultant's evaluation of the applicant's PRA considered the validity and the degree of conservatism in the assumptions, data, and mathematical approach used in the applicant's analysis to estimate the probability of tornado and hurricane-borne missile damage to a specified set of targets at the Seabrook Station. Also included in the evaluation is an assessment of the correctness of the results obtained in the study. Numerical results obtained by the analysis reviewed for the 30 openings in seismic Category I buildings are listed in Tables V-1, V-2, and V-3 of the applicant's report. The probability of missile entrance into the identified openings, where such an event could lead to unacceptable damage (potential release of radioactivity), is estimated by the applicant to be about 1.2×10^{-6} per year. Inherent in this estimate are the following conservative assumptions: (1) The number of potential missiles present on the site during construction of Unit 2 is not reduced when Unit 2 is completed, (2) the tornado windspeeds corresponding to the upper limit (F-scale) are 360 mph, and (3) the estimated probabilities are for missile entrance into the opening and do not include impact and damage probabilities for safety-related equipment or spent fuel. If such calculations were carried out in a more detailed analysis, it is likely that there would be no opening that would lead to a safety-related component missile strike probability greater than about 10^{-8} per year. On the basis of its review, the staff concludes that the probability of unacceptable missile damage indicated in the applicant's analysis is conservative and that the true probability is likely to fall within the range of 10^{-6} to 10^{-7} per year, thereby satisfying the SRP guidelines for tornado missile failure probability. Therefore, the applicant's tornado missile PRA is acceptable.

On the basis of its review of the applicant's tornado missile PRA, the staff concludes that the applicant's analysis is reasonable and utilizes the present state of the art. The staff further concludes that the probability of unacceptable tornado missile damage satisfies its numerical acceptance criterion, 10^{-6} to 10^{-7} per year. Therefore, the applicant has satisfactorily demonstrated compliance with the requirements of GDC 2 and 4 regarding protection of

safety-related plant equipment and spent fuel from the effects of tornado- and high-wind-generated missiles, through the use of probabilistic risk assessment analysis, and the plant design is, therefore, acceptable.

3.6 Protection Against Dynamic Effects Associated With the Postulated Rupture of Piping

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated With the Postulated Rupture of Piping

3.6.2.1 Introduction

In the "Background" to Branch Technical Position (BTP) MEB 3-1 in SRP Section 3.6.2 (NUREG-0800, Rev. 1), the staff position on pipe break postulation is that pipe rupture is a rare event that may only occur under unanticipated conditions such as those that might be caused by possible design, construction, or operation errors, unanticipated loads, or unanticipated corrosive environments. The BTP MEB 3-1 pipe break criteria were intended to utilize a technically practical approach to ensure that an adequate level of protection had been provided to satisfy the requirements of GDC 4. Specific guidelines were developed in BTP MEB 3-1 to define explicitly how the requirements of GDC 4 were to be implemented. The guidelines in BTP MEB 3-1 were not intended to be absolute requirements but rather represent viable approaches considered acceptable by the staff.

The SRP provides a well-defined basis for performing safety reviews of light water reactors. The uniform implementation of design guidelines in BTP MEB 3-1 ensures that a consistent level of safety will be maintained during the licensing process. Alternative criteria and deviations from the SRP are acceptable provided an equivalent level of safety can be demonstrated. Acceptable reasons for deviations from SRP guidelines include changes in emphasis of specific guidelines as a result of new developments from operating experience or plant-unique design features not considered when the SRP guidelines were developed.

The SRP presents the most definitive basis available for specifying NRC's design criteria and design guidelines for an acceptable level of safety for reviews of light water reactor facilities. The SRP guidelines resulted from many years of experience gained by the staff in establishing and using regulatory requirements in the safety evaluation of nuclear facilities. The SRP is part of a continuing regulatory standards development activity that not only documents current methods of review, but also provides a basis for an orderly modification of the review process when the need arises to clarify the content, correct any errors, or modify the guidelines as a result of technical advancements or an accumulation of operating experience. Proposals to modify the guidelines in the SRP are considered for their impact on matters of major safety significance.

The staff has recently received requests (letters dated February 7, February 19, and April 12, 1985) from the applicant for Seabrook Unit 1 to consider an alternate approach to the existing guidelines in SRP Section 3.6.2, BTP MEB 3-1, regarding the postulation of intermediate pipe breaks. For all high energy piping systems at Seabrook, the applicant proposes to eliminate from design considerations those breaks generally referred to as "arbitrary intermediate breaks" (AIBs), which are defined as those break locations that, on the basis of piping stress analysis results, are below the stress and fatigue limits specified in BTP MEB 3-1 but are selected to provide a minimum of two postulated breaks

between the terminal ends of a piping system. The applicant has documented the cost savings and reduced radiation exposure benefits resulting from the elimination of the structures associated with the protection against the effects of pipe rupture. The applicant has further stated that all dynamic effects associated with previously postulated AIBs will be excluded from the plant design basis and that pipe whip restraints and jet shields associated with previously postulated AIBs will be eliminated. However, the applicant has stated that the elimination of AIBs will not downgrade the environmental qualification levels of Class 1E equipment. The break postulation for environmental effects is performed independently of the break postulation for pipe whip and jet impingement.

In the early 1970s when the pipe break criteria in BTP MEB 3-1 were first drafted, the advantages of maintaining low stress and usage factor limits were clearly recognized, but it was also believed that equipment in close proximity to the piping throughout its run might not be adequately designed for the environmental consequences of a postulated pipe break if the break postulation proceeded on a purely mechanistic basis using only high stress and terminal end breaks. As the pipe break criteria were implemented by the industry, the impact of these criteria on plant reliability and costs as well as on plant safety became apparent. Although the overall criteria in BTP MEB 3-1 have resulted in a viable method that ensures that adequate protection has been provided to satisfy the requirements of GDC 4, it has become apparent that the particular criterion requiring the postulation of AIBs can be overly restrictive and may result in an excessive number of pipe rupture protection devices that do not provide a compensating level of safety.

At the time the BTP MEB 3-1 criteria were first drafted, high energy leakage cracks were not being postulated. In NUREG-0800, the concept of using high energy leakage cracks to mechanistically achieve the environment desired for equipment qualification was introduced to cover areas that are below the high stress/fatigue limit break criteria and that would otherwise not be enveloped by a postulated break in a high energy line. In the proposed elimination of AIBs, the staff believes that the essential design requirement of equipment qualification is not only being retained but is being improved because all safety-related equipment is to be qualified environmentally; furthermore, certain elements of construction that may lead to reduced reliability are being eliminated.

In addition, some requirements that have been developed over the years as part of the licensing process have resulted in additional safety margins that overlap the safety margin provided in the pipe break criteria. For example, the criteria in BTP MEB 3-1 include margins to account for the possibility of flaws that might remain undetected in construction and to account for unanticipated piping steady-state vibratory loadings not readily determined in the design process. However, inservice inspection requirements for the life of the plant to detect flaws before they become critical and staff positions on the vibration monitoring of safety-related and high energy piping systems during preoperational testing further reduce the potential for pipe failures occurring from these causes.

Because of the recent interest by the industry to eliminate the AIB criteria and, particularly, in response to the detailed submittals provided by several utilities other than PSNH, the staff has reviewed the BTP MEB 3-1 pipe break criteria and approved deviations from the criteria for Catawba Unit 2 (Novak, April 2, 1984) and Vogtle Units 1 and 2 (Novak, June 28, 1984).

3.6.2.2 Applicant's Bases for the Elimination of Arbitrary Intermediate Pipe Breaks

In letters dated February 7, February 19, and April 12, 1985, the applicant forwarded a request for the elimination of AIBs and the technical bases for the proposal. The applicant's submittals suggest a consensus in the nuclear industry that current knowledge and experience support the conclusion that designing for the AIBs is not justified. The reasons given for this conclusion are discussed in the following paragraphs.

(1) Operating Experience Does Not Support Need for Criteria

The applicant states that the combined operating history of commercial nuclear plants (extensive operating experience in over 80 operating U.S. plants and a number of similar plants overseas)* has not shown the need to provide protection from the dynamic effects of AIBs.

(2) Piping Stresses Well Below Allowable Limits in the American Society of Mechanical Boiler and Pressure Vessel Code (ASME Code)

Currently, AIBs are postulated to provide a minimum of two pipe breaks at the two highest stress locations between piping terminal ends. Consequently, AIBs are postulated at locations in the piping system where pipe stresses and/or cumulative usage factors are well below ASME Code allowable limits. Such postulation necessitates the installation and maintenance of complicated mitigating devices to afford protection from dynamic effects such as pipe whip and/or jet impingement. When these selected break locations have stress levels only slightly greater than the rest of the system, installation of mitigating devices not only lends little to enhance overall plant safety but also provides the potential for inadvertent thermal restraint of piping.

(3) Arbitrary Intermediate Breaks Complicate the Design Process

The applicant states that the design of piping systems is an iterative process; therefore, the location of the highest stress points usually changes several times during design. Although SRP Section 3.6.2 (NUREG-0800) provides criteria intended to reduce the need to relocate the intermediate break locations when high stress points shift as a result of piping reanalysis, in practice, these criteria provide little relief from moving arbitrary break locations because the revised break locations must still be evaluated as to their effects on essential equipment and structures.

(4) Substantial Cost Savings

The cost benefits to be realized from the elimination of the AIB locations center primarily on the elimination of the associated pipe whip restraints and jet shields. Although a substantial reduction in capital and engineering costs for these restraints and structures can be realized in the design and construction stages of the plant, there are also significant operational benefits to be realized over the 40-year life of the plant, as reduced man-hours for inservice inspection and maintenance will result.

*The information within parentheses is provided by the NRC staff and not by the applicant.

(5) Improved Inservice Inspection

Pipe whip restraints are normally located adjacent to or surrounding the welds at changes in pipe direction. Access during plant operation for inservice inspection activities can be improved because of the elimination of congestion created by these pipe rupture protection devices and the supporting structural framing associated with arbitrary pipe breaks.

(6) Reduction in Radiation Exposure

In the event of a radioactive release or spill inside the plant, decontamination operations could be more effective if the pipe whip restraints and jet shields associated with AIBs and the large structural frameworks supporting the restraints were eliminated. Recovery from unusual plant conditions would also be improved by reducing the congestion in the plant. A significant reduction in man-rem exposure can be realized through fewer man-hours spent in radiation areas.

The applicant, as part of his justification for the elimination of AIBs (letters dated February 7 and April 12, 1985), has estimated that the reduction in operational radiation exposure caused by the elimination of AIBs and the resulting decrease in pipe whip restraints and jet deflectors over the 40-year life of the plant will be about 80-160 person-rem.

(7) Improved Operational Efficiency

The elimination of pipe whip restraints associated with arbitrary breaks will preclude the requirement for cutback insulation or special insulating assemblies near the close fitting restraints. This will reduce the heat loss to the surrounding environment, especially inside containment.

3.6.2.3 Staff Evaluation of the Bases for the Elimination of Arbitrary Intermediate Pipe Breaks

The technical bases for the elimination of the AIB criteria as discussed in the preceding section provided many arguments supporting the applicant's conclusion that the current SRP guidelines on this subject should be changed. However, it is not apparent that a unilateral position by the utility concluding in an unconditional deletion of the AIB criteria can be justified without a clear understanding of the safety implications that may result for the various classes of high energy piping systems involved. In this section, the staff will discuss the bases behind the current AIB criteria from an ASME Code design standpoint and put into perspective the uncertainty factors on which the need to postulate AIBs should be evaluated. The staff further evaluates the acceptability of the applicant's proposed deviation from SRP Section 3.6.2. In BTP MEB 3-1, the staff recognizes that pipe rupture is a rare event that may only occur under unanticipated conditions such as those that might be caused by possible design, construction, or operation errors, unanticipated loads, and unanticipated corrosive environments. Furthermore, the staff recognizes that, on those rare occasions when piping failure does occur, the failure is expected to occur at locations of high stress and fatigue such as at terminal ends of piping systems and at local welded attachments to the piping wall. This generalization does not cover situations in which stress corrosion cracking is prevalent. Thus, the staff believes that pipe breaks should be postulated at locations where a

relatively higher potential for failure exists to ensure a practical level of protection. The preceding staff positions are not new.

The SRP guideline that requires that two intermediate breaks be postulated even when the piping stress is low resulted from the need to ensure that equipment qualified for the environmental consequences of a postulated pipe break was provided over a greater portion of the high energy piping run. The staff now proposes to dispense with AIBs on the condition that all equipment in the spaces traversed by the fluid system lines, for which AIBs are being eliminated, is qualified for the environmental (nondynamic) conditions that would result from a nonmechanistic break with the greatest consequences on surrounding equipment.

ASME Code, Class 1 Piping Systems

In accordance with BTP MEB 3-1 (Paragraph B.1.c.(1)), breaks in ASME Code, Class 1 piping should be postulated at the following locations in each piping and branch run

- (1) at terminal ends
- (2) at intermediate locations where the maximum stress range as calculated by Equation (10) and either Equation (12) or (13) of ASME Code NB-3650 exceeds $2.4S_m$
- (3) at intermediate locations where the cumulative usage factor exceeds 0.1
- (4) if two intermediate locations cannot be determined by (2) and (3) above, the two highest stress locations based on Equation (10)

The AIB criteria are stated in Item (4) above. It should be noted that the request for alternative criteria does not propose to deviate from the criteria in (1), (2), and (3) above. Pipe breaks will continue to be postulated at terminal ends irrespective of the piping stresses.

Pipe breaks are to be postulated at intermediate locations where the maximum stress range as calculated by Equation (10) and either Equation (12) or (13) exceeds $2.4S_m$. The stress evaluation in Equation (10) represents a check of the primary plus secondary stress intensity range resulting from ranges of pressure, moments, thermal gradients, and combinations thereof. Equation (12) is intended to prevent formation of plastic hinges in the piping system caused only by moments resulting from thermal expansion and thermal anchor movements. Equation (13) represents a limitation for primary plus secondary membrane plus bending stress intensity excluding thermal bending and thermal expansion stresses; this limitation is intended to ensure that the K_e factor (strain concentration factor) is conservative. The K_e factor was developed to compensate for absence of elastic shakedown when primary plus secondary stresses exceed $3S_m$.

With respect to piping stresses, the pipe break criteria were not intended to imply that breaks will occur when the piping stress exceeded $2.4S_m$ (80% of the primary plus secondary stress limit). It is the staff's belief, however, that if a pipe break were to occur (on one of those rare occasions), it is more likely

to occur at a piping location where there is the least margin to the ultimate tensile strength.

Similarly, from a fatigue strength standpoint, the staff believes that a pipe break is more likely to occur where the piping is expected to experience large cyclic loadings. Although the staff concurs with the industry belief that a cumulative usage factor of 0.1 is a relatively low limit, the uncertainties involved in the design considerations with respect to the actual cyclic loadings experienced by the piping tend to be greater than the uncertainties involved in the design considerations used for the evaluation of primary and secondary stresses in piping systems. The staff finds that the conservative fatigue considerations in the current SRP guidelines provide an appropriate margin of safety against uncertainties for those locations where fatigue failures are likely to occur (e.g., at local welded attachments).

In its presentation to the Advisory Committee on Reactor Safeguards on June 9, 1983, and in an October 5, 1983, meeting of a group of pressurized water reactor near-term operating license utilities and the NRC staff, the staff indicated that the elimination of AIBs was not to apply to piping systems in which stress corrosion cracking, large unanticipated dynamic loads such as steam- or water-hammer, or thermal fatigue in fluid-mixing situations could be expected to occur. In addition, the elimination of AIBs was to have no effect on the requirement to environmentally qualify safety-related equipment; in fact, this requirement was to be clarified to ensure positive qualification requirements.

For Class 1 piping, a considerable amount of quality assurance in design, analyses, fabrication, installation, examination, testing, and documentation is provided which ensures that the safety concerns associated with the uncertainties discussed above are significantly reduced. On the basis of the staff evaluation of the design considerations given to Class 1 piping, the stress and fatigue limits provided in the BTP MEB 3-1 break criteria, and the relatively small degree of uncertainty in the loadings, the staff finds that the need to postulate AIBs in ASME Code, Class 1 piping in which large unanticipated dynamic loads, stress corrosion cracking, and thermal fatigue such as in mixing situations are not present and in which all equipment has been environmentally qualified is not compensated for by an increased level of safety. In addition, systems may actually perform more reliably for the life of the plant if the request to postulate AIB criteria for ASME Code, Class 1 piping is eliminated. The staff has concluded that these requirements are present for those ASME Code, Class 1 piping systems identified in the applicant's submittals of February 7 and February 19, 1985.

ASME Code, Class 2 and 3 Piping Systems

In accordance with BTP MEB 3-1 (Paragraph B.1.c.(2)), breaks in ASME Code, Class 2 and 3 piping should be postulated at the following locations:

- (1) at terminal ends
- (2) at intermediate locations selected by one of the following criteria:
 - (a) at each pipe fitting, welded attachment, and valve

- (b) at each location where the stresses exceed $0.8 (1.2S_h + S_A)$ but at not less than two separated locations chosen on the basis of highest stress

In his request, the applicant has not proposed changing Criterion (1) above. Postulation of pipe breaks at terminal ends will not be eliminated in the requested SRP deviation for Class 2 and 3 piping systems. Breaks are required to be postulated at terminal ends irrespective of piping stresses.

The AIB criterion is stated in Item (2)(b) above where breaks are to be postulated at intermediate locations where the stresses exceed $0.8 (1.2S_h + S_A)$ but "at not less than two separated locations chosen on the basis of highest stress." The stress limit provided in the above pipe break criterion represents the stress associated with 80% of the combined primary and secondary stress limit. Thus, a break is required to be postulated where the maximum stress range, as calculated by the sum of Equations (9) and (10) of Paragraph NC/ND-3652 of the ASME Code, Section III, exceeds 80% of the combined primary and secondary stress limit, when those loads and conditions are considered for which level A and level B stress levels have been specified in the system's design specification (i.e., sustained loads, occasional loads, and thermal expansion) including an operating basis earthquake event. However, the Class 2 and 3 pipe break criteria do not have a provision for the postulation of pipe breaks based on a fatigue limit, since an explicit fatigue evaluation is not required in the ASME Code for these classes of construction because of favorable service experience and lower levels of operating cyclic stresses.

For those Class 2 and 3 piping systems that experience a large number of stress cycles (e.g., main steam and feedwater systems), the ASME Code has provisions that are intended to address these types of loads. The rules governing considerations for welded attachments in ASME Code, Class 2 and 3 piping that do preclude fatigue failure are partially given in Paragraph NC/ND-3645 of the ASME Code. The Code states:

External and internal attachments to piping shall be designed so as not to cause flattening of the pipe, excessive localized bending stresses, or harmful thermal gradients in the pipe wall. It is important that such attachments be designed to minimize stress concentrations in applications where the number of stress cycles, due either to pressure or thermal effect, is relatively large for the expected life of the equipment.

Code rules governing the fatigue effects associated with general bending stresses caused by thermal expansion are addressed in Paragraph NC/ND-3611.2(e) and are generally incorporated into the piping stress analyses in the form of an allowable stress reduction factor.

Thus, it can be concluded that when the piping designers have appropriately considered the fatigue effects for Class 2 and 3 piping systems in accordance with Paragraph NC/ND-3645, the likelihood of a fatigue failure in Class 2 and 3 piping caused by unanticipated cyclic loadings can be significantly reduced.

On the basis of staff's evaluation of the design considerations given to Class 2 and 3 piping, the stress limits provided in the SRP break criterion, and the

Concurrent with, in a letter from J. DeVincentis to G. Knighton dated August 9, 1984, the applicant requested a partial exemption from GDC 4 requirements for postulating breaks in the reactor coolant loop (RCL) piping. The applicant performed an evaluation of the functional capability of ASME Code Class 1 piping systems contingent on the staff's approval of the specific exemption request for eliminating the effects of dynamic loadings resulting from postulated breaks in the RCL on branch lines connected to the RCL. Thus, the functional capability issue for branch lines connected to the RCL was evaluated for seismic and other dynamic transient loadings excluding the loss-of-coolant-accident (LOCA) dynamic loadings.

The applicant performed analyses for these essential Class 1 piping systems for which functionality is required for safe shutdown of the plant. The analyses included approximately 4,300 node points in the piping systems. Of the 4,300 points, only 6 points exceeded the Level C primary stress limit as provided in ASME Code, Section III, Subsection NB (1980 Edition up to and including the Winter 1981 Addenda). Because the staff considers the Level C primary stress limit as revised in the Summer 1981 Code Addenda to be an adequate basis to ensure the functionality of piping systems, the staff focused primarily on the six points that exceeded the Level C primary stress limit. However, no points exceeded the ASME Code Level D stress limit which is required to ensure the piping structural integrity under LOCA conditions. For the six points that exceeded the Level C primary stress limit, the applicant has demonstrated that no mechanism exists that could result in collapse or instability of the piping system. Furthermore, the six points that exceeded the Level C limit were located in five independent piping subsystems and were either reducers or run pipe. Thus, the degree of ovalization of these piping components will not cause a significant reduction in flow area even for relatively large piping deflections. Because the six points are distinctly separated and cannot interact, the potential for the formation of a plastic hinge or total collapse as a result of instability that could lead to a condition adversely affecting the functionality of the piping does not exist.

On the basis of its review of the information provided by the applicant, the staff finds that the applicant has adequately demonstrated that, by excluding the LOCA dynamic loads from the load combinations used for the evaluation of piping systems, the stresses remaining in the piping are sufficiently low to ensure the functionality of the piping system. Thus, contingent on the acceptability and approval of the GDC 4 exemption request, the staff concludes from a technical standpoint that the open issue identified in the SER regarding the functional capability of ASME Code Class 1 auxiliary piping systems is closed.

4 REACTOR

4.4 Thermal-Hydraulic Design

4.4.5 Instrumentation

4.4.5.2 Flow Measurement Uncertainties

The applicant originally proposed a calorimetric flow measurement uncertainty of $\pm 1.5\%$ as stated in the SER. In response to a staff request, the applicant provided additional information in a letter dated April 25, 1983, relative to the flow measurement uncertainty value and to the possibility of fouling of the feedwater venturi meter and crud buildup in the pressure taps for the venturi meter and flow elbow. The applicant stated that his statistical error combination technique was similar to that used by Westinghouse for several other plants (Byron, Braidwood, McGuire, and V. C. Summer). The applicant stated that because of the use of different resistance temperature detectors (RTDs) to measure reactor coolant temperature in the Seabrook Station, the total uncertainty value was increased from the original value of $\pm 1.5\%$ to a range from $\pm 1.9\%$ to $\pm 2.0\%$ depending on the number of elbow taps per loop and whether the reading was by digital voltmeter or computer. The applicant stated that a bias for venturi fouling would not be used and the absence of fouling would be confirmed by visual inspection and by trending plant performance. The applicant provided information on why crud buildup is not expected. Also, because crud buildup has not been detected at any Westinghouse reactors, it is, therefore, not being considered for the Seabrook plant.

The staff requested additional information on the measurement uncertainty analysis and venturi flow meter fouling in a letter dated February 6, 1984. The applicant in a letter dated November 29, 1984, provided a flow measurement uncertainty analysis that included plant-specific instrumentation uncertainties for Seabrook when bounding values for the generic Westinghouse analysis (Rahe, March 31, 1982) were exceeded. The applicant's submittal included a tabulation comparing Westinghouse generic values of flow uncertainties with the ones for Seabrook. Calibration and drift values were included for the instrumentation. The instrumentation for Seabrook is the same as that assumed by Westinghouse with the following exceptions, which were accounted for in the analysis: feedwater pressure transmitter, feedwater RTDs, station computer, and test equipment. Also, the measurement arrangement for Seabrook differs from the standard assumed for Westinghouse in several aspects: the use of digital voltmeters for direct measurement of feedwater flow and elbow taps for reactor coolant system pressure and for the steam pressure and pressurizer pressure. As an aid in checking for venturi fouling, the applicant intends to use trending of several performance parameters: feed flow versus steam flow, reactor power versus core differential pressure, reactor power versus generator output, and venturi flow measurement versus flow measurement by a sonic flow meter in series with each venturi.

The sonic flow meters have a repeatability of 0.1% as specified by the manufacturer. On the basis of the anticipated performance of the sonic flow meter, an

uncertainty of 0.1% for the ability to determine whether venturi fouling exists has been included in the measurement uncertainty analysis. The performance of the sonic flow meters is to be monitored and evaluated in the power ascension phase of startup testing. If and when venturi fouling is detected, either the venturi will be cleaned before the next fuel cycle measurement or corrections to the feedwater flow measurement are to be applied as a bias. The feedwater piping does not include design features specifically to clean the venturi meters. Provisions and procedures to clean the venturi meters are to be established when or if fouling is detected and if it is determined that cleaning is warranted. The applicant intends to use three elbow taps per loop when available and two as a minimum. The applicant's analysis indicates a calorimetric flow measurement uncertainty of $\pm 1.9\%$. The use of elbow taps increases the uncertainty to $\pm 2.0\%$. This value is applicable for two or three elbow taps per loop and when either a computer or digital voltmeter is used with the elbow tap readings.

The staff has reviewed the analysis and found the responses acceptable in regard to concerns on crud buildup and instrument calibration and drift. Described methods of trending fouling of the venturi meter based on other measured performance parameters are acceptable for detecting an effect of fouling. As another means for detection of fouling, a sonic sensor with repeatability and uncertainty of 0.1% is to be installed and will be evaluated during the power ascension phase of startup testing. The applicant will provide for the cleaning of the venturi if it is warranted. However, the staff still requires a 0.1% bias on the measurement uncertainty to account for possible venturi fouling. This value is based on the capability to detect the limiting amount (0.1%). This will then result in an increase of the calorimetric flow measurement uncertainty value for the Seabrook Station to $\pm 2.0\%$ and an increase of the total uncertainty when using elbow taps to $\pm 2.1\%$. The $\pm 2.1\%$ uncertainty value must be implemented by inclusion in the Technical Specifications.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Compliance With Codes and Code Cases

5.2.1.2 Applicable Code Cases

As was noted in the SER, the staff's acceptance of ASME Code cases was contingent on the applicant supplying a confirmatory list of ASME Code cases used in the construction of Section III, Class 1 components within the reactor coolant pressure boundary (RCPB). This information has been supplied in Amendment 52 to the Final Safety Analysis Report (FSAR) in response to Q210.76.

The staff has reviewed the list of Code cases and finds they are acceptable. The staff concludes that compliance with the requirements of these Code cases will result in a component quality level that is commensurate with the importance of the safety function of the RCPB and constitutes an acceptable basis for satisfying the requirements of GDC 1, and is, therefore, acceptable.

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

This section was prepared with the technical assistance of U.S. Department of Energy contractors from the Idaho National Engineering Laboratory.

This evaluation supplements the conclusions in this section of the SER, which addresses the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g).

5.2.4.3 Evaluation of Compliance With 10 CFR 50.55a(g) for Seabrook Unit 1

The staff has completed its review of the information presented in the FSAR through Amendment 53 dated August 1984; the Seabrook Unit 1 Balance-of-Plant (BOP)* Preservice Inspection (PSI) Program, Revision 1, dated January 6, 1984; and the Seabrook Unit 1 Reactor Pressure Vessel (RPV) PSI Program Plan, Revision 3, dated March 15, 1984. The latter two documents were submitted by the applicant in a letter dated June 21, 1985.

On the basis of the construction permit date of July 7, 1976, this section of the regulations requires that a PSI program be developed and implemented using at least the edition and addenda of Section XI of the ASME Code applied to the construction of the particular components. The components (including supports) may meet the requirements in subsequent editions of this Code and addenda that are incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein. The applicant states that the PSI program must comply with requirements in the ASME Code no later than the Summer 1972

*The applicant defines BOP as ASME Code, Class 1, 2, and 3 components other than the reactor vessel.

Addenda of the 1971 Edition. However, the applicant has voluntarily updated the PSI program to meet the requirements of Section XI of the 1977 Edition including Addenda through Summer 1978, except where 10 CFR 50.55a(b) requires or permits the use of the 1974 Edition of Section XI including Addenda through Summer 1975.

The review of Revision 3 of the RPV PSI program has determined that the selection of welds for the reactor vessel examination is consistent with the applicable Code requirements, and the applicant is complying with the recommendations of Regulatory Guide 1.150 for the examination of the vessel welds. The vessel welds are examined by means of a layered technique utilizing different transducer configurations rather than electronic gating or gain differential utilizing the same transducer. The applicant has developed and qualified the inside-diameter, near-surface techniques utilizing dual 60° search units. This technique is used on all vessel welds except some lower head areas where a "full vee," 45° shear wave, near-surface technique is used. The applicant states that the layered approach is more conservative than other implementation concepts and ensures that the amount of unexamined volume is negligible throughout the reactor pressure vessel. The staff considers the RPV PSI program acceptable and the review completed.

On the basis of the review of Revision 1 of the BOP PSI program plan, the staff has determined that systems and components within the reactor coolant pressure boundary are included for examination according to the applicable Code requirements. However, additional information is needed to complete the staff's evaluation. The following items require further input or clarification from the applicant:

- (1) FSAR Table 5.2-2, "Material Specifications Class 1 Primary Components," shows the use of cast stainless steel (SA-351, Gr CF8A) fittings. Paragraph 2.2, Table 2.2, in the PSI program for examination of the reactor coolant system (Document 80A8983) discusses exceptions from the ASME Code-required ultrasonic examination of cast austenitic stainless steel fitting and component welds in the primary piping systems. Table 2.2 states that the wrought pipe-to-cast elbow or cast pump stainless steel welds will be examined with a one-side, 1/2-vee ultrasonic examination from the pipe side, and the cast side may not be examined because of metallurgical constraints (large grain structure that produces a high degree of scatter and attenuation of ultrasound). This table also states that the cast elbow-to-cast pump welds will not receive an ultrasonic examination because of these constraints.

A one-side, 1/2-vee examination does not meet the Code requirements. The staff recognizes that the ultrasonic examination of cast stainless steel fittings and components may be difficult. However, a preservice inspection with the best available instrumentation with straight beam and angle beam techniques should be attempted and documented on all welds in accordance with Section XI requirements. A technical justification is required if the examination is performed from the pipe side only.

In Table 2.4, which lists reference calibration standards, only SA-376, Gr 304N calibration blocks have been identified. The program tables indicate that the cast stainless steel fittings to pipe welds will be examined

with these wrought stainless steel calibration standards. Calibration blocks for fitting to nozzle safe end welds are to be identified "later." Calibration blocks should be available for each type of material (i.e., ferritic, cast, or wrought) that is representative of both sides of the weld. The applicant should describe the measures that were taken (i.e., using the best available refracted longitudinal waves transducers) to determine that a meaningful ultrasonic examination could not be performed from the cast side of these welds. Copies of the ultrasonic testing procedures for the primary piping welds should be provided for staff review.

- (2) Visual Examination Procedure 80A6474 states that "at least one member of a visual examination team shall be certified to at least Level II and shall have passed his/her visual acuity examination in accordance with the minimum qualification requirements of Nuclear Energy Services (NES) document 80A9069 (latest revision)." The applicant should clarify the above, since ASME Code, Section XI, Paragraph IWA-2300 requires that all nondestructive examination personnel for all methods be certified to at least Level I and be examined by qualified personnel to ensure the required near-distance visual acuity. In addition, personnel qualified for VT-2, VT-3, and VT-4 shall receive a visual examination to ensure far-distance acuity. Visual examinations must be conducted annually.

The specific areas where the Code requirements cannot be met will be identified after the examinations are performed. The applicant has committed to identify all plant-specific areas where the Code requirements cannot be met and provide a supporting technical justification for requesting relief. The input will be complete after the applicant

- (1) docket an acceptable resolution to the above issues
- (2) submits all relief requests with a supporting technical justification

The staff considers the review of the Seabrook Unit 1 PSI program an open item subject to the applicant providing an acceptable response to the above issues concerning the ultrasonic examination of the cast stainless steel welds and the visual examination requirements. Evaluation of the response will be reported in a future supplement to the SER.

The initial inservice inspection program has not been submitted by the applicant. This program will be evaluated after the applicable ASME Code edition and addenda can be determined based on 10 CFR 50.55a(b) but before inservice inspection commences during the first refueling outage.

6 ENGINEERED SAFETY FEATURES

6.6 Inservice Inspection of Class 2 and 3 Components

This section was prepared with the technical assistance of U.S. Department of Energy contractors from the Idaho National Engineering Laboratory.

This evaluation supplements conclusions in this section of the SER, which addresses the definition of examination requirements and the evaluation of compliance with 10 CFR 50.55a(g).

6.6.3 Evaluation of Compliance with 10 CFR 50.55a(g) for Seabrook Unit 1

The staff has completed its review of the information presented in the FSAR through Amendment 53 dated August 1984 and the Seabrook Unit 1 BOP* PSI Program, Revision 1, dated January 6, 1984.

On the basis of the construction permit date of July 7, 1976, this paragraph of the regulations requires that a preservice inspection program be developed and implemented using at least the edition and addenda of Section XI of the ASME Code applied to the construction of the particular components. The components (including supports) may meet requirements in subsequent editions of this Code and addenda that are incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein. The applicant states that the PSI program must comply with requirements in the ASME Code no later than the Summer 1972 Addenda of the 1971 Edition. However, the applicant has voluntarily updated the PSI program to meet the requirements of Section XI of the 1977 Edition including Addenda through Summer 1978, except where 10 CFR 50.55a(b) requires or permits the use of the 1974 Edition of Section XI including Addenda through Summer 1975.

On the basis of the review of the above documents, additional information is needed to complete the staff's evaluation. The following items require further input or clarification from the applicant:

- (1) 10 CFR 50.55(e)(2)(iv) requires that ASME Code, Class 2 piping welds in the residual heat removal (RHR), emergency core cooling (ECC), and containment heat removal (CHR) systems shall be examined. The staff review of the PSI program shows extensive numbers of surface-only examinations for Class 2 piping welds in the safety injection, chemical and volume control, and containment spray systems. These systems should not be completely exempted from preservice volumetric examination based on Section XI exclusion criteria contained in Paragraph IWC-1220. To satisfy the inspection requirements of GDC 36, 39, 42, and 45, the PSI program must include volumetric examination of a representative sample of welds in the RHR, ECC, and CHR systems. The PSI program should be revised to include a volumetric examination of a representative sample of the welds in these systems.

*The applicant defines BOP as ASME Code, Class 1, 2, and 3 components other than the reactor vessel.

- (2) Visual Examination Procedure 80A6474 states that "at least one member of a visual examination team shall be certified to at least Level II and shall have passed his/her visual acuity examination in accordance with the minimum qualification requirements of NES document 80A9069 (latest revision)." The applicant should clarify the above, since ASME Code, Section XI, Paragraph IWA-2300 requires that all nondestructive examination personnel for all methods be certified to at least Level I and be examined by qualified personnel to ensure the required near-distance visual acuity. In addition, personnel qualified for VT-2, VT-3, and VT-4 shall receive a visual examination to ensure far-distance acuity. Visual examinations must be conducted annually.

The specific areas where the Code requirements cannot be met will be identified after the examinations are performed. The applicant has committed to identify all plant-specific areas where the Code requirements cannot be met and provide a supporting technical justification for requesting relief. The input will be complete after the applicant

- (1) docket an acceptable resolution to the above issues
- (2) submits all relief requests with a supporting technical justification

The staff considers the review of the Seabrook Unit 1 PSI program an open item subject to the applicant providing an acceptable response to the above issues concerning the volumetric examination of the ASME Code, Class 2 components in the ECC, RHR, and CHR systems and the visual examination requirements. Evaluation of the response will be reported in a future supplement to the SER.

The initial inservice inspection program has not been submitted by the applicant. This program will be evaluated after the applicable ASME Code edition and addenda can be determined based on 10 CFR 50.55a(b) but before inservice inspection commences during the first refueling outage.

9 AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.3 Spent Fuel Pool Cooling and Cleanup System

In the SER, the staff stated that the spent fuel pool cooling system would maintain the fuel pool water temperature at 140°F or less with the loss of either a spent fuel pump or heat exchanger under a heat load based on decay heat generation from the design storage case of 845 fuel assemblies (one-third of a core placed in the pool 150 hours after reactor shutdown plus 12 previous one-third-core annual refueling cycles). In Amendment 52 to the FSAR, the applicant revised Table 9.1-3, "Design Conditions for Normal and Abnormal Operating Conditions." In this table, the applicant indicated that, with either spent fuel pool cooling train (one pump and one heat exchanger) operating, the pool water temperature would be maintained at or below 140°F, with a maximum normal heat load based on decay heat generation from the design storage case of one-third of a core placed in the pool 150 hours after reactor shutdown plus 15 previous one-third-core annual refueling cycles. This normal heat load temperature meets the acceptance criterion of 140°F.

In addition, space is reserved in the pool for the maximum abnormal heat load condition, which includes a full-core unload when the pool contains 1,223 spent fuel assemblies from 16 one-third-core refueling cycles. Under these conditions, both spent fuel pool cooling trains maintain a temperature of 141°F or less. This maximum abnormal heat load temperature is within acceptable limits.

The staff has reviewed the above FSAR changes and has performed an independent calculation of the heat loads for the above normal and abnormal cases in accordance with BTP ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling." On the basis of its review, the staff concludes that the applicant's calculated decay heat load is conservative and, therefore, acceptable. The spent fuel pool cooling and cleanup system meets the applicable acceptance criteria of SRP Section 9.1.3.

9.3 Process Auxiliaries

9.3.4 Chemical and Volume Control System

9.3.4.2 Evaluation

II.B.3 Postaccident Sampling System

After the incident at Three Mile Island Unit 2, the need was recognized for an improved postaccident sampling system (PASS) to determine the extent of core degradation following a severe reactor accident. Criteria for an acceptable sampling and analysis system are specified in NUREG-0737, Item II.B.3. The system should have the capability to obtain and quantitatively analyze reactor coolant and containment atmosphere samples without radiation exposure to any individual exceeding 5 rem to the whole body or 75 rem to the extremities (GDC 19)

during and following an accident in which there is core degradation. Materials to be analyzed and quantified include certain radionuclides that are indicators of severity of core damage (e.g., noble gases, isotopes of iodine and cesium, and nonvolatile isotopes), hydrogen in the containment atmosphere, and total dissolved gases or hydrogen, boron, and chloride in reactor coolant samples.

To comply with NUREG-0737, Item II.B.3, the applicant should (1) review and modify his sampling, chemical analysis, and radionuclide determination capabilities as necessary and (2) provide the staff with information pertaining to system design, analytical capabilities, and procedures in sufficient detail to demonstrate that the criteria are met.

By letter dated December 18, 1984, the applicant provided information on the PASS.

Criterion (1)

The applicant shall have the capability to promptly obtain reactor coolant samples and containment atmosphere samples. The combined time allotted for sampling and analysis should be three hours or less from the time a decision is made to take a sample.

The PASS provides the capability to obtain liquid samples from reactor coolant loops 1 and 3 (hot legs), containment recirculation sumps, and emergency core cooling system pump room sumps and gas samples of the containment atmosphere within 3 hours from the time a decision is made to take a sample. All electrically powered equipment (i.e., solenoid valves and sample pumps), whose operation is required to perform postaccident sampling, is powered from an emergency backup power source. The staff finds that these provisions meet Criterion (1) and are, therefore, acceptable.

Criterion (2)

The applicant shall establish an onsite radiological and chemical analysis capability to provide, within the three-hour time frame established above, quantification of the following:

- (a) certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage (e.g., noble gases, iodines and cesiums, and nonvolatile isotopes)
- (b) hydrogen levels in the containment atmosphere
- (c) dissolved gases (e.g., hydrogen), chloride (time allotted for analysis subject to discussion below), and boron concentration of liquids
- (d) alternatively, have in-line monitoring capabilities to perform all or part of the above analyses

The PASS provides the capability to collect diluted samples of reactor coolant and to analyze them for pH, boron, chloride, radionuclides, and dissolved gases. It also provides the capability to draw samples of the containment atmosphere

for analysis from the hydrogen analyzer system. Background levels will be reduced in the counting room through the use of a shielded cave. Personnel radiation exposure will be maintained as low as is reasonably achievable (ALARA) through the use of lead shield carrying devices and remote handling devices where appropriate.

The staff finds that the applicant partially meets Criterion (2) by establishing an onsite radiological and chemical analysis capability. However, the applicant should provide a procedure, consistent with the clarification of NUREG-0737, Item II.B.3, Attachment 1, to estimate the extent of core damage on the basis of radionuclide concentrations and taking into consideration other physical parameters such as core temperature data, pressure vessel liquid level, containment radiation levels, and hydrogen concentrations. The applicant is a participant in the Westinghouse Owners Group that has developed a methodology for a generic core damage assessment based on measurements of radionuclide concentrations and other plant indicators. A plant-specific procedure based on this methodology would be acceptable.

Criterion (3)

Reactor coolant and containment atmosphere sampling during post-accident conditions shall not require an isolated auxiliary system (e.g., the letdown system or the reactor water cleanup system) to be placed in operation in order to use the sampling system.

Reactor coolant and containment atmosphere sampling during postaccident conditions does not require an isolated auxiliary system to be placed in operation in order to perform the sampling function. The PASS provides the ability to obtain samples from the reactor coolant system, the residual heat removal system, the containment sump, and the containment atmosphere without using an isolated auxiliary system.

All remotely operated valves are environmentally qualified for the conditions in which they need to operate and are cycled from either the control room or the local panel. The staff finds that these provisions meet Criterion (3) and are, therefore, acceptable.

Criterion (4)

Pressurized reactor coolant samples are not required if the licensee can quantify the amount of dissolved gases with unpressurized reactor coolant samples. The measurement of either total dissolved gases or hydrogen gas in reactor coolant samples is considered adequate. Measuring the oxygen concentration is recommended, but is not mandatory.

The amount of dissolved gases in the reactor coolant will be determined by extracting a gaseous sample from the postaccident sampling panel using a shielded syringe if necessary. This sample will be analyzed for the hydrogen and gamma spectrum only. The procedure for this analysis is currently under development and will be available 6 months before fuel load. The staff finds that these provisions meet Criterion (4) and are, therefore, acceptable.

Criterion (5)

The time for a chloride analysis to be performed is dependent upon two factors: (a) if the plant's coolant water is seawater or brackish water and (b) if there is only a single barrier between primary containment systems and the cooling water. Under both of the above conditions the licensee shall provide for a chloride analysis within 24 hours of the sample being taken. For all other cases, the licensee shall provide for the analysis to be completed within 4 days. The chloride analysis does not have to be done on site.

Grab sample analysis for chloride of a diluted liquid sample will be completed within 96 hours of drawing the sample. At Seabrook Station seawater is used for cooling water, but the design incorporates a double barrier between primary containment systems and the cooling water. Samples may be diluted up to a factor of 1,000 at the sample station. However, the analysis employed will provide for a minimum detectable threshold of 10 ppm chloride.

The PASS will provide for the capability of taking an undiluted sample consistent with ALARA principles. This undiluted sample will be retained for analysis within 30 days.

Procedures for drawing both the diluted and undiluted chloride samples and for the analysis of the diluted chloride sample are under development and will be available 6 months before fuel load. The staff finds that these provisions partially meet Criterion (5). The minimum detectable threshold of 0.15 ppm chloride in the reactor coolant is desired. The applicant should provide a more sensitive method for chloride analysis.

Criterion (6)

The design basis for plant equipment for reactor coolant and containment atmosphere sampling and analysis must assume that it is possible to obtain and analyze a sample without radiation exposures to any individual exceeding the criteria of GDC 19 (Appendix A, 10 CFR Part 50) (i.e., 5 rem whole body, 75 rem extremities). (Note that the design and operational review criterion was changed from the operational limits of 10 CFR Part 20 (NUREG-0578) to the GDC 19 criterion (October 30, 1979, letter from H. R. Denton to all licensees.))

The applicant will perform a shielding analysis to ensure that operator radiation exposure from reactor coolant/containment atmosphere sampling and analysis is within the acceptable limits of 5 rem whole body and 75 rem extremities. The operator exposure will include entering and leaving the sample panel area, operating the sample panel manual valves, performing manual sample dilutions, and transferring samples to a shielded cart for analysis. The staff has determined that this issue remains open, and it will evaluate the applicant's compliance with the requirements of Criterion (6) once it has received the applicant's analyses.

Criterion (7)

The analysis of primary coolant samples for boron is required for PWRs. (Note that Rev. 2 of Regulatory Guide 1.97 specifies the need for primary coolant boron analysis capability at BWR plants.)

A diluted grab sample of the reactor coolant will be analyzed for boron. This provision meets the recommendations of RG 1.97, Revision 2, and Criterion (7) and is, therefore, acceptable.

Criterion (8)

If in-line monitoring is used for any sampling and analytical capability specified herein, the licensee shall provide backup sampling through grab samples, and shall demonstrate the capability of analyzing the samples. Established planning for analysis at offsite facilities is acceptable. Equipment provided for backup sampling shall be capable of providing at least one sample per week until the accident condition no longer exists.

No in-line monitoring capability is provided. All chemical analysis will be done on diluted grab samples. However, undiluted grab samples will be taken and retained for chloride analysis within 30 days after the accident. The staff finds that these provisions meet Criterion (8) and are, therefore, acceptable.

Criterion (9)

The applicant's radiological and chemical sample analysis capability shall include provisions to:

- (a) Identify and quantify the isotopes of the nuclide categories discussed above to levels corresponding to the source terms given in Regulatory Guides 1.3 or 1.4 and 1.7. Where necessary and practicable, the ability to dilute samples to provide capability for measurement and reduction of personnel exposure should be provided. Sensitivity of onsite liquid sample analysis capability should be such as to permit measurement of nuclide concentration in the range from approximately 1 $\mu\text{Ci/g}$ to 10 Ci/g .
- (b) Restrict background levels of radiation in the radiological and chemical analysis facility from sources such that the sample analysis will provide results with an acceptably small error (approximately a factor of 2). This can be accomplished through the use of sufficient shielding around samples and outside sources, and by the use of a ventilation system design which will control the presence of airborne radioactivity.

The radionuclides in both the primary coolant and the containment atmosphere are identified and quantified. Provisions are available for diluted reactor coolant and containment atmosphere samples to minimize personnel exposure. The PASS can perform radioisotope analyses at the levels corresponding to the source terms in RGs 1.4, Revision 2, and 1.7. Radiation background levels will be restricted by shielding and ventilation in the radiological and chemical analysis facilities so that analytical results can be obtained within an acceptably small error (approximately a factor of 2). The staff finds that these provisions meet Criterion (9) and are, therefore, acceptable.

Criterion (10)

Accuracy, range, and sensitivity shall be adequate to provide pertinent data to the operator in order to describe the radiological and chemical status of the reactor coolant systems.

The applicant stated that the accuracy, range, and sensitivity of postaccident analyses capabilities will be detailed in the analytical procedures currently under development. The staff has determined that this issue remains open, and it will evaluate the applicant's compliance with the requirements of Criterion (10) once it has received the applicant's analyses.

Criterion (11)

In the design of the postaccident sampling and analysis capability, consideration should be given to the following items:

- (a) Provisions for purging sample lines, for reducing plateout in sample line, for minimizing sample loss or distortion, for preventing blockage of sample lines by loose material in the RCS or containment, for appropriate disposal of the samples, and for flow restrictions to limit reactor coolant loss from a rupture of the sample line. The postaccident reactor coolant and containment atmosphere samples should be representative of the reactor coolant in the core area and the containment atmosphere following a transient or accident. The sample lines should be as short as possible to minimize the volume of fluid to be taken from containment. The residues of sample collection should be returned to containment or to a closed system.
- (b) The ventilation exhaust from the sampling station should be filtered with charcoal adsorbers and high-efficiency particulate air (HEPA) filters.

The applicant has addressed provisions for purging to ensure samples are representative, for limiting reactor coolant loss from a rupture of a sample line, and for ventilation exhaust from PASS filtered through charcoal adsorbers and HEPA filters. The reactor coolant system sampling locations were selected to provide coolant samples that are representative of core conditions. Heat tracing was installed on sampling lines to limit iodine plateout. The staff has determined that these provisions meet Criterion (11) and are, therefore, acceptable.

Conclusion

The applicant's proposed methods to meet 7 of the 11 criteria are acceptable. The four criteria that have not been fully resolved are

- (1) Criterion (2) - provide a core damage estimate procedure to include radionuclide concentrations and other physical parameters as indicators of core damage
- (2) Criterion (5) - provide PASS capability to analyze for chloride by a method with the required sensitivity in the presence of anticipated interfering species and a minimum detectable threshold of 0.15 ppm chloride.

- (3) Criterion (6) - provide information on the collection and analysis of reactor and containment atmosphere samples without radiation exposures to individuals consistent with the criteria in GDC 19.
- (4) Criterion (10) - (a) develop more sensitive analysis methods for chemical and radiological analyses; (b) provide information on ranges and accuracies for these analyses demonstrating applicability of procedures and instrumentation in the postaccident water chemistry and radiation environment; and (c) retrain operators on a semiannual basis.

11 RADIOACTIVE WASTE MANAGEMENT

11.3 Gaseous Waste Management System

In the SER, the staff expressed concern on how the applicant would handle concentrations greater than 4% hydrogen in cubicles containing components of the radioactive gaseous waste system (RGWS). The applicant in response to a staff question stated that some cubicles of the RGWS would be monitored for hydrogen and if concentrations approached 4%,

- (1) the affected components of the process stream would be isolated and/or the affected component purged with nitrogen
- (2) the affected cubicle would be ventilated to reduce the hydrogen concentration
- (3) unnecessary personnel would be evacuated from the area

The staff's concern was that the affected cubicle was not ventilated on a routine basis and that, with a hydrogen concentration greater than 4%, to begin ventilation would present an oxygen source and potentially an explosive mixture. The staff's position was that the cubicle should not be ventilated unless the cubicle's concentration of hydrogen is reduced. This could be done by purging the affected component with nitrogen.

The applicant in a revised response to the staff's original question on the hydrogen concentrations in cubicles of the RGWS components stated that potential leakage from the RGWS components is vented along with normal building exhaust air to the Unit 1 plant vent and that this ventilation flow is maintained in the event of abnormal levels of hydrogen within the cubicles of the RGWS. For abnormal levels of hydrogen within the hydrogen surge tank cubicle, an additional 20,000-scfm purge system will automatically activate on high hydrogen concentrations. The normal and supplemental ventilation flows will dilute and reduce the hydrogen concentration in the affected compartments.

The staff considers that this revised response has satisfied its concerns on the adequacy of diluting hydrogen concentrations in various cubicles housing RGWS components. With the resolution of this item, the staff concludes that the design of the gaseous waste management systems is acceptable and meets the requirements of 10 CFR 20 and 20.106, 10 CFR 50 and 50.34a, GDC 3, 60, and 61, and 10 CFR 50, Appendix I. This conclusion is based on the following:

- (1) The applicant has met the requirements of GDC 60 and 61 with respect to controlling releases of radioactive material to the environment by ensuring that the designs of the gaseous waste management systems include the equipment and instruments necessary to detect and control the release of radioactive materials in gaseous effluents.

- (2) The applicant has met the requirement of Appendix I of 10 CFR 50 by meeting the as low as is reasonably achievable criterion as follows:
- (a) Regarding Sections II.B and II.C of Appendix I, the staff has considered releases of radioactive material (noble gases, radioiodine, and particulates) in gaseous effluents for normal operation, including anticipated operational occurrences, based on expected radwaste inputs over the life of the plant for each reactor on the Seabrook site. The staff has determined that the proposed gaseous waste management systems are capable of maintaining releases of radioactive materials in gaseous effluents so that the calculated individual doses in an unrestricted area from all pathways of exposure are less than 5 mrem to the total body or 15 mrem to the skin and less than 15 mrem to any organ from releases of radioiodine and radioactive material in particulate form.
 - (b) Regarding Section II.D of Appendix I, the staff has considered the potential effectiveness of augmenting the proposed gaseous waste management systems using items of reasonably demonstrated technology and has determined that further effluent treatment will not effect reductions in the cumulative population dose within a 50-mi radius of the reactor at a cost of less than \$1,000 per man-rem or \$1,000 per man-thyroid-rem.
- (3) The applicant has met the requirements of 10 CFR 20, since the staff considered the potential consequences resulting from reactor operation with "1% of the operating fission product inventory in the core being released to the primary coolant" for a pressurized water reactor and determined that, under these conditions, the concentrations of radioactive materials in gaseous effluents in unrestricted areas will be a small fraction of the limits specified in 10 CFR 20, Appendix B, Table II, Column 1.
- (4) The staff has considered the capability of the proposed gaseous waste management systems to meet the anticipated demands of the plant resulting from anticipated operational occurrences and has concluded that the system capacity and design flexibility are adequate to meet the anticipated needs of the plant.
- (5) The staff has reviewed the applicant's quality assurance provisions for the gaseous waste management systems, the quality group classifications used for system components, and the seismic design applied to the design of the systems and of structures housing the radwaste systems. The design of the systems and structures housing these systems meets the criteria in RG 1.143.
- (6) The staff has reviewed the provisions incorporated in the applicant's design to control releases from hydrogen explosions in the gaseous waste management systems and concludes that the measures proposed by the applicant are adequate to prevent the occurrence of an explosion or to withstand the effects of an explosion in accordance with GDC 3.

11.5 Process and Effluent Radiological Monitoring and Sampling Systems

In the SER, the staff indicated that the applicant had committed to incorporate into plant procedures a requirement to obtain periodic grab samples of the service water whenever a leak between the primary component cooling water (PCCW) system and the service water system is confirmed and the PCCW is radioactively contaminated. The staff stated that such a proposal did not satisfy the intent of Table 2 of SRP Section 11.5 and that the staff will require either a radiation monitor or a continuous sampler of the service water.

After a series of meetings and telephone conversations, the applicant and the staff have agreed to an approach that will satisfy the intent of Table 2 of SRP Section 11.5. The applicant has made commitments that will ensure that the potential releases from the service water system are known. These commitments are summarized below:

- (1) weekly sampling and analysis of the PCCW and the service water, daily sampling and analysis of the PCCW and service water if the PCCW radiation monitor is inoperable
- (2) daily sampling and analysis of the service water discharge when the PCCW concentration is $\geq 10^{-3}$ $\mu\text{Ci/cc}$
- (3) sampling and analysis of the service water once every 12 hours when the activity level in the PCCW is $\geq 10^{-4}$ $\mu\text{Ci/cc}$ and leakage is confirmed from the PCCW system to the service water system
- (4) installation of a rate-of-change alarm that will indicate a decreasing liquid level in the PCCW head tank based on detection of a low-level dose (LLD) of 10^{-8} $\mu\text{Ci/cc}$.

The above commitments, which will be incorporated into the Technical Specifications' sampling analysis requirements for the service water system, extend over various operating ranges with the increased sampling and analysis at times when leakage from the PCCW to the service water is occurring and/or the activity level in the PCCW is high.

The rate-of-change alarm will work in conjunction with the PCCW radiation monitor to alert the operator in the main control room of a leak to the service water system from the PCCW system. For the rate-of-change alarm, the applicant will select a setpoint based on detection of an activity level of 10^{-8} $\mu\text{Ci/cc}$ in the combined discharge of the service water system. This activity level was selected because it is the minimum detectable level of a service water monitor if such a monitor were installed.

Weekly sampling and analysis of the service water system will provide effluent data when confirmed leakage from the PCCW system exists and the PCCW activity level is less than 10^{-4} $\mu\text{Ci/cc}$. It will also provide a check of the operability of the rate-of-change monitor's function. Weekly sampling and analysis of the PCCW will confirm the operability of the PCCW radiation monitor.

Should the PCCW radiation monitor be inoperable, daily sampling and analysis of the service water and the PCCW will ensure that any release will be determined

within 24 hours and that a record of effluents from the service water may be maintained if a leak exists from the PCCW to the service water.

The intent of daily sampling and analysis of the service water when the PCCW activity level is $> 10^{-3}$ $\mu\text{Ci/cc}$ is to cover those situations when the responsiveness of the rate-of-change alarm may be slow indicating a leak caused by equal inleakage and outleakage from the PCCW. With this sampling and analysis requirement, the time period before a leak is determined is minimized (24 hours versus 168 hours) and the potential consequences of such a leak are reduced. The PCCW activity level of 10^{-3} $\mu\text{Ci/cc}$ was chosen because release of activity above this level would be unacceptable if allowed to continue for 7 days. The rate-of-change alarm would provide the operator with an alert should a leak develop during this period of time.

When confirmed leakage from the PCCW system exists and the radioactivity level in the PCCW is $\geq 10^{-4}$ $\mu\text{Ci/cc}$, samples of the service water will be taken and analyzed once every 12 hours. The requirement to sample and analyze once every 12 hours is the standard action statement for the service water system if its radiation monitor is inoperable. The staff finds the concentration level of 10^{-4} $\mu\text{Ci/cc}$ acceptable for initiating this twice-daily sampling because at this concentration the leak rate from the PCCW would have to be at least 1.1 gpm so that a service water monitor could detect the leak. At concentrations below 10^{-4} , the weekly sampling and analysis are sufficient.

The rate-of-change alarm provides a reasonable approach to determine leakage from the PCCW system. Its alarm setpoint will be established at a concentration of 10^{-8} $\mu\text{Ci/cc}$. This translates to a leak rate of approximately 1.1 gpm at a concentration of 10^{-4} $\mu\text{Ci/cc}$. With the incorporation of the PCCW radiation monitor and the rate-of-change alarm in the PCCW head level, the fluctuation in the PCCW can be seen in a short period of time. The methodology establishing the setpoint for this alarm will be included in the applicant's Offsite Dose Calculation Manual.

On the basis of its review of the proposed Technical Specifications for the sampling and analysis of the service water system and PCCW system and the utilization of the rate-of-change alarm, the staff concludes that the applicant's proposed approach for determining effluents from the service water system, in lieu of a service water monitor, is acceptable and meets the intent of Table 2 of SRP Section 11.5.

12 RADIATION PROTECTION

12.3 Radiation Protection Design Features

12.3.2 Shielding

II.B.2 Plant Shielding To Provide Access to Vital Areas

The staff has reviewed the Seabrook Station Post-Accident Dose Engineering Manual and has concluded that it is acceptable. The manual provides detailed analysis of mathematical models, using appropriate source terms and geometry of sources, to determine the plant shielding required to provide access to vital areas (NUREG-0737, Item II.B.2). Amendment 48 to the FSAR (p. RB-68) notes that this document was submitted by the applicant for staff review. Because this document was formally submitted by the applicant for staff review and was acceptable, the confirmatory item pertaining to this item is now closed.

12.3.4 Area Monitoring and Airborne Radioactivity Monitoring Instrumentation

12.3.4.1 Area Radiation Monitoring System

II.F.1 (3) Containment High-Range Radiation Monitor

Seabrook Station will have two high-range radiation monitors installed in containment. These monitors will be designed, located, calibrated, and qualified in accordance with Table II.F.1-3 of NUREG-0737 and will have a separate readout in the electronics cabinet located outside containment in the electrical tunnel. Indication and alarm will be provided in the control room. On the basis of a formal submittal by the applicant of the exact location of these monitors, the staff finds that Seabrook meets Item II.F.1, Attachment 3 of NUREG-0737, and this item is closed.

12.3.4.2 Airborne Radioactivity Monitoring Instrumentation

In response to a staff question on the improper installation of ventilation monitors downstream of filters for inplant airborne radioactivity monitoring, by letter dated November 29, 1982, the applicant committed to install portable continuous air monitors, equipped to monitor particulates and noble gases and to sample for iodine, at four locations within the plant as a resolution of the issue. The commitment also included a provision for installation of a system to monitor the primary auxiliary building exhaust air, upstream of filters. The commitments were incorporated into the FSAR by Amendment 48 (p. 12.3-24) and Amendment 49 (p. 12.3-20). Consequently, the staff finds Seabrook's inplant airborne radioactivity monitoring program acceptable, and this item is closed.

12.5 Operational Radiation Protection Program

12.5.3 Equipment and Instrumentation

III.D.3.3 Improved Inplant Iodine Instrumentation Under Accident Conditions

SER Section 12.5.3 stated that the applicant had committed to an improved in-plant iodine instrumentation system under accident conditions in accordance with NUREG-0737, Item III.D.3.3, by referring to a letter from the applicant dated November 29, 1982. This commitment has been incorporated directly into the FSAR in Amendments 49 and 53. The staff, therefore, finds that Seabrook Meets Item III.D.3.3 of NUREG-0737, and this item is closed.

13 CONDUCT OF OPERATIONS

13.5 Station Administrative Procedures

13.5.2 Operating and Maintenance Procedures

13.5.2.3 Reanalysis of Transients and Accidents; Development of Emergency Operating Procedures

In the SER, the staff described the status of the development of generic technical guidelines by the Westinghouse Owners Group (WOG) and the staff's development of guidelines for the long-term upgrading of emergency operating procedures (EOPs). NUREG-0899, "Guidelines for the Preparation of Emergency Operating Procedures," represents the staff's long-term program for upgrading EOPS and describes the use of a procedures generation package (PGP) to prepare EOPs. Submittal of the PGP was made a requirement by Supplement 1 to NUREG-0737 (Generic Letter 82-33). The generic letter requires each applicant to submit a PGP that includes:

- (1) plant-specific technical guidelines
- (2) writer's guide
- (3) description of the program to be used for the validation of EOPs
- (4) description of the training program for the upgraded EOPs.

This report describes the results of the staff's review of the applicant's response to Section 7 of Generic Letter 82-33 related to the development and implementation of EOPs for Seabrook Station, Units 1 and 2.

The staff's review was conducted to determine the adequacy of the applicant's program for preparing and implementing EOPs. Criteria for the review of a PGP are not currently in the Standard Review Plan (SRP). Therefore, this review was based on NUREG-0899, the reference document for the EOP upgrade portion of Supplement 1 to NUREG-0737 (Generic Letter 82-33).

The applicant provided a PGP and supporting materials in letters dated January 17 and September 18, 1984, and March 29, 1985. In addition to an introduction, the PGP provided with the January 17, 1984, letter included the following sections:

- (1) Plant-Specific Technical Guidelines
- (2) Writer's Guide for EOPs
- (3) User's Guide
- (4) EOP Verification
- (5) EOP Validation
- (6) Training

Because the Seabrook Station was the verification and validation site for WOG's Revision 1 to the generic guidelines and because the PGP did not fully describe the EOP efforts at Seabrook Station, the following documents were included in the review materials by the September 18, 1984, letter:

(1) WCAP-10599, Emergency Response Guidelines Validation Program, WOG letter OG-129, dated August 14, 1984

(2) Seabrook Station Procedure, AQ1.002, "Station Operating Procedures"

As additional information, the applicant also provided the following with the March 29, 1985 letter:

(1) PGP amendments

- (a) Writer's Guide
- (b) Plant-Specific Guidelines
- (c) Training

(2) Station Operating Procedure OS1300, Rev. 00 (AQ1.002A), "Generation of Emergency Response Procedures" (revised Writer's Guide)

The discussion of the staff review of these materials is organized according to the four areas of the PGP required by the generic letter.

The materials addressing the plant-specific technical guidelines were reviewed to determine if they provided acceptable methods to meet the objectives of NUREG-0899. The applicant described the methods used to develop EOPs based on Revision 1 to the WOG generic guidelines. The staff recently approved these guidelines for implementation in a letter from D. G. Eisenhut (NRC) to J. J. Sheppard (WOG) dated December 27, 1984. As the reference plant for the WOG validation effort, the generic guidelines were made directly applicable to the Seabrook Station as part of that effort. In the amendment to the PGP, the applicant stated that neither his review nor the vendor's identified any safety-significant deviations from the generic guidelines in the Seabrook procedures. Therefore, the applicant's plant-specific technical guideline program has met the objectives of NUREG-0899 and is acceptable.

As part of the technical basis for operator procedures, Supplement 1 to NUREG-0737 also required an analysis of the operators' tasks to identify the information and controls necessary to support emergency operations. The information and control needs are used in the development of EOPs and in the detailed control room design review. The applicant's submittals and the staff's review and approval are discussed in Section 18 of the SER.

The materials addressing the Writer's Guide were reviewed to determine if they provided acceptable methods for accomplishing the objectives stated in NUREG-0899. The applicant stated that the purpose of the Writer's Guide is to provide administrative and technical guidance on the preparation of the EOPs. The guide provides instructions for writing EOPs, including format, action steps, mechanics of style, and examples. The Writer's Guide states that each EOP shall provide the basic purpose and scope for the procedure, symptoms or entry conditions, and in the body of the procedure, contingent operator actions based on interpretation of parameters and conditions. The operator action steps are described as being in a two-column format with the left-hand column for instructions and expected plant responses, and the right-hand column for contingency actions. With the revision of the Writer's Guide published as Station Operating Procedure OS1300, Revision 00, the Seabrook Writer's Guide provides acceptable methods for accomplishing the objectives stated in NUREG-0899. Use of the Writer's Guide should

help ensure that the EOPs are usable, accurate, complete, readable, convenient to use, and acceptable to control room operators.

The descriptions of the validation and verification programs were reviewed to determine if they address the objectives in NUREG-0899, that is, to establish the accuracy of information and instructions, to determine that the procedures can be accurately and efficiently carried out, and to demonstrate that the procedures are adequate to mitigate transients and accidents.

As the site of the WOG generic guidelines validation effort, the Seabrook-specific EOPs were the bases for the generic evaluation. The evaluation included checking the EOPs with their source documents, exercising the EOPs using the Seabrook-specific simulator, observing operators' performance by multidisciplined observation teams, and debriefing of the operating crews and observation teams for each scenario. These efforts resulted in many recommendations for Seabrook's Writer's Guide, User's Guide, and training program but identified no safety-significant technical deficiencies.

The WOG generic validation was completed before the applicant's submittal of the PGP. As a result, most of the recommendations had already been incorporated and the PGP included both before and after versions of the Writer's Guide and User's Guide. The training program was revised after submittal of the PGP and was described in the March 29, 1985, submittal.

On the basis of the documentation of the validation and verification program results, the program appears to have adequately incorporated the guidance of the Writer's Guide and the plant-specific technical guidelines and will guide the operator in mitigating the consequences of accidents and transients. Therefore, the staff finds it acceptable.

The applicant's amended description of the training program on the EOPs was reviewed to determine if it addressed the objectives in NUREG-0899. The description included the use of classroom instruction, control room walkthroughs, and plant-specific simulator exercises. The applicant also provided some training materials in the form of student handouts. On the basis of the review of the program description and the student handouts, the implementation of this training program should provide operators with the philosophy of the EOP approach, the strategy and technical basis of the EOPs, and a working knowledge of the content of the EOPs sufficient to enable them to execute the procedures under operational conditions. Although the training program on the EOPs was described as part of the overall operator training program, the applicant did not commit to train all operators on the upgraded EOPs. The staff will confirm that all operators have received the described training before licensing.

Contingent on confirmation of the above item, the staff concludes that the training program meets the guidance of NUREG-0899 and should provide assurance that the EOPs will adequately guide the operators in mitigating the consequences of accidents and transients. Therefore, the training program is acceptable.

On the basis of its review, the staff concludes that the PGP for Seabrook Station, Units 1 and 2, meets the requirements of Supplement 1 to NUREG-0737 and provides acceptable methods for accomplishing the objectives of NUREG-0899 for the technical guidelines, Writer's Guide, and the verification and validation and training programs. The staff will confirm that the applicant adequately

completes the operator training. Future changes to the PGP should be made in accordance with 10 CFR 50.59.

15 ACCIDENT ANALYSES

15.4 Reactivity and Power Distribution Anomalies

15.4.3 Rod Cluster Control Assembly Malfunctions

SER Section 15.4.3 indicated that a potential controller problem existed for the dropped control rod event that could lead to the imposition of operating restrictions. It also indicated that it was anticipated that a detailed analysis would show that if the problem should occur, thermal limits would not be exceeded, but that this analysis had not been submitted. Since then Westinghouse has developed a solution for the problem via a new methodology for analyzing the event and has documented it in a topical report (WCAP-10297P). This report and its methodology have been evaluated by the staff and approved. The solution requires a reactor-cycle-specific analysis showing that departure from nucleate boiling (DNB) limits will not be exceeded. This analysis has been done for Seabrook (letter from applicant dated January 7, 1985), and the Seabrook FSAR was revised (Amendment 54 to the FSAR) to include a discussion of this analysis and the results for Cycle One operation that indicate that DNB limits will be met for this cycle. Each future reload cycle will require similar cycle-specific analysis as part of the normal reload analysis. This closes the open item on the dropped control rod event.

15.9 TMI Action Plan Requirements

15.9.5 II.K.2.17 Potential for Voiding in the Reactor Coolant System During Transients

Westinghouse has performed a study that addresses the potential for void formation in Westinghouse-designed nuclear steam supply systems during natural circulation cooldown/depressurization transients. This study has been submitted to the NRC by the Westinghouse Owners Group (Jurgensen, April 20, 1981). The results of this study have been reviewed and accepted by the staff, and this item is closed.

17 QUALITY ASSURANCE

17.1 General

As a result of the applicant's submittal of Amendment 53, which affects the staff's previous SER organizational description for the operation of Seabrook Station, it is necessary that Sections 17.2, "Organization," and 17.4, "Conclusions," and the quality assurance (QA) organizational chart (Figure 17.1) be replaced by the following.

17.2 Organization

The structure of the organization responsible for the operation of Seabrook Station and for the establishment and execution of the operations phase QA program is shown in Figure 17.1. The Public Service Company of New Hampshire (PSNH) President has delegated to the New Hampshire Yankee (NHY) President (a division within Public Service Company of New Hampshire) the direct responsibility for operation, maintenance, modification, and refueling of Seabrook Station, Units 1 and 2. The NHY Vice President of Nuclear Production reports through the NHY Senior Vice President to the NHY President and is responsible for the operation and operational support of Seabrook Station, Units 1 and 2, including QA functions. The Seabrook Station Manager, the Nuclear Quality Manager, and other support groups report to the NHY Vice President of Nuclear Production. The Nuclear Quality Manager is in charge of the Quality Assurance Department, which consists of a Quality Assurance Section, an Audit and Evaluation Section, and a Quality Control Section.

The Nuclear Quality Manager has been delegated the authority for establishing QA program requirements, verifying implementation, and measuring the overall effectiveness of the QA program. The Nuclear Quality Manager and staff (which currently consists of 28 persons) have the responsibility and authority to stop unsatisfactory work and control further processing, delivery, or installation of nonconforming material.

The QA organization has the authority to (1) identify quality problems; (2) initiate, recommend, or provide solutions through designated channels; (3) verify implementation of solutions; and (4) stop unsatisfactory work and control further processing, delivery, or installation of nonconforming items. The QA organization is responsible for (1) reviewing and concurring with documents affecting safety; (2) verifying inplant activities by surveillance inspections and examinations; (3) evaluating suppliers before contracts are awarded; (4) inspecting suppliers' facilities; (5) ensuring that personnel qualifications are current and applicable to the work being performed; (6) ensuring that corrective actions are effective and accomplished in a timely manner; and (7) conducting (a) internal audits of maintenance, modification, and operations activities and (b) external audits of suppliers' activities.

The Seabrook Station Manager reports to the Vice President of Nuclear Production and is responsible for (1) ensuring the safe, reliable, and efficient operation of the plant and (2) ensuring that quality-affecting activities are conducted

in accordance with the QA program. Disputes on any QA matter that arise between QA/Quality Control and other departments are resolved by the management of the involved organizations or, if necessary, with the NHY Vice President of Nuclear Production.

17.3 Quality Assurance Program

The evaluation in the SER on this subject is still valid.

The staff has evaluated Section 17.1 "Quality Assurance During Design and Construction," of Amendment 53 to the FSAR and the applicant's letters of August 31 and October 31, 1984, which discuss the establishment of a new division within PSNH called New Hampshire Yankee (NHY). NHY has been delegated the responsibility for the design, construction, and operation of the Seabrook Station. Under this new organizational arrangement, PSNH continues to delegate to the Yankee Atomic Electric Company (YAEC) responsibility for establishing and implementing the QA program for the construction of Seabrook Station. Also, PSNH continues to retain ultimate responsibility for this program. This arrangement has been acceptable in the past and complies with Criterion I of 10 CFR 50, Appendix B, which permits PSNH to delegate to others, such as contractors, agents, or consultants, the work of establishing and executing the QA program or any part thereof, providing PSNH retains responsibility.

The lines of responsibilities and authority over quality assurance are adequately described in FSAR Sections 1.4, "Identification of Agents and Contractors," 13.1.1.5, "Construction and Construction/Operation Interface," and 17.1, "Quality Assurance During Design and Construction," which includes Section 17.1.1.1(a), "Authority, Responsibilities, and Duties." From these descriptions, it is clear that QA personnel within YAEC who are responsible for establishing and implementing the Seabrook QA programs report to the YAEC Construction QA Manager. The YAEC Construction QA Manager is assigned exclusively to the Seabrook project and is responsible for interfacing with the NHY Vice President in charge of Administrative Services. United Engineers and Constructors and Westinghouse Electric Corporation QA programs are extensions of the YAEC QA program and have been reviewed and accepted by YAEC. YAEC maintains control of these and other contractors by means of audits, surveillance, surveys, investigations, and reviews.

The staff concludes that the establishment of the NHY division and the delegated responsibilities to this division from PSNH have not diluted or weakened the previously approved QA program for design and construction. Therefore, the NHY organization and the QA program for design and construction are acceptable for the remaining construction activities at Seabrook Station.

17.4 Conclusion

On the basis of its detailed review and evaluation of the QA program as described in FSAR Section 17.2, the staff concludes:

- (1) The organizations and persons performing QA functions have the required independence and authority to effectively carry out the QA program without undue influence from those directly responsible for cost and schedules.

- (2) The QA program, including the list of safety-related structures, systems, and components to which it applies as indicated in FSAR Section 17.2.2.2, describes requirements, procedures, and controls that, when properly implemented, comply with the requirements of Appendix B to 10 CFR 50 and with the acceptance criteria in SRP Section 17.2.

Accordingly, the staff concludes that the applicant's description of the QA program is in compliance with applicable NRC regulations.

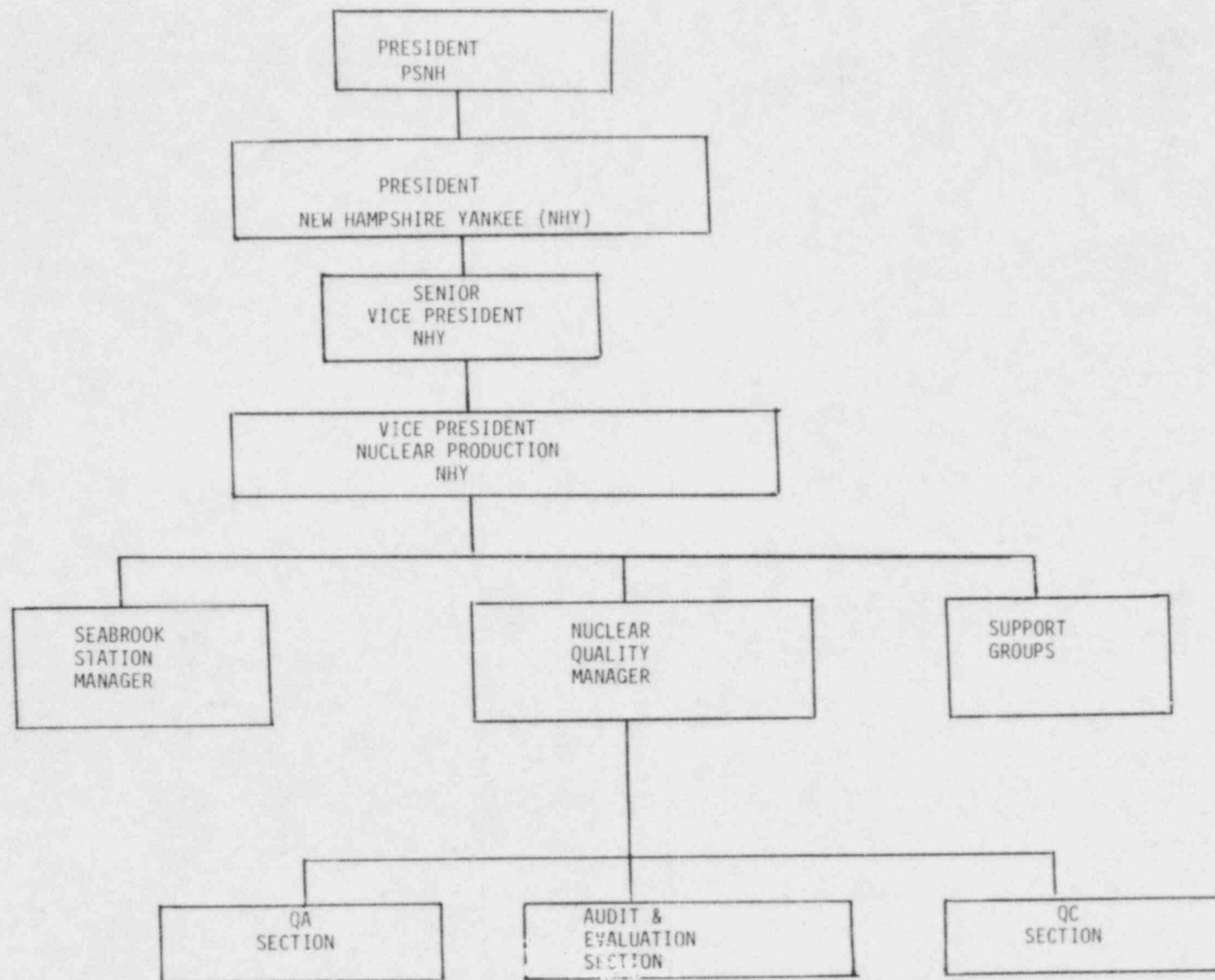


Figure 17.1 Public Service Company of New Hampshire operations quality assurance organization

18 HUMAN FACTORS ENGINEERING

Item I.D.1, "Control Room Design Reviews," of Task I.D, "Control Room Design," of the NRC Task Action Plan (NUREG-0660) developed as a result of the accident at Three Mile Island Unit 2 states that operating licensees and applicants for operating licenses will be required to perform a detailed control room design review (DCRDR) to identify and correct design discrepancies. The objective, as stated in NUREG-0660, is to improve the ability of nuclear power plant control room operators to prevent or cope with accidents if they occur by improving the information provided to them. Supplement 1 to NUREG-0737 confirmed and clarified the DCRDR requirement in NUREG-0660. As a result of Supplement 1 to NUREG-0737, each applicant or licensee is required to conduct a DCRDR on a schedule negotiated with NRC.

NUREG-0700 describes four phases of the DCRDR to be performed by the applicant and licensee. The phases are (1) planning, (2) review, (3) assessment and implementation, and (4) reporting.

Criteria for evaluating each phase are contained in Section 18.1, Revision 0, and Appendix A to Section 18.1 of NUREG-0800.

Supplement 1 to NUREG-0737 requires each applicant and licensee to submit a program plan that describes how the following elements of the DCRDR will be accomplished:

- (1) establishment of a qualified multidisciplinary review team
- (2) function and task analysis to identify control room operator tasks and information and control requirements during emergency operations
- (3) a comparison of display and control requirements with a control room inventory
- (4) a control room survey to identify deviations from accepted human factors principles
- (5) assessment of human engineering discrepancies (HEDs) to determine which HEDs are significant and should be corrected
- (6) selection of design improvements
- (7) verification that selected design improvements will provide the necessary correction
- (8) verification that improvements will not introduce new HEDs
- (9) coordination of control room improvements with changes from other programs such as the safety parameter display system (SPDS), operator training, RG 1.97 instrumentation, and upgrade of emergency operating procedures

Supplement 1 to NUREG-0737 also requires each applicant and licensee to submit a summary report at the end of the DCRDR. The report should describe the proposed control room changes and implementation schedules and provide justification for leaving safety-significant HEDs uncorrected or partially corrected.

The NRC staff will evaluate the organization, process, and results of each DCRDR. The evaluation of the applicant's and licensee's DCRDR effort will consist of the following, as described in NUREG-0800:

- (1) an evaluation of the program plan report submitted by the licensee or applicant
- (2) a visit to some of the plant sites to audit the progress of the DCRDR programs
- (3) an evaluation of the licensee's or applicant's DCRDR summary report
- (4) a possible pre-implementation audit
- (5) preparation of a safety evaluation report that will present the results of the NRC evaluation

Significant HEDs should be corrected. Improvements that can be accomplished with an enhancement program should be done promptly.

In accordance with Supplement 1 to NUREG-0737, the applicant is required to complete the DCRDR before licensing. The DCRDR process for Seabrook is nearly complete. On May 12, 1982, the applicant formally submitted a preliminary program plan report to NRC entitled "Seabrook Station Control Room Design Review Preliminary Report." Following that, the applicant conducted a control room human factors design review. In response to the NRC requirements in Supplement 1 to NUREG-0737, a report entitled "Seabrook Station Control Room Design Review" was transmitted to the NRC by letter dated April 14, 1983.

A human factors engineering in-progress audit of the Seabrook Station control room design review was performed at the Seabrook site on July 26 through July 28, 1983, by an NRC team assisted by consultants from Lawrence Livermore National Laboratory (LLNL). Following the onsite audit, the applicant submitted supplemental information in a letter dated August 10, 1983, to clarify several concerns identified during the audit.

A detailed audit report was transmitted to the applicant on January 12, 1984. The conclusions of that report indicated that the applicant is conducting a DCRDR that substantially meets the requirements of Supplement 1 to NUREG-0737 and the guidelines of NUREG-0700, except for the following:

- (1) system function and task analysis
- (2) comparison of display and control requirements determined by function and task analysis with a control room inventory
- (3) implementation schedule definitions

The function and task analysis documentation did not show evidence of a systematic determination of information and control capability requirements (e.g.,

information parameter type, dynamic range, accuracy, frequency, control capability type, precision, gain, and criticality), which could be objectively compared to the instrumentation available in the control room. Information requirements must be based on the needs of the operator to successfully perform the described task, not on existing instrumentation that is available in the control room.

Some items were not included in the applicant's reports because they were not ready for review.

These items from the applicant's letter of August 10, 1983, are described below:

- (1) Item 13 of the letter lists seven categories of items that will be reviewed and for which information will be provided to the NRC 4 months before the loading of fuel:
 - (a) video alarm system, SPDS, and associated computer aids to the operator
 - (b) hard-wired annunciators
 - (c) radiation monitoring system
 - (d) lighting
 - (e) control room access and architecture relative to supervision, storage of emergency equipment, escape, limiting access of unauthorized persons, rest rooms, and eating facilities
 - (f) storage of operating procedures and keys, tagging, shift turnover, and other administrative procedures
 - (g) remote shutdown panel
- (2) Item 13 lists two categories of items for which preliminary reviews have been accomplished, HEDs have been identified, and acceptable resolutions and implementation schedules have been reviewed by the staff. A final review will be accomplished after a heating and cooling cycle has been experienced.
 - (a) auditory signals, communications within and outside the control room, acoustic noise
 - (b) heating, ventilation, and air conditioning
- (3) Item 11 states that the list of abbreviations under development will be distributed to appropriate potential users, but does not describe how its proper use will be ensured. A plan to ensure its use should be submitted to the NRC for review 120 days before the loading of fuel.
- (4) Item 15 describes two studies requested by the NRC, which must be submitted 120 days before fuel load. These are a tabulation of colors, how they are used, and in what contexts, and a tabulation of lights used in control room indicators with details regarding their redundancy, testability, and replacement procedures.

In addition, Appendices A, C, and D of the audit report contained specific items not yet evaluated or HEDs not fully resolved.

By letter dated July 30, 1984, the applicant provided responses to all unresolved items except for describing a procedure for ensuring the consistent use of abbreviations in procedures, on the main control board, and on the cathode ray tubes (CRTs). The staff disagreed with some responses, and some responses did not fully resolve the issues. A meeting was held on October 12, 1984, in Bethesda, Maryland, between the NRC staff and the applicant's representatives. At this meeting the applicant committed to provide additional information on 19 specific items. By letter dated January 7, 1985, the applicant submitted additional information.

The first item deals with the system function and task analysis to determine the required characteristics for instruments and controls associated with emergency operations. The applicant provided a draft sample of his documentation format and analysis method for staff review and has committed to complete the task analysis on all emergency operations. It is understood that this will be accomplished by July 1, 1985. The staff has reviewed this sample and agrees that the method and format appear satisfactory. However, it is the content and results of a task analysis that are critical, and the letter of January 7, 1985, indicates that the sample is not complete.

The applicant's submittal of January 7, 1985, does not discuss the comparison of the control room inventory with the task analysis results to determine availability and suitability of instrumentation to accomplish all emergency operations. The staff expects that resolutions of any HEDs resulting from this process will be determined and reported to the NRC at least 120 days before the loading of fuel. Implementation of any corrective actions resulting from this process should be on a schedule agreed on with the NRC.

The staff agrees with the applicant's responses to Items 2 through 9 of the January 7, 1985, submittal. Item 2 deals with the applicant's final evaluation of the control room environment after achieving commercial operation. The results of this evaluation will be reviewed by the NRC when it is completed.

Items 10 through 19 of the January 7, 1985, submittal include reviews that are incomplete. The applicant has committed to complete the reviews and determine resolutions for any resulting HEDs 120 days before the loading of fuel. Although this schedule is acceptable to the staff, it appears to be excessive in light of the original date (August 10, 1983) on which the commitments were made. These 10 items will remain open until satisfactory resolutions are determined and an implementation schedule for corrective actions is approved by the staff.

Item 16 refers to four HEDs involving the annunciator system. It is the staff's position that annunciator system HEDs should be corrected before startup so that operators can be trained in the alarm system changes and modifications after startup will be minimized. The applicant has committed to evaluate this issue and submit the results to the NRC 120 days before the loading of fuel. This does not appear to be a prudent schedule in light of corrective actions that might be required.

Conclusions

The applicant is conducting a DCRDR for Seabrook Station that substantially meets the requirements of Supplement 1 to NUREG-0737 and the guidance in NUREG-0700. Several items of the DCRDR will need to be completed and the results submitted to enable the staff to determine acceptability of the DCRDR. Except for these incomplete items (described below), all other control room improvements and implementation schedules, as committed to by the applicant in the letters dated August 10, 1983, July 30, 1984, and January 7, 1985, are acceptable to the staff:

- (1) The function and task analysis method and documentation format, as provided in the draft sample attached to the January 7, 1985, submittal by the applicant, should satisfy the requirements of Supplement 1 to NUREG-0737. This analysis will be performed for all emergency operations and the results made available to the NRC for confirmatory review by July 1, 1985. Any HEDs regarding instrumentation availability and suitability resulting from a comparison of the control room inventory with the task analysis results should have proposed resolutions and a schedule for implementing any corrective actions reported to the NRC at least 120 days before the loading of fuel.
- (2) Final evaluation of the control room environment (temperature, humidity, airflow, acoustic noise, and auditory signals) will be completed and reported to the NRC for confirmatory review within 1 year after commercial operation is achieved. Should any HEDs be identified, proposed resolutions and a schedule for implementing corrective actions should be included in the report.
- (3) The following items should be submitted to the NRC for confirmatory review at least 120 days before the loading of fuel:
 - (a) a plan or administrative procedure and acceptable implementation schedule to ensure that consistent abbreviations are used throughout the control room (i.e., main control board, CRTs, and procedures)
 - (b) a description of the method for ensuring that procedural documents are returned to storage in proper order
 - (c) an acceptable schedule for the addition of hierarchical labeling
- (4) The following are areas in which the review is still incomplete. Each review must be completed and proposed resolutions of any HEDs resulting from the review must be submitted to the NRC at least 120 days before the loading of fuel, along with an acceptable implementation schedule for correcting the HEDs:
 - (a) radiation monitoring system panels
 - (b) main steam isolation valve panel
 - (c) status light matrix - the staff disagrees with the current proposed resolution as stated in the applicant's letter of July 30, 1984 (Appendix C, Item 5.29)

- (d) tabulation of the use of color as a code in all display contexts
- (e) fire panel
- (f) clarification as to how demarcation and labeling assist operators in reading the steam dump meter located 15 ft from the controls
- (g) correction of annunciator system HEDs before startup
- (h) movable bench to reach atmospheric dump valve controllers unacceptable to the staff

Although Items 1, 2 and 3 above are confirmatory in nature in that the staff understands what the applicant is going to accomplish, any resulting HEDs, proposed resolutions, and acceptable schedules for corrective actions should be submitted for staff review. Item 4 requires detailed review by the staff, since several studies must be accomplished by the applicant and results cannot be predicted.

For the staff to complete its review of the Seabrook Station DCRDR in accordance with the requirements of Supplement 1 to NUREG-0737, the applicant is required to provide an acceptable submittal that addresses the items identified above.

APPENDIX A
CONTINUATION OF CHRONOLOGY
OF RADIOLOGICAL REVIEW

November 10, 1982	Letter from applicant transmitting response to request for information.
November 29, 1982	Letter from applicant summarizing certain open items by Effluent Treatment Systems and Radiological Assessment Branches.
April 14, 1983	Letter from applicant providing response to Generic Letter 82-33.
April 25, 1983	Letter from applicant transmitting information on flow measurement uncertainty (proprietary and nonproprietary versions).
June 3, 1983	Generic Letter 83-22 - Safety Evaluation of "Emergency Response Guidelines."
June 7, 1983	Letter from applicant transmitting "Containment Ultimate Capacity of Seabrook Station Units 1 and 2 for Internal Pressure Loads."
June 15, 1983	Letter from applicant transmitting response to Meteorological and Effluent Systems Branch Draft SER items.
June 15, 1983	Letter from applicant forwarding response to SER Section 11.5.2.
June 20, 1983	Letter from applicant transmitting revised report, "NUREG-0612, Control of Heavy Loads."
June 21, 1983	Letter from applicant transmitting "An Evaluation of the Cost and Schedule Estimate of the Seabrook Nuclear Project," by Management Analysis Company.
June 24, 1983	Letter from applicant transmitting revised response to SER Outstanding Issue 1 (from Meteorological and Effluent Treatment Systems Branch).
June 24, 1983	Letter to applicant advising that staff's estimate for Unit 1 fuel loading is first quarter of 1986.
June 27, 1983	Letter from applicant regarding emergency classification system.

July 5, 1983	Generic Letter 83-26 - Clarification of Surveillance Requirements for Diesel Fuel Impurity Level Tests.
July 6, 1983	Generic Letter 83-27 - Surveillance Intervals in Standard Technical Specifications.
July 7, 1983	Letter from applicant forwarding "Seabrook Station Control Room Design Review."
July 8, 1983	Letter to applicant transmitting requests for additional information.
July 8, 1983	Generic Letter 83-28 - Required Actions Based on Generic Implications of Salem ATWS Event.
July 11, 1983	Letter from applicant transmitting Amendment 49 to Final Safety Analysis Report (FSAR).
July 12, 1983	Meeting with applicant to discuss reinforcing bar splicing.
July 13, 1983	Issuance of Supplement 2 to SER.
July 13, 1983	Meeting with applicant to discuss alternative designs for stone revetment.
July 21, 1983	Generic Letter 83-30 - Deletion of Standard Technical Specification Surveillance Requirement 4.8.1.1.2.d.6 for Diesel Generator Testing.
July 22, 1983	Letter to applicant regarding detailed control room design review program plan submittal.
July 26-29 1983	Control room design review audit.
July 27, 1983	Letter from applicant transmitting 1982 Annual Financial Reports.
August 5, 1983	Letter from applicant transmitting correction to FSAR Section 13.2.
August 5, 1983	Letter from applicant transmitting information regarding administrative and procedural controls for radiological monitoring, sampling, and analysis of process and effluent streams.
August 9, 1983	Board Notification 83-111 - Summary Board Notification.
August 10, 1983	Letter from applicant forwarding revised Sections I-IV to July 7, 1983, submittal.
August 12, 1983	Letter from applicant transmitting "Environmental Qualification of Electrical Equipment Important to Safety."

August 12, 1983	Letter from applicant transmitting response to Request for Additional Information (RAI) 260.28 and SER Outstanding Issue 18.
August 18, 1983	Letter to applicant advising of new project manager.
August 23, 1983	Board Notification 83-121 - Allegation Concerning Seabrook Station.
August 25, 1983	Board Notification 83-118 - Materials Supplied to Nuclear Industry Companies by Ray Miller Inc. and Tube-Line Corporation.
August 25, 1983	Letter from applicant transmitting "Seismological and Geological Studies, Miramichi Area, New Brunswick, and Central New Hampshire."
August 30, 1983	Letter from applicant transmitting description of proposed redesign of flood protection features.
August 31, 1983	Letter to applicant forwarding "Draft Technical Evaluation Report on Control of Heavy Loads."
September 6, 1983	Letter from applicant requesting extension for responding to Section 2.2.2 of Generic Letter 83-28.
September 8, 1983	Letter from applicant transmitting Amendment 50 to FSAR.
September 8, 1983	Letter from applicant transmitting revised response to RAI 640.55
September 9, 1983	Letter from applicant withdrawing request to utilize the Dayton bar grip system.
September 12, 1983	Letter from applicant forwarding revised response to RAI 430.14.
September 23, 1983	Meeting with applicant to review closeout status of SER outstanding issues.
September 29, 1983	Board Notification 83-66A - Westinghouse Rod Drop Issue.
October 4, 1983	Board Notification 83-151 - Westinghouse ECSS Actuation Logic.
October 5, 1983	Letter from applicant advising of reduced expenditures for Unit 2.
October 6, 1983	Board Notification 83-128A - Draft Test Report on Qualification Test Program of Class 1E Solenoid Valves.
October 14, 1983	Letter to applicant forwarding "Control of Heavy Loads at Nuclear Power Plants, Seabrook Units 1 and 2 (Phase I)."

October 17, 1983	Letter from applicant forwarding Amendment 51 to application - requesting transfer of ownership share from Vermont Electric Cooperative, Inc. to Vermont Electric Generation and Transmission Cooperative, Inc.
October 18, 1983	Letter to applicant regarding clarification of required actions based on generic implications of Salem anticipated transients without scram (ATWS) events (Unit 1).
October 18, 1983	Letter to applicant regarding clarification of required actions based on generic implications of Salem ATWS events (Unit 2).
October 19, 1983	Generic Letter 83-33 - NRC Positions on Certain Requirements of Appendix R to 10 CFR 50.
October 21, 1983	Letter to applicant regarding quality assurance list.
October 25, 1983	Letter to applicant requesting update of master list to show seismic and dynamic qualification status.
October 28, 1983	Letter from applicant advising that there is no need to install electric heaters in diesel generator air intake plenum.
October 31, 1983	Generic Letter 83-38 - NUREG-0965, "NRC Inventory of Dams."
November 2, 1983	Generic Letter 83-35 - Clarification of TMI Action Plan Item II.K.3.31.
November 4, 1983	Letter from applicant providing response to Generic Letter 83-28.
November 7, 1983	Letter to applicant confirming telephone agreement that Caseload Forecast Panel visit will be held week of February 27, 1984.
November 8, 1983	Board Notification 83-121A - Allegations Concerning Seabrook.
November 14, 1983	Letter to Yankee Atomic regarding request for withholding information submitted August 25, 1983.
November 22, 1983	Letter from applicant regarding solid radwaste handling system.
December 1, 1983	Letter from applicant regarding electrical interconnections between redundant divisions.
December 2, 1983	Generic Letter 83-32 - NRC Staff Recommendations Regarding Operator Action for Reactor Trip and ATWS.

December 5, 1983	Letter from applicant forwarding revised responses to RAIs 430.67 and 460.69.
December 15, 1983	Letter from applicant transmitting information on control of heavy loads.
December 16, 1983	Generic Letter 83-41 - Fast Cold Starts of Diesel Generators.
December 16, 1983	Letter from applicant forwarding response to facility staffing survey.
December 16, 1983	Letter from applicant forwarding "Cadmold Splices for Construction Access Openings."
December 19, 1983	Generic Letter 83-43 - Reporting Requirements of 10 CFR Parts 50.72 and 50.73.
December 19, 1983	Letter from applicant on decay heat removal.
December 19, 1983	Generic Letter 83-42 - Clarification of Generic Letter 81-07 on Heavy Loads.
December 20, 1983	Letter from applicant regarding containment high-range radiation monitor location.
December 20, 1983	Letter from applicant regarding spent fuel pool temperature reduction.
December 20, 1983	Generic Letter 83-44 - Availability of NUREG-1021, "Operator Licensing Examiner Standards."
December 21, 1983	Generic Letter 83-40 - Best Estimate of Need for Operator Licensing Examinations for FY84-87.
January 5, 1984	Board Notification 84-004 - Environmental Qualification Briefing of Chairman by Sandia.
January 5, 1984	Generic Letter 84-01 - Use of Terms "Important to Safety" and "Safety Related."
January 5, 1984	Letter from applicant transmitting response to 230 Series RAIs.
January 6, 1984	Letter from applicant regarding removal of proprietary status for document submitted August 25, 1983.
January 6, 1984	Letter from applicant forwarding revision to response to RAI 430.81.
January 6, 1984	Generic Letter 84-02 - Notice of Meeting Regarding Facility Staffing.

January 12, 1984	Letter to applicant forwarding report on results of in-progress audit of Seabrook detailed control room design review.
January 13, 1984	Generic Letter 84-03 - Availability of NUREG-0933, "General Safety Issues Prioritization."
January 17, 1984	Letter from applicant transmitting "Procedures Generation Package for Seabrook Station."
January 18, 1984	Board Notification 84-011 - NRC Use of Terms "Important to Safety" and "Safety Related."
January 27, 1984	Issuance of Amendment 6 to construction permits authorizing transfer of ownership share from Vermont Electric Cooperative, Inc. to Vermont Electric Generation and Transmission Cooperative, Inc.
January 30, 1984	Letter from applicant transmitting "Seabrook Station Probabilistic Safety Assessment."
February 1, 1984	Letter to applicant advising that Caseload Forecast Panel visit will be rescheduled to week of March 12, 1984.
February 1, 1984	Generic Letter 84-04 - Safety Evaluation of Westinghouse Topicals Dealing With Elimination of Postulated Pipe Breaks in PWR Primary Main Loops.
February 6, 1984	Letter to applicant transmitting RAI on flow measurement uncertainties.
February 8, 1984	Letter from applicant regarding Markowitz allegation.
February 13, 1984	Letter to applicant acknowledging receipt of January 6, 1984, letter regarding removal of proprietary status.
February 13, 1984	Board Notification 84-032 - Additional Information on Environmental Qualification.
February 13, 1984	Letter to applicant regarding March 13-15, 1984, Caseload Forecast Panel visit.
February 13, 1984	Letter from applicant transmitting revision to radiological emergency plan.
February 15, 1984	Letter from applicant transmitting Amendment 52 to FSAR.
February 16, 1984	Board Notification 84-33 - Task Action Plan for USI A-17, "Systems Interaction Program."
February 20, 1984	Letter to applicant regarding deletion of home telephone numbers, unlisted utility numbers, etc. from emergency plans.

March 1, 1984	Letter to applicant regarding concerns on limited operating experience of operating crews and development of acceptable experience profile.
March 13-14, 1984	Caseload Forecast Panel visit.
March 14, 1984	Board Notification 84-050 - Environmental Qualification: Commission Policy Statement and Proposed Rulemaking.
March 14, 1984	Letter from applicant transmitting revision to radiological emergency plan.
March 14, 1984	Letter from applicant forwarding news release regarding March 1, 1984, meeting of joint owners of Seabrook, presenting revised cost estimates and completion schedules.
March 15, 1984	Letter from applicant requesting approval to implement Subsection NF, Paragraphs NF-3324.5 and NF-4427 of ASME Code, Section III.
March 20, 1984	Letter from applicant forwarding information on operator shift experience.
March 21, 1984	Letter from applicant forwarding data package presented to Caseload Forecast Panel representatives.
March 30, 1984	Letter from applicant forwarding marked-up Technical Specifications.
April 2, 1984	Generic Letter 84-05 - Operator Licensing Examiner Standards.
April 2, 1984	Letter to applicant forwarding integrated design inspection report for Unit 1.
April 3, 1984	Letter to applicant advising of acceptability of use of ASME Code, Section III, 1983 Edition, Subsection NF, Paragraphs NF-3324.5 and NF-4437.
April 3, 1984	Letter from applicant regarding Markowitz allegation (Board Notification 83-121A.)
April 4, 1984	Generic Letter 84-08 - Interim Procedures for NRC Management of Plant-Specific Backfitting.
April 16, 1984	Letter from applicant regarding NUREG-0737, Item II.B.3, "Post-Accident Sampling Capability."
April 26, 1984	Generic Letter 84-10 - Administration of Operating Tests Prior to Initial Criticality.
April 30, 1984	Generic Letter 84-12 - Compliance With 10 CFR 61 and Implementation of Radiological Effluent Technical Specifications and Attendant Process Control Program.

May 1, 1984	Issuance of order extending construction completion dates to June 30, 1986, and October 31, 1988.
May 3, 1984	Generic Letter 84-13 - Technical Specifications for Snubbers.
May 7, 1984	Letter to applicant forwarding request for additional information on environmental qualification of electrical equipment important to safety.
May 7, 1984	Letter from applicant regarding schedule for integrated design inspection response.
May 7, 1984	Letter to applicant requesting information regarding probabilistic risk assessment.
May 8, 1984	Generic Letter 84-09 - Recombiner Capability.
May 11, 1984	Generic Letter 84-14 - Requalification Training Program.
May 18, 1984	Letter from applicant regarding May 7, 1984, letter on probabilistic safety assessment.
May 21, 1984	Board Notification 84-099 - Integrated Design Inspection.
May 31, 1984	Letter from applicant forwarding information on flood and erosion protection features - civil plans.
June 1, 1984	Letter from applicant forwarding response to SER Outstanding Issue 9 on fracture toughness of reactor coolant pressure boundary and secondary system materials.
June 20, 1984	Board Notification 84-104 - Westinghouse ECCS Actuation Logic Review.
June 20, 1984	Letter from applicant forwarding information on electrical interconnections between redundant divisions.
June 21, 1984	Letter from applicant transmitting Unit 1 preservice inspection program plan: reactor vessel and balance of plant.
June 22, 1984	Letter from applicant requesting approval to implement ASME Code Case N-368.
June 27, 1984	Generic Letter 84-16 - Adequacy of On-Shift Operating Experience for Near-Term Operating License Applicants.
June 29, 1984	Board Notification 84-123 - Board Notification Regarding Westinghouse Reactor Coolant Pump Seals.
June 29, 1984	Letter from applicant in response to Generic Letter 83-28.

June 29, 1984	Letter from applicant advising that plant-specific design and test information will be submitted by December 31, 1984.
July 2, 1984	Generic Letter 84-15 - Proposed Staff Action To Improve and Maintain Diesel Generator Reliability.
July 3, 1984	Generic Letter 84-17 - Annual Meeting To Discuss Recent Developments Concerning Operator Training, Qualifications and Examinations.
July 6, 1984	Generic Letter 84-18 - Filing of Applications for Licenses and Amendments.
July 11-12 1984	Meeting to hear applicant's discussion of revisions to fire protection report.
July 12, 1984	Letter from applicant confirming August 9, 1984, meeting to discuss status of engineering, licensing, construction, startup and operations, quality assurance, and management actions taken since March 1, 1984, and future schedule.
July 12, 1984	Letter from applicant regarding cable tray loading criteria for heavy power cables.
July 30, 1984	Letter from applicant forwarding response to SER Outstanding Issue 19.
July 30, 1984	Letter to applicant advising of acceptability of use of ASME Code Case N-368.
August 6, 1984	Generic Letter 84-19 - Availability of Supplement 1 to NUREG-0933, "Prioritization of Generic Safety Issues."
August 9, 1984	Meeting of NRC management and applicant management to hear applicant's presentation of status of engineering, construction, licensing, startup operations, and quality assurance and plans for transferring responsibility for construction and operation of Unit 1 to management's agent.
August 9, 1984	Letter from applicant regarding alternate pipe break design criteria.
August 20, 1984	Generic Letter 84-20 - Scheduling Guidance for Licensee Submittals of Reloads That Involve Unreviewed Safety Questions.
August 21, 1984	Letter from applicant summarizing methodology employed to ensure functional capability of essential ASME Code, Class 1 piping.
August 31, 1984	Letter from applicant documenting verbal presentation given at August 9, 1984, meeting regarding present organizational structure.

September 7, 1984	Letter from applicant transmitting response to SER Outstanding Issue 6.
September 11, 1984	Letter from applicant concerning environmental qualification of electrical equipment important to safety.
September 14, 1984	Letter from applicant transmitting "Seabrook Station Fire Protection of Safe Shutdown Capability (10 CFR 50, Appendix R)."
September 18, 1984	Letter from applicant transmitting documents for emergency operating procedures generation package.
September 20, 1984	Letter to applicant requesting additional information regarding seismic qualification program.
September 25, 1984	Letter from applicant transmitting Amendment 53 to FSAR.
September 27, 1984	Letter from applicant regarding interim agreement by New Hampshire Public Utilities Commission to preserve and protect assets of and investment in New Hampshire nuclear units.
September 28, 1984	Letter from applicant transmitting Revision 4 of security plan.
October 6, 1984	Letter to applicant requesting assessment of relevance of the Westinghouse July 13, 1984, letter regarding fluid systems that utilize ASME Code Case N-368.
October 12, 1984	Meeting with applicant to discuss applicant's proposed method to accomplish the task analysis and status of human engineering discrepancies.
October 16, 1984	Letter to applicant requesting additional information regarding proposed organizational changes.
October 17, 1984	Letter from applicant regarding review of probabilistic safety assessment.
October 31, 1984	Letter from applicant in response to October 16, 1984, letter, providing information on organizational changes.
November 1, 1984	Letter from applicant transmitting "Moderate Energy Line Break Study."
November 29, 1984	Letter from applicant forwarding response to request concerning SER Outstanding Issue 7, reactor coolant system flow measurement uncertainty (proprietary and nonproprietary information).
November 29, 1984	Letter from applicant forwarding "Seabrook Station Fire Protection Program: Evaluation and Comparison to Branch Technical Position APCS 9.5-1, Appendix A," Revision 2.

December 13, 1984	Letter to applicant requesting modification of security plan as a result of review of September 28, 1984, submittal.
December 13, 1984	Letter from applicant requesting approval to use ASME Code Case N-411 damping values.
December 13, 1984	Letter to applicant regarding seismic piping analysis.
December 18, 1984	Letter from applicant regarding revisions to postaccident sampling system.
December 18, 1984	Letter from applicant transmitting 1983 Financial Reports.
December 20, 1984	Letter from applicant transmitting "Steam Generator Tube Rupture Analysis Methodology To Determine Margin to Steam Generator Overfill."
December 27, 1984	Generic Letter 84-24 - Certification of Compliance to 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety of Nuclear Power Plants."
January 7, 1985	Letter from applicant advising that response to Generic Letter 83-28 will be provided by June 30, 1985.
January 7, 1985	Letter from applicant transmitting final report on rod drop.
January 7, 1985	Letter from applicant transmitting information on detailed control room design review.
January 9, 1985	Generic Letter 85-01 - Fire Protection Policy Steering Committee Report.
January 28, 1985	Letter from applicant advising that diesel generator auxiliaries will not be automatically tripped on diesel generator start.
January 28, 1985	Letter from applicant forwarding responses to RAI 260.28 and SER Outstanding Issue 18.
January 31, 1985	Letter to applicant advising that the newly established New Hampshire Yankee organization is acceptable and forwarding safety evaluation of design and construction QA program changes.
February 1, 1985	Letter from applicant forwarding information on reactor coolant loop pipe break elimination benefits summary.
February 1, 1985	Letter from applicant requesting meeting on SER Outstanding Issue 5, "Load Combinations, Design Transients, and Stress Limits."

February 7, 1985	Letter from applicant requesting approval of elimination of arbitrary intermediate breaks in all high-energy piping systems, except feedwater system.
February 19, 1985	Letter from applicant forwarding partial replacement for submittal of February 7, 1985.
February 20, 1985	Meeting with applicant to discuss fire protection review.
February 21, 1985	Meeting with applicant to hear presentation of plans for initiation of a cable tray support reevaluation program.
March 12, 1985	Letter from applicant transmitting Amendment 54 to FSAR.
March 22, 1985	Letter from applicant forwarding application for amendment to construction permits regarding exemption from 10 CFR 50, Appendix A, General Design Criterion 4.
March 29, 1985	Letter from applicant submitting procedures generation package and supporting materials.
April 1, 1985	Meeting with applicant to discuss Auxiliary System Branch review regarding safe shutdown capabilities (Appendix R).
April 9, 1985	Meeting with applicant to discuss Technical Specification review.
April 12, 1985	Letter from applicant requesting elimination of feedwater system arbitrary intermediate pipe breaks from design bases.
April 16, 1985	Generic Letter 85-06 - Quality Assurance Guidance for ATWS Equipment Not Safety-Related.
April 18, 1985	Letter from applicant transmitting information on tornado missile analysis.
April 19, 1985	Letter to applicant concerning resolution of integrated design inspection issues.
April 22, 1985	Letter to applicant advising of completion of review of request for authorization to use ASME Code Case N-413.
April 22, 1985	Letter to applicant transmitting request for additional information regarding post-fire safe shutdown capability.
April 23, 1985	Letter to applicant transmitting request for additional information regarding changes to Chapter 14 of FSAR.
April 29, 1985	Letter from applicant transmitting revised information on operator requalification training program.
May 1, 1985	Letter from applicant transmitting Construction Status Report for period ending March 31, 1985.

May 1, 1985	Letter from applicant regarding use of ASME Code Case N-368.
May 2, 1985	Generic Letter 85-07 - Implementation of Integrated Schedules for Plant Modifications.
May 8, 1985	Letter from applicant forwarding additional information for use in review of seismic qualification program.
May 8, 1985	Meeting with applicant to discuss fire protection program and safe shutdown capability.
May 10, 1985	Meeting with applicant to discuss operator experience level.
May 10, 1985	Letter from applicant forwarding information regarding security plan.
May 23, 1985	Generic Letter 85-08 - 10 CFR 20.408 Termination Reports - Formats.
May 23, 1985	Generic Letter 85-09 - Technical Specifications for Generic Letter 83-28, Item 4.3.
May 24, 1985	Meeting with applicant to address cable tray support reevaluation program.
May 24, 1985	Letter to applicant advising of July 9-11, 1985, visit to perform caseload forecast.
May 24, 1985	Letter to Westinghouse advising that information submitted by applicant August 9 and November 29, 1984, will be withheld from public disclosure.
May 29, 1985	Letter from applicant concerning Technical Specification improvement program.
June 3, 1985	Letter from applicant transmitting information on cable raceway system damping.
June 3, 1985	Letter from applicant regarding flood protection features enhancement.
June 3, 1985	Letter from applicant transmitting information on radiation and shielding design for vital area access.
June 7, 1985	Letter from applicant forwarding response to request for additional information.
June 7, 1985	Letter to applicant advising of acceptability of alternative pipe break criteria.
June 10, 1985	Letter from applicant requesting evaluation and approval of program for field fabrication and installation of Class 2 and 3 pipe supports.

June 11, 1985	Letter to applicant concerning TMI Action Item II.K.3.30 for Westinghouse plants.
June 11, 1985	Letter to applicant concerning resolution of integrated design inspection issues.
June 13, 1985	Meeting with applicant regarding environmental qualification program.
June 27, 1985	Meeting with applicant to discuss fire protection program and safe shutdown capability.
July 1, 1985	Letter from applicant concerning functional capability of ASME Code Class 1 piping.

APPENDIX B

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APPENDIX D

ACRONYMS AND INITIALISMS

AIB	arbitrary intermediate break
ALARA	as low as reasonably achievable
ANL	Argonne National Laboratory
ASME	American Society of Mechanical Engineers
ATWS	anticipated transients without scram
BOP	balance of plant
BTP	Branch Technical Position
BWR	boiling water reactor
CFR	<u>Code of Federal Regulations</u>
CHR	containment heat removal
CRT	cathode ray tube
DCRDR	detailed control room design review
ECC	emergency core cooling
ECCS	emergency core cooling system
EOP	emergency operating procedure
ESF	engineered safety feature
FSAR	Final Safety Analysis Report
GDC	General Design Criterion(a)
HED	human engineering discrepancy
HEPA	high efficiency particulate air
IE	Office of Inspection and Enforcement
ISEG	Independent Safety Engineering Group
LLD	low-level dose
LLNL	Lawrence Livermore National Laboratory
LOCA	loss-of-coolant accident
MMI	Modified Mercalli intensity
msl	mean sea level
NBS	National Bureau of Standards
NES	Nuclear Energy Services
NHY	New Hampshire Yankee
NRC	U.S. Nuclear Regulatory Commission
PASS	postaccident sampling system
PCCW	primary component cooling water

PGP	procedures generation package
PRA	probabilistic risk assessment
PSI	preservice inspection
PSNH	Public Service Company of New Hampshire
PWR	pressurized water reactor
QA	quality assurance
RAI	request for additional information
RCL	reactor coolant loop
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	Regulatory Guide
RGWS	radioactive gaseous waste system
RHR	residual heat removal
RPV	reactor pressure vessel
RTD	resistance temperature detector
SER	Safety Evaluation Report
SMA	strong motion accelerograph
SPDS	safety parameter display system
SRP	Standard Review Plan
SSE	safe shutdown earthquake
TER	Technical Evaluation Report
TMI	Three Mile Island
USGS	U.S. Geological Survey
USI	unresolved safety issue
WOG	Westinghouse Owners Group
YAEC	Yankee Atomic Electric Company

APPENDIX F
NRC STAFF CONTRIBUTORS

The NRC staff members listed below were principal contributors to this report.

<u>Name</u>	<u>Title</u>	<u>Review Branch</u>
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APPENDIX J

TECHNICAL EVALUATION OF REPORT C569,
"SEABROOK NUCLEAR POWER PLANT TORNADO MISSILE ANALYSIS"

Technical Evaluation of Report C569
Seabrook Nuclear Power Plant Tornado Missile Analysis
(Prepared by Applied Research Associates, Inc., for
United Engineers and Constructors, Inc. -- Sept. 1983)

Emil Simiu

1. INTRODUCTION

The objective of this evaluation is to assess the validity and the degree of conservatism of the assumptions, data, and mathematical approach used in the Report C569* to estimate the tornado missile risk to a specified set of targets at the Seabrook Nuclear Generating Station.

The Report includes:

- (a) A climatological assessment of tornado wind speeds
- (b) An assessment, based on a site survey, of the number and types of potential tornado-borne missiles at the Seabrook Plant.
- (c) An analysis of tornado-borne missile risks obtained by Monte Carlo simulations of missile trajectories at the plant. The input used in the simulations was based on the assessments mentioned in item (a) and (b) above. The simulation methodology was based on Ref. 1, to which is added a model of missile passage through openings and "labyrinths".

2. CLIMATOLOGICAL ASSESSMENT OF TORNADO WIND SPEEDS

The Seabrook site is located in NRC Tornado Region I, as defined in Ref. 2. For consistency with Ref. 2, the upper bound F-scale wind speed is assumed in the Report to be 360 mph. This assumption is likely to be conservative.**

The tornado occurrence rates for the site are estimated in the Report as follows. It is indicated in the Report that, according to the Seabrook FSAR (Final Safety Analysis Report), the tornado strike probability for a point at

* Hereinafter referred to as the Report

** The term "conservative" is defined as "erring on the side of safety".

the site, as estimated from 28 years of NSSFC (National Severe Storms Forecast Center) data, is 7.8×10^{-5} /year. This is lower by a factor of about 1.5 than the estimated strike probability for tornado Region C (see p. II-4 of Report), which covers an area of about 775,000 sq. mi. and includes the Seabrook site near its eastern periphery. To reflect the site specific information from the Seabrook FSAR, the Report first assumes that the tornado occurrence rates at the Seabrook site are lower by a factor of 1.5 than those estimated in Ref. 1 for Region C. Secondly, the Report assumes that the tornado path length, width, and direction data are the same at the site as those estimated in Ref. 1 for Region C.

The Report does not describe in detail the characteristics of the tornado wind field assumed in the numerical calculations. However, in response to a question raised in a letter by the National Bureau of Standards to the Nuclear Regulatory Commission dated October 4, 1984, it was indicated by the Public Service of New Hampshire that wind speeds were assumed to be reduced below 33 ft. elevation to a value at the ground level equal to about 75% of the speed at 33 ft.

To the reviewer's knowledge, there are not data in the literature that would validate such a reduction. It is therefore possible that this reduction offsets to some degree the conservatism inherent in the assumption that the tornado F-scale wind speeds have the values listed in Ref. 2. The Report does not include an error analysis to reflect the effect on the final estimates of the assumptions on tornado wind speeds and on tornado wind speed reduction near the ground.

For the purpose of estimating approximate tornado wind effects, the assumptions used in the Report appear nevertheless to be reasonable.

3. NUMBER AND TYPES OF POTENTIAL TORNADO-BORNE MISSILES AT THE SITE

The number and type of potential tornado-borne missiles used in the simulations were assessed on the basis of a survey performed at the site. The data obtained in the survey "were used to characterize potential missile sources that might be typical at the site with Unit 1 operating and Unit 2 under construction". It was assumed that none of the potential missiles were substantially blocked by other obstacles or restraints during the passage of a tornado. In the Report, potential missiles assumed to be present at the site include components of buildings that would be damaged or destroyed during a tornado. (For buildings with metal siding, the number of assumed potential missiles is 5 purlins and 10 siding sections per 1,000 sq. ft. of floor area, injected in the tornado flow uniformly from an elevation of 1 ft. to the top of the building; for trailers, the assumed number is 20 wood beams, 10 wood planks, 10 plywood sheets, and 10 metal channel sections; for engineered non-safety related plant structures the number assumed is 5 missiles per 1,000 sq. ft. of floor area, e.g., about 700 missiles for the turbine building.) In addition, 25 potential missiles are assumed for each of four cranes present at the site. Of the total of about 65,000 potential missiles assumed to be present at the site, less than 5% were assumed to originate from damaged structures.

It is noted that the number of missiles assumed to originate from damaged structures, N_d , was estimated subjectively. According to the Report, the choice of this number was based "on past experience of similarly designed structures" and is independent of tornado intensity.

To the extent that information on damaged structures was obtained under conditions corresponding to the less intense tornadoes of the F0-F6 Fujita scale, it is legitimate to surmise that the assumption that N_d is independent

of tornado intensity may be unconservative. Otherwise, the assumptions regarding potential tornado-borne missiles used in the Report appear to be reasonable.

4. ANALYSIS OF TORNADO-BORNE MISSILE RISKS

The analysis of tornado-borne missile risks uses the methodology described in Ref. 1, an assessment of which was presented in Ref. 3. In addition to this methodology, the Report uses a model of missile passage through openings and "labyrinths".

In order to improve the precision of the simulations, the area of any target opening is increased by a factor $K_a > 1$ (p. III-11). The probability of passage through the actual opening is then obtained by reducing the estimated probability of passage through the increased opening by the factor K_a . This procedure is judged to be acceptable.

Missiles are assumed to pass through a rectangular opening with area A only if they do not hit the edges of an equivalent circular opening with the same area, whose center coincides with the center of the rectangular opening. It is noted that this assumption is not necessarily conservative. Some missiles can hit the edges of the equivalent circular opening and still pass through the rectangular opening. However, estimates based on this assumption are judged to provide a reasonably close approximation to average conditions. The number of missiles that pass through the rectangular opening is assumed in the Report to be equal to the number of missiles whose centroid hits the opening, multiplied by a reduction factor, K_m . This factor depends, for any given area, upon the probability distribution of the missile length, L , and the maximum value of L , denoted by L_m (Fig. III-5 of the Report). The method used in the Report to obtain the factor K_m is judged to be acceptable.

To model passage of missiles through "labyrinths", that is, through a succession of openings, the Report uses an overall reduction factor

$$K_m = \frac{1}{2^{s-1}} \prod_{j=1}^s K_{mj}$$

where K_{mj} is the reduction factor for the j -th opening, and s is the total number of openings in the labyrinth. The factor $1/2^{s-1}$ in the above expression accounts, subjectively but not unreasonably, for loss of momentum following each passage through an opening.

Numerical results obtained by the analysis reviewed above are listed in Tables V-1, V-2, and V-3 of the Report. The probability of missile entrance in spaces where such an event would lead to undesirable safety-related consequences is estimated in the Report to be about 1.2×10^{-6} per year.

5. CONCLUSIONS

As indicated in the preceding section, the probability of missile entrance in spaces where such an event could lead to undesirable conditions from a safety point of view is estimated in the Report to be about 1.2×10^{-6} per year. A review of the assumptions, data, and approaches used in the Report suggests that this estimate is a reasonable one given the present state of the art. Inherent in this estimate are the following conservative assumptions:

- the tornado wind speeds corresponding to the upper limit of the F-scale are 360 mph
- the number of potential missiles present on the site during construction of Unit 2 is not reduced when Unit 2 is completed.

Other assumptions used in the Report and reviewed herein appear to be reasonable, although it is difficult to judge whether they are conservative. On the other hand, the subjective assumptions regarding the numbers of missiles

originating from damaged structures may be unconservative. The tornado speed reduction near the ground may also be unconservative.

Since it appears that some assumptions are conservative while others are not, it is difficult to determine whether the estimate of the probability of missile entrance presented in the Report is conservative. However, it is noted that wind speed values have a very strong influence on missile transport in air (see Ref. 4). It would therefore appear that the effect of the conservatism inherent in the assumption that wind speeds are scaled in accordance with Ref. 2 is stronger than the possible lack of conservatism inherent in other assumptions used in the Report, including the assumption that wind speeds at the ground level are about 25% lower than wind speeds at 33 ft. elevation. To the extent that this is the case, it is concluded that the probability of missile entrance for the entire plant estimated in the Report is likely to be conservative, and that the "true" probability is likely to fall within the range of 10^{-6} to 10^{-7} per year.

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