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Docket Nos.: 50-443
and 50-444

AUG 13 1982

Mr. William C. Tallman
Chairman and Chief Executive Officer
Public Service Company of New Hampshire
P. O. Box 330
Manchester, New Hampshire 03105

Dear Mr. Tallman:

Subject: Draft Safety Evaluation Report (SER) for Seabrook Station

Reference: NRC Letter to PSNH (Tedesco to Tallman), Subject: Status of Seabrook Licensing Review, July 2, 1982

With regard to the referenced letter, sufficient progress has now been made in some areas of the safety review to permit the issuance of related SER sections in draft form.

Enclosure 1 provides a copy of those SER sections that are sufficiently complete for issuance of a draft edition. Enclosure 2 identifies the additional information required to complete the SER sections in Enclosure 1. Enclosure 3 provides a brief statement describing the present status of the review of those major sections not included in Enclosure 1.

In some past OL application reviews, it has been a worthwhile effort on the part of the applicant to send a representative, authorized to make binding commitments for applicant action, to meet with the staff to participate in the resolution of open items. You may wish to consider making a similar effort for the Seabrook safety review.

The NRC Project Manager (Mr. L. Wheeler, 301/492-7792) is available to provide additional assistance and coordination for such a meeting, and to respond to any other concerns you may have regarding the SER.

Sincerely,

Original Signed By:

Thomas M. Novak, Assistant Director
for Licensing
Division of Licensing

Enclosures:
As stated

OFFICE	cc w/enclosures: See next page	DL:LB#3	DL:LB#3	DL:AD:L
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SAFETY EVALUATION

REPORT (PARTIAL)

RELATED TO THE OPERATION OF

SEABROOK STATION

UNITS NOS. 1 AND 2

DOCKET NOS. 50-443 AND 50-444

This partial draft SER has been assembled to identify the open items for each enclosed section to permit the NRC and applicant staffs to work toward the objective of resolving them prior to issuance of the final SER. The following sections are included in this part of the SER:

2.3 Meteorology 2.3.1 through 2.3.5
2.4 Hydrologic Engineering 2.4.1 through 2.4.9
2.5 Geology and Seismology 2.5.1 through 2.5.5
3.3 Wind and Tornado Loadings 3.3.1 through 3.3.2
3.4 Water Level (Flood) Design 3.4.1 through 3.4.2
3.5 Missile Protection 3.5.1.1, 3.5.1.2 & 3.5.1.4
3.6 Protection Against Pipe Ruptures 3.6.1 through 3.6.2
3.7 Seismic Design 3.7.1 through 3.7.3
3.8 Seismic Category I Structures 3.8.1 through 3.8.6
3.9 Mechanical Systems and Components 3.9.1 & 3.9.4 through 3.9.6
4.2 Fuel Design 4.2.1 through 4.2.3
4.3 Nuclear Design 4.2.1 through 4.3.4
4.4 Thermal-Hydraulic Design 4.4.1 through 4.4.8
4.6 Functional Design of Reactivity Control System
5.4 RCS Subsystem Analysis 5.1.11
6.1 ESF Materials 6.1.1 through 6.1.2
6.4 Control Room Habitability
9.3 Process Auxiliaries 9.3.2 through 9.3.3
9.4 HVAC 9.4.2 through 9.4.5
10.4 Steam/Power Conversion Features 10.4.5 and 10.4.7
11.1 Source Terms
12.1 ALARA Program 12.1.1 through 12.1.3
12.2 Radiation Sources 12.2.1 through 12.2.2
12.3 Radiation Protection Features 12.3.1 through 12.3.3

2.3 METEOROLOGY

Evaluation of regional and local climatological information, including extremes of climate and severe weather occurrences which may affect the design and siting of a nuclear plant, is required to assure that the plant can be designed and operated within the requirements of Commission regulations. Information concerning atmospheric diffusion characteristics of a nuclear power plant site is required for a determination that radioactive effluents from postulated accidental releases, as well as routine operational releases, are within Commission guidelines. Sections 2.3.1 through 2.3.5 have been prepared in accordance with the review procedures described in the Standard Review Plan (NUREG-0800), utilizing information presented in Section 2.3 of the FSAR, responses to requests for additional information, and generally available reference materials as described in the appropriate sections of the Standard Review Plan.

2.3.1 REGIONAL CLIMATOLOGY

The climate of the region in which the plant is located can be described as generally continental,

characterized by rapid changes in temperature and marked extremes, but with marked martitime influences. The Gulf of Maine exerts a moderating influence on temperature extremes in the immediate coastal region. The mean annual temperature in the area is about 7.2°C (45°F), ranging from about -5.0°C (23°F) in January to about 20.0°C (68°F) in July. Annual precipitation in the area is about 1092 mm (43 inches).

The Seabrook plant is located near the principal tracks of cyclonic storms that move across the continental United States and southern Canada, as well as storms that move northeastward along the East Coast, resulting in frequently changing conditions and a variety of severe weather phenomena. About 26 thunderstorms can be expected on about 19 days each year, being most frequent in June, July, and August. Considering the frequency of thunderstorms and the size of the structures of the plant, the applicant has estimated that large structures such as the containment building will average about one lightning strike every two years. Hail often accompanies severe thunderstorms. In

the period 1955-1967, only 6 occurrences of hail with diameters 19 mm (3/4 inch) or greater were reported in the one-degree latitude-longitude square containing the site. However, in the same period, 28 such occurrences were reported in the adjacent one-degree square to the west.

Tornadoes also occur in the area. The applicant has reported that 69 tornadoes have occurred within 50 miles of the plant site in the period 1950-1977, resulting in an annual tornado frequency of 2.5 occurrences. The applicant has reported that the initial touchdown point for one tornado (July 1968) was approximately 2 miles from the site, but the wind associated with the tornado was probably 100 mph or less. Assuming an average tornado path area of 2.8 square miles, a recurrence interval for a tornado at the plant site was computed to be about once in 1100 years. The applicant has determined that the mean path area for tornadoes in the region of the Seabrook plant is much smaller than 2.8 square miles, which is based on tornadoes in Iowa. Use of a smaller mean tornado path area will result in a correspondingly longer computed recurrence interval.

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The applicant has also examined occurrences of waterspouts off the New Hampshire coast, and determined that 14 waterspouts were reported off the coast between Boston and Portsmouth in the period 1959-1973, of which 3 probably caused coastal damage. The design basis tornado characteristics selected by the applicant conform to the recommendations of Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," for this region of the country. These characteristics are: rotational speed - 290 mph; translational speed - 70 mph; and a total pressure drop of 3 psi occurring at a rate of 2 psi/sec.

Hurricanes or remnants of hurricanes pass through the region occasionally. The applicant has determined that during the period 1871-1977, 43 tropical cyclones passed within 100 nautical miles of the site. Of these 43 tropical cyclones, only 3 were classified as hurricanes within 100 nautical miles of the site.

High wind speeds also occur in the area as a result of extratropical cyclones. The highest "fastest mile"

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wind speed reported at Boston was 87 mph in September 1938. The applicant has selected an operating basis wind speed (defined as the "fastest mile" wind speed at a height of 30 feet with a return period of 100 years) of 110 mph for consideration in plant designs.

The applicant has examined 10 years (1961-1970) of hourly data from Pease AFB and 29 years (1945-1973) of hourly data from Boston to determine the meteorological design conditions of the ultimate heat sink. The applicant determined the 30-day period with the largest difference between the daily average dry bulb temperature and daily average wet bulb temperature to evaluate the maximum evaporative and drift loss for the mechanical draft cooling tower used as the ultimate heat sink. The applicant also determined the maximum daily average and maximum consecutive 30 day average wet bulb temperature to evaluate minimum heat transfer to the atmosphere for the ultimate heat sink. Because the maximum 24-hour wet bulb temperature considered in the design of the ultimate heat sink 75°F has been exceeded in

the region (a maximum 24 hour wet bulb temperature of 75.5oF was observed at Boston), the applicant performed a supplemental analysis to demonstrate that cooling water temperatures remain within acceptable limits with wet bulb temperatures in excess of 75.5oF under postulated accident conditions. (see Sections 2.4.4 and 9.X.X of this report for a determination of the adequacy of the ultimate heat sink.) The meteorological conditions selected by the applicant for the design of the ultimate heat sink appear reasonably conservative. The applicant appears to have analyzed available long-term meteorological data in accordance with the procedures described in Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Power Plants".

Heavy snowfall is not uncommon in the region. Average annual snowfall at Portsmouth is about 1830 mm (72 inches). The applicant has estimated the weight of the 100 year return period snowpack at ground level to be 42 psf. The snowload for design basis consideration (the weight of the 100 year snowpack plus the weight of the 48 hour Probable Maximum Winter Precipitation) was determined to be

125.7 psf at ground level. Ice storms are not uncommon in the area, with about 8 ice storms expected each year. About 26 hours of freezing rain occur each year at Portsmouth. Freezing rain is most likely in December and January.

The applicant has considered extreme outdoor temperatures of 31.1°C (88°F) and -17.8°C (0°F) in the design of the HVAC systems for all safety-related buildings. The basis for the selection of these temperatures were the 2 1/2% probability of occurrences (summer) and 97 1/2% (winter) probability of occurrence values from the distributions presented by ASHRAE. The diesel generators are designed for temperatures of 35.0°C (95°F) to -34.4°C (-30°F). Extreme temperatures of 37.2°C (99°F) and -30.6°C (-23°F) have been reported at Portsmouth. An extreme maximum temperature of 40.0°C (104°F) has been reported at Boston. Temperatures in excess of 32.2°C (90°F) are expected at Boston about 12 days each year. Temperatures of 95°F or higher are expected every two years. Temperatures of -17.8°C (0°F) or lower are expected about 11 days each year at Portsmouth. Conditions with return periods of 100 years are normally considered for design of safety-related

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auxiliary systems and components. The 100 year return period extreme temperatures in the Seabrook area are approximately 41.1°C (106°F) and -35.6°C (-32°F). Further justification of the adequacy of the extreme temperatures considered by the applicant for the design of safety-related auxiliary systems and components is required.

Large-scale episodes of atmospheric stagnation are not common in the region. About 8 atmospheric stagnation cases totaling about 35 days were reported in the area in the period 1936-1970. One of these cases lasted 7 days or more.

As discussed above, the staff has reviewed available information relative to the regional meteorological conditions of importance to the safety design and siting of this plant in accordance with the criteria contained in Section 2.3.1 of the Standard Review Plan. Based on this review, the staff concludes that, with the exception of design basis temperatures for auxiliary systems and components, the applicant has identified and considered appropriate regional

meteorological conditions in the design and siting of this plant, and, therefore, meets requirements of 10 CFR Part 100.10 and 10 CFR Part 50, the Appendix A, General Design Criterion 2. The design basis tornado characteristics selected by the applicant conform to the position set forth in Regulatory Guide 1.76, and, therefore, meets the requirement of 10 CFR Part 50, Appendix A, General Design Criterion 4 to determine an acceptable design basis tornado for missile generation.

2.3.2 LOCAL METEOROLOGY

Climatological data from Boston, Massachusetts, Portsmouth, New Hampshire (Pease AFB), and available onsite data have been used to assess local meteorological characteristics of the plant site.

Precipitation is well-distributed throughout the year, ranging from about 69 mm (2.7 inches) in June and August, to about 117 mm (4.6 inches) in November. Maximum and minimum monthly amounts of precipitation observed in the area have been 350.5 mm (13.8 inches) in January 1958 at Portsmouth, and a trace (less than

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0.01 inch) in March 1915 at Boston. The maximum amount of precipitation in a 24-hour period at Boston was 213.4 mm (8.4 inches) in August 1955, although 171.2 mm (6.74 inches) has been recorded in a 12-hour period at Boston. Monthly snowfall is usually heaviest in February (18.9 inches). The maximum amount of snowfall in a 24-hour period was 548.6 mm (21.6 inches) at Portsmouth in December 1954.

Wind data taken from the 13.1 m level of the onsite meteorological tower for a two-year period (April 1979 - March 1980 and June 1980 - May 1981) indicate prevailing winds from the west-northwest (about 18% of the time). Winds from the southwest clockwise through northwest occur almost 60% of the time. Winds from the north-northeast occur least frequently at about 2% of the time. The median wind speed at the 13.1 m level is about 3 m/sec.

Neutral (Pasquill type "D") conditions predominate, occurring about 45% of the time. Moderately stable (Pasquill type "F") and extremely stable (Pasquill type "G") conditions occur about 7% and 6% of the time, respectively.

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As discussed above, the staff has reviewed available information relative to local meteorological conditions of importance to the safe design and siting of this plant in accordance with the criteria contained in Section 2.3.2 of the Standard Review Plan. Based on this review, the staff concludes that the applicant has identified and considered appropriate local meteorological conditions in the design and siting of this plant and, therefore, meets the requirements of 10 CFR Part 100.10 and 10 CFR Part 50, Appendix A, General Design Criterion 2.

2.3.3 ONSITE METEOROLOGICAL MEASUREMENTS PROGRAM

An onsite meteorological measurements program at the Seabrook site was initiated in November 1971. Meteorological measurements were made on a 45.7 m (150 ft) tower located along the Browns River north-northwest of the reactor area. Wind speed and wind direction were measured at the 9.1 m (30 ft) and 39.6 m (130 ft) levels and vertical temperatures gradient was measured between the 9.1 m and 39.6 m levels. Two years (December 1971 - November 1972 and December 1972-November 1973) of data from this tower were submitted with the Construction Permit application, and one

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year (April 1972 - March 1973) of data from this program was included in the FSAR. However, wind sensors were changed at the end of 1972 making a composite data set somewhat suspect.

A new 64 m (210 ft) tower was erected at the same location as the old tower in April 1979. Wind speed and wind direction are measured at the 13.1 m (43 ft) and 63.7 m (209 ft) levels. Because the base elevation of the meteorological tower is slightly lower than plant grade, the applicant believes the 13.1 m level represents approximately the 10 m level in the vicinity of plant structures. Vertical temperature gradient is measured between the 13.1 m and 45.7 m levels and between the 13.1 m and 63.7 m levels. Ambient dry bulb and dew point temperatures are measured at the 13.1 m level, and precipitation and solar radiation are measured near the ground. A digital recording system is in use and analog strip charts are utilized as backup. The entire data collection system is calibrated quarterly and the calibration activities are performed under a quality assurance program which meets the requirements of 10 CFR Part 50, Appendix B.

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The Applicant has provided two years (April 1979 - March 1980 and June 1980 - May 1981) of meteorological data. These data have been combined into a composite set of joint frequency distributions of wind speed and wind direction by atmospheric stability. Wind speed and wind direction data were based on measurements at the 31.1 m level, and atmospheric stabilty was defined by the measurement of vertical temperature gradient between the 13.1 m and 63.7 m levels. Data recovery for the joint frequency distribution for the composite two-year period was in excess of 95%.

Although the measurement of vertical temperature gradient between the 13.1 m and 63.7 m levels was considered as the primary indicator of atmospheric stability, the staff also examined atmospheric stability determined by the measurement of vertical temperature gradient between the 13.1 m and 45.7 m levels for consistency, particularly with the earlier data collection periods. A comparison of the available onsite meteorological data sets showed a significant variability in the annual frequency of unstable (Pasquill types "A", "B", and C) and slightly

stable (Pasquill type "E") conditions using the vertical temperature gradient between the 13.1 m and 45.7 m levels. Atmospheric stability conditions were considerably less variable when based on the measurement of vertical temperature gradient between the 13.1 m and 63.7 m levels. Concurrent measurements of vertical temperature gradient between the two intervals since April 1979 have also indicated very different atmospheric stability conditions on certain occasions, particularly just after sunrise when the lower measurement interval indicated extremely unstable conditions while the higher measurement interval indicated neutral conditions. The applicant has been requested (Q 451.14) to provide an explanation of the year-to-year variability in atmospheric stability conditions considering the atmospheric mechanisms for generating thermal instability, the classification scheme used, the location of the meteorological tower and the surface characteristics around the tower, and the location of the site. The applicant should also address the differences in atmospheric stability conditions indicated by concurrent measurements of vertical temperature gradient between the two

intervals. Such differences could have a significant impact on real-time assessments for emergency planning and preparedness if one interval was to be used as a backup to the other.

The staff has reviewed the onsite meteorological measurement system in accordance with the criteria contained in Section 2.3.3 of the Standard Review Plan. The onsite meteorological measurement system conforms to the guidance of Regulatory Guide 1.23, "Onsite Meteorological Programs," and has provided adequate data to represent onsite meteorological conditions as required in 10 CFR Part 100.10. The site data provide an acceptable basis for making conservative estimates of atmospheric dispersion conditions used for estimating consequences of design basis accidents and routine releases from the plant.

To address the meteorological requirements for emergency preparedness planning outlined in 10 CFR Part 50.47 and Appendix E to 10 CFR Part 50, the applicant will be required to upgrade the operational meteorological measurements program to meet the criteria in NUREG-0654, Appendix 2, "Criteria for Preparation and Evaluation of Radiological Emergency

Response Plans and Preparedness in Support of Nuclear Power Plants." The upgrades must be in accordance with the schedule of NUREG-0737, III.A.2, "Clarification of TMI Action Plan Requirements." The upgrades will consist of the addition of a viable backup meteorological measurements system and the demonstrated capability for remote interrogation of meteorological information by the utility, emergency response organizations, and the NRC staff during emergency situations. The incorporation of current meteorological data into a real-time atmosheric dispersion model for dose assessments will also be considered as part of the upgraded capability.

2.3.4 SHORT-TERM (ACCIDENT) DIFFUSION ESTIMATES

To audit the applicant's estimates, the staff has performed an independe t assessment of short-term (less than 30 days) accidental releases from buildings and vents using the direction-dependent atmospheric dispersion model described in Regulatory Guide 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," with consideration of increased lateral dispersion during

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stable conditions accompanied by low wind speed conditions. Two years (April 1979 - March 1980 and June 1980 - May 1981) of onsite data were used for this evaluation. Wind speed and direction data were based on measurements at the 13.1 m level and atmospheric stability was defined by the measurement of vertical temperature gradient between the 13.1 m and 63.7 m levels. A ground-level release with a building wake factor, cA, of 1045 m^2 was assumed. The relative concentration (X/Q) value for the 0-2 hour time period was determined to be 2.7×10^{-4} sec/m^3 at an exclusion boundary distance of 914 m in the east sector. A nearly identical value was calculated at the same distance in the east-southeast sector. The X/Q values for appropriate time periods at the outer boundary of the low population zone (2012 m) are:

Time Period	X/Q (sec/m^3)
0-8 hours	6.4×10^{-5}
8-24 hours	4.5×10^{-5}
1-4 days	2.0×10^{-5}
4-30 days	6.7×10^{-6}

The applicant has calculated similar X/Q values for the 0-2 hour period at the exclusion area boundary.

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and for the 0-8 hour time period at the LPZ distance. The applicant has calculated substantially lower values for longer time periods at the LPZ distance. Based on the results of a study being performed by the applicant on long-term atmospheric dispersion conditions (see Section 2.3.5), the X/Q values for the various time periods may be increased slightly. Based on the above evaluation performed in accordance with the criteria contained in Section 2.3.4 of the Standard Review Plan, the staff concludes that the applicant has considered appropriate atmospheric dispersion estimates for assessments of the consequences of radioactive releases in accordance with the requirements of 10 CFR Part 100.11. The atmospheric dispersion estimates provided in this section have been used by the staff in an independent assessment of the consequences of radioactive releases for design basis accidents.

2.3.5 LONG-TERM (ROUTINE) DIFFUSION ESTIMATES

To audit the applicant's estimates, the staff has performed an independent calculation of annual average relative concentration (X/Q) and relative deposition

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(D/Q) values using the straight-line Gaussian atmospheric dispersion model described in Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors." The applicant is examining spatial and temporal variations in airflow, particularly airflow reversals during the onset of the seabreeze and curved trajectories during the decay of the seabreeze, to determine appropriate modifications to the results of the straight-line model.

Releases through the two unit vents have been considered to be partially elevated based on the criteria contained in Regulatory Guide 1.111. Intermittent releases through each vent have been evaluated using the methodology described in NUREG-0324, "XOQDOQ Program for the Meteorological Evaluation of Routine Effluent Releases at Nuclear Power Stations." Only one year (April 1979 - March 1980) was available for the calculation of annual average X/Q and D/Q included in the Draft Environmental Statement. These estimates will be updated to reflect the additional year of onsite meteorological data now available.

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Based on the above evaluation performed in accordance with the criteria contained in Section 2.3.5 of the Standard Review Plan, the staff concludes that the applicant has considered representative atmospheric dispersion estimates for demonstrating compliance with the numerical guides for doses contained in 10 CFR Part 50, Appendix I. The atmospheric dispersion estimates developed by the staff are included in the assessment of the radiological impact to man resulting from routine releases to the atmosphere contained in the staff's environmental statement.

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Revised Draft Hydrologic Engineering Evaluation

Seabrook Nuclear Power Station

Units 1 and 2

Docket Nos. 50-443/444

2.4 HYDROLOGIC ENGINEERING

2.4.1 Introduction

The staff has reviewed the hydrologic engineering aspects of the applicant's design, design criteria and design bases of safety-related facilities at the Seabrook Station. The acceptance criteria include the applicable GDC reactor site criteria (10 CFR 100), and standards for protection against radiation (10 CFR 20, Appendix B, Table II). Guidelines for implementation of the requirements of the acceptance criteria are provided in Regulatory Guides, ANSI Standards and Branch Technical Positions identified in SRP Section 2.4-1 through 2.4-14. Conformance to the acceptance criteria provides the bases for concluding that the site and facilities meet the requirements of Parts 20, 50 and 100 of 10 CFR with respect to hydrologic engineering.

2.4.2 Hydrologic Description

The site is about 2 miles inland (west) from the Hampton and Seabrook Beaches, and about 1 mile east of the town of Seabrook, on the shore of the marsh area surrounding Hampton Harbor. The site lies within the Hampton Harbor watershed between Browns River and Hunts Island Creek. The proposed plant grade

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is 20 feet above mean sea level (MSL). Normal high tide is about 4.6 feet MSL. Significant hydrologically related plant features include intake and discharge tunnels from the site into the Atlantic Ocean, an onsite intake structure at the terminus of the tunnels, a mechanical draft cooling tower (ultimate heat sink), and various slope protection features around the plant site.

Normal operation will utilize once-through cooling, extracting water from and returning it to the Atlantic Ocean. The intake and discharge tunnels have been excavated through bedrock beneath the marsh, Hampton Harbor and the waterfront beach area to prevent disturbance of these areas. The discharge outfall will be about 5500 feet and the intake structure will be about 7000 feet offshore from Hampton Beach.

Ground water is a major resource of the area. The plant will use ground water supplied by the City of Seabrook, NH. Plant operation or normal and accidental effluents will not affect ground water users.

The applicant has provided hydrologic-descriptions of the site. The staff has reviewed the applicant's information in accordance with procedures in SRP Sections 2.4.1 and 2.4.2, and concludes that it is sufficient to meet the requirements of GDC-2.

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2.4.3 Flood Potential

Flooding near the site has historically been caused by "northeasters" and hurricanes. The applicants reported that the maximum tidal flooding was the result of a northeaster on February 7, 1978, which produced an estimated tidal elevation of 9.1 feet MSL on the New Hampshire coast; the maximum hurricane induced high water of 7.7 feet MSL was on December 29, 1959. The applicants considered several flooding sources in establishing the flooding design basis for the site. These events include stream flooding, precipitation induced flooding, flooding caused by seismically induced dam failure, ice flooding, tsunami induced flooding, surge and seiche flooding and combination of surge and stream flooding. The applicants state that the flooding design basis would be established by the Probable Maximum Hurricane (PMH) open coast surge with wind generated waves coincident with a Standard Project Flood (SPF), which is [redacted] assumed equivalent to one-half of the Probable Maximum Flood (PMF) on the Hampton Harbor watershed. The staff does not concur that the SPF/PMH combined event is the appropriate combination for this site. Current criteria for sites on watersheds less than 300 square miles and subject to hurricane surge (ANS N170, 9.2.2.2 Alternate IV) is a combination of a PMF and a PMH event.

2.4.3.1 Probable Maximum Flood (PMF) on Streams and Rivers

The applicants estimated the PMF for the Hampton Harbor watershed. The total drainage area to the Harbor is 47.4 square miles, of which 12.2 square miles are submerged. The probable maximum precipitation (PMP), used to

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estimate the PMF, was based on Hydrometeorological Report No. 33, "Seasonal Variation of the Probable Maximum Precipitation East of the 105th Meridian for Areas from 10 to 1000 Square Miles and Durations of 6, 12, 24, and 48 Hours", U.S. Weather Service.

Precipitation losses assumed in the analysis were conservative. The runoff models for the watershed were developed using standard methods. The flood hydrograph and astronomical tide were routed into the Harbor such that the peak effect (high water) was identified. The resultant water level at the plant site was 8.9 feet MSL. Wave effects were not evaluated because this stillwater level resulting from the PMF was well below the design basis water level, discussed in Section 2.4.3.6 of this report.

We have reviewed the FSAR material presented in accordance with the procedures described in SRP Sections 2.4.2 and 2.4.3, and conclude that the plant meets the requirements of GDC-2 with respect to flooding on streams and rivers caused by intense precipitation.

2.4.3.2 Intense Local Precipitation

Safety-related structures, systems, and components, including the roofs of safety-related buildings, are protected from the effects of the local Probable Maxixum Precipitation (PMP).

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The applicant has indicated that roofs of safety-related buildings are designed to dispose of local severe precipitation up to and including the local Probable Maximum Precipitation. All safety-related buildings (except containment enclosure) are designed with relatively flat roofs having an inclination of approximately 1%. Parapets around the perimeters of safety-related buildings have been designed low enough (approximately 9 inches higher than the roof crown) so that if internal roof drains become clogged, thereby causing impoundment, water would overflow before basic roof loading would be exceeded.

There is an underground storm drainage system designed to carry runoff from the local PMP and the site is graded so that there would be less than 1/2 foot of ponding. At the request of the staff, the applicant analyzed for the case of complete blockage of the storm drainage system. They concluded that ponding with blocked site drainage would be less than 0.6 feet.

Entrances to safety-related buildings are at least Elevation 21.0 (one foot above plant grade) or are provided with one foot high curbs. During early stages of the plant design, the applicant assumed plant grade at El 20.0 ft (MSL). Later refinements of the plant grade have established drainage patterns which are controlled by the crown elevation of the roads at Elevation 20.5 ft (MSL). We have asked the applicant to reanalyze the effects of intense local precipitation on the safety related facilities, using the current controlling plant design grades and building layout at the site.

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Using the procedures described in SRP Section 2.4.2, we have reviewed the applicant's methods of runoff analysis for site drainage during the local PMP. The staff concludes that the applicant must reanalyze the effects of intense local precipitation with respect to current controlling plant grade elevations and layout of buildings on the site.

2.4.3.3 Potential Dam Failures ~~(Seismically Induced)~~

The applicant reports only two small artificial ponds upstream of the site and concludes that they represent no hazard to the safety related facilities of the plant. Using the procedures described in SRP Section 2.4.4, we have reviewed the applicant's submittal, and conclude that the plant meets the requirements of GDC-2 with respect to hydrologic aspects of dam failures.

2.4.3.4 Ice Flooding

The applicants reported that ice up to one foot thick has occurred in the vicinity of the site. However, due to topographic considerations, ice blockage of waterways would not produce water levels in excess of the design basis water level (PMH).

Based upon our review of the FSAR using SRP Section 2.4.7, the staff concludes that ice would not cause the design basis flood, and that the plant meets the guidelines of R.G. 1.102 and the requirements of GDC-2 with respect to ice flooding.

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2.4.3.5 Probable Maximum Tsunami Flooding

The applicants evaluated historical tsunamis on the Atlantic Coast and the causal mechanisms. The applicants conclude that while tsunamis may occur, they will not exceed the design basis water level of the PMH.

Based upon our analyses using SRP Section 2.4.6, the staff concludes that the plant ~~design for tsunamis~~ meets the guidelines of R.G. 1.102 and the requirements of GDC-2 with respect to tsunami flooding.

2.4.3.6 Probable Maximum Surge and Seiche Flooding

Major historical surge flooding on the New Hampshire coast has been induced by "northeasters" rather than by hurricanes. The applicants, however, analyzed the potential surge flooding from both the probable maximum northeaster (PMN) and the Probable Maximum Hurricane (PMH) and concluded that the PMH would produce the design basis water level at the plant site. The parameters for the PMH were obtained from HUR 7-97, "Interim Report - Meteorological Characteristics of the Probable Maximum Hurricane, Atlantic and Gulf Coast of the United States," E.S.S.A. (now NOAA), May 7, 1968. The reference provides a range of characteristics to define the size (radius to maximum winds) and speed (forward translation) of the hurricane. The PMH results from the most critical combination of these, and other characteristics, including a postulated critical track to the site. Characteristics identifying the PMH for the site are a Central

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Pressure Index (CPI) of 27.42 inches of mercury, asymptotic pressure of 30.42 inches of mercury, radius of maximum winds of 56 nautical miles, maximum forward speed of translation of 52 knots, and maximum 10-minute sustained wind speed of 129 knots. Coincidentally the applicants assumed an astronomical tide of 10.6 ft. mean low water.

The applicants coastal surge estimate also includes consideration of an initial rise of 0.9 feet (sea level anomoly). The design basis open coast surge near the site was calculated by the applicant to have a still water level of 14.5 feet MSL. The staff concurred with this estimate in the CP review on the basis of an independent assessment.

Surge ~~flood~~ on the open coast would affect the site by increasing the water level in Hampton Harbor, adjacent to the site. The flood in Hampton Harbor was calculated by "routing" the open coast storm surge and including the effects of runoff from heavy precipitation, wind setup across the bay and wind waves coming from offshore, or generated within the bay. The applicant assumed that the precipitation runoff was from the Standard Project Flood (SPF), which was ~~is generally~~ taken as ~~about~~ half ~~as great as~~ that for the PMF. Flow into Hampton Harbor from the Atlantic Ocean was calculated on the basis of the inlet width, open coast water level and harbor water level. The analysis included, in addition to the highest open coast surge, a hurricane

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which gave a smaller, but longer lasting surge, in order to explore the possibility that this combination would result in a higher flood level in Hampton Harbor. This, however, proved not to be the case. Setup caused by hurricane winds blowing across the harbor was added to the water level computed from the SPF runoff and PMH surge to give a design basis stillwater level of 15.6 feet MSL at the site. The staff concurred at the CP stage with the applicant's prediction of stillwater level at the site.

The applicants estimated the waves and wave runup at the site coincident with the SPF/PMH surge. The estimate was based on consideration of modified deep water waves entering the harbor through the entrance and regenerated waves inside the harbor. The maximum runup elevation on the north and northeast sides of the site are 19.6 feet MSL and 18.0 feet MSL, respectively. The southeast side of the site, particularly at the vertical seawall, would be subject to extensive wave overtopping in such a situation. The revetments and seawalls themselves have been designed to survive the PMH, including wave impact forces and erosion of their foundations. Based on our review of the design the staff concludes that these shore protection features would survive the Design Basis Flood.

The height of the shore protective structures relative to the design basis stillwater level would preclude transmission of waves from this event across plant grade and safety related structures will, therefore, not be subjected to wave forces. However, runoff through the site from waves overtopping

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the seawall combined with that from local intense precipitation will result in water elevations above plant grade. At the CP licensing stage the applicant indicated that the effects of local intense SPF precipitation or plant grade (Elevation 20.0 ft MSL) was used in combination with the overtopping waves. The resulting maximum flooding elevation occurring at the time of peak overtopping was 20.6 feet (MSL). This value utilized available drainage capacity in the storm drainage system.

Since the issuance of the SER-CP, ANSI Guide N170 has suggested that for drainage basins of less than 300 square miles, the Probable Maximum Flood (PMF) resulting from the PMP should be combined with the Probable Maximum Hurricane for the determination of Design Basis Flood. At the request of the staff, the applicant analyzed the potential increased surge level over the originally determined level in Hampton Harbor for the case of the PMF combined with the PMH. They determined that the combined event would increase the stillwater level less than 0.1 feet above that calculated for the PMF/PMH event, primarily because the storm surge dominates the flood still water level. The staff has requested the applicant to provide a detailed analysis to support this contention. The staff has further requested the applicant to evaluate the effect on the wave overtopping rate resulting from the PMF/PMH event and determine the maximum flooding level resulting from this wave overtopping runoff and concurrent PMP runoff and to justify the credit, if taken, for flow through the storm drainage system.

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Based upon our analyses using SRP Section 2.4.2, 2.4.5, and 2.4.10, the staff cannot conclude that the plant design for direct storm surge flooding meets the guidelines of R.G. 1.102 and the requirements of GDC-2, because the flooding level on plant grade resulting from wave overtopping runoff and concurrent PMP runoff associated with the PMF/PMH event remains an open item.

2.4.4 Flood Protection Requirements

The flood design bases for the site comprise intense local precipitation and the PMH induced stillwater level with coincident waves, wave runup, and wave overtopping.

Flood protection from the Design Basis Flood and associated wave runup is provided by riprap revetments with slopes of 1 on 1.5, a vertical seawall, and retaining walls. This shore protection has been designed to withstand the effects of direct wave attack and the hydrostatic and hydrodynamic forces associated with the Design Basis Flood. The vertical seawall and certain sections of the riprap revetment (to a lesser extent) are subjected to wave overtopping. The wave overtopping at the vertical seawall is the major source of runoff through the plant site and the resultant ponding. Problems that remain outstanding all relate to the wave overtopping during the PMF/PMH event. They are as follows:

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a) scour protection of plant grade behind the vertical seawall and adjacent to electrical manholes #13/14 and #15/16,

b) routing of the combined runoff from PMP and PMF/PMH wave overtopping through the plant site particularly as associated with buildings, crown of roads, and/or obstacles constricting flow and increasing ponding level, and

c) identification of plant access that are effected by locally increased ponding level.

Based upon our analysis using SRP Sections 2.4.2, 2.4.5 and 2.4.10 the staff concludes that the applicant has not shown that the plant design flood protection meets the guidelines of R.G. 1.102 and the requirement of GDC-2 as noted above regarding scour protection, ponding levels, and plant access affected by ponding.

2.4.5 Cooling Water Supply

2.4.5.1 Description of Normal and Emergency Supply

Condenser cooling water and service cooling water are supplied to the plants from the Atlantic Ocean through a tunnel bored in rock. The intake structure is located offshore in about 60 feet of water. Normal once through condenser and service cooling water for both units is provided through the 19 foot diameter intake tunnel at a flow rate of 824,000 gallons per minute

(412,000 gallons per minute per unit). Normal full power operation of service cooling water for both units is supplied at the flow rate of 42,000 gallons per minute (21,000 gallons per minute per unit). A minimum flow rate of 19,600 gallons per minute for both units (9,800 gallons per minute per unit) is required for emergency shutdown.

The service water pump bay is supplied with two lines from the intake structure. The discharge structure, also located in deep water, may be used to supply service water in the event of the intake line being unavailable. The applicant has also provided mechanical draft cooling towers which could be used to supply emergency cooling water for plant shutdown irrespective of the normal system.

2.4.5.2 Adequacy of Cooling Water Supply

The applicant analyzed the adequacy of the cooling water supply from the effects of severe natural phenomenon. While ice up to one foot thick has occurred near the site, the submerged offshore location of the intake and discharge structures would make them immune to ice blockages. Thus we conclude that the intake and discharge structures are adequately protected against ice effects.

The applicants reported the minimum historical astronomical tide for Hampton Harbor is -6.3 feet MSL. The maximum set down due to the PMH wind

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field blowing offshore was estimated to be 3 feet and was assumed coincident with the minimum astronomical tide. The resultant water level would be -9.3 feet MSL. The ocean intake and discharge structures would be well submerged and thus functional during this event.

The applicant's describe the Ultimate Heat Sink as comprising the Atlantic Ocean via the intake and discharge tunnels and a mechanical draft cooling tower. The tunnels are not designed for the safe shutdown earthquake (SSE). However, the ocean intake and tunnel could provide the required emergency water flow with as much as 95 percent of the flow area blocked. An alternate seismic Category I source of cooling is provided from a mechanical draft cooling tower. The applicants state that the system will meet the natural phenomena criteria and design meteorology suggested in Regulatory Guide 1.27. The adequacy of this system is still under review by the staff.

Based upon our review, using the procedures described in SRP Sections 2.4.5, 2.4.7 and 2.4.11, and the guidance of R.G. 1.27, we conclude that with respect to low water levels and ice blockages of the water intakes, the plant design is acceptable, and meets the requirements of GDC 2. Our evaluation of the Ultimate Heat Sink's ability to provide emergency cooling in accordance with the guidance in Regulatory Guide 1.27 is still under review.

2.4.6 Groundwater

The site is underlain by relatively impermeable marine sediments. Beneath these sediments are shallow unconsolidated surficial deposits which constitute the principal aquifer in the area. The aquifer is composed of beach deposits, swamp deposits, and glacial drift. The drift is composed of till, ice contact, margin and outwash deposits. Beneath these materials is bedrock in which groundwater occurs only in fractures.

The largest producing wells in the area are in the ice contact deposits of stratified sand and gravel. The formation provides the water supply for the communities of Seabrook, Salisbury, and Hampton.

The outwash deposits provide adequate supply for smaller users. Yields generally do not exceed 100 gpm. A few wells, delivering only a few gallons per minute, are developed in the till and beach sands. Most bedrock wells provide yields of less than 10 gpm.

The applicants will have no wells on site, but will obtain water from the town of Seabrook. Normal operational usage will be about 200 gpm, which will include potable and sanitary supply and makeup for the demineralized water system and fire system. The estimated plant startup demand will be about 350 gpm. The groundwater supply is not essential for the safe shutdown of the plant.

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There are no public water supply wells within 2 miles of the plant and only 2 private wells within 1 mile of the site. The groundwater gradient is generally eastward at the plant site toward the neighboring marsh. All of the existing wells are upgradient from the plant, and therefore could not be affected by accidental contamination of the ground water on the site.

The design basis for groundwater hydrostatic loading of safety-related structures is conservatively taken as plant grade of +20 feet MSL, even though the water table normally is between about 10 and 17 feet MSL.

Our review was based on the guidance of SRP Section 2.4.12. We have determined that the site does not affect the safety of neighboring groundwater supplies, that emergency shutdown of the plant does not depend on groundwater supplies, and that safety related structures have been designed to conservative groundwater levels. We conclude, therefore, that the site meets requirements of GDC-2, 10 CFR Part 100, and Appendix A thereto, 10 CFR Part 50, and GDC-4.

2.4.7 Accidental Releases of Liquid Effluents in Ground and Surface Waters

The accidental release of radioactive liquids to the ground or directly to surface water would not affect drinking water users. Radioactive water which might be accidently released to the circulating water system would be carried to the Atlantic Ocean where it would be highly mixed by the discharge diffusor and be carried away by ambient ocean currents.

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Radioactive contamination of groundwater would be transported in the direction of the saltwater marsh and away from areas of groundwater usage. The staff has conservatively estimated the travel time from the site to the waters of the marsh to be on the order of 170 days. Most radionuclides would travel even slower because of sorption with sediments. Concentrations of radionuclides in surface waters from such an improbable release would be well below hazardous levels, for accidents within the design basis. Furthermore, migration could be largely arrested before radionuclides reached surface water.

The staff therefore concludes that accidental releases of liquid radioactivity from accidents within the design basis would not pose a threat to public health and safety, and that the plant meets the requirements of 10 CFR Part 100 with respect to potential accidental releases of radioactive effluent. The staff relied on the guidance of SRP 2.4.12, 2.4.13, R.G. 1.113, 10 CFR Part 20 and 10 Part 100 in performing its analysis.

2.4.8 Technical Specifications and Emergency Operation Requirements

Based upon our review in accordance with SRP Section 2.4.14, the staff can not conclude at this time whether technical specifications or emergency operating plans are required for the flood protection.

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2.4.9 Conclusions

Based on our independent review and analysis as described above, the staff has requested the applicant to reanalyze the flood design bases and to provide protection to the Seabrook Station from both locally severe precipitation (up to and including the localized PMP) and from the stillwater, waves, and wave runup and overtopping induced by hurricanes as severe as a combined Probable Maximum Flood/Probable Maximum Hurricane event. We, therefore, cannot conclude that the plant meets the requirements of GDC-2 with respect to flooding. The staff concludes the plant will not adversely affect groundwater users and any accidental spill will migrate toward the adjacent tide water (marsh) areas rather than toward existing wells. Such spills, if allowed to proceed, would be diluted in Hampton Harbor and be further diluted in the Atlantic Ocean. The staff continues to review the conformance of the proposed ultimate heat sink design to the suggested criteria of Regulatory Guide 1.27, and the hydrologic criteria of GDC-44.

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GEOLOGY AND SEISMOLOGY

2.5 Geology and Seismology

In accordance with Sections 2.5.1, 2.5.2, and 2.5.3 of the revised Standard Review Plan (NUREG-0800), the staff has reviewed pertinent seismological and geological information that has become available since the issuance of the construction-permit (CP) Safety Evaluation Report (SER) and SER supplement in 1974. This new information includes evaluation of faults and other geologic features found in excavations at the site, recently published literature (including both published and unpublished geological and seismological information obtained from the NRC-sponsored New England Seismotectonic Study), and a set of independently derived site-specific spectra. New data resulting from the recent central New Brunswick earthquake of January 9, 1982 and the Gaza, New Hampshire earthquake of January 18, 1982, is not yet available and may be important for our review. We have asked the applicant to document these events and, specifically, to address possible implications these events may have on the choice of the maximum historical earthquake for the safe shutdown earthquake, the appropriate attenuation model for the northeastern United States, the assumed relationship between peak vertical and horizontal acceleration, and the probability of exceeding the operating basis earthquake during the operating life of the plant.

Considering the information available at this time, the staff reaffirms their earlier conclusion, stated in the CP-SER, that the applicant has adequately investigated and characterized the seismological and geologic hazards at the site and, from the standpoint of those hazards, the site is acceptable. In particular, we conclude that:

(1) Geological and seismological investigations and information provided by the applicant and required by Appendix A to 10 CFR Part 100 provide an

adequate basis to establish that no capable faults exist in the plant site area which would cause earthquakes to be localized there.

(2) There is no evidence that a potential exists for surface faulting at the plant.

(3) The acceleration level (0.25 g) proposed for the safe shutdown earthquake is an acceptable acceleration level to anchor a Regulatory Guide 1.60 spectrum for the seismic design of the plant in conformance with Appendix A to 10 CFR Part 100.

In addition, the staff finds that the applicant has satisfied the requirements of:

(1) Regulatory Guide 1.70 (Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, Revision 2).

(2) 10 CFR 50, Appendix A (General Design Criterion 2) with respect to protection against natural phenomena such as earthquakes, faulting, and collapse.

(3) 10 CFR Part 100 (Reactor Site Criteria) with respect to the identification of geologic and seismic characteristics used in determining the suitability of the site.

(4) 10 CFR Part 100, Appendix A (Seismic and Geologic Siting Criteria for Nuclear Power Plants) with respect to obtaining the geologic and seismic information necessary to determine (1) site suitability, and (2) the appropriate design of the plant. In complying with this regulation the applicant also meets the staff's guidance described in Regulatory Guide 1.132 (Site Investigations for Foundations of Nuclear Power Plants), Regulatory Guide 4.7 (General Site Suitability Criteria for Nuclear Power Stations), and Regulatory Guide 1.60 (Design Response Spectra for Seismic Design of Nuclear Power Plants).

A review of the bases for the staff's conclusions is given in the following sections.

2.5.1 Basic Geologic and Seismic Information

2.5.1.1 Regional Geology

The Seabrook site is in the Seaboard Lowland Section of the New England Physiographic Province (Fenneman, 1938 and Thornbury, 1965). The Seaboard Lowland Section ranges in elevations from mean sea level (msl) to +500 feet msl near its boundary with the Upland Section.

The New England Physiographic Province is a northern extension of the Appalachian Mountains which has been modified by glaciation. Bedrock is generally overlain by a few feet to a few hundred feet of glacial deposits.

Based on our review of the Seabrook site, and past reviews of the Pilgrim 2, Montague and New England 1 and 2 sites, the staff conludes that the Seabrook site is within the New England-Piedmont Tectonic Province. This is in accord with the tectonic province concept of King (1969), Rodgers (1970), Eardley (1974) and Hadley and Devine (1974). The New England-Piedmont Province is comprised of Precambrian and Paleozoic basement and sedimentary rocks that have been extensively folded, faulted, metamorphosed, and intruded by igneous rocks during successive episodes of orogenic activity. Although we accept the larger tectonic province, the New England-Piedmont Province in New England can be further subdivided based on geology into the Southeastern New England Platform, the White Mountain Plutonic Series, and the New England fold belt. The Southeastern New England Platform is separated from the rest of the New England-Piedmont Province in the site region by the Clinton-Newbury and Bloody Bluff fault systems. The boundary farther to the south and west is the Honey Hill and Lake Char thrust fault systems. It has been suggested (Rodgers 1972) that these generally northerly dipping thrust faults and associated rocks of high grade metamorphism represent a Paleozoic collision zone between a plate containing the Southeastern New England platform and a plate containing the New England fold belt.

The Southeastern New England Platform is composed of Precambrian granitic basement rocks, Silurian and Devonian volcanic and instrusive rocks, Cambro-Permian basins, an area with Late Paleozoic intrusive and metamorphic rocks, and the zone of mid-Paleozoic, post-metamorphic thrust faulting represented in the site region by the Clinton-Newbury and Bloody Bluff fault systems.

The Southeastern New England Platform has undergone relatively little structural deformation or metamorphic alteration since the Paleozoic (240 million years before present mybp). Known faulting is related to basin development during the Cambrian-Permian (570 mybp to 240 mybp). These basins include the Narragansett, Boston, North Scituate, Woonsocket, and Norfolk.

The White Mountain Plutonic series is an elongate, north-northwest oriented group of alkaline intrusives that extend from northeastern Massachusetts through New Hampshire. They were emplaced from Permian to Cretaceous. As a result of reviews of the Indian Point 3, Seabrook, Montague, Pilgrim 2, and New England sites, the staff concluded that there was a spatial relationship between the zone defined by these intrusives, which represent the youngest significant deformation features in New England, and historic seismicity. The largest New England earthquakes occurred within this zone, referred to herein as the "New Hampshire-Cape Ann seismic zone" (Section 2.5.2.3.). The Seabrook site is located within this zone.

The New England fold belt of the New England-Piedmont Province consists of major northeast-southwest striking anticlinoria and synclinoria composed of metamorphic rocks and plutonic bodies. From the west in Vermont and western Massachusetts to the Atlantic Coast these major folds are: the Green Mountains - Sutton Mountain anticlinorium, the Connecticut Valley - Gaspe, synclinorium, the Bronson Hill-Boundary Mountain anticlinorium, the Merrimack synclinorium, and the Coastal anticlinorium. The site lies within the White Mountain Plutonic belt where it cuts across the southern end of the Coastal anticlinorium.

The site lies about 7 miles north of the Clinton-Newbury fault system. This fault system and other regional faults are discussed in greater detail in Section 2.5.3.

2.5.1.2 Site Geology

The site is located on a small wooded peninsula which, before removal of surficial soil, rose to an elevation of +20 to +30 ft msl. The peninsula is bounded by marshes and tidal tributaries.

The site was underlain by a veneer of glacial till, which was removed from the area where plant structures are located during site preparation activites. Groundwater before construction was at a depth of 5 to 10 feet below ground surface.

Bedrock on which the plant is founded consists of two principal lithologies: metasediments of the Merrimack group, and quartz diorite of the Newburyport pluton. Throughout the Cenozoic age and prior to glaciation, the site rock had been exposed to subaerial weathering and erosion. The metasediments of the Merrimack Group were weathered and eroded much more deeply than the more competent quartz diorite. Glaciation removed most of the weathered soil and rock thereby creating an irregular, rolling topography with valleys and depressions where the Merrimack rocks outcropped and ridges and knobs where the quartz diorite outcropped. Maximum relief of bedrock surface in the site area was found to be on the order of 200 feet. Overlying this irregular bedrock surface are various thicknesses of till, outwash, and marine clays and silts. Recent deposits of beach sands and gravels and peat are also present.

The Merrimack group (Rye, Kittery, and Eliot formations) is a fine-grained rock consisting of quartzite, schist, granulite slate, and phyllite. Recent interpretations as to the age of these rocks are that they are no younger than Ordovician (435 mybp), and could be as old as Precambrian. However, previous investigators have assigned them to the period from Ordovician to Devonian.

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The Newburyport Pluton at the site is a gneissoid quartz diorite containing intrusion breccia and inclusions of meta-sedimentary rock. It is currently interpreted as having intruded into the rocks of the Merrimack Group during the Acadian Orogeny in the Devonian Period (410 mybp to 360 mybp). The pluton has also been interpreted in the past as being older. The Newburyport pluton is truncated to the south by the Scotland Road fault, which is the northernmost major splay of the Clinton-Newbury fault system. A different rock type of unknown age is juxtaposed across the fault.

The Exeter pluton lies to the north of the site. Like the Newburyport, it is a quartz diorite that has intruded the Merrimack Group. It has been dated as late Silurian to Early Devonian age (420 mybp to 390 mybp).

The plant is located just south of the contact between the meta-sedimentary rocks of the Merrimack Group and the quartz diorite of Newburyport Pluton. Excavation for the plant is within the quartz diorite pluton, however, two large xenoliths of Kittery formation rock were encountered in the excavation: one in the area of the Unit 2 containment; and the other in the pumphouse excavation.

The site area is crossed by numerous mafic dikes that range in thickness from 1 inch to 20 feet, and are spaced from 30 to 300 feet apart. The dikes generally strike from N35° to 45°E and dip from 80°N to 75°S. Most of the dikes were discontinuous in exposure, of short length, and formed an en echelon, left-stepping pattern. Radiometric dating of these dikes indicates at least two periods of emplacement, Early Triassic (236 mybp to 212 mybp) and Upper Paleozoic (295 mybp to 255 mybp).

Structurally the site lies near the southwest flank of the southwest plunging Rye anticline. The Rye anticline strikes northeast and has influenced the structural fabric of both the Merrimack rocks and the Newburyport Pluton.

Numerous minor faults were mapped in the site excavations, these faults were shown to be 200 million years old. Site faulting is discussed in Section 2.5.3.

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Based on the applicant's mapping of rocks in outcrop in the site area and in site excavations, the most common joint trends within the Newburyport Pluton were N 30° to 45°E with steep dips to the northwest. Secondary trends of N30° to 50°W that dip steeply to the northeast were also mapped. Dominant joint trends in the metamorphic rocks ranged from N 50° to 60° W to the north away from the pluton, then swing to N 30° to 45° E near the margin of the Newburyport Pluton. Foliation in both rock types strikes east-west and dips vertical.

In the plant area, jointing in the rock is relatively frequent in the upper ten to twenty feet, but decreases substantially with depth. The deeper foundations are as much as -63 feet below sea level. Weathering, which also characterizes the upper ten to twenty feet, does not penetrate to the area of foundations except along some of the joints, faults and shear zones. These features are narrow, usually only a fraction of an inch wide. The applicant stated that where these features were found to have undergone severe weathering below foundation levels, the weathered material was removed. There are no potentially unstable natural or cut slopes on the site.

Extensive geologic mapping was done in the circulating water system tunnels. The circulating water system is made up of 2 tunnels extending from the plant to the east: a discharge tunnel and an intake tunnel. The tunnels are about 16,500 feet and 17,000 feet long, respectively, and run parallel for about 10,000 feet at a distance apart of 90 feet on centers. Beneath the state park at the coastline, the intake tunnel diverges to the northeast. The tunnels slope landward from a depth below sea level of about 170 feet at the eastern ends to 250 feet below sea level under the pumping house at the site. The tunnels were machine bored by moles at a diameter of 22 feet.

The tunnel alignments are located along the contact between diorite of the Newburyport pluton and metasedimentary rocks of the Kittery formation (Merrimack group). The Kittery is basically 2 lithologies in the tunnel areas, quartzite and mica schist. These rocks contain joints, folds and faults, and have been intruded by steeply dipping diabase dikes that are generally oriented in a northeast-southwest direction.

In the intake tunnel quartzite is the most abundant rock type. Diorite is present in the immediate site vicinity and land shaft area, and beneath the state park. Mica schist is the least abundant. Diabase dikes are present throughout the length of the intake tunnel.

Diorite is the most abundant rock type exposed in the discharge tunnel. It was penetrated between the plant and Hampton Marsh, under the eastern half of Hampton Harbor, and under the far offshore area including the discharge diffusers. Mica schist is the least common rock, and is present under the western half of Hampton Harbor. Quartzite occurs in the tunnel associated with or alternating with the diorite. As in the intake tunnel, diabase dikes are present throughout the length of the discharge tunnel.

In the circulating water system tunnels, jointing in the quartz diorite is similar to that at the plant site, but orientations of joints in the metamorphic rocks are more diverse. This is believed to be true because the tunnels are located close to the contact between the Newburyport pluton and the Merrimack country rock. Intrusion breccia is also common in the metamorphic rocks exposed in the tunnel. Jointing appears to be most pronounced in association with contacts between diorite and metasediments (especially schist). Joints are widely spaced away from these contacts particularly in the diorite.

Weathering varied from rust-stained joints to total disintegration of rocks. Intense weathering seems be concentrated along steeply dipping fractures or fracture zones and diabase dikes. In the intake tunnel, zones of widespread weathering are beneath the east side of Hampton Harbor within zones of extensive jointing and lithologic contacts. In the discharge tunnel the most extensive weathering was found near the shaft in the plant vicinity at the diorite and quartzite contact, and in badly jointed mica schist beneath the west side of Hampton Harbor.

Based on our review, which included 3 site visits to examine geologic features exposed in the excavations and tunnels, we conclude that the applicant has performed an adequate investigation, prior to and during construction, to define the regional and site geology. There are no geologic hazards in the site vicinity,

and the rock is of high quality as demonstrated by the high percentage of core recovery in borings, laboratory test results, and high field compression and shear wave velocities.

2.5.2 Vibratory Ground Motion

The staff has reviewed the seismological and geological investigations performed by the applicant to establish the acceleration for seismic design of the plant, the procedures and analyses used by the applicant to determine the safe shutdown earthquake and the operating basis earthquake, and the resulting seismic design bases for foundations. All areas of review and review procedures identified in Section 2.5.2 of the Standard Review Plan (NUREG-0800) were followed. Our conclusions resulting from this review may be summarized as follows: (1) Seismological information provided by the applicant and required by Appendix A to 10 CFR Part 100 provides an adequate basis to establish that no capable faults exist in the plant site area which would cause earthquakes to be localized there. (2) The acceleration level (0.25g) proposed for the safe shutdown earthquake is the appropriate acceleration level for anchoring a Regulatory Guide 1.60 spectra for the seismic design of the plant in conformance with Appendix A to 10 CFR Part 100.

A review of the bases for the staff's conclusions regarding vibratory ground motion is presented in Sections 2.5.2.1 through 2.5.2.7.

2.5.2.1 Seismicity

In Section 2.5.2.1 of the Final Safety Analysis Report (FSAR) the applicant presents the results of a thorough study of historical seismicity in New England up to June, 1979. The study shows that New England is characterized by the infrequent occurrence of low to moderate intensity earthquakes; considering past patterns of population density and other factors, the early record can be considered complete only for events with epicentral Modified Mercalli (MM) Intensities of VII or greater. The largest earthquakes in New England have occurred in a region extending from Cape Ann and Boston in Massachusetts to central New Hampshire. These include the 1727 Newbury Event (intensity VII (MM)), the 1755 Cape Ann Event (intensity VIII (MM)), the 1817

Woburn Event (intensity VI (MM)), and two intensity VII (MM) events in 1940 at Lake Ossippee, New Hampshire. The MM intensities of some of these events have been reduced since the issuance of the CP-SER, following re-evaluation by the U.S. Geological Survey (1980) and other investigators (Chiburis, 1981).

The Northeastern United States Seismic Network (NEUSSN), partially funded by the NRC, has been in operation since 1975 and provides locations and magnitudes of the more numerous, smaller earthquakes. Since the start-up of the NEUSSN, the closest event to the site has been a magnitude 2.3 earthquake at a distance of 3.4 km. This event is part of the low level of microearthquake activity occurring throughout New England.

The staff considers the applicant's documentation of New England seismicity to be excellent. However, since the issuance of the FSAR, significant seismic activity has occurred in New England that also needs to be documented: the central New Brunswick earthquake sequence beginning January 9, 1982 and the Gaza, New Hampshire earthquake of January 18 (January 19, Greenwich Mean Time), 1982. The New Brunswick and New Hampshire earthquakes both produced no greater than Intensity VI (MM) shaking, well below the design intensity for the Seabrook site. However, the records from these events--the only strong-motion records ever obtained in New England--will provide important new information on the character of earthquake ground motion in New England. The applicant is documenting this new information and the staff will evaluate it as it becomes available.

2.5.2.2 Geologic and Tectonic Characteristics of Site and Region

See Section 2.5.1.1.

2.5.2.3 Correlation of Earthquake Activity with Geologic Structure or Tectonic Provinces

In the CP-SER the staff determined that the site lay within a northwest-southeast trending zone (called the "Boston-Ottawa seismic belt") of potential intensity VIII (MM) shaking, extending from the Canadian Shield through

Montreal and Boston and out to sea. Evidence for the Boston-Ottawa seismic belt included the observation that seismicity within the belt (with the exception of central Vermont) is high compared to the rest of New England, the apparent coincidence with the belt of a zone of P-wave travel-time anomalies relative to adjacent areas (Fletcher, et al. 1972), the inclusion within the belt of the Monteregian Hills and White Mountain intrusives (plutons), and the possible association of the belt with the Kelvin seamount chain through an ancient fracture zone (LePichon and Fox, 1971). New evidence, however, indicates that the Boston-Ottawa seismic belt may not be a continuous feature. After further research, Fletcher, et al. (1978) concluded that there is no apparent relationship between P-wave travel-time anomalies and the seismic belt. A recent study by Sbar and Sykes (1977) suggests that the area of low seismicity in central Vermont may be due to a mismatch (for the generation of earthquakes) between the local direction of maximum compressive crustal stress (north-south) and the probable orientation of unhealed faults in central Vermont (also north-south, parallel to the predominant structural trends).

On the basis of the new evidence, the staff concludes that the Seabrook site lies within a seismic zone that extends from Cape Ann to central New Hampshire (and not beyond), and that is anomalous with respect to the New England Piedmont Tectonic Province as a whole. Various hypotheses of the configuration and nature of the New Hampshire-Cape Anne seismic zone have been advanced. Hadley and Devine (1974) believe that epicenters may correlate with northeast-trending faults in the area. In an extensive investigation conducted for the Pilgrim Station Unit 2 (USNRC, 1977), Boston Edison Company concluded that the larger earthquakes in this region are spatially and causally related to cylindrical Mesozoic mafic plutons and tangential fault zones. The applicants for Seabrook also subscribe to this position.

As stated in the Safety Evaluation Reports for Montague Station, Units 1 and 2 (USNRC, 1976) and Pilgrim Unit 2 (USNRC, 1977), the staff has found that the New Hampshire-Cape Ann seismic zone is spatially associated with structures represented by [illegible] shallow intrusives and volcanic rocks of predominantly Jurassic-Cretaceous age. This zone extends in a north-northwest direction from Cape Ann and includes the White Mountain magma series. However, as stated in

the Pilgrim 2 SER, we find, for the purposes of power-plant licensing, the existing geologic and seismic data base is not sufficiently developed to allow correlation of the larger earthquakes with structures near specific known or inferred mafic plutons, as proposed by the applicants, or with specific northeast-trending structures. Therefore, in conformance with Appendix A to 10 CFR Part 100, future earthquakes must be assumed to occur throughout the New Hampshire-Cape Ann seismic zone.

2.5.2.4 Maximum Earthquake

The 1755 Cape Ann event is the maximum historical earthquake within the New Hampshire-Cape Ann seismic zone, which includes the Seabrook site. This event, which caused intensity VIII (MM) shaking, had a magnitude (m_{bLg}) of approximately 6.0 (Street and Lacroix, 1979). Following Appendix A to 10 CFR Part 100, the staff considers this to be the controlling earthquake in determining the Safe Shutdown Earthquake. As mentioned above, the applicant maintains that the larger earthquakes in New England are related to known tectonic structures that are confined to a tectonic province different than that of the site. However, for licensing and design purposes, the applicant has assumed that the 1755 earthquake could occur at the site.

2.5.2.5 Seismic Wave Transmission Characteristics of the Site

All seismic Category I structures are founded on sound bedrock or on engineered backfill extending to sound bedrock. The bedrock compressional wave velocities range from 16,500 to 18,500 ft/sec and the bedrock shear wave velocities range from 8,000 to 10,000 ft/sec, indicating quite competent rock in which no local amplification or deamplification effects are expected. Fill concrete was used as the engineered backfill beneath the foundations of all seismic Category I structures except for safety-related electrical duct banks, five electrical manholes, and the service-water pipes, all of which are founded on offsite borrow or tunnel cuttings. The material properties determined by the applicant for the bedrock and engineered backfill are reviewed in Section 2.5.4. The effect of the offsite borrow and tunnel cuttings on the seismic response of the

manholes, ductbanks and water pipes is incorporated in the seismic analysis of systems and subsystems, reviewed in Sections 2.5.4.7, 3.7.2 and 3.7.3.

2.5.2.6 Safe Shutdown Earthquake

Following Appendix A to 10 CFR Part 100, the staff considers the 1755 Cape Ann earthquake to be the controlling earthquake in determining the Safe Shutdown Earthquake (SSE). We see no evidence such as the occurrence of numerous earthquakes of size similar to the 1755 event, or earthquake-localizing structures near the site, that would warrant the choice of an earthquake larger than the 1755 event for the SSE. Because the $m_{bLg} \approx 5.7$ central New Brunswick earthquake occurred within the New England Piedmont Tectonic Province and was, apparently, the largest earthquake to occur in central New Brunswick in historical times, the staff has requested the applicant to document this event and specifically, to assess the implications of its occurrence for the choice of the (maximum historical) 1755 earthquake as the controlling earthquake. The applicant is currently preparing his response.

Given that the SSE corresponds to the occurrence of the intensity VIII (MM) 1755 Cape Ann earthquake at the site, the applicant claims and the staff concurs that a Reg. Guide 1.60 response spectrum, asymptotic to ("anchored at") 0.25g peak spectral acceleration at high frequencies, is an adequate and conservative representation of the SSE at the reactor foundation level with (no structure present). The 0.25g acceleration value fits the empirical relation of Trifunac and Brady (1975) for the trend of the means of peak horizontal acceleration versus intensity. As stated in Section 2.5.2.6 of the Standard Review Plan, the staff generally views a "reference acceleration for seismic design" (high frequency asymptote of a Reg. Guide 1.60 design response spectrum) as conservative if, when applied at the ground surface, it results in a value at the foundation free-field level as large as would be obtained from the empirical relation of Trifunac and Brady (1975).

For vertical ground motion, the applicant assumes a reference acceleration for seismic design of 0.175g, somewhat greater than would be obtained from the relation of Trifunac and Brady (1975). Studies of western U.S. earthquakes

(e.g., Shannon & Wilson and Agbabian Associates, 1979) have shown that the assumption is generally conservative that vertical ground motion levels are 2/3 those of horizontal motion. Relatively high peak vertical accelerations recorded during past earthquakes appear to be related to site and fault conditions not present at the Seabrook site. For instance, the high peak vertical acceleration recorded at Station #6 during the 1979 Imperial Valley earthquake is thought to be due to soil amplification from a strong near-surface P-wave velocity gradient (Mueller et al., 1982) and the interaction of the propagating rupture with the thick sedimentary sequence (Archuleta and Spudich, 1981). In the case of the 1976 Gazli earthquake, high vertical accelerations were recorded at a site over a fault which ruptured vertically towards the surface (Hartzell, 1980). The applicant is evaluating the levels of vertical and horizontal ground motion recorded during the recent New Brunswick and New Hampshire earthquakes; the staff will review the new information as it becomes available.

Several studies have aided the staff in judging of the adequacy of representing the SSE by a Regulatory Guide (R.G.) 1.60 design response spectrum anchored at 0.25g. In one study, a "historical analysis" of past earthquakes that have affected the site was performed for the staff by Lawrence Livermore National Laboratory (LLNL). LLNL (Bernreuter, 1982) estimated the probability that the ground motion at the Seabrook site has exceeded various levels, based on the seismic history of the northeastern U.S. and some assumed attenuation (decay of intensity with distance) model. No uncertainty was assumed for the historical record though,in reality, there is considerable uncertainty in both the locations and magnitudes of the events. The probabilities arise from the assumed dispersion (expected scatter) of the data about the attenuation model. The historical analysis, using the "Ossippee" attenuation model (discussed below), indicated probabilities of 1.0×10^{-2}, 1.0×10^{-3}, and 2.5×10^{-4} that, in any random one year time interval from 1700 to 1978, the peak ground acceleration at the Seabrook site exceeded 0.1g, 0.3g and 0.5g, respectively. Assuming that future seismicity can be characterized by the seismicity of the past 280 years, the calculated probabilities represent estimates that these peak acceleration levels will be exceeded in any one-year period during the operating life of the plant. It should be pointed out that peak ground acceleration is usually ~~lower~~ greater than the reference acceleration for seismic design (.025g for

Seabrook), which is used to anchor the R.G. 1.60 design response spectrum. LLNL assumed very conservative values for the magnitudes of the historical earthquakes that contribute most to the calculated seismic hazard, and currently is performing a sensitivity study, incorporating more recent, lower magnitude estimates into the analysis. The staff will review the results of this study as soon as they are available.

The staff has compared the 0.25g-R.G. 1.60 response spectrum with other, independently derived spectral estimates appropriate for the Seabrook site. The first comparison is with response spectra calculated from the records of a set of Intensity VIII (MM) earthquakes that occured in the western U.S. (O'Brien, 1980). Figure 2.5-1 shows ~~shos~~ the upper bound, mean, and lower bound of pseudo-relative velocity (PSRV) response spectra (for 5% damping) of the data set plotted with the 0.25g-R.G. 1.60 response spectrum; the Seabrook design spectrum lies at about the upper bound of the data. Uncertainties in this comparison result from the facts that: (1) intensity is a subjective, analyst-dependent parameter; (2) it is not clear that the intensity at the recording site was the same as in the vicinities which were used to characterize the records; and (3) these records were made at alluvial sites, whereas Seabrook is a rock site.

The second comparison is with spectra of earthquakes of nearly the same magnitude as the postulated SSE, recorded at nearby rock sites. Street and Lacroix (1979), using the falloff-of-intensity-with-distance technique developed by Nuttli (1973) and the isoseismal map produced by Weston Geophysical Research, Inc. (1976), estimated for the 1755 Cape Ann earthquake a body-wave magnitude (m_{bLg}) of approximately 6.0. LLNL, in a study performed for the staff (Bernreuter, 1982), has collected fourteen rock-site records, obtained at distances (to the fault trace) from 3 to 25 km, of U. S. and Italian earthquakes ranging in local magnitude (M_L) from 5.5 to 6.4, with a mean value of 6.0. The range in magnitude is designed to account for the uncertainty in all of the magnitude estimates, the relation between m_{bLg} and M_L, and the difference between eastern U.S. and western U.S. earthquakes. Figure 2.5-2 shows the Seabrook SSE response spectrum plotted with the upper bound, 84th percentile, 50th percentile, and lower bound of PSRV response spectra (for 5% damping) of

the LLNL data set; the SSE spectrum envelopes the 84th percentile spectrum. Another study, performed by the applicant's consultants (Holt, 1981), shows that the Seabrook SSE spectrum also envelopes the 84th percentile of spectra from a similar suite of records. In previous applications of site-specific spectra for OL reviews (e.g., Sequoyah Units 1 and 2 and Enrico Fermi Unit 2), the staff has accepted the 84th percentile spectrum as adequately conservative, given the appropriateness of the suite of records with respect to the postulated SSE and the site conditions.

The third comparison is with probabilistic "uniform hazard" spectra. The uniform hazard methodology, as developed during the Site Specific Spectra Project (SSSP), produces pseudo-relative velocity response spectra that have, at all frequencies, equal probability of being exceeded within given time periods, considering the integrated effect of earthquakes of different size (up to some maximum size) recurring at different rates in different source zones (TERA Corp. and LLNL, 1981). The calculated probabilities are intended to reflect uncertainty due to the inherent randomness of earthquake occurrences and earthquake effects. However, the probabilities themselves are uncertain due to unknown errors in the attenuation model and input parameters used in the calculations. The SSSP methodology attempts to account for the uncertainty in the input parameters by incorporating a range of "expert" judgements and then evaluating the effect of the uncertainties on the calculated seismic hazard at the site. For the Systematic Evaluation Program (SEP), the SSSP methodology was applied to several sites in the northeastern U. S. (TERA Corp. and LLNL, 1981). Expert opinions were obtained regarding the configuration of seismic source zones (regions of uniform seismic activity), the largest earthquake expected in each zone, the appropriate attenuation model (for predicting ground motion from an earthquake of a given size at a given distance), and explicit descriptions of the uncertainty in each of the input parameters.

Using the models of the northeastern U. S. developed for the SEP program, LLNL generated uniform hazard spectra for the Seabrook site (Bernreuter, 1982) on behalf of the staff. Spectra were generated for each of ten experts' overall

earthquake occurrence models, assuming several different attenuation relationships, then a single "synthesis" hazard spectrum was calculated using a complex weighting scheme using self weights supplied by the individual experts. In Figure 2.5-3 the Seabrook SSE response spectrum is plotted with the synthesis LLNL uniform hazard spectra (5% damping) for "1,000 year" and "4,000 year" return periods, assuming the Ossippee attenatuion relationship. The Ossippee attenuation model is based on data from the 1940 Lake Ossippee, New Hampshire earthquakes and has been recommended previously by the staff for use in determining ground motion in the northeastern U. S. (Jackson, 1980).

Figure 2.5-3 shows that the Seabrook SSE spectrum falls between the synthesis curves for a "1000 year" return period and a "4000 year" return period and, thus, that the return period (inverse of annual risk of exceedance) for the SSE lies between 1,000 and 4,000 years. For a number of reasons (Jackson, 1980), the staff believes that "1000 year spectra," generated by the SSSP methodology, are conservative and represent loads with true return periods greater than 1,000 years. Design spectra that were assumed to have return periods on the order of 1,000 to 10,000 years have been accepted implicitly by the NRC in recent licensing decisions (Jackson, 1980).

The staff considers that the combined weight of the "historical" analysis, the ensemble of Intensity VIII (MM) spectra, the ensemble of close-in, rock-site, magnitude 5.5-to-6.4 spectra, and the uniform hazard spectra is to support the conclusion that the 0.25g-R.G. 1.60 response spectrum is an adequate and conservative representation of the Seabrook SSE. The applicant presently is evaluating the strong motion data from the recent New Hampshire and New Brunswick earthquakes with respect to current assumptions about earthquake ground motion in the northeastern U.S. The staff will review this information and, if necessary, reassess the appropriateness of the Seabrook SSE response spectrum.

The applicant has assumed a duration of strong ground shaking (acceleration greater than 0.05g) of 10 to 15 seconds for the SSE. This is a conservative estimate for Intensity VIII (MM) shaking at rock sites near the causative earthquake (e.g. Chang and Krinitzsky, 1977). The adequacy of the time history used for seismic analysis is reviewed in Section 3.7.1.

2.5.2.7 Operating Basis Earthquake

The applicant has characterized the Operting Basis Earthquake (OBE) by a reference acceleration for seismic design of 0.125g; this acceleration level corresponds to Intensity VII (MM) shaking, according to the relationship of Trifunac and Brady (1975). The OBE acceleration level is one-half that of the SSE, in accordance with Appendix A to 10 CFR Part 100, and is acceptable to the staff. A preliminary seismic hazard analysis by the applicant indicates a probability of exceeding the OBE of 1.97×10^{-3} per year, or 7.6×10^{-2} over the expected 40-year operating life of the plant; the staff finds this to be conservative for an event that "could reasonably be expected to affect the plant site during the operating life of the plant" (10 CFR Part 100, Appendix A, Section III (d)). In response to a staff request for additional information (RAI 230.3), the applicant is repeating the seismic hazard analysis, incorporating information gained from the recent New Hampshire and New Brunswick earthquakes. The staff will review the updated probability analysis when it is submitted by the applicant.

close up

2.5.3 Surface Faulting

There are many faults in the site vicinity. Some of these faults are in the rock beneath the site. The faults that have been discussed in the literature, and faults that have been discovered during the applicant's investigations have been investigated and shown to be noncapable according to Appendix A. The following paragraphs present the bases for that conclusion.

The most significant regional faults to the site are major thrust faults of the Clinton-Newbury fault system and the Bloody Bluff fault system located within an arcuate, generally east-west trending zone between the area 7 to 30 miles south of the site. The zone is characterized by many closely spaced east-northeast striking, north dipping thrust faults, which also have a component of dextral strike slip movements (Dennen, 1976). This zone of faulting is interpreted to represent a collision boundary between a plate containing the southeastern New England Platform to the south and a plate containing the New

England fold belt to the north. The zone of deformation curves to the southwest and extends into eastern Connecticut. It is of Late Paleozoic age (at least 240 mybp) and contains Early Paleozoic volcanoclastic rocks similar to those in the Merrimack Group, and Early Paleozoic intrusive rocks and slices of Precambrian rocks, and rocks like those found in the southeastern New England Platform. (Cameron and Naylor, 1976; Nelson, 1976, and Schutts et al, 1976).

The thrust fault complex has been extensively investigated and mapped during the NRC-sponsored New England Seismotectonic Study (Dennen, 1979, 1980), and no evidence has been found that indicates recent movement. The northern-most fault of this system, the Scotland Road fault, has been investigated by the Seabrook applicant using borings, trenching and geologic mapping.

The NRC staff evaluated the applicant's work on the Scotland Road fault during the CP review and concluded that it was not capable:

> "The Scotland Road fault, as defined by A.F. Shride (1971), is interpreted to be the Northeastern projection of the regional Clinton-Newbury fault, an apparent thrust fault on which the hanging wall plate moved from north to south over the footwall block. Various workers related this fault to the Acadian orogeny or post-orogenic adjustments prior to the end of the Paleozoic era. Shride's extension of this fault projects it to Plum Island on the New Hampshire coast some 7 miles to the south of the site. Results of more recent investigations by J.R. Rand substantiate Shride's interpretation. Subsurface investigations located the fault within 150 feet of the location inferred by Shride in his regional field studies. Deformed rock within the fault zone has been annealed and radiometric age dating of several samples indicate the fault to be of early to middle Permian age."

A large fault had been postulated north of the site by Novotny (1963). Novotny interpreted it as a normal fault of unknown displacement which formed the contact between the Kittery and Rye formations. He projected the fault trend for a distance of about 9 miles from New Castle, near Portsmouth,

to North Hampton, where it is shown to die out against the Newburyport pluton about one and one-half miles from the site. The postulate was based, in part, on an apparent unconformable stratigraphic relationship between the Kittery and Rye formations. Time of faulting was interpreted by Novotny to be during the Acadian orogeny, about 330 to 360 million years ago. J.R. Rand, consultant for the applicants conducted an extensive investigation in the area of the inferred structure including borings, trenching and geologic mapping. This investigation failed to encounter any evidence of the fault. The NRC staff evaluated this fault during the CP review. We conclude that the fault, if it exists, is at least 330 million years old and not capable because it apparently does not cut the Newburyport pluton.

During construction activities for the Seabrook plant and excavation of the circulating water tunnels, numerous faults were encountered. It had been expected that faults would be found based on geologic investigations of the site and region around the site. The faults were investigated in considerable detail by the applicant. The staff has completed its review of the data and the applicant's analysis and concludes that these faults are not capable.

Faults exposed in the plant excavation were examined by NRC geologists on 8 June, 1978 and 17 July, 1978. Some of the faults and other features exposed in the circulating water tunnels were examined by an NRC geologist on 19 October, 1981.

Sixty-one minor faults were mapped in the excavation for the plant. None are considered to be throughgoing in that all but one of them terminate with at least one end in the excavation, and many are of limited vertical extent. They appear to be controlled by pre-existing joints or foliation planes. Displacements range from a few inches to several feet, and sense of movement is generally normal.

The applicant has categorized all faults in the excavation into seven sets based on orientations, attitudes, sense of displacement, physical characteristics of the fault and lithologic relationships. The applicant demonstrated by cross-cutting relationships with other faults and/or diabase dikes that the

youngest faults last moved about the same time as dike emplacement (more than 200 mybp). By crosscutting relationships and radiometric age dating, the applicant showed that all of the faults were related to two periods of deformation, one in early Paleozoic, about the time of intrusion of the Newburyport Pluton (400 mybp); and the other during the early Mesozoic at the time of intrusion of the diabase dikes, more han 200 mybp. Additional evidence of antiquity was documented by mapping unfaulted Pleistocene sediments overlying the faults.

More than 100 faults were mapped in the circulating water tunnels. According to the applicant all of these faults have similar orientations, attitudes, and relationships to diabase dikes as the seven sets of faults mapped in the plant excavations, and therefore are interpreted to have been formed by the same tectonic mechanisms in the Paleozoic and Mesozoic Eras. NRC geologists visited the site on several occasions to examine geologic features exposed in excavations, including the circulating water tunnels. Based on our review of the applicant's data, the scientific literature, including the results of the New England seismotectonic study, and our observations during the site reconaissances, we conclude that there are no capable faults in the site vicinity.

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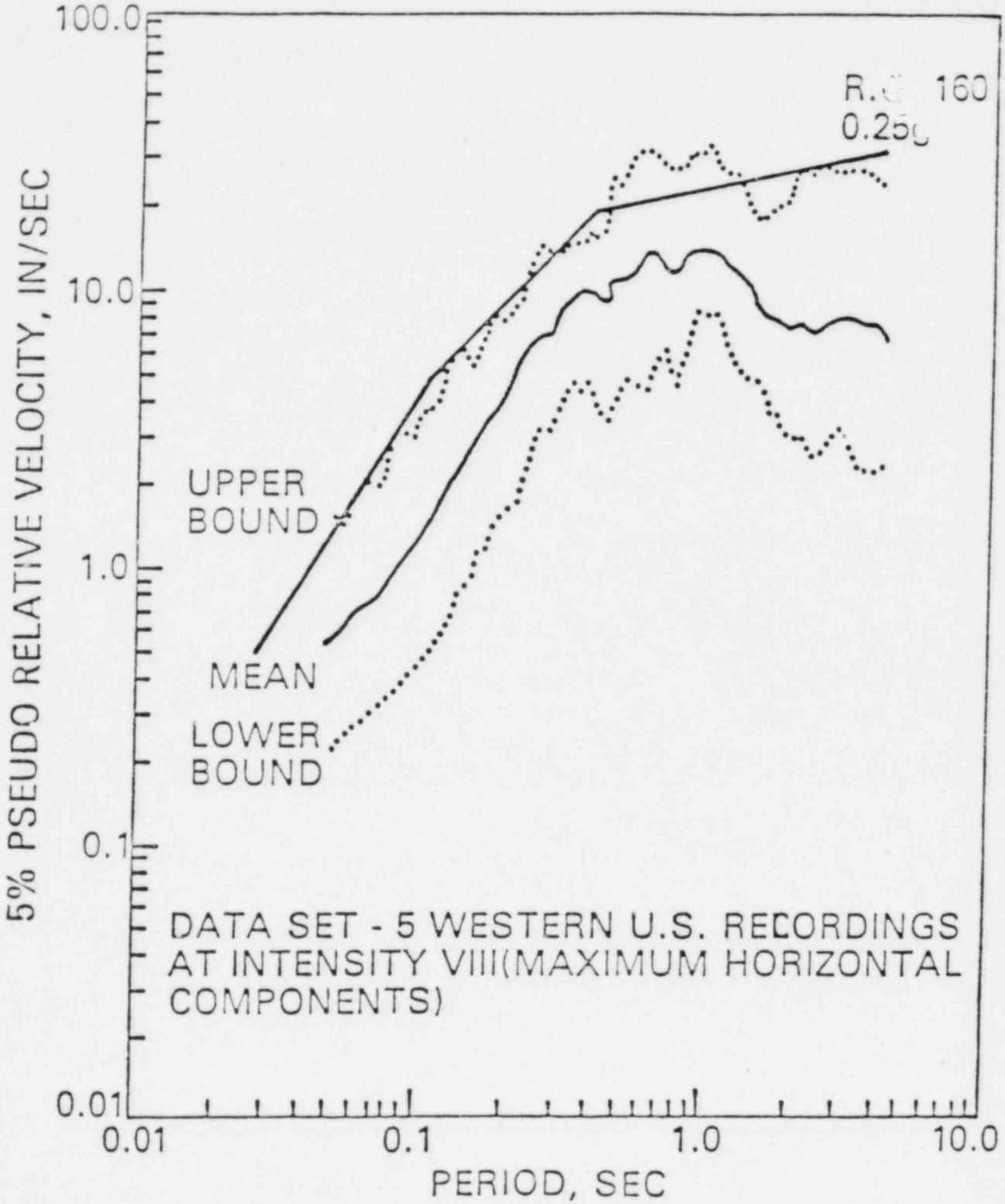

FIGURE 2.5-1 Average PSRV Spectrum Compared with the Upper and Lower Bounds of the Observed Data, Modified Mercalli Intensity Level VIII (NUREG/CR-1259)

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50th & 84th Percentile & Envelope of Spectra of Fourteen Rock Site Records, $5.5 \leq M_L \leq 6.4$

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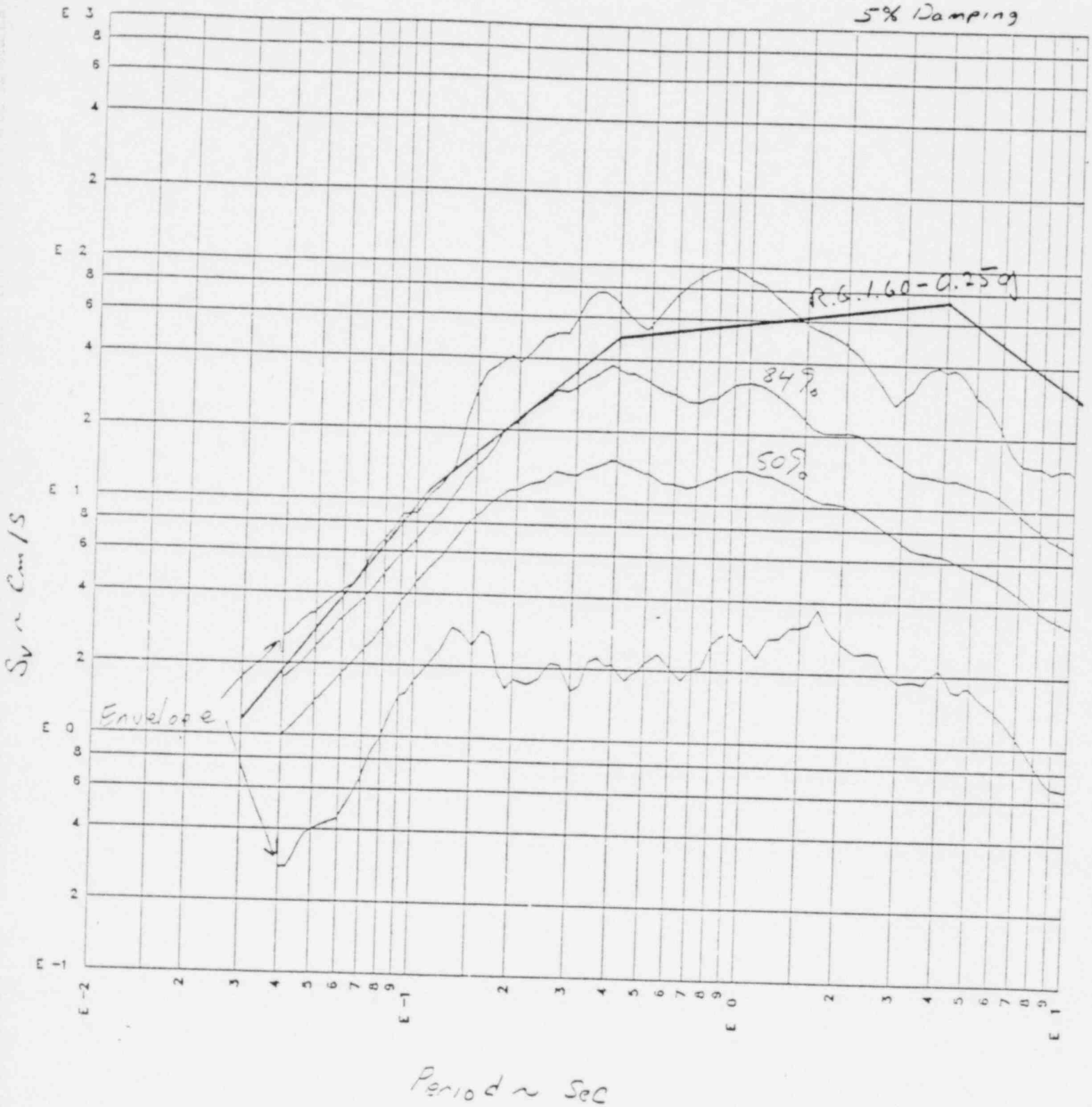

Fig. 2.5-2

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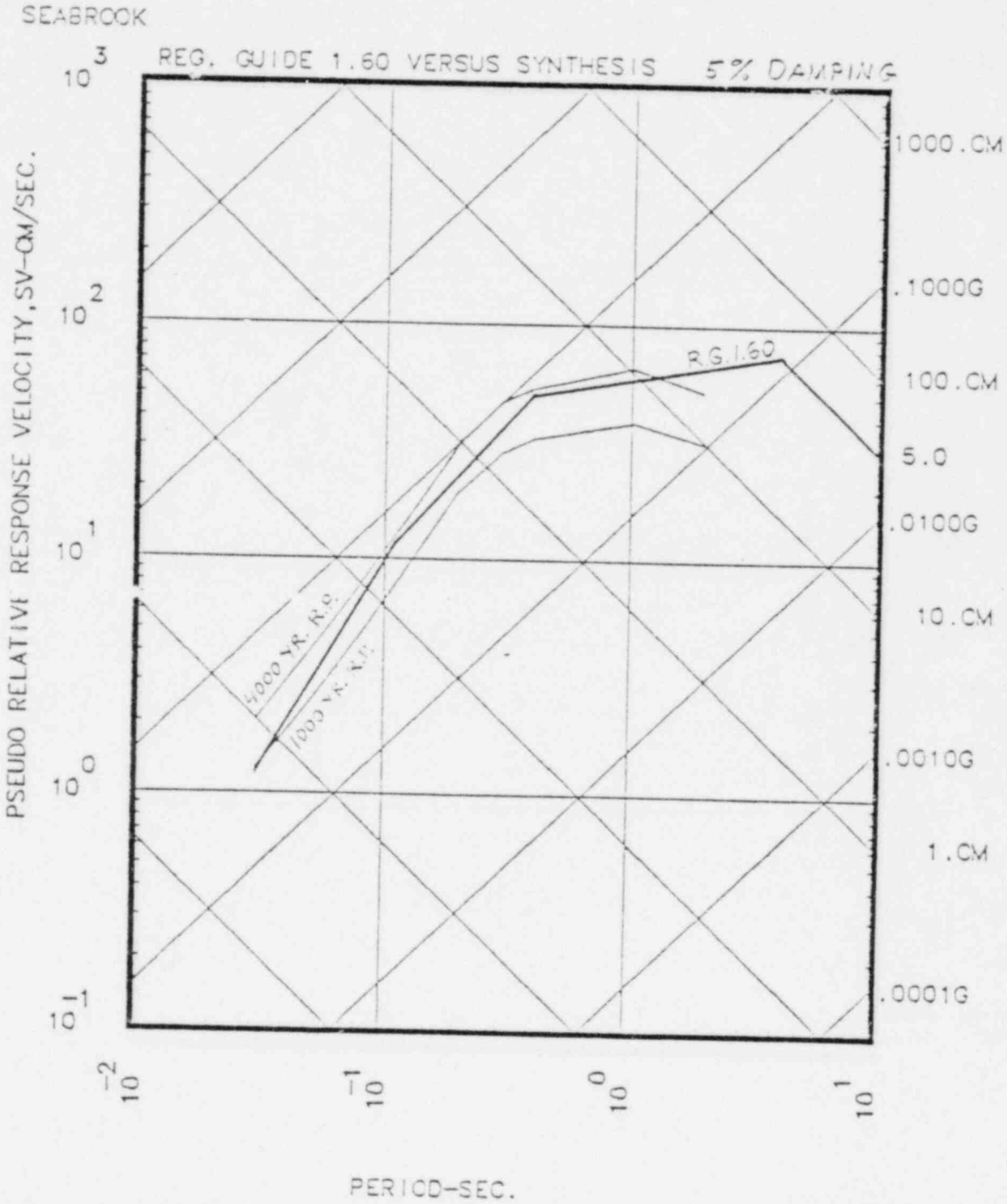

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Fig. 2.5-3

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2.5.4 Stability of Subsurface Materials and Foundations

2.5.4.1 Geologic Features

2.5.4.1.1 General Plant Description

The Seabrook Plant site is located 2 miles inland from the open Atlantic Ocean coast of New Hampshire about 13 miles south of the Maine state line and 1.5 miles north of the Massachusetts state line. The site is situated within the Seabrook Lowland section of the New England Physiographic Province. The topography of the Seabrook Lowland section is gently undulating rising gradually from the seacoast to an elevation of 500' approximately 30 miles inland. The topography of the general Seabrook Station site area is flat, consisting of

broad open areas of level tidal marshes. The plant site itself is located on an outcrop of bedrock which was overlain by a thin veneer of glacial till prior to construction. The outcrop rises out of the tidal marsh at this location to form a peninsula composed of quartz diorite and included quartzitic bedrock. All of the main seismic Category I structures were founded on sound bedrock or on 3000-psi concrete backfill extending to sound bedrock. Table 2.5.4 below lists the major seismic Category I structures, the approximate foundation dimensions and the approximate bearing elevation of each foundation.

Table 2.5.4 Main seismic Category I structures foundation data

Main Category I structures	Approximate foundation dimension	Approximate bearing elevation (MSL)
Containment structure	153' - OD*	-40'
Containment enclosure	153' - ID** 173' - OD*	-40'
Control building	138' x 90'	+18'
Diesel generator building	95' x 90'	-20' to +18'
Primary auxiliary building	145' x 79'	-30' to + 3'
Fuel storage building	98' x 98'	-21' to + 7'
Condensate storage tank	64' - OD*	+17'
Circulating water pumphouse	114' x 130'	-43'
Service water pumphouse	97' x 80'	-43'
Service water cooling tower	312' x 61'	-12
Intake transition structure	82' x 81'	-49'
Discharge transition structure	77' x 77'	-62'

*OD = Outside Diameter
**ID = Inside Diameter

Electrical ductbanks, five electrical manholes and service water piping at the site were founded on compacted granular backfill extending to sound bedrock. The finished plant grade has been established at 20 feet above mean sea level (MSL).

In order to ensure the protection of plant safety-related structures during the period of peak probable maximum hurricane (PMH) surge activity, protective structures including armor stone covered rip-rap revetments; a vertical seawall; and a concrete retaining wall are being placed along the portions of the site perimeter which will be exposed to wave action.

2.5.4.1.2 Foundation Material

Category I structures are founded either on dioritic igneous rocks, quartzitic metamorphic rocks, or on compacted granular backfill placed over competent bedrock. The rocks underlying Category I facilities are not generally subject to deep weathering effects and are not readily soluble or cavernous. The bedrock surface is overlain by relatively thin unconsolidated deposits of glacial till which is, in turn, locally overlain by sandy outwash deposits and thin marine clay. Organic marsh accumulations and sandy beach deposits are the youngest materials in the area. The bedrock in foundation excavations is generally fresh, hard, and unweathered. Weathering is significant only in a 10-20 foot zone associated with the top of bedrock. All surface materials were removed in the area of Category I facilities in order to found the structures on competent bedrock, concrete backfill over competent bedrock or compacted backfill over bedrock.

The largest portion of the site, including Unit 1 and some Unit 2 facilities are founded on a gneissoid phase of a quartz diorite intrusive, a hard, durable crystalline igneous rock. The rock consists of a medium to coarse-grained quartz diorite matrix enclosing inclusions of fine grained diorite. The balance of the Unit 2 foundations are founded on metamorphic rock consisting of metaquartzite and granulite occurring as an inclusion in the enclosing igneous mass. The physical, chemical and mechanical properties of the metamorphic rock are comparable for foundation purposes to those of the igneous rock. Sections 2.5.1 and 2.5.2 of the FSAR contain details of the geologic characteristic of the site bedrock.

2.5.4.2 Properties of In-Situ Materials

The applicant has conducted investigative programs to determine the engineering properties of the foundation bedrock materials at the site. The programs

included both laboratory and field in situ testing efforts. The reported properties were derived from the following field and laboratory efforts accomplished in conformance with the procedures identified below:

- Rock Quality Designation (RQD) - (Reference 1)
- Permeability of Rock - (Reference 2)
- Rock Density Testing - (ASTM D-2845)
- Unconfined Compression - (ASTM D-2938)
- Young's Modulus - (ASTM D-3148)
- Shear Modulus - (calculated)
- Poisson's Ratio - (ASTM D-3148)
- In situ Rock Stress - (Reference 3)
- Rock Hardness - (Reference 4)

2.5.4.2.1 Static Properties

The results of the laboratory unconfined compression tests, Poisson's Ratio tests and rock density testing provided the basis for the applicant's selection of the static engineering properties of the bedrock materials. The range and average of the test results are reported in Table 2.5.12 of the FSAR. The applicant has taken due consideration of the effect of in situ geologic discontinuities existing at the site as evidenced by the reported range of RQD values of 49 to 87 percent and reduced the modulus values of the intact rock specimen by 90 percent to establish appropriate conservative design properties for the in situ rock mass.

Resultant statically determined properties representative of the intact rock as reported by the applicant include:

Density: 2.8 g/cm (average)
Unconfined Compressive Strength: 6,000 - 34,000 psi
Young's Modulus (Tangent): 1.3 - 6.3 x 10^6 psi
Poissons's Ratio: 0.17 - 0.36

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2.5.4.2.2 Dynamic Properties

Properties of the foundation rock determined from geophysical surveys are discussed in subsection 2.5.4.4 below.

2.5.4.3 Exploration

The applicant has reported that during the period 1968 through 1979 a total of 345 borings and 200 seismic refraction and reflection surveys were accomplished at and in the vicinity of the site. Additional in situ testing accomplished included seismic cross-hole and up-hole surveys, in situ rock stress measurements and water pressure permeability tests. Plate load tests were performed on test fill sections of compacted backfill materials to measure the in situ modulus of the test fill materials. Within the immediate plant site area a total of 112 borings were drilled to obtain data on the bedrock and overburden soils. Soil and rock samples were obtained from each of the borings and rock core was oriented in many of the borings to determine the strike and dip of joints, fractures, and foliations. Compression and shear wave velocity measurements were accomplished in boreholes in the vicinity of the reactor locations to obtain velocity data related to the rock mass properties of the bedrock using up-hole and cross-hole techniques. The up-hole data were obtained by detonating small charges in Boring No. B 38 located adjacent to the Unit 1 reactor site and recording at the surface. Cross-hole data was obtained from surveys performed in an array of 7 boreholes at the site. Bedrock density values were obtained from representative core samples within the area.

In situ rock stress measurements were performed using the overcoring technique in a borehole drilled adjacent to the Unit 1 reactor site for this purpose. A series of 11 in situ borehole expansion measurements were taken during overcoring. The modulus of elasticity of the rock was measured by testing sections of overcored annular cylinders of rock removed from the hole and the magnitude and direction of the largest and smallest normal stresses in the horizontal plane were then computed using the data generated. Results of horizontal compressive stress measurements are presented in Section 2.5.4.2 of the FSAR.

Additional site investigations reported as accomplished by the applicant relevant to the geotechnical aspects of the site include a comprehensive office review of available published reports and geologic maps of the area.

Based upon the information presented in the FSAR it is the finding of the staff that the applicant's site investigation efforts provide adequate coverage of the site area in sufficient detail to provide a high level of confidence that specific subsurface conditions have been adequately defined. The staff's review of data presented reveals no evidence of significant areas of landsliding, subsidence, uplift, collapse or solutioning in the vicinity of the site.

2.5.4.4 Geophysical Surveys

The applicant has accomplished a series of compressional "P" wave and shear "S" wave velocity measurement at the site. Up-hole and cross-hole "P" and "S" wave velocity measurements were made at 7 boreholes in the vicinity of the reactor location. Laboratory sonic testing was also accomplished. Density values used in estimating elastic properties of the in situ bedrock were obtained from rock core samples also taken in the vicinity of the reactor sites. The results of these investigation are presented in Table 2.5-12 of the FSAR. A compressional "P" wave velocity range of 13,000 - 16,000 fps was measured by the surface seismic procedure. A range of 14,500 to 20,000 fps was measured using laboratory sonic techniques on intact rock cores. Shear "S" wave velocities measured using up-hole and cross-hole geophysical tests ranged between 8,000 to 10,000 fps. The density of rock core specimens tested ranged between 2-63 and 3.01 grams/cm^3. The range of Poisson's ratio calculated using "P" and "S" wave values obtained from up-hole and cross-hole testing ranged between 0.29 and 0.35. The staff considers these values representative of the igneous and metamorphic rocks of the site area (References 5, 6, and 7), and suitable for use in design calculations, as appropriate.

2.5.4.5 Excavation and Backfill

2.5.4.5.1 Excavation

Excavation in soil overburden and rock was required at the site to establish the planned foundation grade for plant structures. Overburden was removed by

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conventional means. Rock excavation in partially weathered and sound rock was accomplished by controlled blasting and was nominally vertical except where joint patterns and bedding planes controlled. After excavation all bedrock surfaces were throughly cleaned, inspected and mapped in detail by qualified geologists. A summary map of site bedrock geology is presented in Figure 2.5-15 of the FSAR. A thorough discussion of the geology of exposed site foundation excavation is presented in subsection 2.5.1.2.b.6 of the FSAR. Areas of overbreak or of overexcavation to remove weathered rock were backfilled with 3000 psi fill concrete produced and tested in accordance with Category I structural concrete procedures. Because calculated maximum expected rock heave at the bottom of the reactor excavations was less than 0.25 inches, the applicant did not implement a rock movement monitoring program. The applicant reports that no instances or evidence of rock behavior or foundation movement attributable to heave were observed during construction.

2.5.4.5.2 Backfill

The applicant has reported that approximately 500,000 cubic yards of engineered backfill were used under and around all Category I structures in safety-related areas. An additional 500,000 cubic yards of engineered backfill and random fill material were used in nonsafety-related areas. Five types of engineered backfill materials were used.

a) Fill Concrete - Fill concrete batched to attain a 28-day compressive strength of 3000 psi was placed under all Category I structures except ductbanks, manholes and service water piping runs from the top of sound bedrock to the bottom of the structure. Locations and typical depth of placement of fill concrete are presented in Figures 2.5-42 through 2.5-42d of the FSAR. Aggregrates conformed to ASTM C-33. Portland Cement conformed to ASTM C-150 Type II. Concrete samples were taken in accordance with ASTM C-172 and cylinders were made in accordance with ASTM C-31. Compressive strength was tested in accordance with ASTM C-39. The minimum 28-day strength reported by the applicant was greater than 3000 psi and the minimum 90-day strength was greater than 5000 psi.

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b) Backfill Concrete - Backfill concrete proportioned to have a 28-day compressive strength of 2000 psi was placed between rock excavation walls and structure walls below the bedrock surface for all Category I structures. Typical sections for placement locations are presented in Figures 2.5-42 through 2.5-42d of the FSAR. All backfill concrete conformed to the same standards as fill concrete. The applicant reports that the minimum 28-day compressive strength of the backfill concrete tested was greater than 3000 psi with a minimum 90-day strength of greater than 4000 psi.

c) Offsite Borrow - Offsite borrow was placed under, around, and above safety-related ductbanks, four manholes, and ajacent to all Category I structures above bedrock. The maximum depth of borrow beneath ductbanks was 25 feet, beneath manholes 18 feet, and beneath service water piping 37 feet. The maximum depth of borrow adjacent to structural walls was 63 feet. Offsite borrow is described as a 1-1/2-inch maximum, gravelly sand material containing no more than 10 percent passing a #200 sieve (washed), with a coefficient of uniformity C_u greater than 3. Placement of the borrow was controlled to obtain an in place density of greater than 95 percent Modified Proctor (ASTM D1557). The applicant reports that any compaction layer (8" or less) that did not initially test at 95 percent was recompacted until a minimum of 95 percent compaction was obtained. The engineering properties of offsite borrow are presented in Table 2.5-15 of the FSAR. A peak friction angle (undrained triaxial - 95 percent compaction) of 36° was reported as representative of the borrow. A test fill section was constructed of the offsite borrow and in situ Young's Modulus "E" values were measured by plate test (Reference 8) to range between 10,000 - 30,000 psi (reloading) when tested in a field drained condition at 97 percent Modified Proctor density.

d) Tunnel Cuttings - Tunnel cuttings produced by boring machines during excavation of the circulating water tunnels were placed over fill concrete up to the foundation for one manhole in the vicinity of the turbine building for Unit 2, and in nonsafety-related areas at the site. The cuttings were not placed closer than 10 feet to any Category I structure. The cuttings consisted mainly of quartzite, quartz diorite, and shist. The cuttings are described as gravelly sand or sandy gravel and conformed to

a gradation of 3" maximum size with 0-12 percent passing a #200 seive (washed) with a coefficient of uniformity Cμ equal or greater than 5. Tunnel cuttings placed in safety-related areas were compacted to 95 percent Modified Proctor (ASTM-D1557) density. Engineering properties of the tunnel cuttings are presented in Table 5-16 of the FSAR. Results of plate load tests performed upon test fill sections constructed of these materials to determine representative in situ Young's Modulus "E" are also reported (Reference 8). Typical values measured were 54,000 - 67,000 psi (reloading) when placement was controlled to greater than 95 percent compaction and 25,000 - 40,000 psi (reloading) when no control was exercised over placement.

e) Sand Cement - Sand cement batched to develop a 28-day compression strength of 100 psi was used to provide embedment for a 180' section of 38 Ø service water pipe placed in a rock trench between the service water pumphouse and the discharge transition structure. The pipe was placed on top of offsite borrow extending from the top of sound bedrock. The sand cement was placed from the invert of the pipe to a height of approximately 6 feet above the top of the pipe. Offsite borrow was then placed above the sand cement embedment to finished grade. All sand-cement materials used conformed to the same standards as fill concrete. The applicant reports that the minimum 28-day compressive strength of the sand cement tested was greater than 130 psi and the minimum 90-day strength was greater than 180 psi.

Random fill backfill materials for nonsafety-related areas consisted of offsite borrow, tunnel cuttings, and materials obtained from onsite excavations. Placement of random fill material was controlled to provide in place densities of 90 percent of Modified Proctor (ASTM D-1557). The maximum depth of random fill material placed within the general plant site area was approximately 40 feet. Random fill was not placed closer than 10 feet to a safety-related facility.

2.5.4.6 Groundwater Conditions

The water table in the site area ranges between 10 and 17 feet below the natural ground surface. The ground water is principally sustained by infiltrating

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precipitation and no major aquifers underlie the site. The movement of groundwater in the area of the site is toward the adjoining tidal areas. The applicant reports the average permeability of the glacial till overlying the bedrock and the bedrock itself to be less than 10 gallons per day per square foot. During construction total inflow into excavation sites varied between 0 to 15 gallons per minute. A permanent dewatering system is not required for groundwater control at the site and all safety-related structures have been designed for hydrostatic pressure and uplift based upon an assumed groundwater level at +20 feet MSL. A discussion of the relevant groundwater conditions, flow and monitoring programs is presented in subsection 2.4.13 of the FSAR.

2.5.4.7 Response of Soil and Rock to Dynamic Loading

For all seismic analyses for structures founded on rock, the applicant treated the rock as a fixed boundary. All mathematical models used in the seismic analyses were, therefore, fixed against translation and rotation at their base. Therefore, no dynamic rock properties were required in the analyses.

Four seismic Category I electrical manholes were founded on offsite borrow and one manhole was founded on tunnel cuttings. Based upon results of tests accomplished in a test fill area the applicant developed representative strain dependent shear moduli for the foundation borrow material. These values which are reported in subsection 2.5.4.7 of the FSAR were used to cross check the original seismic design and the results were reported by the applicant to be satisfactory.

Buried seismic Category I piping systems were analyzed by the applicant by demonstrating their capability to withstand soil strain and internal pressures due to dynamic loading using the procedure of Iybal and Gooding (Reference 9). Elastic properties used in the analysis for the embankment backfill material have been reported by the applicant in Section 2.5.4.7 of the FSAR. The values reported were determined using results of field and laboratory testing or calculated using the procedure presented in Reference 9.

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Based upon the information presented in the FSAR, the staff concurs with the position of the applicant that all soil properties values used to estimate soil loadings in the dynamic analysis were conservative and representative of the inplace soils.

2.5.4.8 Liquefaction Potential

All Category I structures, electrical duct banks, manholes and piping will be supported on competent bedrock or on engineered backfill extending to competent bedrock. Engineered backfill will also be placed around all Category I structures extending up to plant finished grade. Five types of engineered backfill material in various combinations were used in construction: fill concrete, backfill concrete, sand cement, offsite borrow, and tunnel cuttings. Placement of the offsite borrow and the tunnel cuttings was controlled beneath and adjacent to Category I structures to achieve an in place minimum density of 95 percent Modified Proctor (ASTM 1557). Laboratory testing of the offsite borrow and tunnel cutting materials to determine their cyclic strength characteristics was not performed as the applicant concluded that such materials, when compacted to at least 95 percent Modified Proctor, would not be susceptible to liquefaction when subjected to the postulated Safe Shutdown Earthquake (SSE) event. This conclusion was based upon the results of laboratory triaxial tests which indicated that these materials when compacted to 95 percent Modified Proctor are dilative during shear and thus a pore pressure increase would not be sustained during earthquake induced shearing action.

The staff considers compaction to 95 percent Modified Proctor to be a significant index of a low potential for liquefaction of in placed backfill materials. The staff also recognizes that the backfill materials have other favorable properties such as a relatively coarse grain size (significant amount of gravel included with the sand) and good gradation (SW-GW) which also tend to increase the resistance of the backfill material to potential liquefaction (Reference 10). Considering these points, the staff accepts the position of the applicant and concludes with high confidence that the gravelly sand and sandy gravel backfill materials compacted to 95 percent Modified Proctor are not significantly susceptible to liquefaction and would not be susceptible to liquefaction when exposed to the postulated SSE.

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2.5.4.9 Earthquake Design Basis

The applicant has analyzed the stability of foundations under earthquake loading conditions. A Safe Shutdown Earthquake (SSE) peak horizontal acceleration of 0.25g has been selected as appropriate for the site. A peak horizontal acceleration of 0.125g has been selected for the Operating Basis Earthquake (OBE). The applicant has scaled the Design Response Spectra for horizontal motions for both the SSE and the OBE based upon the above peak horizontal accelerations and using the amplification factors and control points provided in Nuclear Regulatory Commission Regulatory Guide 1.60. The derevation of the Design Response Spectra for this site is discussed in FSAR Sections 2.5.2.6 and 3.7(B).1.1. The staff's evaluation of the applicant's selected earthquake design basis is discussed in Section 2.5.2 of this SER.

2.5.4.10 Static Stability

2.5.4.10.1 Bearing Capacity and Settlement

All major Category I structures are founded on sound bedrock or on concrete backfill extending to sound bedrock. The applicant has estimated that the containment enclosure structure with a maximum bearing pressure of 72 ksf represents the most severe loading condition in bearing on the site rock. The staff has estimated the bearing capacity of the foundation rock using the procedures contained in Reference 7 and has conservatively determined that the foundation rock provides a factor of safety of greater than 6 for all proposed static loading conditions.

The applicant has estimated a maximum settlement for any Category I structure founded on rock of less than 0.5 inches based upon elastic theory and considering recompression of rock rebound which occurred due to overburden excavation. The staff has independently verified these findings using the procedures in Reference 11 and concludes that an adequate margin of safety exists to assure the static stability of plant structures founded on competent bedrock.

The applicant has identified the most severe static bearing conditions for loads supported by compacted backfill material to be associated with the

foundation for the electric manholes with a maximum bearing pressure of 1.4 ksf. The staff has independently estimated the bearing capacity of the inplace compacted backfill when subjected to the maximum structure loading using the procedures in Reference 7. Results indicate that the backfill material can be expected to safely support a loading of 1.4 ksf with a factor of safety of greater than 10 and with a settlement of less than 1 inch. The staff concludes that such a margin of safety is acceptable.

2.5.4.10.2 Lateral Loading

For static loading conditions, the applicant has designed rigid Category I structure subsurface walls to resist at rest earth pressures determined using Rankine's Theory (Reference 12), full static ground water pressure at all levels below a grade elevation of 20-ft MLS, pressure due to surcharge loadings and an additional pressure increment to account for the compaction of the backfill material adjacent to the subsurface walls. An at-rest lateral earth pressure coefficient of 0.5 was selected for design. MSL

For dynamic loading condition, the applicant has considered the maximum dynamic pressure to be equal to the sum of the static at-rest soil pressure, the hydrostatic pressure, the static pressure due to surcharge loadings, the additional static pressure due to compaction of the backfill, the dynamic component of the surcharge pressure, and a dynamic pressure increment due to earthquake effects on the backfill. A dynamic coefficient of 0.28, which considers the influence of both the horizontal and vertical acceleration components induced in the soil by the SSE, was used by the applicant in estimating inertial forces.

The staff has evaluated the conservativeness of the applicant's estimate using the procedures in References 12 and 13 and concludes that the applicant's methods of estimating lateral earth pressures are in accordance with the current state of the art and are sufficiently conservative to allow compliance with the requirements of the Commission criteria identified in 10 CFR 50, Appendix A and 10 CFR 100, Appendix A.

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2.5.4.11 Conclusion

Based on the applicant's design criteria and construction reports and on the reported results of the applicant's investigations, laboratory and field tests, and analyses, the staff concurs with the position of the applicant that the site and plant foundation materials will be adequate to safely support the Seabrook plant facilities during the planned life of the plant. The staff concludes that the geotechnical engineering related site and plant foundation efforts of the applicant meet the requirements of the applicable rules and basic acceptance criteria of the Commission contained in 10 CFR 50 and in 10 CFR 100 and in the regulatory positions contained in Regulatory Guides pertinent to Section 2.5.4 of the Standard Review Plan (NUREG-0800) and are therefore acceptable.

2.5.5 Stability of Slopes

There are no offsite natural or man-made slopes whose failure under any condition would adversely affect the safety of the plant. There are no onsite slopes whose failure under a seismic event would adversely affect the safety of the plant. There are however onsite slopes associated with stone revetments which are required for the protection of the site during peak Probable Maximum Hurricane (PMH) surge activity. The applicant has designed the protective stone revetments using Armor stone and riprap stone materials with stone size, layer thickness, and slope geometry governed by the maximum wave conditions and coincident stillwater levels expected during a PMH event. Details of the wave design of the revetments are described in Subsection 2.4.5.5 of the FSAR. In addition the applicant has analyzed the stability of the revetment slopes and has presented information in Section 2.5.5 of the FSAR to demonstrate the static and dynamic stability of these slopes under conditions to be expected during the operating life of the plant.

2.5.5.1 Slope Characteristics

Two stone revetment designs were provided to protect portions of the perimeter of the plant that could be exposed to wave action during peak PMH surge. Design "A" to be used along portions of the northeast perimeter, consists of a

6-foot layer of Armor stone each weighing between 1.5 to 3 tons, overlying a 3-foot layer of stone weighing 300 to 600 lbs each, overlying a 1.2-foot layer of stone each weighing 15 to 30 lbs. Each stone layer of revetment "A" is to be constructed on a 1.5 to 1 slope. Design "B", to be used along portions of the east, southeast and southern perimeter of the plant consists of a 3.6-foot layer of Armor stone weighing between 700 to 1200 lbs each, overlying a 1.6-foot layer of stone weighing between 70 to 120 lbs. Each stone layer of revetment "B" will also be constructed on a 1.5 to 1 slope. The depth to bedrock beneath the revetments varies from 0 to approximately 40 feet. Prior to placing the revetment stone the applicant has committed to removing all topsoil and to preparing the supporting surface consisting of either natural bedrock and/or inplace dense glacial till, or offsite borrow compacted to 90% Modified Proctor (ASTM D-1557) or greater to a nominal 1.5 to 1 slope. The applicant has also committed to the placement of a polypropylene filter cloth between the revetment stone and the supporting slope and toe materials to reduce potential for erosion and scour.

2.5.5.2 Design Criteria and Analysis

The applicant has analyzed the static stability of the highest section of the revetment using the wedge analysis procedures published by the U.S. Army Corps of Engineers (Reference 14). Properties of the revetment stone and the supporting slope materials used in the analysis are presented in Table 2.5-20 of the FSAR. The effective strength parameters used are as follows:

Revetment Stone:	Cohesion = 0; $\emptyset = 36°$
Offsite Borrow:	Cohesion = 0; $\emptyset = 34°$
Glacial Till:	Cohesion = 0; $\emptyset = 36°$

Results of the applicant's analysis indicate a minimum factor of safety against shear failure under design static conditions of 1.51. Based upon these results and using the criteria of Terzaghi and Peck (Reference 15) the staff concludes that, for the static design loads considered, the revetment slopes will remain stable.

Seismic stability analyses of the revetment designs were also accomplished by the applicant to determine potential revetment displacements which might occur during seismic events. The analyses were performed using the 2-dimensional finite element program "FLUSH" (Reference 16) with displacements computed using Newmark theory for a wedge failure surface occurring under conditions corresponding to Safe Shutdown Earthquake (SSE) loading. Properties of the revetment stone and the supporting slope materials used in the analyses are also presented in Table 2.5-20 of the FSAR. Results reported by the applicant indicate the maximum expected displacement of the Armor stone would result in a less than 2 ft settlement of any portion of the revetment crest under seismic events of magnitude up to the design SSE.

2.5.5.3 Conclusion

The staff finds that the methods and procedures used by the applicant to analyze the stability of the slopes supporting the revetments under static and seismic conditions are acceptable. Based upon the analyses presented by the applicant the staff also finds that the slopes possess an adequate margin of safety against failure during required plant operating conditions. To further assure the safe functioning of the revetment slopes during the life of the plant the staff requires that the applicant commit to developing an appropriate inservice inspection and surveillance program for the revetments and to include the revetments as items to be inspected under the purview of USNRC Regulatory Guide 1.127--"Inspection of Water Control Structures Associated with Nuclear Power Plants."

The staff concludes that the geotechnical engineering related slope stability efforts of the applicant meet the requirements of the applicable rules and basic acceptance criteria of the Commission contained in 10 CFR 50 and in 10 CFR 100 and in the regulatory positions contained in Regulatory Guides pertinent to Section 2.5.5 of the Standard Review Plan (NUREG 0800) and are therefore acceptable.

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ENCLOSURE 2

PUBLIC SERVICE COMPANY OF NEW HAMPSHIRE

SEABROOK NUCLEAR STATION

STRUCTURAL ENGINEERING BRANCH

DOCKET NOS. 50-443/444

DRAFT SAFETY EVALUATION REPORT

3.~~4~~3.1 Wind Design Criteria

All Category I structures exposed to wind forces were designed to withstand the effects of the design wind. The design wind specified has a velocity of 110 mph, at 30 feet above the nominal ground elevation based on a recurrence interval of 100 years.

The procedures that were used to transform the wind velocity into pressure loadings on structures and the associated vertical distribution of wind pressures and gust factors are in accordance with "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures," ANSI Standard A 58.1 and ASCE paper #3269.

The staff concludes that the plant design is acceptable and meets the requirements of General Design Criterion 2. This conclusion is based on the following:

The applicant has met the requirements of GDC 2 with respect to the capability of the structures to withstand design wind loading so that the design reflects

(1) appropriate consideration for the most severe wind recorded for the site with an appropriate margin
(2) appropriate combinations of the effects of normal and accident conditions with the effects of natural phenomena
(3) the importance of the safety function to be performed.

The applicant has met these requirements by using ANSI A58.1 and ASCE paper 3269, which the staff has reviewed and found acceptable, to transform the wind velocity into an effective pressure on structures and for selecting pressure coefficients corresponding to the structural geometry and physical configuration.

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The applicant has designed the plant structures with sufficient margin to prevent structural damage during the most severe wind loadings that have been determined appropriate for the site so that the requirements of item (1) above are met. In addition, the design of seismic Category I structures, as required by item (2) above, has included, in an acceptable manner, load combinations which occur as a result of the most severe wind load and the loads resulting from normal and accident conditions.

The procedures used to determine the loadings on structures induced by the design wind specified for the plant are acceptable because these procedures have been used in the design of conventional structures and been proven to provide a conservative basis which, together with other engineering design considerations, ensures that the structures will withstand such environmental forces. The use of these procedures provides reasonable assurance that in the event of design-basis winds, the structural integrity of the plant structures that have to be designed for the design wind will not be impaired and, in consequence, safety-related systems and components located within these structures are adequately protected and will perform their intended safety functions if needed. Thus, requirement of item (3) above is satisfied.

3.3.2 Tornado Design Criteria

All Category I structures exposed to tornado forces and needed for the safe shutdown of the plant were designed to resist a tornado of 290 mph tangential wind velocity and a 70 mph translational wind velocity. The simultaneous atmospheric pressure drop was assumed to be 3 psi at a rate of 2 psi/second. Tornado missiles are also considered in the design, as discussed in Section 3.5 of this report.

The procedures that were used to transform the tornado wind velocity into pressure loadings are similar to those used for the design wind loadings as discussed in Section 3.3.1 of this report. The tornado missile effects were determined using procedures to be discussed in Section 3.5 of this report. The total effect of the design tornado on Category I structures is determined by the appropriate combinations of the individual effects of the tornado wind pressure, pressure drop, and tornado-associated missiles. Structures are arranged on the plant site and protected in such a manner that collapse of structures not designed for the tornado will not affect other safety-related structures.

The staff concludes that the plant design is acceptable and meets the requirements of General Design Criterion 2. This conclusion is based on the following:

The applicant has met the requirements of GDC 2 with respect to the structural capability to withstand design tornado wind loading and tornado missiles so that the design reflects

(1) appropriate consideration for the most severe tornado recorded for the site with an appropriate margin

(2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena

(3) the importance of the safety function to be performed

The applicant has met these requirements by using ANSI A58.1 and ASCE paper 3269 to transform the wind velocity generated by the tornado into an effective pressure on structures and for selecting pressure coefficients corresponding to the structural geometry and physical configuration.

The applicant has designed the plant structures with sufficient margin to prevent structural damage during the most severe tornado loadings that have been determined appropriate for the site so that the requirements of item (1) above are met. In addition, the design of seismic Category I structures, as required which occur as a result of the most severe tornado wind load and the loads resulting from normal and accident conditions.

The procedures used to determine the loadings on structures induced by the design-basis tornado specified for the plant are acceptable bucause these procedures have been used in the design of conventional structures and been proven to provide a conservative basis which together with other engineering design considerations ensures that the structures will withstand such environmental forces. The use of these procedures provides reasonable assurance that in the event of design-basis tornado, the structural integrity of the plant structures that have to be designed for the tonadoes will not be impaired and in consequence, safety-related systems and components located within these structures are adequately protected and will perform their intended safety functions if needed. Thus the requirement of item (3) above is satisfied.

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3.4.1 Flood Protection

The design of the facility for flood protection was reviewed in accordance with Section 3.4.1 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria, except as noted below, formed the basis for our evaluation of the design of the facility for flood protection with respect to the applicable regulations of 10 CFR 50.

The acceptance criteria are General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and 10 CFR 100, Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants," Section IV.C. The second criterion concerning seismic and geologic siting criteria and seismically induced flooding was not used. Our review for flood protection is concerned not with the criteria for determining flood level; but rather, once the flood level is established, we evaluate safety-related structures, systems, and components to assure that they are protected from flooding.

In order to assure conformance with the requirements of General Design Criterion 2, "Deisgn Bases for Protection Against Natural Phenomena," we reviewed the overall plant flood protection design including all systems and components whose failure due to flooding could prevent safe shutdown of the plant or result in the uncontrolled release of significant radioactivity. The applicant has provided protection from inundation and the static and dynamic effects of flooding for safety-related structures, systems, and components by the use of exterior and incorporated barriers as defined in Regulatory Guide 1.102, "Flood Protection for Nuclear Power Plants," Position C.1 as described below.

The exterior barriers consist of a seawall, a retaining wall, and revetments which protect the site at the perimeters exposed to seawater action. Plant grade is at elevation 20' msl. The maximum standing water elevation above plant grade is 20'6" msl. The recommendations of Regulatory Guide 1.59, "Design Basis Floods for Nuclear Power Plants," have been followed in determining the maximum water level.

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Seismic Category I (safety-related) structures housing safety-related equipment are designed to withstand the 20'-6" msl water elevation, and all penetrations in those structures below this level are watertight. The only access opening in any exterior wall that is at the maximum flood level is the movable steel door in the fuel storage building (located at elevation 20'-6" msl). Flood protection for this building is provided by a curb at elevation 20'-6" msl located behind this door. Groundwater seepage is prevented by the use of waterproofing membranes and water stops in the structural walls with sump pumps in the buildings provided as a secondary means of protection.

Within plant structures, safety-related equipment is protected against flooding from failures in tanks, vessels, and fluid piping systems as identified in the guidelines of Branch Technical Position ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," by equipment location and drainage as described under Section 9.3.3 of this SER.

Based on our review of the design criteria and bases, and safety classification of safety-related systems, structures, and components necessary for a safe plant shutdown during and following flood conditions, we conclude that the design of the facility for flood protection conforms to the requirements of General Design Criterion 2 with respect to protection against natural phenomena and the guidelines of Regulatory Guides 1.59, Positions C.1 and C.2, and 1.102, Position C.1, concerning design-basis floods and protection and is, therefore, acceptable. The design of the facility for flood protection meets the applicable acceptance criteria of SRP Section 3.4.1.

3.4.2 Water Level (Flood) design Procedures

The design flood level resulting from the most unfavorable condition or combination of conditions that produce the maximum water level at the site is discussed in Section 2.4. The hydrostatic effect of the flood was considered in the design of all Category I structure exposed to the water head.

The procedures utilized to determine the loadings on seismic Category I structures induced by the design flood or highest groundwater level specified for the plant are acceptable because these procedures provide a conservative basis for engineering design to ensure the structures will withstand such environmental forces.

The staff concludes that the applicant has met the requirements of GDC 2 with respect to the structural capability to withstand the effects of the flood or highest groundwater level so that the design reflects

(1) appropriate consideration for the most severe flood recorded for the site with an appropriate margin

(2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena

(3) the importance of the safety function to be performed

The applicant has designed the plant structures with sufficient margin to prevent structural damage during the most severe flood or groundwater and the associated dynamic effects that have been determined appropriate for the site so that the requirements of item (1) above are met. In addition, the design of seismic Category I structures, as required by item (2) above, has included in an acceptable manner load combinations which occur as a result of the most severe flood or groundwater-related load and the loads resulting from normal and accident conditions.

The procedures utilized to determine the loadings on seismic Category I structures induced by the design flood or highest groundwater level specified for the plant are acceptable because these procedures have been used in the design of conventional structures and have been proven to provide a conservative basis which together with other engineering design considerations ensures that the structures will withstand such environmental forces.

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The use of these procedures provides reasonable assurance that in the event of floods or high groundwater, the structural integrity of the plant seismic Category I structures will not be impaired and, in consequence, seismic Category I systems and components located within these structures will be adequately protected and may be expected to perform necessary safety functions, as required, thus satisfying the requirement of item (3) above.

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3.5.1.1 Internally Generated Missiles (Outside Containment)

The design of the facility for providing protection from internally generated missiles (outside containment) was reviewed in accordance with Section 3.5.1.1 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria except as noted below formed the basis for our evaluation of internally generated missile protection outside containment with respect to the applicable regulations of 10 CFR 50.

Our review of internally generated missiles does not include turbine missiles, since turbine missiles are evaluated separately by MTEB in Section 3.5.1.3.

Protection of plant structures, systems, and components outside containment that are requried for safe plant shutdown against postulated internally generated missiles associated with plant operation was considered. Protection provided against missiles generated by rotating or pressurized equipment as identified in the requirements of General Design Criterion 4, "Environmental and Missile Design Bases,"is provided by any one or a combination of compartmentalization, barriers, separation, and equipment design. The primary means utilized by the applicant to provide protection to safety-related equipment from potential damage from internally generated missiles is through the use of plant physical arrangement and by the design adequacy of plant equipment to prevent missile generation. Safety-related systems are physically separated from nonsafety-related systems, and redundant components of safety-related systems are physically separated so that a potential missile could not damage both trains of the safety-related systems. Stored fuel is protected from damage by internal missiles which could result in radioactive release as identified in the guidelines of Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis," by the fuel pool walls and by locating new and spent fuel in an area with no high-energy piping system or rotating machinery in the vicinity.

The applicant has provided an evaluation of potential missile sources from rotating component failures and pressurized component failures. This evaluation included internal missile sources such as valve stems, valve bonnets,

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temperature element-sensing wells, motor-generator sets, and pump impellers. Based on the design of these components, the applicant concluded that none of these are credible missiles. The applicant confirmed that pressurized components designed in accordance with ASME Section VIII are not credible missile sources since the components designed to ASME Section VIII (e.g., accumulators) have either appropriate impact test of material-operating temperatures that preclude brittle fracture. Remote location and separation of safety-related systems trains provide further protection against the effects of potential internally generated missiles.

Although the applicant has provided information which indicates that no credible missiles should be postulated outside containment, we requested that further assurance be provided that the turbine drive of the emergency feedwater (EFW) turbine-driven pump is not a missile source or that missiles from the turbine cannot damage safety-related equipment. The applicant responded that the turbine is of solid wheel, single-stage design with an overspeed trip system and therefore not considered a missile source. In addition, the turbine is oriented perpendicular to the redundant train of the EFW system which is further protected by a partial barrier. However, we remain concerned about (1) the proximity and exposure of the main EFW header which is aligned opposite and parallel to the turbine at a distance of approximately 12 ft and without any barrier protection, and (2) the ability of the partial barrier to protect the motor-driven emergency feedwater pump.

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Protection of safety-related equipment and stored fuel from the effects of turbine missiles including compliance with the guidelines of Regulatory Guide 1.115, "Protection Against Low Trajectory Turbine Missiles" is discussed in Section 3.5.1.3 of this SER.

We have reviewed the adequacy of the applicant's design to maintain the capability for a safe plant shutdown in the event of internally generated missiles outside containment. Based on the above, we cannot conclude that the design is in conformance with the requirements of General Design Criterion 4 with respect to internally generated missile protection outside containment until resolution of the above concern. We will report resolution of this concern in a supplement to this SER.

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3.5.1.2 Internally Generated Missiles (Inside Containment)

The design of the facility for providing protection from internally generated missiles (inside containment) was reviewed in accordance with Section 3.5.1.2 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of internally generated missile protection inside containment with respect to the applicable regulations of 10 CFR 50.

Protection of plant structures, systems, and components inside containment that are required for safe plant shutdown against postulated internally generated missiles associated with plant operation such as missiles generated by rotating or pressurized equipment as identified in the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," is provided by any one or a combination of barriers, separation and equipment design. The primary means of providing protection to safety-related equipment from potential damage from internally generated missiles is provided by shield walls and separation within the containment.

The applicant has provided an evaluation of potential missile sources inside containment. The only credible potential missile sources identified are from high-energy systems as follows:

1. Control Rod Drive Mechanism Components: drive shaft, and drive shaft and control rod cluster together.

2. Pressurizer: valves and heaters.

3. Temperature and pressure sensor assemblies.

Characteristics were determined for each of the above potential missiles. The applicant's analysis verified either that structures, shields, barriers, and/or equipment orientation provide protection for safety-related equipment from the

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above primary missiles and any secondary missiles generated by their impact, or that these missiles are of insufficient energy to cause unacceptable impact or damage. The applicant's analysis also confirmed that no non-seismically supported components within the containment result in gravitational missiles with potentially adverse consequences to safety-related equipment. We have reviewed the applicant's analysis and concur with the applicant's assumptions and evaluation for potential missiles inside containment.

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The applicant has analyzed the potential for the reactor coolant pump flywheel to become a missile source as a result of flywheel failures in accordance with the guidelines of Regulatory Guide 1.14, "Reactor Coolant Pump Flywheel Integrity." The applicant's analysis evaluated the integrity of the flywheel under assumed overspeed conditions of the pump as a result of pipe break at the pump discharge. The analysis verified that failure of the flywheel does not occur and thus it is not a postulated missile source. Refer to Section 5.4.1.1 of this SER for further discussion of reactor coolant pump flywheel integrity and compliance with the criteria of Regulatory Guide 1.14.

We have reviewed the adequacy of the applicant's design to maintain the capability for a safe plant shutdown in the event of internally generated missiles inside containment. Based on the above, we conclude that through the use of barriers, separation, and equipment design, the design is in conformance with the requirements of General Design Criterion 4 with respect to missile protection and is, therefore, acceptable. The design of the facility for providing protection from internally generated missiles meets the acceptance criteria of SRP Section 3.5.1.2.

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3.5.1.4 Missiles Generated by Natural Phenomena

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The tornado missile spectrum was reviewed in accordance with Section 3.5.1.4 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section except as noted below. Conformance with the acceptance criteria formed the basis for our evaluation of the tornado missile spectrum with respect to the applicable regulations of 10 CFR 50.

The portions of the "Review Procedures" concerning the probability per year of damage to safety-related systems due to missiles was not used in our review. Our review for this section of the SRP is concerned with establishing the missile spectrum, not with calculating the probability of damage.

General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," requires that structures, systems and components essential to safety be designed to withstand the effects of natural phenomena, and General Design Criterion 4, "Environmental and Missile Design Bases," requires that these same plant features be protected against missiles. Tornado-generated missiles are the only missiles arising from natural phenomena that are of concern. The applicant has identified the plant site in tornado Region I as defined in Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants," Positions C.1 and C.2, and has selected as the design basis missiles those given for Spectrum A of Standard Review Plan (SRP) Section 3.5.1.4 (Revision 2). The spectrum includes the weight, velocity, dimensions, and impact area and is in accordance with Regulatory Guide 1.76. We have reviewed this spectrum and conclude that it is representative of missiles at the site and is, therefore, acceptable. Discussion of the protection (barriers and structures) afforded safety-related equipment from the identified tornado missiles including compliance with the guidelines of Regulatory Guide 1.117, "Tornado Design Classification," Positions C.1 through C.3, is provided in Section 3.5.2 of this SER.

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A discussion of the adequacy of barriers and structures designed to withstand the effects of the identified tornado missiles is provided in Section 3.5.3 of this SER. Based on our review of the tornado-missile spectrum, we conclude

that the spectrum was properly selected and meets the requirements of General Design Criteria 2 and 4 with respect to protection against natural phenomena and missiles and the guidelines of Regulatory Guides 1.76, Positions C.1 and C.2, and 1.117, Positions C.1 through C.3, with respect to identification of missiles generated by natural phenomena and is, therefore, acceptable. The tornado missile spectrum meets the acceptance criteria of SRP Section 3.5.1.4.

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3.5.2 Structures, Systems and Components to be Protected from Externally Generated Missiles

The design of the facility for providing protection from tornado-generated missiles was reviewed in accordance with Section 3.5.2 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria, except as noted below, formed the basis for our evaluation of the design of the facility for providing protection from tornado-generated missiles with respect to the applicable regulations of 10 CFR 50.

General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," requires that all structures, systems, and components essential to the safety of the plant be protected from the effects of natural phenomena and General Design Criterion 4, "Environmental and Missile Design Bases," requires that all structures, systems, and components essential to the safety of the plant be protected from the effects of externally generated missiles. The spectrum of tornado missiles is discussed in Section 3.5.1.4 of this SER. The applicant has identified all safety-related structures, systems, and components requiring protection from externally generated missiles. All safety-related structures (including containment, the primary auxiliary building, the fuel storage building, the emergency feedwater pump building, and the main steam and feedwater pipe chase) are designed to withstand postulated tornado-generated missiles without damage to safety-related equipment with the exception of portions of the ultimate heat sink mechanical draft cooling towers. Since these cooling towers are required to function only in the case of a seismic failure of the main circulating water tunnels to and from the Atlantic Ocean, only those parts of the cooling tower structure which protect essential service water piping up to and including the cooling tower pump discharge valves require missile protection.

Areas of the cooling towers not protected against missiles include the nonessential air intake, tower pumps, fans, and gear boxes. The valve pit areas of the intake and discharge transition structures and all other piping and equipment associated with the service water system are protected from tornado

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missiles. The portion of the intake and discharge transition structure not protected against tornado missiles which can affect vital station service water is sufficiently below water level to preclude the blockage of the service water intakes by missiles.

FSAR Section 3.5.2 indicates that all safety-related systems and components including outside air intakes and exhausts in safety-related structures and stored fuel are located within tornado-missile-protected structures or are provided with tornado-missile barriers or other protection or are oriented such that tornado missiles do not present a safety hazard. However, FSAR Section 6.4 indicates that the control room intakes are not protected from tornado missiles (see SER Section 9.1.4). We require that the applicant clarify this discrepancy.

Portions of safety-related systems which are buried, such as piping and electrical circuits, are adequately protected by the overlying earth. The requirements of General Design Criteria 2 and 4 with respect to missile protection and the guidelines of Regulatory Guides 1.13, "Spent Fuel Storage Facility Design Basis," Position C.2, 1.27, "Ultimate Heat Sink for Nuclear Power Plants," and 1.117, "Tornado Design Classification," Positions C.1 through C.3, concerning tornado-missile protection for safety-related structures, systems, and components including stored fuel and the ultimate heat sink are met. Protection from low-trajectory turbine missiles, including compliance with Regulatory Guide 1.115, "Protection Against Low-Trajectory Turbine Missiles," is discussed in Section 3.5.1.3 of this SER.

Based on the above, we conclude that the applicant's list of safety-related structures, systems, and components to be protected from externally generated missiles and the provisions in the plant design providing this protection are in accordance with the requirements of General Design Criteria 2 and 4 with respect to missile and environmental effects and the guidelines of Regulatory Guides 1.13, Position C.2, 1.27, Positions C.1 through C.3, 1.115 and 1.117, Positions C.1 through C.3, concerning protection of safety-related plant features including stored fuel and the ultimate heat sink from tornado missiles and are, therefore, acceptable, subject to clarification of the discrepancy

between FSAR Sections 3.5.2 and 6.4 regarding the tornado missile protection afforded the control room air intake. The design of the facility for providing protection from tornado-generated missiles meets the applicable acceptance criteria of SRP Section 3.5.2 pending resolution of the above discrepancy.

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The use of these procedures provides reasonable assurance that in the event of floods or high groundwater, the structural integrity of the plant seismic Category I structures will not be impaired and, in consequence, seismic Category I systems and components located within these structures will be adequately protected and may be expected to perform necessary safety functions, as required, thus satisfying the requirement of item (3) above.

3.5 Missile Protection

3.5.3 Barrier Design Procedures

The plant Category I structures, systems, and components are shielded from, or designed for, various postulated missiles. Missiles considered in the design of structures include turbine-and tornado-generated missiles and various containment internal missiles, such as those associated with a loss-of-coolant accident.

Information has been provided indicating procedures that were used in the design of the structures, shields, and barriers to resist the effect of missiles. The analysis of structures, shields, and barriers to determine the effects of missile impact is accomplished in two steps. In the first step, the potential damage that could be done by the missile in the immediate vicinity of impacts is investigated. This is accomplished by estimating the depth of penetration of the missile into the impacted structures. Furthermore, secondary missiles are prevented by fixing the target thickness well above that determined for penetration. In the second step of the analysis, the overall structural response of the target when impacted by a missile is determined using methods of impactive analysis. The equivalent loads of missile impact, whether the missile is environmentally generated or accidentally generated within the plant, are combined with other applicable loads as discussed in Section 3.8 of this report.

the staff concludes that the barrier design is acceptable and meets the requirements of GDC 2 and 4 with respect to the capabilities of the structures, shields, and barriers to provide sufficient protection to equipment that must withstand

the effects of natural phenomena (tornado missiles) and environmental effects including the effects of missiles, pipe whipping, and discharge fluids.

The procedures utilized to determine the effects and loadings on seismic Category I structures and missile shields and barriers induced by design-basis missiles selected for the plant are acceptable because these procedures provide a conservative basis for engineering design to ensure that the structures or barriers are adequately resistant to and will withstand the effects of such forces.

The use of these procedures provides reasonable assurance that in the event of design-basis missiles striking seismic Category I structures or other missile shields and barriers, the structural integrity of the structures, shields, and barriers will not be impaired or degraded to an extent that will result in a loss of required protection. Seismic Category I systems and components protected by these structues are, therefore, adequately protected against the effects of missiles and will perform their intended safety function, if need be. Conformance with these procedures is an acceptable basis for satisfying in part the requirements of GDC 2 and 4.

3.6.1 Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment

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The design of the facility for providing protection against postulated piping failures outside containment was reviewed in accordance with Section 3.6.1 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the design of the facility for providing protection against postulated piping failures outside containment with respect to the applicable regulations of 10 CFR 50.

General Design Criterion 4, "Environmental and Missile Design Bases," requires that systems and components important to safety be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures. In order to meet this requirement with regard to protection against pipe breaks outside containment, the applicant has designed the plant in accordance with the high-energy pipe criteria of Appendix B of Branch Technical Position (BTP) ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment," and moderate-energy pipe criteria of Section B.3 of this position. Compliance with these sections of BTP ASB 3-1 is compatible with the tendering date for the Seabrook construction application (April 2, 1973). These criteria concern failures in high- and moderate-energy fluid systems. The applicant has identified the high- and moderate-energy piping systems in accordance with these guidelines and has also identified those systems requiring protection from postulated piping failures.

The plant design accommodates the effects of postulated pipe breaks in high-energy fluid piping systems outside containment with respect to pipe whip, jet impingement and resulting reactive forces, and environmental effects, and the effects of postulated cracks in moderate-energy fluid systems outside containment with respect to jet impingement, flooding, and other environmental effects. The means used to protect safety-related systems and components throughout the plant include physical separation, enclosure in suitably designed structures or compartments, drainage systems, pipe whip restraints, equipment shields, and equipment environmental qualification as required. DRAFT DRAFT

The applicant analyzed high-energy piping systems for the effects of pipe whip, jet impingement, and environmental effects on safety-related systems and structures. For moderate-energy systems, protection of safety-related systems from the jet, flooding, and other environmental effects due to critical cracks is incorporated into the plant design. We have reviewed the applicant's analysis and we conclude that the protection provided against pipe failure outside containment is in conformance with the guidelines of Branch Technical Position ASB 3-1.

The main steam and feedwater lines between the first pipe whip restraint inside containment and the first pipe whip restraint outside containment have been classified as part of the break exclusion boundary. Outside containment, this portion of these lines is located in two pipe chases (east and west).

The applicant has performed a subcompartment analysis for the pipe chases and the main steam and feedwater lines in order to assure that the resulting jet impingement and environmental effects from a postulated full flow area pipe break in the main steam line will not result in adverse consequences. The results of this analysis indicate that the pipe chase structural integrity is not affected by the pressure increase to 7 psig from the resulting blowdown. Main steam isolation valve (MSIV) functional capability is assured by the environmental qualification of system components to the expected condition of 325°F for 5 hours. The analysis also indicated that the MSIV closure will terminate the blowdown from the steam generator in 5 seconds or less.

The nonsafety-related portions of the main steam and feedwater pipes leave the pipe chases on elevated supports. Failures of the main steam lines leaving the east pipe chase present no problem to the emergency feedwater pump house and other safety-related areas because of building/pipe orientations. However, the main steam and feedwater lines leaving the west pipe chase are routed adjacent to the east wall of the control building. A crack in the main steam line which runs nearest to the control building wall could cause jet impingement which might result in failure of the 2-foot-thick reinforced concrete wall, with formation of missiles inside the control building. This line is sleeved along this wall and a pipe whip restraint bumper is provided at the line elbow to prevent damage to the control building wall. However, the seismic category

of the sleeve is not known. A staff evaluation of the design requirements for this sleeve has not been completed and will be presented in a supplement to this SER.

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The plant has the ability to sustain a high-energy pipe break coincident with a single-active failure in essential systems and retain the capability for safe cold shutdown. For postulated pipe failures, the resulting environmental effects do not preclude the habitability of the control room, the accessibility of other areas that have to be manned during and following an accident, and the loss of function of electric power supplies and controls and instrumentation needed to complete a safety action. Further discussion of the environmental qualification of safety-related equipment is contained in Section 3.11 of this SER.

~~With regard to the design criteria for the emergency feedwater (EFW) system~~ piping, the applicant originally indicated that the piping was not considered either high or moderate energy. In response to our concerns, the applicant responded that the EFW discharge lines upstream of the manual stop-check valves would be considered moderate energy lines, and that a moderate energy failure study is now being performed. However, the applicant's position on the steam supply lines to the EFW turbine downstream of the normally closed steam admission valves MSV 127 and 128 is that it does not qualify as moderate energy piping since it is vented to the atmosphere. It is our position that the piping downstream of valves MSV 127 and 128 is moderate energy in accordance with BTP MEB 3-1 and is subject to through-wall leakage cracks, since it will be periodically pressurized for test and maintenance, but for less than two percent of ~~total operational time.~~

Based on our review, ~~we find that with the above exception~~, the applicant has adequately designed and protected areas and systems required for safe plant shutdown following postulated events, including the combination of pipe failure and single-active failure and for the environmental effects of moderate-energy pipe cracks. We conclude that with the above exceptions, the plant design meets the requirements of General Design Criterion 4 and the criteria set forth in Branch Technical Position ASB 3-1 with regard to the protection of safety-related systems and components from a postulated high energy line break and with regard to the protection of safety-related systems and components from a postulated

moderate-energy line failure. We further conclude that the plant design for the protection of safety-related equipment against dynamic effects associated with the postulated rupture of piping outside containment is acceptable, ~~subject to satisfactory resolution of the above identified concerns. We will report on the resolution of these items in a supplement to this SER.~~

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

General Design Criterion 4, "Environmental and Missile Design Bases", of 10 CFR Part 50, Appendix A, requires that structures, systems, and components important to safety shall be designed to be compatible with and to accommodate the effects of the environmental conditions due to normal operations, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be adequately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power plant.

Our review, conducted in accordance with Standard Review Plan (NUREG 0800 dated July 1981), Section 3.6.2, "Determination of Break Locations and Dynamic Effects Associated with the Postulated Rupture of Piping", pertains to the methodology used for protecting safety-related structures, systems, and components against the effects of

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postulated pipe breaks both inside and outside containment. We have used the review procedures identified in SRP 3.6.2 to evaluate the effect that breaks in high energy fluid systems would have on adjacent safety-related structures, systems, or components with respect to jet impingement and pipe whip. We also reviewed the location, size, and orientation of postulated failures and the methodology used to calculate the resultant pipe whip and jet impingement loads which might affect nearby safety-related structures, systems, or components. The details of our review follows.

Pipe whip need only be considered in those high energy piping systems having fluid reservoirs with sufficient capacity to develop a jet stream. The criteria for determining high and moderate energy lines is found in Branch Technical Position ASB3-1 of Standard Review Plan 3.6.1, "Plant Design for Protection Against Postulated Piping Failures in Fluid System Outside Containment". This criteria has been used correctly by the applicant. A list of all high energy systems is included in the FSAR.

For high energy piping within the containment penetration break exclusion region, Standard Review Plan

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3.6.2 sets forth certain criteria for the analysis and subsequent augmented inservice inspection requirements. Breaks need not be postulated in those portions of piping within the containment penetration region that meet the requirements of the ASME Code, Section III, Subarticle NE-1120 and the additional requirements outlined in Branch Technical Position MEB 3-1 of Standard Review Plan 3.6.2. Augmented inservice inspection is required for those portions of piping within the break exclusion region. The applicant has committed to perform a 100% volumetric testing examination of all welds except socket welds on high energy piping greater than 1-inch nominal pipe diameter in the break exclusion region during each inspection interval. Socket welds will be examined using surface examination methods. We find the applicant's commitment meets our criteria for the augmented inservice inspection program.

Additionally, the applicant has used an integrally forged flued head on all hot high energy piping containment penetrations (greater than 200 F) and seamless pipe welded to flat plate heads on cold lines and lines less than 1" nominal diameter. The flat plates are welded to sleeves anchored in the containment structure. All penetration

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sleeves are seal welded to the steel containment liner. There are no process pipe welds within these protective assemblies. The applicant has performed a detailed stress analysis where welding to the outer pipe surface was required. We conclude that the applicant's containment penetration designs meet the criteria of Branch Technical Position MEB 3-1.

For ASME Section III Class 1 high energy fluid system piping, not in the containment penetration area, SRP (NUREG-0800) Section 3.6.2 states that breaks are to be postulated at every location where the cumulative usage factor, as determined by the ASME Code, is greater than or equal to 0.1. In addition, breaks are to be postulated at those locations where the primary or secondary stress intensity range (including the zero load set) as calculated by equation (10) and either equation (12) or (13) in Paragraph NB-3653 of ASME Section III exceeds 2.4 Sm for normal and upset conditions including the OBE.

In addition to postulating breaks in accordance with the above stress intensity and cumulative usage criteria, the applicant has postulated breaks at every fitting, weld and terminal end.

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For those ASME Class 1 piping systems where breaks would not be postulated on the basis of stress or usage factor, breaks were postulated at two intermediate locations where the most severe effects would result, except for primary loop piping. Primary loop piping breaks were postulated according to Westinghouse topical report WCAP-8082. We have reviewed this report and find its procedures acceptable. The applicant has postulated breaks in ASME Code Class 2 and 3 piping at all terminal ends and at intermediate locations where the stress due to plant operation conditions plus the OBE exceeds 0.8 (SH + SA). Where this stress criteria is not exceeded, the applicant has postulated breaks at two intermediate locations based on the highest stress. Breaks in non-seismic Category I piping have been postulated at terminal ends and at all structural discontinuities such as valves, tees, elbows and reducers. In addition all piping located inside containment is seismic Category I.

Based on the break postulation methodology used by the applicant for safety related piping systems, we conclude that the applicant's procedures meet the requirements of SRP (NUREG-0800) 3.6.2.

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We have reviewed the applicant's method for determining break opening areas and break opening times. The applicant has taken credit for limited break opening areas in the primary coolant loop where pipe whip restraints are designed to limit the break opening area. Specifically, the applicant has limited the break opening areas of the reactor pressure vessel inlet nozzles to 80 square inches and the reactor pressure vessel outlet nozzles to 30 square inches. Pipe whip restraints at these locations are designed for a break opening area of 144 square inches. Break opening times are instantaneous in the case of circumferential breaks and one millisecond for longitudinal breaks. We find the applicant's break opening area and break opening time procedures to be technically sound and consistant with the requirements of SRP (NUREG-0800) 3.6.2.

In general, the applicant has not taken credit for strain rate and strain hardening effects in pipe whip restraint system analysis. With the exception of U-bolts and crush pads the pipe whip restraints are designed to remain elastic.

The applicant has provided drawings of break locations showing types of breaks, structural barriers, restraint locations and constrained directions for each restraint for

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the primary coolant loop and has committed to providing such drawings for all break locations inside containment in the FSAR.

We have reviewed the applicant's methodology for analysis of jet impingement forces and we find it acceptable.

Based on our review of FSAR Section 3.6.2, our findings are as follows:

Our evaluation concludes that the pipe rupture postulation and the associated effects are adequately considered in the plant design, and, therefore, are acceptable and meet the requirements of General Design Criterion 4. This conclusion is based on the following.

1. The proposed pipe rupture locations have been adequately assumed and the design of piping restraints and measures to deal with the subsequent dynamic effects of pipe whip and jet impingement provide adequate protection to the structural integrity of safety-related structures, systems and components.

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2. The provision for protection against dynamic effects associated with pipe ruptures of the reactor coolant pressure boundary inside containment and the resulting discharging fluid provide adequate assurance that design basis loss-of-coolant accidents will not be aggravated by sequential failures of safety-related piping, and emergency core cooling system performance will not be degraded by these dynamic effects.

3. The proposed piping and restraint arrangement and applicable design considerations for high and moderate energy fluid systems inside and outside of containment, including the reactor coolant pressure boundary, will provide adequate assurance that the structures, systems, and components important to safety that are in close proximity to the postulated pipe rupture will be protected. The design will be of a nature to mitigate the consequences of pipe ruptures so that the reactor can be safely shut down and maintained in a safe shutdown condition in the event of a postulated rupture of a high or moderate energy piping system inside or outside of containment.

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3.7.1 Seismic Input

The input seismic design response spectra for OBE and SSE are defined at the bedrock. These spectra comply with Regulatory Guide 1.60. All seismic Category I structures, with the exception of electrical manholes and ductbanks, are founded on the bedrock or concrete fill extending to the bedrock. The manholes and ductbanks are founded on soil with maximum depth of soil between the foundation and the bedrock being 18 feet. The soil amplification phenomenon has been accounted for in the seismic analysis for manholes but not for ductbanks. The NRC staff is awaiting for further submittals on ductbank analyses which account for the soil amplification phenomenon.

The damping ratios (expressed as a percentage of critical) used in the analysis of various seismic Category I structures, systems, and components are in compliance with those listed in Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants."

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The staff concludes that the seismic design parameters used in the plant structure design are acceptable and meet the requirements of General Design Criterion 2 and Appendix A to 10 CFR Part 100. This conclusion is based on the following:

The applicant has met the relevant requirements of GDC 2 and Appendix A to 10 CFR Part 100 by appropriate consideration for the most severe earthquake recorded for the site with an appropriate margin and considerations for two levels of earthquakes, the SSE and OBE. The applicant has met these requirements by the use of the methods and procedures indicated below:

The seismic design response spectra (OBE and SSE) applied in the design of seismic Category I structures, systems, and components comply with the recommendations of Regulatory Guide 1.60. The specific percentage of critical damping values used in the seismic analysis of Category I structures, systems, and components are in conformance with Regulatory Guide 1.61. The artificial synthetic time history used for seismic design of Category I plant structures, systems, and components is adjusted in amplitude and frequency content to obtain response spectra that envelop the design response spectra specified for the site. Conformance with the recommendations of Regulatory Guides 1.60 and 1.61 assures that the seismic inputs to Category I structures, systems, and components are adequately defined so as to form a conservative basis for the design of such structures, systems, and components to withstand seismic loadings.

3.7.2 Seismic System Analysis, and

3.7.3 Seismic Subsystem Analysis

The scope of review of the seismic system and subsystem analysis for the plant included the seismic analysis methods for all Category I structures, systems, and components. It included review of procedures for modeling, seismic soil-structures interaction, development of floor response spectra, inclusion of torsional effects, evaluation of Category I structure overturning, and determination of composite damping. The review has included design criteria and procedures for evaluation of interaction of non-Category I structures with Category I structures and effects of parameter variations on floor response spectra. The review has also included criteria and seismic analysis procedures for Category I buried piping outside the containment.

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The system and subsystem analyses were performed by the applicant on an elastic basis. Modal response spectrum and time history methods form the basis for the analyses of all major Category I structures, systems, and components. When the modal response spectrum method was used, all modes except the closely spaced modes are combined by the SRSS method. The method used to combine the closely spaced modes was in compliance with the intent of Regulatory Guide 1.92. The applicant has used a fixed-base lump mass model for all major seismic Category I structures. The square root of the sum of the squares of the maximum co-directional responses was used in accounting for three components of earthquake motion. Floor spectra inputs used for design and test verifications of structures, systems, and components were generated from the time history methods, taking into account variation of parameters by peak widening. A vertical seismic system dynamic analysis is employed for all structures, systems, and components. Torsional effects and stability against overturning are considered. However, the accidental torsion was not accounted for in the analysis.

The present staff position requires that the accidental torsion, based on the eccentricity of 5% of the base dimension, be included in the design of structures. This is an addition to that which results from the actual geometry and mass distribution of the building. The applicant believed that the added accidental torsion would not result in a need of modifying its structures, and the confirmatory analyses subsequently performed have verified that.

The staff concludes that the plant design is acceptable and meets the requirements of GDC 2 and Appendix A to 10 CFR Part 100 with respect to the capability of the structures to withstand the effects of the earthquakes so that the design reflects

(1) appropriate consideration for the most severe earthquake recorded for the site with an appropriate margin (GDC2); consideration of two levels of earthquakes (Appendix A, 10 CFR Part 100)

(2) appropriate combination of the effects of normal and accident conditions with the effect of the natural phenomena

(3) the importance of the safety functions to be performed (GDC 2); the use of a suitable dynamic analysis or a suitable qualification test to demonstrate that SSC can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate consideration (Appendix A, 10 CFR Part 100)

The applicant has met the requirements of item (1) listed above by use of the acceptable seismic design parameters, as per SRP Section 3.7.1. The combination of earthquake-resultant loads with those resulting from normal and accident conditions in the design of Category I structures as specified in SRP Sections 3.8.1 through 3.8.5 will be in conformance with item (2) listed above.

The staff concludes that the use of the seismic structural analysis procedures and criteria delineated above by the applicant provides an acceptable basis for the seismic design, which are in conformance with the requirements of item (3) listed above.

3.7.4 Seismic Instrumentation Program

The type, number, location, and utilization of strong motion accelerographs to record seismic events and to provide data on the frequency, amplitude, and phase relationship of the seismic response of the containment structure comply with Regulatory Guide 1.12. Supporting instrumentation is being installed on Category I structures, systems, and components in order to provide data for the verification of the seismic responses determined analytically for such Category I items.

The staff concludes that the seismic instrumentation system provided for the plant is acceptable and meets the requirements of General Design Criterion 2, 10 CFR Part 100, Appendix A and 10 CFR Part 50, § 50.55a. This conclusion is based on the following:

The applicant has met the requirements of GDC 2 by providing the instrumentation that is capble of measuring the effects of an earthquake. The applicant has met the requirements of 10 CFR 50.55a by providing the inservice inspection program that will verify operability by performing channel checks, calibrations, and functional test at acceptable intervals. In addition, the installation of the

specified seismic instrumentation in the reactor containment structure and other Category I structures, systems, and components constitutes an acceptable program to record data on seismic ground motion as well as data on the frequency and amplitude relationship of the seismic response of major structures and systems. A prompt readout of pertinent data at the control room can be expected to yield sufficient information to guide the operator on a timely basis for the purpose of evaluating the seismic response in the event of an earthquake. Data obtained from such installed seismic instrumentation will be sufficient to determine that the seismic analysis assumptions and the analytical model used for the design of the plant are adequate and that allowable stresses are not exceeded under conditions where continuity of operation is intended. Provision of such seismic instrumentation complies with Regulatory Guide 1.12, "Instrumentation on Earthquakes.

3.8.1 Concrete Containment

The reactor coolant system is enclosed in a reinforced concrete containment, as described in Section 3.8.1 of the Seabrook FSAR. The major code used for materials, design, fabrication, construction, examination, testing and surveillance of the concrete containment is the ASME Code Section III, Division 2, "Concrete Reactor Vessels and Containments."

The containment war designed to resist various combinations of dead loads; live loads; environmental loads including those due to OBEs, and SSEs; and loads generated by the design-basis accident, including pressure and temperature. The design and analysis procedures that were used for the containment are the same as those approved for previously licensed applications.

The containment structure was designed, and proportioned to remain within elastic limits under the various postulated load combinations. The criteria for allowable stresses and strains are in accordance with those delineated in the ASME Section III Code, Division 2.

The materials of construction, the quality control procedures, and the fabrication and construction requirements are similar to those used for previously accepted facilities.

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Before plant operation, the containment will be subject to a structural integrity acceptance test in accordance with Regulatory Guide 1.18. During the test, the internal pressure will be raised to 1.15 times the containment design pressure.

The applicant has committed to perform an ultimate capacity analysis for the containment. The NRC staff will review the analysis.

The following conclusions are subject to the review and approval of the ultimate capacity analysis for the containment yet to be received from the applicant.

The staff concludes that the design of the concrete containment is acceptable and meets the relevant requirements of 10 CFR Part 50, §§ 50.55a, and General Design Criteria 1, 2, 4, 16 and 50. This conclusion is based on the following:

(1) The applicant has met the requirements of Section 50.55a and GDC 1 with respect to assuring that the concrete containment is designed, fabricated, erected, constructed, tested and inspected to quality standards commensurate with its safety function to be performed by meeting the guidelines of regulatory guides and industry standards indicated below.

(2) The applicant has met the requirements of GDC 2 by designing the concrete containment to withstand the most severe earthquake that has been established for the site with sufficient margin and the effects of normal and accident condition with the effects of environmental loadings such as earthquakes and other natural phenomena.

(3) The applicant has met the requirements of GDC 4 by assuring that the design of the concrete containment is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.

(4) The applicant has met the requirements of GDC 16 by designing the concrete containment so that it is an essentially leaktight barrier to prevent the uncontrolled release of radioactive effluents to the environment.

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(5) The applicant has met the requirements of GDC 50 by designing the concrete containment to accommodate, with sufficient margin, the design leakage rate, calculated pressure and temperature conditions resulting from accident conditions, and by assuring that the design conditions are not exceeded during the full course of the accident condition. In meeting these design requirements, the applicant has used the recommendations of Regulatory Guides and industry standards indicated below. The applicant has also performed appropriate analysis which demonstrates the ultimate capacity of the containment subjected to internal pressure load.

The criteria used in the analysis, design, and construction of the concrete containment structure to account for anticipated loadings and postulated conditions that may be imposed upon the structure during its service lifetime are in conformance with established criteria, and with codes, standards, guides, and specifications acceptable to the Regulatory staff. These include meeting the positions of Regulatory Guides 1.10, 1.15, 1.18, 1.19, 1.35, 1.55, 1.90, 1.94, 1.103, 1.107, 1.136 and industry standard ASME Boiler and Pressure Vessel Code, Section III, Division 2.

The use of these criteria as defined by applicable codes, standards, guides, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and inservice surveillance requirements, provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within and outside the containment, the structure will withstand the specified design conditions without impairment of structural integrity or safety function of limiting the release of radioactive material.

3.8.2 Steel Containment

Not applicable.

3.8.3 Concrete and Structural Steel Internal Structures

The containment interior structures consist of walls, compartments, and floors. The major code used in the design of concrete internal structures is ACI 318-71, "Building Code Requirements for Reinforced Concrete." For steel internal structures the AISC specification, "Specifications for the Design, Fabrication, and Erection of Structural Steel for Buildings," is used.

The containment concrete and steel internal structures were designed to resist various combinations of dead and live loads, accident-induced loads including pressure and jet loads, and seismic loads. The load combinations used cover those cases likely to occur and include all loads which may act simultaneously. The design and analysis procedures that were used for the internal structures are the same as those on previously licensed applications and, in general, are in accordance with procedures delineated in the ACI 318-71 Codes and in the ASIC specification for concrete and steel structures, respectively.

The containment internal structures were designed and proportioned to remain within limits established by the staff under the various load combinations. These loads are, in general, based on the ACI 318-71 Code and on the AISC specification for concrete and steel structures, respectively, modified as appropriate for load combinations that are considered extreme.

The materials of construction, their fabrication, construction, and installation, are in accordance with ACI 318-71 Code and AISC specification for concrete and steel structures, respectively.

The staff concludes that the design of the containment internal structures is acceptable and meets the relevant requirements of 10 CFR 50.55a, and GDC 1, 2, 4, 5, and 50. This conclusion is based on the following:

(1) The applicant has met the requirements of Section 50.55a and GDC 1 with respect to ensuring that the containment internal structures are designed, fabricated, erected, constructed, tested, and inspected to quality standards commencurate with its safety function to be performed by meeting the guidelines of Regulatory Guides and industry standards indicated below.

(2) The applicant has met the requirements of GDC 2 by designing the containment internal structure to withstand the most severe earthquake that has been established for the site with sufficient margin and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

(3) The applicant has met the requirements of GDC 4 by ensuring that the design of the internal structures is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.

(4) The applicant has met the requirements of GDC 5, "Sharing the Structures, Systems, and Components," by demonstrating the structures, systems, and components are not shared between units or that if shared they have demonstrated that sharing will not impair their ability to perform their intended safety function.

(5) The applicant has met the requirements of GDC 50 by designing the containiment internal structures to accommodate, with sufficient margin, the design leakage rate and calculated pressure and temperature conditions resulting from accident conditions and by ensuring that the design conditions are not exceeded during the full course of the accident condition. In meeting these design requirements, the applicant has used the recommendations of Regulatory Guides and industry standards indicated below. The applicant has also performed appropriate analyses which demonstrate that the ultimate capacity of the structures will not be exceeded and establishes the minimum margin of safety for the design.

The criteria used in the design analysis, and construction of the containment internal structures to account for anticipated loadings and postulated conditions that may be imposed during their service lifetime are in conformance with established criteria, and with codes, standards, and specifications acceptable to the staff. These include meeting the positions of regulatory Guide 1.10, 1.15, 1.55, 1.57, and 1.94.

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The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and inservice surveillance requirements provide reasonable assurance that, in the event of earthquakes and various postulated accidents occurring within the containment, the interior structures will withstand the specified design conditions without impairment of structural integrity of the performance of required safety functions.

3.8.4 Other Category I Structures

Category I structures other than containment and its interior structures are all of structural steel and concrete. The major code used in the design of concrete Category I structures is ACI 318-71. For steel Category I structures, the AISC, "Specifications for the Design, Fabrication, and Erection of Structural Steel for Buildings," is used.

The concrete and steel Category I structures were designed to resist various combinations of dead loads; live loads; environmental loads including winds, tornadoes, OBE, and SSE; and loads generated by postulated ruptures of high-energy pipes such as reaction and jet impingement forces, compartment pressures, and impact effects of whipping pipes.

The various Category I structures are designed and proportioned to remain within limits established by the staff under the various load combinations. These limits are, in general, based on the ACI 318-71 Code and on the AISC specification for concrete and steel structures, respectively, modified as appropriate for load combinations that are considered extreme.

The applicant has been requested to assess and demonstrate that the design of Category I structures comply with the requirements of ACI 349 code as amended by Regulatory Guide 1.142.

The materials of constructuion, their fabrication, construction and installation, are in accordance with the ACI 318-71 Code and AISC specification for concrete and steel structures, respectively.

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There are no safety-related masonry walls for the Seabrook faciltiy.

The staff concludes that the design of safety-related structures other than containment is acceptable and meets the relevant requirements of 10 CFR 50.55a and GDC 1, 2, 4, and 5. This conclusion is based on the following:

(1) The applicant has met the requirements of 10 CFR 50.55a and GDC 1 with respect to ensuring that the safety-related structures other than containment are designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with its safety function to be performed by meeting the guidelines of Regulatory Guides and industry standards indicated below.

(2) The applicant has met the requirements of GDC 2 by designing the safety-related structures other than containment to withstand the most severe earthquake that has been established for the site with sufficient margin and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

(3) The applicant has met the requirements of GDC 4 by ensuring that the design of the safety-related structures is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.

(4) The applicant has met the requirements of GDC 5 by demonstrating that SSC are not shared between units or that if shared they have demonstrated that sharing will not impair their ability to perform their intended safety function.

(5) The applicant has met the requirements of Appendix B of 10 CFR 50 because his quality assurance program provides adequate measures for implementing guidelines relating to structural design audits.

The criteria used in the analysis, design, and construction of all the plant Category I structures to account for anticipated loadings and postulated conditions that may be imposed on each structure during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the staff. These include meeting the positions of Regulatory Guides 1.10, 1.15, 1.94.

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The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and spcial construction techniques; and the testing and inservice surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes, and various postulated accidents occurring within the structures, the structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions.

3.8.5 Foundations

Foundations of Category I structures are described in FSAR Section 3.8.5. Primarily, these foundations are reinforced concrete and of the mat type. The major code used in the design of these concrete mat foundations is ACI 318-71 Code. These concrete foundations have been designed to resist various combinations of dead loads; live loads; environmental loads including winds, tornadoes, OBE, and SSE; and loads generated by postulated ruptures of high-energy pipes.

The design and analysis procedures that were used for these Category I foundations are the same as those approved on previously licensed applications and, in general, are in accordance with procedures delineated in the ACI 318-71 Code. The various Category I foundations were designed and proportioned to remain within limits established by the staff under the various load combinations. These limits are, in general, based on the ACI 318 Code and on the AISC specification for concrete and steel structures, respectively, modified as appropriate for load combinations that are considered extreme.

The materials of construction, their fabrication, construction and installation are in accordance with ACI 318-71 Code and AISC specification for concrete and steel structures, respectively.

The criteria that were used in the analysis, design, and construction of all the plant Category I foundations to account for anticipated loadings and postulated conditions that may be imposed upon each foundation during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the NRC staff.

The staff concluges that the design of the seismic Category I foundations is acceptable and meets the relevant requirements of 10 CFR 50.55a, and GDC 1, 2, 4, and 5. This conclusion is based on the following:

(1) The applicant has met the requirements of 10 CFR 50.55a and GDC 1 with respect to ensuring that the seismic Category I foundations are designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with its safety function to be performed by meeting the guidelines of Regulatory Guides and industry standards indicated below.

(2) The applicant has met the requirements of GDC 2 by designing the seismic Category I foundation to withstand the most severe earthquake that has been established for the site with sufficient margin and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

(3) The applicant has met the requirements of GDC 4 by ensuring that the design of seismic Category I foundations is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.

(4) The applicant has met the requirements of GDC 5 by demonstrating that SSC either are not shared between units or that, if shared, they have demonstrated that sharing will not impair their ability to perform their intended safety function.

The criteria used in the analysis, design, and construction of all the plant seismic Category I foundations to account for anticipated loadings and postulated conditions that may be imposed upon each foundation during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the staff.

The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptatance criteria; the materials, quality control,

and special construction techniques; and the testing and inservice surveillance requirements provide reasonable assurance that, in the event of winds, tornadoes, earthquakes, and various postulated events, seismic Category I foundations will withstand the specified design conditions without impairment of structural integrity and stability or the performance of required safety functions.

3.8.6 Structural Audit

From March 29 through April 2, 1982, the staff met with the applicant and his consultants to conduct the seismic and structural audit. The audit covered each major safety-related structure at Seabrook facility.

The staff conducted the audit in order to

(1) investigate in detail how the applicant has implemented the structural and seismic design criteria that he committed to use, prior to obtaining construction permits for the facility

(2) verify that the key structural and seismic design and the related calculations have been conducted in an acceptable way

(3) identify and assess the safety significance of these areas where the plant structures were designed and analyzed using methods other than those recommended by the Standard Review Plan.

As a result of the audit, the staff identified 14 action items. The review and evaluation of the information resulting from these action items also provided a basis for the conclusions reached and reported in this safety evaluation report.

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3.9 Mechanical Systems and Components

The review performed under Standard Review Plan Sections 3.9.1 through 3.9.6 of NUREG-0800, "Standard Review Plan", dated July 1981, pertains to the structural integrity and functional capability of various safety-related mechanical components in the plant. Our review is not limited to ASME Code components and supports, but is extended to other components such as control rod drive mechanisms, certain reactor internals, and any safety-related piping designed to industry standards other than the ASME Code. We review such issues as load combinations, allowable stresses, methods of analysis, summary of results, and preoperational testing. Our review must arrive at the conclusion that there is adequate assurance of a mechanical component performing its safety-related function under all postulated combinations of normal operating conditions, system operating transients, postulated pipe breaks, and seismic events.

3.9.1 Special Topics for Mechanical Components

The review of this section was performed following Standard Review Plan 3.9.1, "Special Topics for Mechanical

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Components". All areas of review and review procedures identified in SRP Section 3.9.1 were followed. We have reviewed the design transients and methods of analysis used for all seismic Category I components, component supports, core support structures and reactor internals designated as Class 1 and CS under the American Society of Mechanical Engineers Code, Section III, and those not covered by the Code. The assumptions and procedures used for the inclusion of transients in the fatigue evaluation of American Society of Mechanical Engineers Code Class 1 and CS have been reviewed. Our review also covered the computer programs used in the design and analysis of seismic Category I components and their supports and experimental and inelastic analytical techniques. We found the plant acceptable in these areas. Details of our review follow.

The applicant has provided a list of the design transients and the number of cycles for each design transient used for design. Fatigue evaluation of NSSS components utilized five OBEs of ten cycles per earthquake. Balance-of-plant piping utilized 5 OBEs with twenty cycles per OBE. This is in conformance with the requirements of SRP 3.9.1. We conclude from our review of the design transients and their respective number of transients that they are acceptable.

Analysis of mechanical components by the use of computer programs was performed by the applicant. A list showing all computer programs used by the applicant for static and dynamic analyses to determine the structural integrity and functional capability along with a description of the program is included in the FSAR. Design control measures to verify the adequacy of the design of safety-related components is required by 10 CFR Part 50, Appendix B. The Seabrook FSAR includes the methods of verification for all computer programs used in the design of safety-related mechanical components. Our review of the list of computer programs and their verification methods finds them generally acceptable. In the case of the computer program "ADLPIPE", used by the applicant to evaluate loads and stresses in piping systems, the applicant has committed to expanding the verification methods to include the solution of benchmark problem number four from NUREG/CR 1677. Based on our review and the applicant's commitment, we find the list of computer programs and their verification methods acceptable.

In conjunction with contract personnel from the Pacific Northwest Laboratory, an audit of the seismic design

procedures used by United Engineers and Constructors (UEC) for the Seabrook Nuclear Power Plant was performed. Results of this audit provided reasonable assurances that the analytical procedures used by UEC in their design of piping for the Seabrook Nuclear Power Plant were acceptable. Acceptance of the analysis results was based on 1) the technical soundness of the analytical methods used, 2) the degree of conformance to our acceptance criteria in the Standard Review Plan, the General Design Criteria of 10 CFR 50, and current Regulatory Guides, and 3) verification of stresses for selected piping systems under required load combinations being within allowable limits of the applicable industry codes (i.e., ASME Section III) and NRC acceptance criteria.

The applicant has not used the elastic-plastic method of analysis to evaluate safety-related code or non-code items for the faulted condition.

Material yield strengths used in elastic design were taken from Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. The applicant has used stress allowables from the AISC "Manual of Steel

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Construction" in the design of certain mechanical component supports. For the faulted condition, tensile and bending stresses were limited to 90% of the material yield strength, and shear stresses were limited to 60% of the material yield strength. No increase or multiplication factors were used for bolts or buckling analysis. We find this methodology to have a sound technical basis.

Based upon our review of FSAR Section 3.9.1 our findings are as follows.

The staff concludes that the design transients and resulting load and load combinations with appropriate specified design and service limits for mechanical components is acceptable and meets the relevant requirements of General Design Criteria 1, 2, 14, 15, 10 CFR Part 50, Appendix B, and 10 CFR Part 100, Appendix B. This conclusion is based on the following:

1. The applicant has met the relevant requirements of General Design Criteria 14 and 15 by demonstrating that the design transients and resulting loads and load combinations with appropriate specified design and

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service limits which the applicant has used for designing Code Class 1 and CS components and supports, and reactor internals provide a complete basis for design of the reactor coolant pressure boundary for all conditions and events expected over the service lifetime of the plant.

2. The applicant has met the relevant requirements of General Design Criteria 2 and 10 CFR Part 100, Appendix A by including seismic events in design transients which serve as design bases to withstand the effects of natural phenomena.

3. The applicant has met the relevant requirements of 10 CFR Part 50, Appendix B, and General Design Criteria 1 by having submitted information that demonstrates the applicability and validity of the design methods and computer programs used for the design and analysis of seismic Category I Code Class 1, 2, 3 and CS structures, and non-Code structures within the present state-of-the-art limits and by having design control measures which are acceptable to assure the quality of the computer programs.

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3.9.4 Control Rod Drive Systems

Our review under Standard Review Plan Section 3.9.4 covers the design of the control rod drive system up tc its interface with the control rods. We reviewed the information in FSAR Section 3.9.4 relative to the analyses and tests performed to assure the structural integrity and functionality of this system during normal operation and under accident conditions. We also reviewed the life-cycle testing performed to demonstrate the reliability of the control rod drive system over its 40-year life.

A detailed review of the design of the control rod drive system with respect to its capability of controlling reactivity and cooling the reactor core with appropriate margin in conjunction with either the emergency core cooling system or the reactor protection system was not performed because of the system similarity with other Westinghouse plants which were found to be acceptable. We are not aware of any significant design changes in the control rod drive system for the Seabrook plant.

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Based on our review of the above information, we conclude that the design of the control rod drive system is acceptable and meets the requirements of General Design Criteria 1, 2, 14, 26, 27, and 29, and 10 CFR Part 50, §50.55a. This conclusion is based on the following:

1. The applicant has met the requirements of GDC 1 and 10 CFR Part 50, §50.55a, with respect to designing components important to safety to quality standards commensurate with the importance of the safety functions to be performed. The design procedures and criteria used for control rod drive systems are in conformance with the requirements of appropriate ANSI and ASME codes.

2. The applicant has met the requirements of GDC 2, 14, and 26 with respect to designing the control rod drive system to withstand effects of earthquakes and anticipated normal operation occurrences with adequate margins to assure its structural integrity and functional capability and with extremely low probability of leakage or gross rupture of reactor coolant pressure boundary. The specified design transients, design and service loadings, combination of

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loads, and limiting the stresses and deformations under such loading combinations are in conformance with the requirements of appropriate ANSI and ASME Codes and acceptable regulatory positions specified in SRP Section 3.9.3.

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3.9.5 Reactor Pressure Vessel Internals

Our review under Standard Review Plan 3.9.5 is concerned with the load combinations, allowable stress limits and other criteria used in the design of the Seabrook reactor internals. We have limited our review of SRP Section 3.9.5 to include the design and analysis of the reactor internals and the deformation limits specified for those components. A detailed review of the configuration and general arrangement of the mechanical and structural internal elements was not performed because of the similarity with other Westinghouse plants which were found acceptable. We are not aware of any significant design changes in the reactor internals for the Seabrook plant.

The reactor internals for the Seabrook plant were procurred before the implementation of Subsection NG.

The reactor internals have been designed to a level equivalent with the requirements of Subsection NG, "Core Support Structures", of the American Society of Mechanical Engineers Code, Section III using the loads, load combinations, and allowable stress limits as provided in Section 3.9.5 of the Seabrook FSAR.

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Based on our review of FSAR Section 3.9.5, we conclude that the design of reactor internals is acceptable and meets the requirements of General Design Criteria 1, 2, 4, and 10 and 10 CFR Part 50, §50.55a. This conclusion is based on the following:

1. The applicant has met the requirements of GDC 1 and 10 CFR Part 50, §50.55a with respect to designing the reactor internals to quality standards commensurate with the importance of the safety functions to be performed. The design procedures and criteria used for the reactor internals are equivalent to the requirements of Subsection NG of the ASME Code, Section III.

2. The applicant has met the requirements of GDC 2, 4, and 10 with respect to designing components important to safety to withstand the effects of earthquake and the effects of normal operation, maintenance, testing, and postulated loss-of-coolant accidents with sufficient margin to assure that capability to perform its safety functions is maintained and the specified acceptance fuel design limits are not exceeded.

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The specified design transients, design and service loadings, and combination of loadings as applied to the design of the reactor internals structures and components provide reasonable assurance that in the event of an earthquake or of a system transient during normal plant operation, the resulting deflections and associated stresses imposed on these structures and components would not exceed allowable stresses and deformations under such loading combinations provides an acceptable basis for the design of these structures and components to withstand the most adverse loading events which have been postulated to occur during service lifetime without loss of structural integrity or impairment of function.

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3.9.6 Inservice Testing of Pumps and Valves

In Sections 3.9.2 and 3.9.2 of this SER, the staff discussed the design of safety-related pumps and valves in the Seabrook plant. The load combinations and stress limits used in the design of pumps and valves assure that the component pressure boundary integrity is maintained. In addition, the applicant will periodically test and perform periodic measurements of all its safety-related pumps and valves. These tests and measurements are performed in accordance with the rules of Section XI of the ASME Code. The tests verify that these pumps and valves operate successfully when called upon. The periodic measurements are made of various parameters and compared to baseline measurements in order to detect long-term degradation of the pump or valve performance. The staff reviews the applicant's program for preservice and inservice testing of pumps and valves using the guidelines of SRP Section 3.9.6, and gives particular attention to those areas of the test program for which the applicant requests relief from the requirements of Section XI of the ASME Code.

The applicant has not yet submitted its program for the preservice and inservice testing of pumps and valves. Therefore, the staff's review of this program will be discussed in a supplement to this SER.

One area of concern during our review is the periodic leak testing of pressure isolation valves.

There are several safety systems connected to the reactor coolant pressure boundary that have design pressure below the rated reactor coolant system (RCS) pressure. There are also some systems which are rated at full reactor pressure on the discharge side of pumps but have pump suction below RCS pressure. In order to protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high pressure RCS and the low pressure systems. The leaktight integrity of these valves must be ensured by periodic leak testing to prevent exceeding the design pressure of the low pressure systems and thus cause an intersystem LOCA.

The applicant has not yet provided a response to our concerns regarding the periodic leak testing of the pressure isolation valves.

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We will report the resolution of the above issues in a supplement to this Safety Evaluation Report.

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4.2 Fuel Design

The Seabrook fuel assembly described in the FSAR is a 17x17 array of fuel rods having a diameter of 0.374 in. This design will be referred to as the Standard Fuel Assembly (SFA) in the following paragraphs.

Section 4.2 of the FSAR presents the design bases for the SFA. For the Westinghouse analysis, plant design conditions are divided into four categories of operation that are consistent with traditional industry classification (ANSI Standards N18.2-1973 and N-212-1974): Condition I is Normal Operation, Condition II is Incidents of Moderate Frequency, Condition III is Infrequent Incidents, and Condition IV is Limiting Faults. Fuel damage is then related to these conditions of operation, which are coupled to the fuel design bases and design limits. The subsections of the design bases section address topics such as (a) cladding, (b) fuel material, (c) fuel rod performance, (d) spacer grids, (e) fuel assemblies, (f) reactivity control and burnable poisons ("core components"), and (g) testing, irradiation, and surveillance. Thus, as part of the discussion of the cladding design bases, material and mechanical properties, stress-strain limits, vibration and fatigue, and chemical properties are also presented. A similar approach is taken for the other major subtopics.

The staff review and safety evaluation will follow SRP Section 4.2. The objectives of this fuel system safety review are to provide assurance that (a) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (b) fuel system damage is never so severe as to prevent control rod insertion when it is required, (c) the number of fuel rod failures is not underestimated for postulated accidents, and (d) coolability is always maintained. "Not damaged" is defined as meaning that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. This objective implements GDC 10 and the design limits that accomplish this are called specified acceptable fuel design limits (SAFDLs). "Fuel rod failure" means that the fuel rod leaks and that the first fission

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product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR Part 100 for postulated accidents. "Coolability," which is sometimes termed "coolable geometry," means, in general, that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channeling to permit removal of residual heat event after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the General Design Criteria (GDC 27, and 35). Specific coolability requirements for the loss-of-coolant accidents are given in 10 CFR Part 50.46.

To meet the above-stated objectives of the fuel system review, the following specific areas are critically examined: (a) design bases, (b) description and design drawings, (c) design evaluation, and (d) testing, inspection, and surveillance plans. In assessing the adequacy of the design, several items involving operating experience, prototype testing, and analytical predictions are weighed in terms of specific acceptance criteria for fuel system damage, fuel rod failure, and fuel coolability. Recently Westinghouse developed an improved fuel assembly design, which is described in WCAP-9500 and is called Optimized Fuel Assembly (OFA). WCAP-9500 was approved by NRC (Rubenstein, 1981). The OFA design also consists of a 17x17 array of fuel rods having a diameter of 0.360 in., which is somewhat smaller than the standard assembly. Because the format of WCAP-9500 followed Regulatory Guide 1.70, some of the fuel design bases and design limits for the OFA were not presented in WCAP-9500 in a form that permitted cross-checking with the acceptable criteria provided in Section 4.2 of the SRP. Therefore several questions were issued (Rubenstein, August 8, 1980) to clarify the design bases and limits. Responses to those questions are contained in letters from Westinghouse (Anderson, August 15, 1980 and April 21, 1981). These responses are applicable to the Standard Assembly as well (Petrick, September 9, 1981). Reference to these questions and answers will be made at several places in the review that follows.

4.2.1 Design Bases

Design bases for the safety analysis address fuel system damage mechanisms and suggest limiting values for important parameters such that damage will be

limited to acceptable levels. For convenience, acceptance criteria for these design limits are grouped into three categories in the Standard Review Plan: (a) fuel system damage criteria, which are most applicable to normal operation (W plant Condition I), including anticipated operational occurrences (W plant Condition II), (b) fuel rod failure criteria, which apply to normal operation (W plant Condition I), anticipated operational occurrences (W plant Condition II), and accidents (W plant Conditions III and IV), and (c) fuel coolability criteria, which apply to accidents (W plant Conditions III and IV).

4.2.1.1 Fuel System Damage Criteria

The following paragraphs discuss the NRC staff's evaluation of the design bases and corresponding design limits for the damage mechanisms listed in the SRP. These design limits along with certain criteria that define failure (see Section 4.2.1.2 of this SER) constitute the SAFDLs required by GDC 10. The design limits in this section should not be exceeded during normal operation including anticipated operational occurrences.

(1) Cladding Design Stress

The design basis for fuel rod cladding stress as given in the response to Q 231.2* is that the fuel system will not be damaged due to excessive fuel rod cladding stresses. The design limit for fuel rod cladding stress under Condition I and II modes of operation is that the volume-averaged effective stress calculated with the von Mises equation, considering interference due to uniform cylindrical pellet-to-cladding contact (caused by pellet thermal expansion and swelling, uniform cladding creep, and fuel rod/coolant system pressure differences), is less than the Zircaloy 0.2 percent offset yield stress as affected by temperature and irradiation. This is a traditional limit consistent with previous Westinghouse design practice and is, therefore, acceptable without further comment except with respect to the credit that is taken by Westinghouse for irradiation-induced strengthening.

The NRC does not routinely grant credit for irradiation-induced strengthening of the cladding, although the Standard Review Plan does not specificially preclude such practice. The staff has been considering such a credit in its ongoing review of a Westinghouse topical report on fuel material properties,

*All questions and responses referred to in this manner will be found in the correspondence cited above.

WCAP-9179, but that credit has not yet been approved. Notwithstanding, Westinghouse does, indeed, take irradiation strengthening into account in the design analysis of fuel rod cladding.

Because current staff understanding is that typical design cladding stresses under Condition I and II modes of operation are significantly below the 0.2 percent offset yield stress, the difference between irradiated and unirradiated values appear to be unimportant in this application, and the staff concludes that the cladding stress design-basis limits are acceptable.

(2) Cladding Design Strain

With regard to cladding strain, a design limit for fuel rod cladding plastic tensile creep (due to uniform cladding creep and uniform cylindrical fuel pellet swelling and thermal expansion) of less than 1 percent from the unirradiated condition is given in response to Q 231.2. Furthermore, the total tensile strain transient limit (due to uniform cylindrical pellet thermal expansion during the transient) is stated to be less than 1 percent from the pretransient value. While the staff has not explicitly reviewed the supporting data for normal operation (Condition I), that value appears to be consistent with past practice (no numerical value for normal operation cladding strain is provided as an acceptance criterion in the Standard Review Plan), and thus there is reasonable assurance that 1 percent total plastic creep strain is an acceptable design limit for normal operation, including Condition I power changes (load following). For transient-induced deformation, the Standard Review Plan indicates that 1 percent uniform cladding strain is an acceptable damage limit that should preclude some types of pellet/cladding interaction (PCI) failures. Such a limit, however, while consistent with past practice, should not be construed to be a broadly applicable PCI damage limit because there is ample evidence (Tokar, November 14, 1979) that PCI failures can occur at less than 1 percent uniform cladding strain. Westinghouse has indicated in response to staff question 231.24 that 1 percent plastic strain from the pretransient value is not meant to serve as a broadly applicable PCI criterion. Nevertheless, the staff finds the 1 percent cladding transient plastic strain criterion to be an acceptable design limit for the type of application indicated in SRP Section 4.2. For fuel assembly structural design, Westinghouse set design

limits on stresses and deformations due to various nonoperational, operational, and accident loads. As indicated in the FSAR, the stress categories and strength theory presented in Section III of the ASME Code are used as a general guide. This is consistent with acceptance criterion II.A.1(a) of SRP Section 4.2 and is acceptable.

(3) Strain Fatigue

The strain fatigue criteria given in response to Q 231.2 are the same as those described in SRP Section 4.2, viz., a safety factor of 2 on stress amplitude or of 20 on the number cycles and are, therefore, acceptable.

(4) Fretting Wear

While the Standard Review Plan does not provide numerical bounding-value acceptance criteria for fretting wear, it does stipulate that the allowable fretting wear should be stated in the safety analyis report and that the stress and fatigue limits should presume the existence of this wear.

From the response to Q 231.5, it can be seen that the Westinghouse design basis for fretting wear is that fuel rods shall not fail during Condition I and II events. Furthermore, Westinghouse does not use an explicit fretting wear limit in their stress and fatigue analysis for fuel rods. However, Westinghouse does use a value (proprietary) of wall thickness as a general guide in evaluating cladding imperfections, including fretting wear. Cladding imperfections including fretting wear are thus considered in the stress and fatigue analysis, albeit in a very qualitative, nonrigorous manner. In view of the apparently small effects of these defects and large stress and fatigue margins (see Section 4.2.3.1(4) of this Safety Evaluation Report), this design method is acceptable.

The design basis for guide thimble tubes is treated differently by Westinghouse, as described in the response to Q 231.41. The design basis is that the thinning of the guide thimble tube walls should not result in the failure of the fuel assembly structural integrity or functionability of the guide thimble tubes. The staff finds this to be an acceptable design basis.

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With regard to a design limit for guide thimble tube wear, Westinghouse has determined from stress analyses that the most limiting load on the fuel assembly structure is that which might occur during a fuel handling accident. For the analysis of this accident, Westinghouse uses a design criterion of 6 g. This design limit is therefore used for degraded guide thimble tubes and has been previously accepted for Westinghouse fuels.

(5) Oxidation and Crud Buildup

The SFA design basis for cladding oxidation and crud buildup is that the increase in cladding temperature due to cladding oxidation and crud buildup is not excessive (see Overheating of Cladding, below).

Section 4.2 of the Standard Review Plan identifies cladding oxidation and crud buildup as potential fuel system damage mechanisms. Because of the increased thermal resistance of these layers, there is an increased potential for elevated temperature within the fuel as well as the cladding. Because the effect of oxidation and crud layers on fuel and cladding temperature is a function of several different parameters (such as heat flux and thermal-hydraulic boundary conditions), a design limit on oxide or crud layer thickness does not, per se, preclude fuel damage as a result of these layers. Rather, it is necessary that these layers be appropriately considered in other temperature-related fuel system damage and failure analyses. This is, indeed, the approach taken by Westinghouse in the design of the Standard Fuel Assembly. The staff finds this approach acceptable.

(6) Rod Bowing

Fuel rod bowing is a phenomenon that alters the pitch dimensions between adjacent fuel rods. Bowing affects local nuclear power peaking and the local heat transfer to the coolant. Rather than placing design limits on the amount of bowing that is permitted, the effects of bowing are included in the safety analysis. This is consistent with the Standard Review Plan and is acceptable. The methods used for predicting the degree of rod bowing are evaluated in Section 4.2.3.1(6), and the impact of the resulting bow magnitude is evaluated in Sections 4.3 and 4.4.

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(7) Axial Growth

In the SFA design the core components requiring axial-dimensional analyses are the control rods, neutron source rods, burnable poison rods, fuel rods, and fuel assemblies (thimble plugging rods are omitted because they are short and not axial-growth limited). The axial growth of the first three of these components is primarily dependent upon the behavior of poison, source, or spacer pellets and their 304 stainless-steel cladding. The growth of the latter two is mainly governed by the behavior of fuel pellets, Zircaloy-4 cladding, and Zircaloy-4 guide thimble tubes.

The Westinghouse design bases for core component rods are that (a) dimensional stability and cladding integrity are maintained during Condition I and II events and (b) these components do not interfere with shutdown during Condition III and IV events.

Westinghouse does not, per se, have design limits on the axial growth of their control, source, and burnable poison rods. However, allowances are made to accommodate (a) pellet swelling due to gas production and (b) relative thermal expansion between the stainless-steel cladding and the encapsulated material. Westinghouse does not account for irradiation growth of the stainless-steel cladding and has cited experiments (Foster and Strain, October 1974) as justification for the insignificance of irradiation growth of stainless-steel at PWR operating conditions.

For the Zircaloy cladding and fuel assembly components, the axial-dimensional behavior is governed by creep (due to mechanical or hydraulic loading) and irradiation growth. The critical tolerances that require controlling are (a) the spacing between the fuel rods and the fuel assembly (shoulder gap) and (b) the spacing between the fuel assemblies and the core internals. Failure to adequately design for the former may result in fuel rod bowing, and for the latter may result in collapse of the holddown springs. With regard to inadequately designed shoulder gaps, problems have been reported (Schenk, October 1973; Kuffer and Lutz, 1973; and FSAR of R. E. Ginna Unit 1, 1972) in foreign (Obrigheim and Beznau) and domestic (Ginna) plants that have necessitated predischarge modifications to fuel assemblies.

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With regard to a design basis for shoulder gap spacing, Westinghouse stated in the responses to Q 231.2, 231.8, 231.25, and 231.40 that interference is precluded by having clearance between the fuel rod end and the top and bottom nozzles. The design clearance accommodates the differences in growth, fabrication tolerances, and the differences in thermal expansion between the fuel cladding and the thimble tubes. Westinghouse does not have specific limits on growth, but does provide a gap spacing that is equal to or greater than a percentage (the specific value is proprietary) of the fuel rod length. The percentage value used by Westinghouse provides gap spacings that are similar to those employed in other fuel vendor designs.

With regard to fuel assembly growth, Westinghouse has a design basis that there shall be no axial interference between the fuel assembly and upper and lower core plates caused by temperature or irradiation. As a design limit, Westinghouse provides a minimum gap (proprietary value that is a fraction of the fuel assembly length) between the fuel assembly and the reactor internals.

The above design bases and limits dealing with axial growth are acceptable.

(8) Fuel and Poison Rod Pressure

For Condition I and II events, the mechanical design basis for core component rods described in the FSAR is that dimensional stability and cladding integrity are maintained. A necessary corollary of this design basis is that the driving force, rod internal pressure, is never so great as to result in loss of dimensional stability and cladding integrity.

Section 4.2 of the Standard Review Plan identifies rod internal pressure as a potential fuel system damage mechanism. In this sense, damage is defined as an increased potential for elevated temperatures within the rod as well as an increased potential for cladding failure. Although the Standard Review Plan mentions only fuel and burnable poison rods, the mechanism also applies to control rods, neutron source rods, and other core component rods. Because rod internal pressure is a driving force for, rather than a direct mechanism of, fuel system damage, it is not necessary that a damage limit be specified. It is only necessary that the phenomenon be appropriately considered in other

fuel system damage and fuel failure analyses. In other words, rod internal pressure must be considered in calculating the temperature of the rod internals, cladding deformation, and cladding bursting.

In order to simplify the analysis of fuel system damage due to excessive rod internal pressure, the Standard Review Plan states that rod internal gas pressure should remain below the nominal system pressure during normal operation unless otherwise justified. Westinghouse has elected to justify limits other than that provided in the Standard Review Plan.

For the fuel rods, revised internal rod pressure criteria as described in an approved topical report (WCAP-8963) were used in the FSAR. Briefly stated, these criteria allow the fuel rod internal pressure to exceed the system pressure under certain conditions: (a) the internal pressure is limited such that the fuel-to-cladding gap does not increase during steady-state operation, and (b) extensive departure from nucleate boiling (DNB) propagation does not occur for postulated transients and accidents. These criteria have been previously approved and remain acceptable.

For nonfueled rods, the rod internal pressure is limited such that the mechanical design limits, discussed in Section 4.2.1.5 of the FSAR, are not exceeded for Condition I and II events. This implies a stress limit of 2/3 of the material yield stress and a strain limit of 1 percent. These limits are unchanged from previously approved Westinghouse fuel designs and remain acceptable for this FSAR.

(3) Assembly Liftoff

The Standard Review Plan calls for the fuel assembly holddown capability (gravity and springs) to exceed worst-case hydraulic loads for normal operation, which includes anticipated operational occurrences. The SFA design basis provides for positive holddown for Condition I, but allows momentary liftoff during one Condition II event. This design basis is acceptable provided that it can be shown that the affected fuel assemblies will reseat properly without damage and without other adverse effects during the event. The ability of the affected fuel assemblies to satisfy this provision will be discussed in paragraph 4.2.3.1.

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(10) Control Material Leaching

The Standard Review Plan and General Design Criteria require that control rod reactivity be maintained. Control rod reactivity can sometimes be lost by leaching of certain poison materials if the control rod cladding has been breached. The mechanical design basis for the control rods is stated in the FSAR to be consistent with the loading conditions of Section III of the ASME Code. Thus, the design basis for the SFA control rods is to maintain cladding integrity; because cladding integrity would ensure that reactivity is maintained, this design basis might appear to be acceptable. However, under some circumstances, unexpected breaches might go undetected, so the staff does not normally accept control rod cladding integrity as a sufficient design basis. A discussion will be presented under Design Evaluation, paragraph 4.2.3.1, that shows that adequate surveillance will be provided to ensure maintenance of reactivity.

4.2.1.2 Fuel Rod Failure Criteria

The NRC staff's evaluation of fuel rod failure threshold for the failure mechanisms listed in the SRP is presented in the following paragraphs. When these failure thresholds are applied to normal or transient operation, they are used as limits (and hence SAFDLs), since fuel failures under those conditions should not occur (according to the traditional conservative interpretation of GDC 10). When these thresholds are applied to accident analyses, the number of fuel failures must be determined for input to the radiological dose calculations required by 10 CFR 100. The basis or reason for establishing these failure thresholds is thus predetermined, and only the threshold values are reviewed below

(1) Internal Hydriding

Hydriding as a cladding failure mechanism is precluded by controlling the level of moisture and other hydrogenous impurities during fabrication. As described in the revised response to Q. 231.6, the moisture levels in the uranium dioxide fuel are limited by Westinghouse to less than or equal to 20 ppm. This specification is compatible with the ASTM specification for sintered Uranium Dioxide Pellets, which allows 2 µg hydrogen per gram of

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uranium (2 ppm), and they are the same as the limits provided in the Standard Review Plan; they are therefore acceptable.

(2) Cladding Collapse

If axial gaps in the fuel pellet column were to occur due to densification, the cladding would have the potential of collapsing into a gap (flattening). Because of the large local strains that would result from collapse, the cladding is assumed to fail. As indicated in the FSAR and responses to Q. 231.2, 231.9 and 231.34, it is a Westinghouse design basis that cladding collapse is precluded during the fuel rod design lifetime. This design basis is the same as that in the Standard Review Plan and is therefore acceptable.

(3) Overheating of Cladding

The design basis as given in the FSAR for the prevention of fuel failures due to overheating is that there will be at least 95 percent probability that departure from nucleate boiling (DNB) will not occur on the limiting fuel rods during normal operation or any transient conditions arising from faults of moderate frequency (Condition I and II events) at a 95 percent confidence level. This design basis is consistent with the thermal margin criterion of SRP Section 4.2 and is, thus, acceptable. The specific DNBR limits and methods of analysis are reviewed in Section 4.4.

(4) Overheating of Fuel Pellets

As a second method of avoiding cladding failure due to overheating, Westinghouse avoids centerline fuel pellet melting as a design basis. This design basis is the same as given in the Standard Review Plan and is thus acceptable.

The design limit corresponding to the design basis given above is that, during modes of operation associated with Condition I and Condition II events, there is at least a 95 percent probability that the peak kW/ft fuel rod will not exceed the UO_2 melting temperature. This design limit is an acceptable representation of the design basis given previously.

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(5) Pellet/Cladding Interaction

As indicated in SRP Section 4.2, there are no generally applicable criteria for PCI failure. However, two acceptance criteria of limited application are presented in the SRP for PCI: (a) less than 1 percent transient-induced cladding strain and (b) no centerline fuel melting. The response to Q. 231.2 indicates that the 1 percent cladding plastic strain limit is met for the SFA design, and as stated in Section 4.2.1.2 of the FSAR, the SFA design ensures that UO_2 centerline melting will not occur through selection of a calculated fuel centerline temperature of 4700°F as an overpower limit. Thus the SFA design basis and limits agree with the only existing licensing criteria for PCI.

(6) Cladding Rupture

In the LOCA analysis for SFA-designed plants, an empirical model is used to predict the occurrence of cladding rupture. The failure temperature is expressed as a function of differential pressure across the cladding wall. There are no specific design limits associated with cladding rupture, and the rupture model is a portion of the ECCS evaluation model, which is documented in WCAP-8301 and WCAP-8302.

4.2.1.3 Fuel Coolability Criteria

For major accidents in which severe fuel damage might occur, core coolability must be maintained as required by several GDCs (e.g., GDC 27 and 35). The following paragraphs discuss the staff's evaluation of limits that will assure that coolability is maintained for the severe damage mechanisms listed in Section 4.2 of the SRP.

(1) Fragmentation of Embrittled Cladding

For LOCA analysis, Westinghouse uses the acceptance criteria of 2200°F on peak cladding temperature and 17 percent on maximum cladding oxidation as prescribed by 10 CFR 50.46.

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For events other than the LOCA, the NRC staff does not have separately established temperature or oxidation criteria. Yet it is clear that for short-term events such as locked rotor, the 2200°F peak cladding temperature and 17 percent oxidation LOCA criteria are not really meaningful, because the temperature history for such an event is much shorter than that of a LOCA. For events such as locked rotor, therefore, Westinghouse uses a unique peak-cladding-temperature (PCT) criterion of 2700°F.

The Westinghouse 2700°F PCT limit was selected taking into consideration the short time (a few seconds) that the fuel is calculated to be in DNB for a locked-rotor type event and the fact that the PCT and total metal-water reaction at the fuel hot spot would not be expected to impact fuel coolable geometry. While this limit has been used by Westinghouse for several years, the basis for the limit has only recently been reviewed. However, a recent assessment (Van Houten, February 23, 1981) of the available experimental information indicates that fuel rod cladding will, indeed, retain its rod-like geometry after exposure to short-term (a few seconds) peak cladding temperature of 2700°F. That conclusion is based on four Japanese reports (Shiozawa, March 1979; Hoshi, May 1980; JAERI, September 1980; and Fukishiro, October 1980) that describe experimental results for reactor test programs reported since 1979. The staff, therefore, concludes that there is reasonable assurance that the 2700°F PCT limit for short-term events such as locked rotor is an acceptable coolability limit for the Westinghouse SFA design.

It should be noted that staff acceptance of the 2700°F PCT limit for fuel rod coolability is currently restricted to undercooling events such as locked rotor. For overpower events such as control rod ejection, which involve a pellet-to-cladding mechanical interaction, the staff has not determined the applicability of a PCT limit and currently uses a fuel rod enthalpy criterion of 280 cal/g for coolability of a rod-ejection accident.

(2) Violent Expulsion of Fuel Material

The design bases that there should be little or no possibility of fuel dispersal in the coolant, gross lattice distortion, or severe shock waves are given in Section 15.4.8.1.2 and are equivalent to those in the Standard Review Plan.

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The design limits given in the FSAR are:

(a) Average fuel pellet enthalpy at the hot spot will be below 225 cal/g for unirradiated fuel and 200 cal/g for irradiated fuel.

(b) Average cladding temperature at the hot spot will be below the temperature at which cladding embrittlement may be expected (2700°F).

(c) Peak reactor coolant pressure will be less than that which could cause pressures to exceed the faulted condition stress limits.

(d) Fuel melting will be limited to less than 10 percent of the fuel volume at the hot spot even if the average fuel pellet enthalpy is below the limits above.

These limits are more conservative than the single 280 cal/g limit given in Regulatory Guide 1.77, they have been previously approved in the review of WCAP-7588, and they remain acceptable.

(3) Cladding Ballooning and Flow Blockage

In the LOCA analyses for SFA-designed plants, empirical models are used to predict the degree of cladding circumferential strain and assembly flow blockage at the time of hot-rod and hot-assembly burst. These models are each expressed as functions of differential pressure across the cladding wall. There are no specific design limits associated with ballooning and blockage, and the ballooning and blockage models are portions of the ECCS evaluation model, which is documented in WCAP-8301 and WCAP-8302.

(4) Structural Damage from External Forces

Section 4.2.3.5 of the FSAR states that the fuel assembly will maintain a geometry that is capable of being cooled under the worst case accident Condition IV event and that no interference between control rods and thimble tubes will occur during a safe shutdown earthquake. This is equivalent to the design basis as presented in the Standard Review Plan and is therefore acceptable.

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4.2.2 Description and Design Drawings

The description of fuel system components, including the fuel rods, bottom and top nozzles, guide and instrument thimble tubes, grid assemblies, rod cluster control assemblies, burnable poison rods, neutron sources, and thimble plugs, is contained in the FSAR Section 4.2.2. In addition, in Table 4.3-1 numerical values are provided for various core component parameters. While each parameter listed in SRP subsection 4.2.2 is not provided in the FSAR, enough information is provided in sufficient detail to provide a reasonably accurate representation of the SFA design and this information is thus acceptable.

4.2.3 Design Evaluation

Design bases and limits were presented and discussed in SER Section 4.2.1. In this section we review W methods of demonstrating that the SFA fuel design meets the design criteria that have been established. This SER subsection will, therefore, correspond to subsection 4.2.1 of the SER point by point. The methods of demonstrating that the design criteria have been met include operating experience, prototype testing, and analytical predictions.

4.2.3.1 Fuel System Damage Evaluation

The following paragraphs discuss the NRC staff's evaluation of the ability of the SFA fuel to meet the fuel system damage criteria described in Section 4.2.1.1. Those criteria apply only to normal operation and anticipated transients.

(1) Cladding Design Stress

As indicated in the response to Q. 231.2, Westinghouse used its Performance-Analysis and Design (PAD) code to analyze cladding stress (WCAP-8720). That code has been reviewed and found acceptable (Stolz, February 9, 1979). Typical calculated design values for cladding effective stress provided in response to Q.231.2 are considerably below the 0.2 percent offset yield stress design limit. Hence, the staff concludes that the SFA cladding stress design limit has been met.

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(2) Cladding Design Strain

The NRC-approved Westinghouse fuel performance code (PAD) was used in the strain analysis, as indicated in the response to O. 231.2. Typical design values of steady-state and transient creep strain, as calculated by that code, are found to be below the 1 percent strain criterion. Hence, the staff concludes that the SFA cladding strain design limits have been met.

(3) Strain Fatigue

As indicated in the response to O 231.2, Westinghouse used their approved PAD code for the strain range and strain fatigue life usage analysis. Experimental data (proprietary) obtained from Westinghouse testing programs were used to derive the Westinghouse Zircaloy fatigue design curve, according to the response to O. 231.4. For a given strain range, the number of fatigue cycles is less than that required for failure, considering a minimum safety factor of 2 on stress amplitude or a minimum safety factor of 20 on the number of cycles, (the fatigue usage factor is less than 1.0). And the computations were performed with an approved code. Therefore, the staff concludes that the SFA fatigue design basis has been met.

(4) Fretting Wear

With regard to the Westinghouse fretting analysis of the fuel cladding, the staff concludes the following:

(a) The out-of-pile flow tests and analyses (WCAP-9401) to determine the magnitude of fretting wear that is anticipated for the OFA design have been previously reviewed and found acceptable (Rubenstein, April 23, 1981). These analyses are also acceptably conservative for SFA applications.
(b) LWR operating experience demonstrates that the number of fretting-induced fuel failures is insignificant.
(c) There should be only a small dependence of cladding stresses on fretting wear because this type of wear is local at grid-contact locations and relatively shallow in depth.
(d) The built-in conservatisms (that is, safety factors of 2 on the stress amplitudes and 20 on the number of cycles) in the strain fatigue analysis

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as well as the calculated margin to fatigue life limit adequately offset the effect of fretting wear degradation.

Therefore, the staff concludes that the SFA fuel rods will perform adequately with respect to fretting wear.

Fretting wear has also been observed on the inner surfaces of guide thimble tubes where the fully withdrawn control rods reside. Significant wear is limited to the relatively soft Zircaloy-4 guide thimble tubes because the Inconel or stainless steel control rod claddings are relatively wear-resistant. The extent of the wear is both time-dependent and plant-dependent and has, in some non-Westinghouse cases, extended completely through the guide thimble tube wall.

Westinghouse has predicted that an SFA can operate under a rod cluster control assembly (RCCA) for a period of time (proprietary) that exceeds the amount of rodded time expected with current 3-cycle fuel schemes before fretting wear degradation would result in exceeding the present margin to the 6 g load criterion for the fuel handling accident. However, the NRC required several applicants to perform a surveillance program because of the uncertainties in predicting wear rates for the standard 17x17 fuel assembly design. The objective of this program is to demonstrate that there is no occurrence of hole formation in rodded guide thimble tubes, thus providing some confidence that scrammability is ensured. These applicants formed an owners' group, which has submitted a generic report (Leasburg, March 1, 1982) that provides post-irradiation examination results on guide thimble tube wear in the Westinghouse 17x17 fuel assembly design. Based on this report, the staff has concluded (Rubenstein, April 19, 1982) that the Westinghouse 17x17 fuel assembly design is resistant to guide thimble tube wear.

(5) <u>Oxidation and Crud Buildup</u>

In the FSAR, there is no explicit discussion of cladding oxidation, hydriding, and crud buildup. The applicable models for cladding oxidation and crud buildup are discussed in the supporting documentation (Salvatori, January 4, 1973) for the Westinghouse fuel performance code PAD-3.1. These models were

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models were previously approved by the NRC staff. A new temperature-dependent cladding oxidation model is also presented in WCAP-9179. Because the temperature-independent model in PAD-3.1 is conservative with respect to the unapproved model in WCAP-9179, the staff continues to find the older models applicable. These models affect the cladding-to-coolant heat transfer coefficient and the temperature drop across the cladding wall. Mechanical properties and analyses of the cladding are not significantly impacted by oxide and crud buildup. On the basis of the Westinghouse discussion (Anderson, January 12, 1981) of the impact of cladding hydriding on fuel performance, and on previous staff review of the oxidation and crud buildup models, the staff concludes that these effects have been adequately accounted for in the Standard Fuel Design.

(6) Rod Bowing

The NRC has previously approved (Meyer, March 2, 1978) the rod bowing correlation (Anderson, April 19, 1978) that was used by the applicant.

(7) Axial Growth

Relative to the discussion above (4.2.1) on stainless steel growth, the staff is aware of supporting information (Bloom, April 1972, and Appleby, April 1972) that was not cited by Westinghouse, but which also implies that irradiation growth of stainless steel should not be significant at the temperatures and fluences that are associated with PWR operation. Furthermore, because the staff is unaware of any operating experience that indicates axial-growth-related problems in Westinghouse NSSS plants, the staff concludes that Westinghouse has made sufficient accommodations for control, source, and burnable poison rod axial rod growth in their NSSS designs.

The Westinghouse analysis of shoulder gap spacing for the SFA has found that interference will not occur until achieving burnups beyond traditional values. The staff, therefore, finds that the required shoulder gap spacing has been

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reasonably accommodated. However, for extended burnup applications, the adequacy of the spacing should be reverified. Furthermore, because stress-free irradiation growth of zirconium-bearing alloys is sensitive to texture (preferred cystallographic orientation) and retained cold work, which, in turn, are strongly dependent on the specific fabrication techniques that are employed during component production, reverification of the design shoulder gap should be performed if Westinghouse current fabrication specifications are significantly altered.

Finally, the staff finds the Westinghouse analysis of fuel assembly growth to be acceptable. However, as stated in the above discussion on shoulder gap spacing, reverification of the fuel assembly growth should be performed if significant changes are made in the Westinghouse current fabrication techniques.

(8) Fuel and Poison Rod Pressure

The analysis of fuel rod internal pressure for the Standard Fuel Design is described in an approved topical report, WCAP-8963-A. The evaluation relies on the Westinghouse PAD-3.3 fuel performance code, which has also been approved (Stolz, February 9, 1979) by the staff.

The analysis of nonfueled rod internal pressure for the SFA is generally based on Section III, Article NG-3000, of the ASME Code. Control rod, neutron source rod, and burnable poison rod cladding is 10 percent cold-worked 304 stainless steel, which is not covered by the Code. Westinghouse therefore defines as the stress limit an intensity value Sm equal to 2/3 of the material yield stress. The yield for this material occurs at about 62,000 psi. A strain limit of 1 percent also applies to the cladding. Predicted maximum values of rod internal pressure have been provided in an answer to NRC question Q 231.2 and they are well below those imposed by the cladding stress and strain limits.

The staff concludes that there is adequate assurance that nonfueled core component rods can operate safety during Conditions I and II modes of operation even though maximum internal rod pressure may exceed system pressure because appropriate stress and strain limits are met.

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(9) Assembly Liftoff

In response to the staff's question on this topic, Westinghouse has confirmed that momentary liftoff will occur only during a turbine overspeed. Westinghouse has further found that (a) proper reseating will occur after momentary liftoff, (b) damage to adjacent assemblies will not occur even if one assembly is fully lifted and the adjacent one remains seated, and (c) no ill consequences of momentary liftoff are expected. The staff concludes, therefore, that fuel assembly liftoff has been adequately addressed for the SFA design.

(10) Control Material Leaching

While the design basis for the SFA control rods is to maintain cladding integrity, and while the probability of control rod cladding failures appears to be quite low, the staff has considered the corrosion behavior of the Seabrook control material and concludes that a breach in the cladding should not result in serious consequences because the hafnium or Ag-In-Cd absorber material is relatively inert.

We have reviewed the use of the new Westinghouse hafnium control rods for Commanche Peak and the Callaway Units (see paragraph 4.2.3.1(10) of NUREG-0830) and found the hafnium rods acceptable subject to some required surveillance. Surveillance of the new hafnium control rods will not be required for Seabrook because of the first-use surveillance programs that will be performed at Comanche Peak and Callaway Unit 1.

4.2.3.2 Fuel Rod Failure Evaluation

The following paragraphs discuss the staff's evaluation of (a) the ability of the SFA fuel to operatre without failure during normal operation and anticipated transients, and (b) the accounting for fuel rod failures in the applicant's accident analysis. The fuel rod failure criteria described in Section 4.2.1.2 were used for this evaluation.

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(1) Internal Hydriding

Westinghouse has used moisture and hydrogen control limits in the manufacture of earlier fuel types and has found that typical end-of-life cladding hydrogen levels are less than 100 ppm -- a level below which hydride blister formation is not anticipated in fuel cladding.

The staff therefore concludes that reasonable evidence has been provided that hydriding as a fuel failure mechanism will not be significant in the SFA.

(2) Cladding Collapse

In calculating the time at which cladding collapse will occur, Westinghouse uses the generic methods described in WCAP-8377, which is approved (Stello, January 14, 1975) for licensing applications. Inputs to the analysis include cladding ovality, helium prepressurization, free volume of the fuel rod, and limiting power histories.

Westinghouse adjusts the fuel rod pressure so that cladding collapse will not occur at a residence time that is less than the design lifetime. Consequently, the staff expects that cladding collapse will not occur, but confirmation should be provided by the applicant by showing that the calculated cladding collapse time for Seabrook using WCAP-8377 methods is more than the expected lifetime of the fuel.

(3) Overheating of Cladding

As stated in SRP Section 4.2, adequate cooling is assumed to exist when the thermal margin criterion to limit the departure from nuclear boiling (DNB) or boiling transition in the core is satisfied. The method employed to meet the DNB design basis is reviewed in Section 4.4 and will not be discussed here.

(4) Overheating of Fuel Pellets

The design evaluation of the fuel centerline melt limit is performed with the Westinghouse fuel performance code, PAD-3.3 (WCAP-8720). This code, which has

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been approved by the NRC (Stolz, February 9,1979), is also used to calculate initial conditions for transients and accidents described in Chapter 15 of the Standard Review Plan (see paragraph 4.2.3.3(1) below for further comments on PAD-3.3).

In applying the PAD-3.3 code to the centerline melting analysis, the melting temperature of the UO_2 is assumed to be 5081°F unirradiated and is decreased by 58°F per 10,000 MWd/t. This relation has been almost universally adopted by the industry and has been accepted by the NRC staff in the past. The expressions for thermal conductivity and gap conductance, described in Section 4.4.2.11 of the FSAR, are unchanged from that originally described in the PAD code. The staff considers it unnecessary to further review these models.

In order to avoid using the PAD code to calculate a continuous set of burnup-dependent conditions necessary to cause centerline melt, Westinghouse has performed the calculation for a single case. This was done by assuming a UO_2 melting temperature of 4701°F, which corresponds to the melting temperature at 65,000 MWd/t, and melting occurred at a linear power rating of approximately 21 kW/ft. The limiting local power for the worst Condition II transient, boron dilution with automatic rod control, is less than or equal to 18 kW/ft for Westinghouse plants with 17x17 fuel. Thus, the centerline melt criterion is satisfied in an acceptable manner.

(5) Pellet/Cladding Interaction

The only two PCI criteria in current use in licensing (1 percent cladding strain and no fuel melting), while not broadly applicable, are easily satisfied. As noted in the discussion of the cladding stress and strain evaluation, Westinghouse uses an approved code (PAD) to calculate creep strain, and the values calculated by that code are found to be below the 1 percent strain criterion. And, as indicated in the discussion on overheating failures, the no-centerline-melt criterion is satisfied based on an analysis (described in Chapter 15.4.6) of the boron dilution event, which is analyzed with an approved code. Therefore, the two existing licensing criteria for PCI have been satisfied.

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In addition to the SRP-type treatment of PCI, however, responses to Q 231.23 and FSAR Section 4.2.3.3(a) address PCI from the standpoint of its effect on fatigue life. Thus, PCI produces cyclic stresses and strains that can affect fatigue life of the cladding. Furthermore, gradual compressive creep of the cladding onto the fuel pellet occurs due to the differential pressure exerted on the fuel rod by the coolant. Westinghouse contends that, by using prepressurized fuel rods, the rate of cladding creep is reduced, thus delaying the time at which fuel-to-cladding contact first occurs. The staff agrees that fuel rod prepressurization should improve PCI resistance, albeit in a presently unquantified amount.

In conclusion, Westinghouse has used approved methods to demonstrate that the present PCI acceptance criteria have been met.

(6) Cladding Rupture

Although a revised cladding rupture temperature correlation has recently been approved (Ref. 1) as an integral part of the 1981 ECCS evaluation model, we previously concluded (Ref. 2) that the old correlation in the earlier ECCS evaluation model was non-conservative over some regions of applicability. To compensate for this deficiency, supplemental calculations have been required for each plant application that uses the earlier Westinghouse ECCS evaluation model. Since the Seabrook analysis was done with this earlier model, supplemental calculations should be provided to demonstrate that Seabrook would conform to the ECCS acceptance criteria of 10 CFR 50.46 if the NRC staff cladding rupture temperature model (Ref. 2) was substituted for the Westinghouse model contained in WCAP-8301.

This requirement for supplemental ECCS calculations is the same as the present requirement made for all operating license applications and all ECCS reanalyses of operating reactors (Eisenhut November 9, 1979, and Denton November 26,1979). See Section 4.2.3.3(3) for a concurrent requirement on cladding ballooning and flow blockage models. The supplemental calculations have not been provided for Seabrook. The overall impact of cladding rupture on the response of the SFA design to the loss-of-coolant accident is evaluated in Section 15.6.5 and not reviewed further in this section.

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4.2.3.3 Fuel Coolability Evaluation

The following paragraphs discuss the staff's evaluation of the ability of the SFA fuel to meet the fuel coolability criteria described in Section 4.2.1.3. Those criteria apply to postulated accidents.

(1) Fragmentation of Embrittled Cladding

The primary degrading effect of a significant degree of cladding oxidation is embrittlement of the cladding. Such embrittled cladding will have a reduced ductility and resistance to fragmentation. The most severe occurrence of such embrittlement is during a LOCA. The overall effects of cladding embrittlement on the SFA design for the loss-of-coolant accident are analyzed in Section 15.6.5 and are not reviewed further in this section.

One of the most significant analytical methods that is used to provide input to the analysis in Section 15.6.5 is the steady-state fuel performance code, which is reviewed in Section 4.2. This code provides fuel pellet temperatures (stored energy) and fuel rod gas inventories for the ECCS evaluation model as prescribed by Appendix K to 10 CFR 50.46. The code accounts for fuel thermal conductivity, fuel densification, gap conductance, fuel swelling, cladding creep, and other phenomena that affect the initial stored energy.

Westinghouse uses a relatively new fuel performance code called PAD-3.3 (WCAP-8720). This new Westinghouse code was approved with four restrictions as described in the staff's safety evaluation (Stolz, February 9, 1979). Three of those restrictions deal with numerical limits and have been met. The fourth restriction relates to the use of the PAD-3.3 code for the analysis of fission gas release from UO_2 for power-increasing conditions during normal operation. This restriction applies to the SFA. However, Westinghouse has stated that this restriction does not adversely affect the results of the as-submitted safety analyses. Although the staff believes that this is essentially correct, Westinghouse has prepared and submitted a detailed evaluation (Anderson, October 22, 1979) of this restriction.

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At this time, the staff has not completed its review of the Westinghouse evaluation of this restriction. However, the review has progressed to the point where the following conclusions can be made:

(a) The Westinghouse evaluation of our restriction on the use of the PAD-3.3 code supports the earlier statement that the restriction does not adversely affect the results of the safety analyses performed for SFA.

(b) Based on additional information submitted by Westinghouse to confirm this conclusion, the staff continues to find this result is essentially correct.

(c) Because the restriction pertains to the release of fission gases from the fuel, any change in the staff's conclusion would not have significant impact at low burnup, when the fission gas inventory in the fuel is low.

At this time, the staff can therefore state that for the first cycle operation at full power, the restriction for PAD-3.3 is not significant and the analyses presented in the FSAR are acceptable. The staff anticipates completion of its review of the Westinghouse evaluation prior to the attainment of extended burnup at the Seabrook plant.

For non-LOCA events, the locked rotor accident (one-pump seizure with three loops operating) is the most severe undercooling event that is analyzed. This event is analyzed in Section 15.3.3 of the FSAR, where it is found that the peak cladding temperature is 2250°F, which is well below the 2700°F design limit. The analysis of this event is reviewed in Section 15.3.3 of this report, but it is clear that the SFA meets the non-LOCA peak cladding temperature design limit.

(2) Violent Expulsion of Fuel Material

The analysis that demonstrates that the design limits are met for this event for the SFA is presented in Section 15.4.3 of the FSAR and is reviewed in that section of the report.

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(3) Cladding Ballooning and Flow Blockage

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Although revised cladding strain and assembly flow blockage correlations have recently been approved (Ref. 1) as integral parts of the 1981 ECCS evaluation model, we previously concluded (Ref. 2) that both correlations in the earlier ECCS evaluation model were non-conservative over some regions of applicability. To compensate for these deficiencies, supplemental calculations have been required for each plant application that uses the earlier Westinghouse ECCS evaluation model. Since the Seabrook analysis was done with this earlier model, supplemental calculations should be provided to demonstrate that Seabrook would conform to the ECCS acceptance criteria of 10 CFR 50.46 if the NRC staff cladding strain and assembly flow blockage models (Ref. 2) were substituted for the Westinghouse models contained in WCAP-8301 and WCAP-8302.

This requirement for supplemental ECCS calculations is the same as the present requirement made for all operating license applications and all ECCS reanalyses of operating reactors (Eisenhut, November 9, 1979, and Denton, November 26, 1979). (See paragraph (6) of Section 4.2.3.2 for a concurrent requirement on the cladding rupture model.) The applicant has not yet provided supplemental ECCS calculations for the Seabrook plant.

The overall impact of cladding ballooning and assembly flow blockage models on the responses of the SFA design to the loss-of-coolant accident is evaluated in Section 15.6.5 and is not reviewed further in this section.

(4) Structural Damage from External Forces

Section 4.2.3.5 of the FSAR refers to WCAP-8236 for this analysis. The staff has reviewed and approved another report (WCAP-9401) which essentially augments the information presented in WCAP-8236 because both WCAP reports apply to similar assemblies. For the Seabrook application, however, the applicant must demonstrate compliance with Appendix A of SRP Section 4.2. The applicant may make reference to WCAP-8236 and WCAP-9401 to accomplish this.

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4.3 Nuclear Design

The Seabrook Units 1 and 2 power plants have a reactor core consisting of 193 fuel assemblies of the Westinghouse 17x17 design. The core has a design heat output of 3800 thermal Megawatts and is essentially identical to the W. B. McGuire reactor and other recent Westinghouse 4 loop reactors. We have reviewed the nuclear design of the Seabrook Units 1 and 2 reactors. Our review was based on information contained in the Final Safety Analysis Report (FSAR), amendments thereto, and the referenced topical reports. Our review was conducted in accordance with the guidelines provided by the Standard Review Plan, Section 4.3.

4.3.1 Design Bases

Design bases are presented which comply with the applicable General Design Criteria. Acceptable fuel design limits are specified (GDC 10), a negative prompt feedback coefficient is specified (GDC 11) and tendency toward divergent operation (power oscillation) is not permitted (GDC 12). Design bases are presented which require a control and monitoring system (GDC 13) which automatically initiates a rapid reactivity insertion to prevent exceeding fuel design limits in normal operation or anticipated transients (GDC 20). The control system is required to be designed so that a single malfunction or single operator error will cause no violation of fuel design limits (GDC 25). A reactor coolant boration system is provided which is capable of bringing the reactor to cold shutdown conditions (GDC 26) and the control system is required to control reactivity changes during accident conditions when combined with the engineered safety features (GDC 27). Reactivity accident conditions are required to be limited so that no damage to the reactor coolant system boundary occurs (GDC 28).

We find the design bases presented in the FSAR to be acceptable.

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4.3.2 Design Description

The FSAR contains the description of the first cycle fuel loading which consists of three different enrichments and has a first cycle length of approximately one year. The enrichment distribution, burnable poison distribution, soluble poison concentration and higher isotope (actinide) content as a function of core exposure are presented. Values presented for the delayed neutron fraction and prompt neutron life-time at beginning and end of cycle are consistent with those normally used and are acceptable.

Power Distribution

The design bases affecting power distribution are:

- The peaking factor in the core will not be greater than 2.50 during normal operation of full power in order to meet the initial conditions assumed in the loss of coolant accident analysis.

- Under normal conditions (including maximum overpower) the peak fuel power will not produce fuel centerline melting.

- The core will not operate during normal operation or anticipated operational occurrences, with a power distribution that will cause the departure from nucleate boiling ratio to fall below 1.3 (W-3 correlation with modified spacer factor).

The applicant has described the manner in which the core will be operated and power distribution monitored so as to assure that these limits are met. The core will be operated in the Constant Axial Offset Control (CAOC) mode which has been shown to result in peaking factors less than 2.32 for both constant power and load following operation. The applicant has elected to use an improved load follow package, developed by Westinghouse, in Seabrook Units 1 and 2.

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CAOC is described in WCAP-8385 (Proprietary) and WCAP-8403 (non-Proprietary), "Power Distribution Control and Load Following Procedures." This report contains methodology for operation with and without part length control rods. The former mode allows better return to power capability than the latter. Use of part length rods has been withdrawn from Westinghouse reactors. The improved load follow strategy provides a return to power capability during operation without part length rods comparable to the level previously obtainable from operation with part length rods.

The improved load follow strategy involves a redesigned control rod bank and modified overlap that allows greater reactivity insertion than the former design bank within the constraints of a widened, asymmetric CAOC band. The control band has been changed from nine to five rods. The four rods removed from the control bank have been redesigned as a shutdown bank, thus maintaining shutdown margins. (There are also an extra four rods assigned to control group A, compared to other Westinghouse reactors.) The CAOC band has been changed from ±5 to +3, -12, Δ I (delta flux difference). The greater inserted reactivity is available for return to power capability upon control rod withdrawal. Another element in the load follow strategy is the use of moderator temperature reductions to augment return to power capability. The temperature reduction adds reactivity during rapid return to power through the inherently negative moderator temperature coefficient.

The analysis used to calculate the maximum peaking factor which can occur using the improved strategy expands the set in the CAOC topical report to 18 calculational cases. However, with the redesigned control bank, maneuvers resulting in greater control rod insertion for a longer duration become operationally practical but tend to become slightly more limiting in terms of total peaking factors. Therefore, simulated load follow maneuvers which return ΔI to the target value (and thereby reduce control rod insertion) have been replaced by load follow strategies which maintain the deeper rod insertion. As a result of our evaluation, we agree with Westinghouse's conclusion that substitution of these more conservative cases will maintain the limiting nature of the 18 case load following analysis.

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The analysis performed by Westinghouse indicated that the peaking factor limit could not be met at BOL of Cycle 1 due to the wide ΔI band. This resulted in limiting the width of the band for the first 20% of the cycle typically, and until 3,000 MWD/MTU burnup for Seabrook Units 1 and 2 to the value of $\pm 5\% \Delta I$. This $\pm 5\% \Delta I$ is the value previously justified by the CAOC analysis. These features will be incorporated in the Seabrook Technical Specifications.

We conclude, for the reasons stated above, that the improved load follow package will continue to prevent the peaking factor limit from being exceeded in normal operation of the power plant, and therefore is acceptable.

Two types of instrumentation systems are provided to monitor core power distribution. Excore detectors are used to monitor core power, axial offset and azimuthal tilt, and movable incore detectors permit detailed power distributions to be measured. These systems are used in operating reactors supplied by Westinghouse and we find their use acceptable for Seabrook. In addition, the Seabrook incore system also contains a set of fixed self powered neutron detectors not normally a part of a Westinghouse design, but similar to systems in Combustion Engineering and Babcock and Wilcox reactors. Initial operation of the reactor will not use these as part of the measurement system but will use the movable system in a standard way. Eventual use of the fixed system will be contingent on accumulating data and approval of future license amendments with supporting analyses.

Reactivity Coefficients

The reactivity coefficients are expressions of the effect on core reactivity of changes in such core conditions as power, fuel and moderator temperature, moderator density, and boron concentration. These coefficients vary with fuel burnup and power level. The applicant has presented values of the coefficients in the FSAR and has evaluated the uncertainties of these values. We have reviewed the calculated values of reactivity coefficients and have

concluded that they adequately represent the full range of expected values. We have reviewed the reactivity coefficients used in the transient and accident analyses and conclude that they conservatively bound the expected values, including uncertainties. Further, moderator and power Doppler coefficients along with boron worth are measured as part of the startup physics testing to assure that actual values are within those used in these analyses.

Control

To allow for changes in reactivity due to reactor heatup, load following, and fuel burnup with consequent fission product buildup, a significant amount of excess reactivity is built into the core. The excess reactivity is controlled by a combination of full length control rods and soluble boron. Soluble boron is used to control changes due to:

- Moderator density and temperature changes from ambient to operating temperatures.
- Equilibrium xenon and samarium buildup.
- Fuel depletion and fission product buildup - that portion not controled by lumped burnable poison.
- Transient xenon resulting from load following.

Control rods are used to control reactivity change due to:

- Moderator reactivity changes from hot zero to full power.
- Fuel temperature changes (Doppler reactivity changes).

Burnable poison rods placed in some fuel assemblies are used for radial flux shaping and to control part of the reactivity change due to fuel depletion and fission product buildup.

The applicant has provided data to show that adequate control exists to satisfy the above requirements with enough additional control rod worth to provide a hot shutdown effective multiplication factor less than the design basis value of 0.987 during initial and equilibrium fuel cycles with the most reactive control rod stuck out of the core. In addition, the chemical and volume control system will be capable of shutting down the reactor by adding soluble boron and maintaining it shut down in the cold, xenon free condition at any time in core life. These two systems satisfy the requirements of General Design Criterion 26.

Comparisons have been made between calculated and measured control rod bank worth in operating reactors and in critical experiments. These comparisons lead to the conclusion that bank worths may be calculated to within approximately ten percent. In addition bank worth measurements are performed as part of the startup test program to assure that conservative values have been used in safety analyses.

Based on these comparisons, we conclude that the applicant has made suitably conservative assessments of reactivity control requirements and that adequate control rod worths have been provided to assure shutdown capability.

The applicant is using hybrid B_4C control rods rather than the Ag-In-Cd alloy rods more commonly used in Westinghouse reactors. This rod uses B_4C pellets rather than Ag-In-Cd in the upper regions of the rod but is otherwise similar to the usual rod. The physics aspects of this rod were reviewed in connection with the Westinghouse topical report on the B_4C control rod, WCAP-8846, and it was concluded that the design is acceptable (D. Ross, August 26, 1977), and it is thus acceptable for use in the Seabrook reactors.

Control Rod Patterns and Reactivity Worths

The control rods are divided into two categories - shutdown rods and regulating rods. The shutdown rods are always completely out of the core when the reactor is at operating conditions. Core power changes are made with regulating rods

which are nearly out of the core when it is operating at full power. Regulating rod insertion will be controlled by power-dependent insertion limits required in the Technical Specifications to assure that:

- There is sufficient negative reactivity available to permit rapid shutdown of the reactor with adequate margin.

- The worth of a control rod that might be ejected is not greater than that which has been shown to have acceptable consequences in the safety analyses.

We have reviewed the calculated rod worths and the uncertainties in these worths, and conclude that rapid shutdown capability exists at all times in core life assuming the most reactive control rod assembly is stuckout of the core.

Stability

The stability of the Seabrook Units 1 and 2 cores to xenon induced spatial oscillations is discussed in the FSAR. The overall negative reactivity (power) coefficient provides assurance that the reactor will be stable against total power oscillation. The applicant also concluded that sustained radial or azimuthal xenon oscillations are not possible. This conclusion is based on measurements on an operating reactor of the same dimensions which showed stability against these oscillations. We concur with this conclusion.

This core is predicted to be unstable with respect to axial xenon oscillations after about 12000 Megawatt days per ton of exposure. The applicant has acceptably shown that axial oscillations may be controlled by the regulating rods to prevent reaching any fuel damage limits.

Criticality of Fuel Assemblies

Criticality of fuel assemblies outside the reactor is precluded by adequate design of fuel transfer and storage facilities. The applicant presents information on calculational techniques and assumptions used to assure that criticality is avoided. We have reviewed this information and the criteria which will be employed and find them to be acceptable.

Vessel Irradiation

Values are presented for the neutron flux in various energy ranges at mid-height of the pressure vessel inner boundary. Core flux shapes calculated by standard design methods are input to a transport theory calculation (Sn) which results in a neutron flux of 2.1×10^{10} neutrons per square centimeter per second having energy greater than 10^6 electron-volts at the inner vessel boundary. This results in a fluence of 2.2×10^{19} neutrons per square centimeter for a forty year vessel life with an 80 percent use factor. The methods used for these calculations are state of the art, and we conclude that acceptable analytical procedures have been used to calculate the vessel fluence. The Materials Engineering Branch will review the requirements for surveillance programs and the pressure-temperature limits for operation.

4.3.3 Analytical Methods

The applicant has described the computer programs and calculational techniques used to obtain the nuclear characteristics of the reactor design. The calculations consist of three distinct types, which are performed in sequence: determination of effective fuel temperatures, generation of macroscopic few-group parameters, and space-dependent few-group diffusion calculations. The programs used (e.g., LASER, TWINKLE, LEOPARD, TURTLE and PANDA) have been applied as part of the applications for most earlier Westinghouse designed nuclear plant facilities and the predicted results have been compared with measured characteristics obtained during many startup tests for first cycle and reload cores. These results have validated the ability of these methods to predict experimental results.

We, therefore, conclude that these methods are acceptable for use in calculating the nuclear characteristics of the Seabrook Units 1 and 2.

4.3.4 Summary of Evaluation Findings

The Seabrook nuclear design was reviewed according to Section 4.3 of the Standard Review Plan (NUREG-0800). All areas of review and review procedures from that section have been followed either for this reactor or for previous similar reactors (e.g., Trojan and McGuire-1) or for Topical Report reviews.

The applicant has described the computer programs and calculational techniques used to predict the nuclear characteristics of the reactor design and has provided examples to demonstrate the ability of the analyses to predict reactivity and physics characteristics of the Seabrook Units 1 and 2 plants.

To allow for changes of reactivity due to reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, a significant amount of excess reactivity is designed into the core. The applicant has provided substantial information relating to core reactivity balances for the first cycle and has shown that means have been incorporated into the design to control excess reactivity at all times. The applicant has shown that sufficient control rod worth is available to make the reactor subcritical with an effective multiplication factor no greater than 0.987 in the hot condition at any time during the cycle with the most reactive control rod stuck in the fully withdrawn position. On the basis of our review, we conclude that the applicant's assessment of reactivity control requirements over the first core cycle is suitably conservative, and that adequate negative worth has been provided by the control system to assure shutdown capability. Reactivity control requirements will be reviewed for additional cycles as this information becomes availabe. We also conclude that nuclear design bases, features, and limits have been established in conformance with the requirements of General Design Criteria 10, 11, 12, 13, 20, 25, 26, 27, and 28.

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This conclusion is based on the following:

1. The applicant has met the requirements of GDC 11 with respect to prompt inherent nuclear feedback characteristics in the power operating range by:

   a. Calculating a negative Doppler coefficient of reactivity, and

   b. Using calculational methods that have been found acceptable.

   The staff has reviewed the Doppler reactivity coefficients in this case and found them to be suitably conservative.

2. The applicant has met the requirements of GDC 12 with respect to power oscillations which could result in conditions exceeding specified acceptable fuel design limits by:

   a. Showing that such power oscillations are not possible and/or can be easily detected and thereby remedied, and

   b. Using calculational methods that have been found acceptable.

3. The applicant has met the requirements of GDC 13 with respect to provisions of instrumentation and controls to monitor variables and systems that can affect the fission process by:

   a. Providing instrumentation and systems to monitor the core power distribution, control rod positions and patterns, and other process variables such as temperature and pressure, and

   b. Providing suitable alarms and/or control room indications for these monitored variables.

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4. The applicant has met the requirements of GDC 26 with respect to provision of two independent reactivity control systems of different designs by:

   a. Having a system that can reliably control anticipated operational occurrences,

   b. Having a system that can hold the core subcritical under cold conditions, and

   c. Having a system that can control planned, normal power changes.

5. The applicant has met the requirements of GDC 27 with respect to reactivity control systems that have a combined capability in conjunction with poison addition by the emergency core cooling system of reliably controlling reactivity changes under postulated accident conditions by:

   a. Providing a movable control rod system and a liquid poison system, and

   b. Performing calculations to demonstrate that the core has sufficient shutdown margin with the highest-worth stuck rod.

6. The applicant has met the requirements of GDC 28 with respect to postulated reactivity accidents by (reviewed under Section 15.4.8):

   a. Meeting the regulatory position in Regulatory Guide 1.77,

   b. Meeting the criteria on the capability to cool the core, and

   c. Using calculational methods that have been found acceptable for reactivity insertion accidents.

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7. The applicant has met the requirements of GDC 10, 20, and 25 with respect to specified acceptable fuel design limits by providing analyses demonstrating:

   a. That normal operation, including the effects of anticipated operational occurrences, have met fuel design criteria,

   b. That the automatic initiation of the reactivity control system assures that fuel design criteria are not exceeded as a result of anticipated operational occurrences and assures the automatic operation of systems and components important to safety under accident conditions, and

   c. That no single malfunction of the reactivity control system causes violation of the fuel design limits.

REFERENCES FOR SECTION 4.3

D. F. Ross (NRC), Memorandum to D. B. Vassallo, "Review of Topical Report WCAP-8846," August 26, 1977.

WCAP-8385, T. Morita, et al., "Power Distribution Control and Load Follow Procedure, September 1974.

WCAP-8846, "Hybrid B_4C Absorber Control Rod Evaluation Report," September 1976.

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4.4 Thermal-Hydraulic Design

4.4.1 Design Criteria

The performance and safety criteria for the Seabrook units, as stated in Section 4.4.1 of the FSAR, are:

(1) "Fuel damage (defined as penetration of the fission product barrier, i.e. the fuel rod clad) is not expected during normal operation and operational transients (Condition I) or any transient conditions arising from faults of moderate frequency (Condition II). It is not possible, however, to preclude a very small number of rod failures. These will be within the capability of the plant design bases."

(2) "The reactor can be brought to a safe state following a Condition III event with only a small fraction of fuel rods damaged (see above definition) although sufficient fuel damage might occur to preclude resumption of operation without considerable outage time."

(3) "The reactor can be brought to a safe state and the core can be kept subcritical with acceptable heat transfer geometry following transients arising from Condition IV events."

4.4.2 Design Bases

The performance and safety criteria listed above are implemented through the following design bases.

4.4.2.1 Departure from Nucleate Boiling

The margin to departure from nucleate boiling at any point in the core is expressed in terms of the departure from nucleate boiling ratio (DNBR). The DNBR is defined as the ratio of the heat flux required to produce departure from nucleate boiling at the calculated local conditions to the actual heat flux.

The thermal-hydraulic design basis, as stated in Section 4.4.1 of the Seabrook FSAR, for the prevention of departure from nucleate boiling is as follows:

> "Departure from nucleate boiling (DNB) will not occur on at least 95 percent of the limiting fuel rods during normal operation and operational transients and any transient conditions arising from faults of moderate frequency (Condition I and II events) at a 95 percent confidence level."

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4.4.2.2 Fuel Temperature

The fuel temperature design basis given in Section 4.4.1.2 is:

> "During modes of operation associated with Condition I and II events, there is at least a 95% probability that the peak KW/FT fuel rods will not exceed the UO_2 melting temperature at the 95 percent confidence level."

This design basis is evaluated in the Safety Evaluation Report. Section 4.2, "Fuel System Design."

4.4.2.3 Core Flow

Section 4.4.1.3 of the FSAR has the following core flow design basis.

> "A minimum of 94.2 percent of the thermal flow rate will pass through the fuel rod region and be effective for fuel rod cooling."

4.4.2.4 Hydrodynamic Stability

The hydrodynamic stability design basis given in Section 4.4.1.4 is:

> "Modes of operation associated with Conditions I and II events shall not lead to hydrodynamic instability."

4.4.3 Thermal-Hydraulic Design Methodology

4.4.3.1 Departure from Nucleate Boiling

The thermal-hydraulic design analyses were performed using the W-3 critical heat flux (CHF) correlation in conjunction with the THINC-IV computer code.

The W-3 correlation was developed from data obtained from experiments conducted with fluid flowing inside single heated tubes. As test procedures progressed to the use of rod bundles instead of tubes, the correlation was modified to include the effects of "R" mixing vane grids and axially nonuniform power distributions.

The applicant has proposed a minimum DNBR of 1.30 to ensure that there is a 95 percent probability at a 95 percent confidence level that DNB will not occur on the limiting fuel rod. The proposed use of the W-3 CHF correlation with a minimum DNBR of 1.30 has been previously approved by the staff.

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THINC-IV is an open channel computer code which determines the coolant density, mass velocity enthalpy, vapor void, static pressure, and DNBR distributions along parallel flow channels within a reactor core. A description of the THINC-IV code is given in WCAP-7956 and the design application of the code is described in WCAP-8054. Both WCAP-7956 and WCAP-8054 have been reviewed and approved by the staff (Ross, April 10, 1978).

The staff has reviewed and approved the use of a uniform core exit pressure gradient in the thermal-hydraulic design of Westinghouse reactors (NUREG-0847). This approval is based on THINC-IV analyses which showed that the effects of a core exit pressure distribution on minimum DNBR are negligible, (Eicheldinger, November 2, 1977). The staff has also reviewed and approved the Westinghouse design approach used to bound future cycles (NUREG-0847).

Based on the information given above the staff concludes that the DNB design methodology used for the Seabrook units is acceptable.

4.4.3.2 Core Flow

The core flow design basis requires that the minimum flow which will pass through the fuel rod region and be effective for fuel rod cooling is 94.2% of the primary coolant flow rate or 134.0×10^6 lbm/hr. The remainder of flow, called bypass flow, will be ineffective for cooling since it will take the following bypass paths:

(1) flow through the spray nozzles into the upper head;

(2) flow into the rod cluster control rod guide thimbles;

(3) leakage from the vessel inlet nozzle directly to the vessel outlet nozzle;

(4) flow between the baffle and barrel; and

(5) flow in the gaps between the fuel assemblies.

The amount of bypass flow is determined by a series of hydraulic resistance calculations on the core and vessel internals and verified by model flow tests. Since the amount of bypass flow is consistent with approved plants of similar design, the staff concludes that the core bypass flow used in the design analysis, 5.8%, is acceptable.

4.4.3.3 Hydrodynamic Instability

For steady-state heated flow in parallel channels, the potential for hydrodynamic instability exits.

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The applicant stated that the core was stable because Westinghouse reactors will not experience any Ledinegg instability over Condition I and II events and open channel configurations, which are a feature of Westinghouse PWRs, are more stable than closed channel configurations. This was shown by flow stability experiments conducted by Kakac, et.al. Additional experiments conducted by Kao, Morgan, and Parker, on closed parallel channels at pressures up to 2200 psia, showed that no flow oscillations could be induced above 1200 psia. Finally, data from numerous rod bundle tests performed over a wide range of operating conditions snow no evidence of premature DNB or of inconsistent data which might be indicative of flow instabilities in the rod bundles.

The staff is conducting a generic study of the hydrodynamic stability of light water reactors. Limitations to the thermal-Hydraulic design resulting from the staff study will be compensated for by appropriate operating restrictions; however, none are anticipated.

In the interim, the staff concludes that past operating experience, flow stability experience, and the inherent thermal-hydraulic design of Westinghouse PWRs serve as a basis for issuance of an operating license.

4.4.4 Fuel Rod Bowing

A significant parameter which affects the thermal hydraulic design of the core is rod-to-rod bowing within fuel assemblies. The Westinghouse methods for predicting the effects of rod bow on DNB, WCAP-8691, Revision 1, are under review by the staff. Therefore, the magnitude of rod bow as a function of burnup was evaluated based on interim methods which have been previously approved by the staff (Ross and Eisenhut, December 16, 1976; Ross and Eisenhut, February 16, 1977: Meyer, March 2, 1978). The resultant reduction in the departure from nucleate boiling ratio due to rod bow is given in Table 4.4.1.

Table 4.4.1 Rod bow penalties

Burnup (MWD/MTU)	DNBR penalty (%)
0	0
3500	0
5000	0
10000	2.15
15000	4.64
20000	6.74
25000	8.59
30000	10.27
35000	13.07
40000	19.09

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Prior to issuance of the Technical Specifications, the staff will ensure that the thermal margin reductions given above have been accommodated using an acceptable method.

For Plants designed by Westinghouse, the staff has approved the following generic margins, (Table 4.4-2), which may be used to offset the reduction in DNBR due to rod bowing.

Plant-specific margins which could be available are:

(1) the Technical Specification minimum flow rate is greater than the design flow rate:

(2) the Technical Specification maximum T_{ave} is less than the design T_{ave}; and

(3) the trip setpoints are more limiting than the thermal-hydraulic analysis indicates.

The applicant should insert into the basis of the Technical Specification any of the generic or plant-specific margins that may be used to offset the reduction in DNBR due to rod bowing.

Table 4.4.2 Generic margins

Margin	% reduction in rod bow penalties
The use of a design minimum DNBR of 1.30 instead of the 95/95 DNBR limit of 1.28.	1.6
A reduction in fuel rod pitch for the hot channel analysis.	1.7
The use of a Thermal Diffusion Coefficient (TDC) of 0.038 instead of a TDC of 0.051.	1.2
The addition of an extra grid in the design of the Westinghouse 17 X 17 fuel assembly relative to the 15 X 15 fuel design.	2.9
The use of a 0.86 multiplier on the modified spacer factor (F's) of the W-3 correlation instead of a 0.88 multiplier.	1.7
Total generic margin which may be claimed.	9.1

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4.4.5 Instrumentation

4.4.5.1 Detection of Crud

Crud deposits in the core and an associated change in core pressure drop and flow have been observed in some pressurized water reactors, not of Westinghouse design. The staff requested that the applicant provide a description of the procedures to detect flow degradation due to crud buildup. The applicant responded that except for steam generator tube plugging, there have been no reports of a significant flow reduction in a relatively short period of time at any Westinghouse plant.

The staff will ensure that the Seabrook Technical Specifications contain the requirement that the actual reactor coolant system (RCS) flow rate be verified to be greater than or equal to the minimum design flow rate plus uncertainties at least once per 12 hours. In addition, we will ensure that the applicant perform a channel calibration at least once per 18 months.

With inclusion of the above requirements into the Technical Specifications, the staff concludes that the applicant response to our concerns on crud deposits is acceptable.

4.4.[illegible] ow Measurement Uncertainties

In response to our question on crud deposits in the core, the applicant provided a description of the calorimetric flow measurement system which will be used by the Seabrook units. Individual coolant loop flow is calculated using the steam generator heat output, adjusted for pump heat input, and the enthalpy rise of the coolant. The total flow rate is calculated by summing the individual flows and is then used to calibrate the flow measured by the elbow taps located in each coolant loop.

By using a statistical error combination technique, the applicant has proposed an uncertainty of $\pm$ 1.5% on the calorimetric measurement. The staff is presently reviewing the 1.5% calorimetric measurement uncertainty on a generic basis (Rahe, March 31, 1982). Although the calorimetric measurement system is an acceptable means of measuring reactor flow, the staff has raised a number of concerns on the values of the components comprising the calorimetric flow measurement uncertainty and how drift associated with the elbow taps is accounted for between channel calibrations. Prior to issuance of our formal SER the applicant should supply the same information for the Seabrook units or reference the generic submittal.

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4.4.5.3 Loose Parts Monitoring Systems

To date, the applicant has not procured the loose parts monitoring system (LPMS) to be installed at the Seabrook units. However, the applicant has committed to provide a system which will comply with Regulatory Guide 1.133 and to provide an equipment description by June 1982. This submittal should include a committment to supply a final design report on the LPMS which contains a description of how the system is in conformance with Sections C.1, C.2, C.3, C.4, C.5, and C.6 of Regulatory Guide 1.133. We will address the acceptability of the LPMS after we have reviewed the June submittal.

4.4.5.4 ICC Instrumentation

The applicant did not provide information in response to NUREG-0737 Item II.F.2, "Instrumentation for Detection of Inadequate Core Cooling". Therefore, the staff will require that the applicant provide the documentation itemized in Item II.F.2 of NUREG-0737. Acceptable documentation must be provided and approved by the staff prior to issuance of an operating license.

4.4.6 Thermal-Hydraulic Comparison

The thermal-hydraulic design parameters for the Seabrook units are compared to values for the McGuire and Trojan plants in Table 4.4.3. All three plants were designed using the W-3 correlation and the THINC-IV computer code.

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Table 4.4.3 Reactor design comparison

		Seabrook Units 1 & 2	Trojan SER	McGuire SER
I.	Performance characteristics:			
	Reactor core heat output (MWT)	3,411	3,411	3,411
	System pressure, psia	2,250	2,250	2,250
	Minimum departure from nucleate boiling ration			
	Typical cell	2.06	2.04	2.08
	Thimble cell	1.72	1.71	1.79
	Minimum DNBR	1.30	1.30	1.30
	Critical heat flux correlation	W-3	W-3	W-3
II.	Coolant flow:			
	Total flow rate (10^6 lb/hr)	142.1	132.7	140.3
	Effective flow rate for heat transfer (10^6 lb/hr)	134.0	125.7	134.0
	Average velocity along fuel rods (ft/s)	16.7	15.7	15.6
	Effective core flow area (ft^2)	51.1	51.1	51.1
III.	Coolant temperature, °F			
	Nominal reactor inlet	558.8	552.7	558.1
	Average rise in core	59.4	66.9	62.7
	Pressure drop across core (psi)	27.9± 5.6		27.9± 5.6
5.6				
IV.	Heat transfer, 100 percent power:			
	Active heat transfer surface area (ft^2)	59,700	59,700	59,700
	Average heat flux (BTU/hr-ft^2)	189,800	189,800	189,800
	Maximum heat flux (Btu/hr-ft^2)	440,300	493,500	440,500
	Average linear heat rate (kw/ft)	5.44	5.44	5.44
	Peak linear power resulting from overpower transients and operator errors (kW/ft)	12.6	13.6	12.2

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The major differences between the Trojan and Seabrook designs are an increase in the flow rates, an increase in the inlet temperature, and a decrease in the average temperature rise across the core. The increase in inlet temperature results in a decrease in thermal margin; however, the increase in flow rates and the decrease in the temperature rise compensate for the inlet temperature effect. Therefore, the net result is an increase in the DNBR for Seabrook and an increase in the thermal margin.

The differences between McGuire and Seabrook are negligible.

The comparability of the Seabrook design to that of Trojan and McGuire support the conclusion that the Seabrook thermal-hydraulic design is acceptable.

4.4.7 N-1 Loop Operation

N-1 loop operation is when one reactor coolant loop is out of service. Thus only three loops are available to supply coolant to the reactor core.

In response to a staff question, the applicant stated that the initial operating license application did not include provisions for three loop operation but that they do intend to submit the necessary information to support N-1 loop operation in the future.

The staff will insure that the Technical Specifications contain the appropriate provisions to prohibit N-1 loop operation until the applicant submits the necessary information which justifies this mode of operation.

4.4.8 Summary and Conclusion

The thermal-hydraulic design of the Seabrook units was reviewed. The acceptance criteria used as a basis for our evaluation are set forth in Section 4.4.II, "Thermal and Hydraulic Design Acceptance Criteria," of the Standard Review Plan (SRP), NUREG-0800. The scope of the review included the design criteria, core design, and the steady-state analysis of the core thermal-hydraulic performance. The review concentrated on the differences between the proposed core design and those designs which have been previously reviewed and found acceptable by the staff.

Based on our review, the staff concludes that the reactor core has been designed with appropriate margin to ensure that acceptable fuel design limits are not exceeded during steady-state operation or anticipated operational occurrences. Therefore, the thermal-hydraulic design of the core meets the requirements of General Design Criterian 10 of 10 CFR Part 50, Appendix A. The staff also concludes that with the exceptions of the LPMS requirement and the documentation requirements of Item II.F.2 of NUREG-0737, the applicants FSAR is in conformance with SRP Section 4.4. Prior to issuance of our formal SER the applicant should supply the information required for the LPMS, supply the documentation required by Item II.F.2 of NUREG-0737, and provide the flow measurement information discussed in Section 4.4.5.2 of this report.

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Finally, the staff will prohibit operation in the N-1 mode by including appropriate provisions in the Technical Specifications unless that mode of operation is evaluated and justified prior to issue of the license.

4.4.8 References

4.4.8.1 NRC Reports

NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1981.

NUREG-0847 "SER Related to the Operation of Watts Bar Nuclear Plant, Units 1 and 2," April, 1982.

4.4.8.2 Westinghouse Reports

WCAP-7956,"THINC-IV -- An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores," June 1973.

WCAP-8054, L. E. Hochreiter, "Application of the THINC-IV Program to PWR Design," September 1973.

4.4.8.3 Other Reference

C. Eicheldinger (Westinghouse) letter to John F. Stolz (NRC), Untitled letter on core exit pressure gradients, November 2, 1977.

R. O. Meyer (NRC) memorandum to D. F. Ross, "Revised Coefficients for Interim Rod Bowing Analysis," March 2, 1978 (Proprietary Information, not publicly available).

E. P. Rahe, Jr. (Westinghouse) letter to Carl H. Berlinger (NRC), Untitled letter on instrumentation uncertainties, March 31, 1982.

D. F. Ross and D. G. Eisenhut (NRC) memorandum to D. B. Vassallo and K. R. Goller, "Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing in Thermal Margin Calculations for Light Water Reactors," December 3, 1976.

D. F. Ross and D. G. Eisenhut (NRC) memorandum to D. B. Vassallo and K. R. Goller, "Revised Interim Safety Evaluation Report on the effects of Fuel Rod Bowing in Thermal Margin Calculations for Light Water Reactors," February 16, 1977.

D. F. Ross (NRC) memorandum to D. B. Vassallo (NRC) "Topical Report Evaluations For; WCAD-7956,, WCAD-8054, WCAD-8762, WCAD-8567/8568," April 10, 1978.

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4.6 Functional Design of Reactivity Control Systems

The functional design of reactivity control systems was reviewed in accordance with Section 4.6 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the functional design of reactivity control systems with respect to the applicable regulations of 10 CFR 50.

The functional designs of the reactivity control systems for the facility have been reviewed to confirm that they meet the various reactivity control conditions for all modes of operation. These are:

1. The capability to operate in the unrodded, critical, full-power mode throughout plant life.

2. The capability to vary power level from full power to hot shutdown and assure control of power distributions within acceptable limits at any power level.

3. The capability to shut down the reactor in a manner sufficient to mitigate the effects of postulated events discussed in Section 15 of this SER.

The control rod drive system (CRDS), the safety injection system (SIS), and the chemical and volume control system (CVCS) constitute the reactivity control systems.

The CRDS is composed of control rod drive mechanisms (CRDMs) to which the rod cluster control assemblies (RCCAs) are attached. The CRDM is a magnetically operated jack. The magnetic jack is an arrangement of three electromagnets which are energized in a controlled sequence to insert or withdraw the RCCAs in discrete steps. The RCCAs are divided into two categories: control and shutdown.

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The control category RCCAs may be automatically inserted or withdrawn to compensate for changes in reactivity associated with power level changes and power distribution, variations in moderator temperature, or changes in boron concentration. The shutdown category RCCAs, which are fully withdrawn during power operations, are used solely to insert large amounts of negative reactivity to shut down the reactor. Refer to Section 4.3 of this SER for further discussions on these features.

The RCCAs are the primary shutdown mechanism for normal operation, accidents, and transients. They insert automatically upon a reactor trip signal. Concentrated boric acid solution is injected by the SIS in the event of a loss of coolant accident, steamline break, loss of normal feedwater flow, steam generator tube rupture, or RCCA ejection, thereby complying with the requirements of General Design Criterion 29, "Protection Against Anticipated Operational Occurrences."

Failure of electrical power to an RCCA will result in the insertion of that assembly as will shearing of the connection between the rod cluster control assembly and control rod drive mechanism. Single failure of a rod cluster control assembly is considered in transient and accident analyses which include the most reactive rod cluster control assembly stuck outside the core. Analysis of accidental withdrawal of a rod cluster control assembly is found to have acceptable results. This conforms to the requirements of General Design Criteria 23, "Protection System Failure Modes," and 25, "Protection System Requirements for Reactivity Control Malfunctions."

The CRDM operating coils are cooled by a flow of containment air provided by four fans dedicated to that duty only. High-temperature alarms in the cooling air outlet are provided to alert the operator to inadequate CRDM cooling. An alarm is provided should a CRDM cooling fan motor trip or if an insufficient number of cooling fans are running. If all cooling capability were lost, the reactor could be tripped and safely shut down. The cooling function does not influence the safety of the CRDMs or their ability to trip the reactor when required.

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The SIS is automatically actuated to inject borated water into the reactor coolant system (RCS) upon receipt of a safety injection actuation signal (SIAS). High pressure safety injection is accomplished by the centrifugal charging pumps which take suction from the refueling water storage tank (RWST) and inject the boric acid solution into the reactor coolant system via the boron injection tanks (BITs). The SIS is further discussed in Section 6.3 of this SER.

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The CVCS is designed to accommodate slow or long-term reactivity changes such as those caused by changes in reactor coolant temperature between cold shutdown and operating temperature, burnup of fuel and burnable poisons and xenon transients. The CVCS is used to control reactivity by adjusting the dissolved boron concentration in the reactor coolant system. The boron concentration is controlled to allow maintaining the RCCA control bank within a prescribed band of travel to compensate for reactivity changes associated with variations in coolant temperature, core burnup, and xenon concentration, and to provide shutdown margin for maintenance and refueling operations or emergencies. A portion of the CVCS (the charging pumps, the boric acid transfer pumps, and the boric acid makeup tanks) injects the desired concentration of boric acid solution into the reactor coolant system for reactivity control. During normal operation the boric acid concentration in the reactor coolant system is controlled by the CVCS boron thermal regeneration subsystem and by the reactor makeup control subsystem. The boron thermal regeneration subsystem is primarily designed to compensate for xenon transients occurring during load follow but can also be used for diluting the primary coolant during startup.

The reactor makeup control subsystem is designed to provide a manually preselected makeup boron concentration to the charging pumps, by blending reactor makeup water and boric acid solution. Under normal plant conditions the subsystem operates in the "automatic makeup" mode to compensate for minor leakage of reactor coolant without causing significant changes in primary coolant boron concentration. Operation in the dilution mode permits addition of a preselected quantity of reactor makeup water and is performed to decrease the boron concentration during startup and to compensate for fuel burnup. Operation in the boration mode permits the addition of a preselected quantity of concentrated boric acid solution for hot shutdown, and to compensate for xenon decay.

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In summary, the concentration of boron in the reactor coolant system is changed manually under the following operating conditions:

1. Startup - boron concentration decreased to compensate for moderator temperature and power increase

2. Load follow - boron concentration increased or decreased to compensate for xenon transients following load changes

3. Fuel burnup - boron concentration decreased to compensate for burnup

4. Cold shutdown - boron concentration increased to compensate for increased moderator density due to cooldown and xenon delay.

The CVCS is discussed further in Section 9.3.4 of this SER.

Soluble poison concentration is used to control slow operating reactivity changes. If necessary, RCCA movement can also be used to accommodate such changes, but assembly insertion is used mainly to control anticipated operational occurrences even with a single malfunction, such as a stuck rod. In either case, fuel design limits are not exceeded. The soluble poison control is capable of maintaining the core subcritical under conditions of cold shutdown, which conforms to the requirements of General Design Criterion 26, "Reactivity Control System Redundancy and Capability."

The reactivity control systems, including the addition of concentrated boric acid solution by the SIS, are capable of controlling anticipated operational changes, transients, and accidents. For further information on the performance of the SIS and charging and borating portion of the CVCS with respect to loss-of-coolant accidents (LOCAs) and major secondary system pipe breaks, refer to Section 6.3 of this SER. All accidents are calculated with the assumption that the most reactive RCCA is stuck out and cannot be inserted, which complies with the requirements of General Design Criterion 27, "Combined Reactivity Control Systems' Capability."

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Compliance with the requirements of General Design Criterion 28, "Reactivity Limits," is discussed in Sections 4.3 and 15 of this SER.

Based on our review, we conclude that the reactivity control system functional design meets the requirements of General Design Criteria 23, 25, 26, 27, 28, and 29 with respect to its fail-safe design, malfunction-protection design, redundancy and capability, combined systems' capability, reactivity limits, and protection against anticipated operational occurrences, and is therefore acceptable.

The control rod drive system meets the acceptance criteria of SRP Section 4.6.

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5.4.11 Pressurizer Relief Tank (Pressurizer Relief Discharge System)

The pressurizer relief tank was reviewed in accordance with Section 5.4.11 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Area of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the pressurizer relief discharge system with respect to the applicable regulation of 10 Part 50.

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The pressurizer relief discharge system consists of the pressurizer relief tank, the discharge piping from the pressurizer relief and safety valves, the relief tank internal spray header, the tank nitrogen supply, the vent to containment, and the drain to the equipment and floor drain system. The system is nonsafety related (Quality Group D, nonseismic Category I) and is not part of the reactor coolant pressure boundary since all of its components are downstream of the reactor coolant system safety and relief valves. Therefore, its failure would not affect the integrity of the reactor coolant pressure boundary.

The pressurizer relief tank is sized to absorb the energy content of 110% of the full-power pressurizer steam volume through the primary relief and safety valves. Other relief valves which discharge to the pressurizer relief tank are from the residual heat removal system and from the chemical and volume control system. Releases from these sources are less than the design basis release from the pressurizer. The internal spray and bottom drain on the pressurizer relief tank are used to cool the water within the tank. A nitrogen blanket is also provided in the tank to permit expansion of entering steam and to control the tank internal atmosphere. If a discharge exceeding the design basis should occur, the rupture discs on the tank would pass the discharge through the tank to the containment.

The contents of the tank can be drained to the primary drain tank in the equipment and floor drain system via the reactor coolant drain tank pumps. The rupture discs on the pressurizer relief tank have a capacity equal to or greater than the combined capacity of the pressurizer safety valves. The tank and the rupture disc holders are designed for full vacuum to prevent collapse if the contents cool following a discharge without nitrogen being added. The pressurizer relief tank is provided with instrumentation to indicate pressure and temperature and alarms for high or low level, high pressure, and temperature.

The tank is separated from safety-related equipment so that its failure would not compromise the capability to safely shut down the plant, and further, possible rupture disc fragments do not present a missile hazard when the disc ruptures. Thus, the requirements of General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena," and 4, "Environmental and Missile

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Design Bases," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Positions C.2 and C.3, are satisfied.

Based on our review, we conclude that the pressurizer relief discharge system meets the requirements of General Design Criteria 2 and 4 with respect to the need for protection against natural phenomena and internal missile protection as its failure does not affect safety system functions. It meets the guidelines of Regulatory Guide 1.29, Positions C.2 and C.3, concerning its seismic classification and is therefore acceptable. The pressurizer relief tank meets the acceptance criteria of SRP Section 5.4.11.

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6.1.1 Post Accident Emergency Cooling Water Chemistry

I. INTRODUCTION

This review is related to providing and maintaining the proper pH of the containment sump water following a design basis accident to reduce the likelihood of stress corrosion cracking of austenitic stainless steel.

During the containment spray injection phase, the applicant will educt 20 weight percent sodium hydroxide into the containment spray solution which is supplied from the refueling water storage tank at a nominal concentration of 2000 ppm boron as boric acid. The injection phase containment spray pH versus time is discussed in SRP 6.5.2.

During the containment spray recirculation phase a final pH of greater than 7.0 will be achieved in

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the sump once the borated water has thoroughly mixed with the educted sodium hydroxide.

II. EVALUATION

The post-accident cooling water chemistry has been reviewed in accordance with Section 6.1.1 of the Standard Review Plan (NUREG-0800, Revision 2).

In the event of a LOCA, a sufficient quantity of NaOH is used in the containment spray to increase the pH of liquids in the containment sump to above 7.0 after thorough mixing of the containment spray solution and containment sump water has occurred.

We evaluated the pH of the containment sump water following mixing in the containment sump with the educted sodium hydroxide. We verified by independent calculations that sufficient sodium hydroxide is available to raise the containment sump water pH to greater than 7. This is consistent with the minimum pH of >7.0 required by BTP-MTEB 6-1 of SRP Section 6.1.1 to reduce the probability of stress-corrosion cracking of austenitic stainless steel components.

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The applicant has taken exception to the SRP recommendation to provide an inert cover gas on the sodium hydroxide storage tank to prevent deterioration of the chemicals. However, the applicant has not provided the information necessary to complete our review. We will report the resolution of this item after we receive the necessary information.

6.1.2 Organic Materials

I. INTRODUCTION

This evaluation is conducted to verify that protective coatings applied inside containment meet the testing requirements of ANSI 101.2 (1972) and the quality assurance guidelines of Regulatory Guide 1.54. Compliance with these requirements provides assurance that the protective coatings will not fail under DBA conditions and generate significant quantities of solid debris or combustible gas which could complicate the accident conditions.

The protective coatings used inside containment have been demonstrated to withstand the design basis accident conditions (consistent with the requirements

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of ANSI 101.2 (1972). Additionally, the protective coatings have been applied in accordance with quality assurance requirements of Regulatory Guide 1.54, except in certain cases where non-LOCA qualified coatings are used on small components with limited surface area.

The non-qualified coatings have been quantified by the applicant. The control of combustible gases that can potentially be generated from the organic materials and from qualified and unqualified paints is reviewed under Section 6.2.5. The consequences of solid debris that can potentially be formed from unqualified paints are reviewed under Section 6.2.2.

II. EVALUATION AND FINDINGS

The organic materials inside containment have been reviewed in accordance with Section 6.1.2 of the Standard Review Plan (NUREG-0800, Revision 2).

Based on the applicant's compliance with the applicable Regulatory Guide and ANSI Standard, we conclude that the protective coating systems and their applications are acceptable and meet the requirements of Appendix B to 10 CFR Part 50. This conclusion is

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based on the applicant having met the quality assurance requirements of Appendix B to 10 CFR Part 50 since the coating systems and their applications meet the positions to Regulatory Guide 1.54, "Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants" and the requirements of ANSI N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities." These measures demonstrate their suitability to withstand a postulated design basis accident (DBA) environment.

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6.4 Control Room Habitability

The requirements for the protection of control room personnel against radioactive and toxic hazards are specified in GDC-4, "Environmental and Missile Design Bases", GDC-5, "Sharing of Structures, Systems and Components", and GDC-19, "Control Room". The applicant's design to meet the requirements includes shielding and the provision of a control room emergency ventilation and filtration system.

The staff review of control room habitability was performed in accordance with Standard Review Plan Section 6.4, "Control Room Habitability System" of NUREG-0800, Regulatory Guide 1.78, "Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release," and Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release". In addition, the review also satisfied the guidance of Task Action Plan Item III.D.3.4, "Control Room Habitability" of NUREG-0737. No review exceptions were applied by the staff for the control rooms at Seabrook.

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Two remote air intakes are used to furnish makeup air to both control rooms. The locations were selected to preclude both inlets from being simultaneously susceptible to either airborne radioactivity or toxic chemicals. The eastern makeup air intake, which is associated with Unit 1 but which can be used by either or both Units 1 and 2 at any time, is located 350 feet northeast from the center of the Unit 1 containment structure. Similarly, the western intake is located 350 feet northwest of the center of the Unit 2 containment structure.

The placement of the inlets has been designed so that at least one of the inlets will be free of radioactive contamination. One inlet will provide sufficient makeup air to maintain both control rooms at a positive pressure.

Meteorological instruments are provided to allow the operators to maintain a continuous supply of makeup air. Self-contained breathing apparatus is supplied to provide breathing protection in the presence of toxic chemicals.

Following a LOCA, meteorological information in the control room will indicate the wind direction and the affected intake can be manually isolated by the operator, or a high radiation signal will automatically shutdown the makeup air intake fans and close their discharge dampers. Once isolated, the makeup air line from the isolated intake can be purged by operation of a unit fan drawing from the opposite intake. Controls for the four fans are located in the control rooms of each unit.

The HEPA and carbon filtering system is automatically initiated by the safety injection signal, or can be manually initiated by the operator from the control room at any time to prevent the build up of airborne particulates and radioactive iodine in the control room complex. Use of the system is beneficial in the removal of smoke and other toxic chemicals.

The licensee has designed the system to eliminate potential adverse interactions between the control room ventilation zones and adjacent zones that could transfer toxic or radioactive gases. The ventilation systems for the control room complex are entirely

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contained within the complex and no ducts or ventilation piping from any other zone penetrate the control room envelope.

The dual intake air fans and their associated discharge dampers are powered from the emergency electrical distribution system, to ensure operating power during all modes of operation. The instrumentation systems monitoring the air intakes are fed from the vital busses. This instrumentation includes radiation monitoring and smoke detection.

When only one remote makeup air intake can be used, 550 to 800 cfm makeup air will be available, 390 cfm of which will be exfiltrated and the balance of 160 to 410 cfm which will be exhausted. Because the diesel generator building and the control building are seismic Category I structures, earthquakes up to and including SSE event will not provide an exfiltration path for control room complex air that will negate the ability to maintain a positive pressure. The intake air equipment and welded piping to the control room complex and between the control rooms is seismic Category I.

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The staff finds that the control room habitability system will provide safe, habitable conditions within the control rooms under both normal and accident radiation and toxic gas conditions, including loss-of-coolant accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of an accident. Therefore, the design meets the criteria identified in GDC-4, GDC-5 and GDC-19, as well as that contained in Item No. III.D.3.4, "Control Room Habitability" of NUREG-0737, and is acceptable.

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9.3.2 Process Sampling System

I. INTRODUCTION

Process sampling is accomplished by four sub systems: Reactor coolant sampling, steam generator blowdown sampling, auxiliary system sampling and secondary steam and water sampling. These four sub systems combined, provide an overall primary and secondary sampling capability. The primary sampling systems are designed to collect water and gaseous samples contained in the reactor coolant system and associated auxiliary system process steams during all normal modes of operation. The secondary sampling systems are designed to collect water and steam from the secondary cycle.

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Continuous secondary samples are analyzed automatically for such parameters as pH, dissolved oxygen and conductivity. Additionally grab samples are obtained for confirmatory analysis of other chemical species. Provisions are made to assure that representative samples are obtained from well mixed streams or volumes of effluent by the selection of proper sampling equipment and location of sampling points as well as proper sampling procedures. The primary sample lines penetrating the containment are each equipped with two normally closed seismic category 1 pheumatically operated isolation valves which close on a containment isolation actuation signal.

II. EVALUATION AND FINDINGS

The process sampling system has been reviewed in accordance with Section 9.3.2 of the Standard Review Plan (NUREG-0800, Revision 2).

The process sampling system includes piping, valves, heat exchangers, and other components associated with the system from the point of sampling station, or local sampling point. Our review included the provisions proposed to sample all principal fluid process streams

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associated with plant operation and the applicant's proposed design of these systems including the location of sampling points, as shown on piping and instrumentation diagrams.

We determined that the proposed process sampling system meets (1) the requirements of GDC 13 in Appendix A to 10 CFR Part 50, by sampling the reactor coolant, the safety injection tanks, the refueling water storage tank, the boric acid mix tank, and the boron injection tank for boron concentration, which can affect the fission process for normal operation, anticipated operational occurrences, and accident conditions; (2) the requirements of GDC 14, by sampling the reactor coolant and the secondary coolant for chemical impurities to ensure that the reactor coolant pressure boundary will have a low probability of abnormal leakage, rapidly propagating failure, and gross rupture; (3) the requirements of GDC 26 by sampling the reactor coolant, the refueling water storage tank, and the boric acid mix tank for boron concentrations for controlling the rate of reactivity changes; (4) the requirements of

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GDC 63 by sampling the fuel pool and the gaseous radwaste storage tank for radioactivity to detect conditions that may result in excessive radiation levels; and (5) the requirements of GDC 64 by sampling the reactor coolant, the pressurizer, the steam generator blowdown, the sump inside containment, the containment atmosphere and the gaseous radwaste storage tank, for radioactivity that may be released from normal operations, including anticipated operational occurrences and from postulated accidents.

We further determined that the proposed process sampling system meets (a) the standards of ANSI N13.1-1969 for obtaining airborne radioactive samples; (b) the requirements of 10 CFR Part 20.1(c) and regulatory positions 2.d(2), 2.f(3), 2.g.(8) and 2.i.(6) of Regulatory Guide 8.8 revision 3, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low as Is Reasonably Achievable," to maintain radiation exposures to as low as is reasonably achieveable, by providing (1) ventilation systems and gaseous radwaste

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treatment system to contain airborne radioactive materials; (2) liquid radwaste treatment system to contain radioactive material in fluids; (3) spent fuel pool cleanup system to remove radioactive contaminants in the spent fuel pool water; and (4) remotely operated containment isolation valves to limit reactor coolant loss in the event of rupture of a sampling line; (c) the requirements of GDC 60 to control the release of radioactive materials to the environment by providing isolation valves that will fail in the closed position; and (d) regulatory positions C.1, C.2, and C.3 of Regulatory Guide 1.26, revision 3, "Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and C.1, C.3, and C.4 of Regulatory Guide 1.29, revision 3, "Seismic Design Classification," by designing the sampling lines and components of the process sampling system to conform to the classification of the system up to and including the first isolation valves to which each sampling line and component is connected, and thus meets the quality standards requirements of GDC 1 and the seismic requirements of GDC 2.

Based on the above evaluation, we find the proposed process sampling system acceptable.

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9.3.3 Equipment and Floor Drainage System

The equipment and floor drain system was reviewed in accordance with Section 9.3.3 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the equipment and floor drain system with respect to the applicable regulations of 10 CFR 50.

The nonsafety-related (Quality Group D, nonseismic Category I) equipment and floor drainage system includes all piping from equipment or floor drains to the sump, sump pumps, and piping necessary to carry potentially radioactive and non-potentially radioactive effluents through separate subsystems. Potentially radioactive drainage is collected in floor and equipment drain sumps in each building and discharged to the radwaste processing system, thus satisfying the requirements of General Design Criterion 60, "Control of Releases

of Radioactive Materials to the Environment." Drainage from non-potentially radioactive sources such as turbine building liquid waste or roof drains is processed in the industrial waste system or discharged directly offsite. The containment penetration for the containment sump pump discharge line is designed to seismic Category I and Quality Group B requirements and is located in seismic Category I, flood- and tornado-protected structures.

Our review considered those safety systems needed to provide safe plant shutdown and the physical location of those systems with regard to potential in-plant flooding. Because of their location at the lowest elevation in the primary auxiliary building, the vaults which contain pumps and equipment necessary for safe plant shutdown under accident conditions have been reviewed in detail relative to their provisions for prevention of water accumulation There are two vaults in the primary auxiliary building of each unit. Each vault is watertight and contains one train of residual heat removal, safety injection, and containment building spray equipment. Each vault has a sump equipped with redundant pumps and level switches and the separate discharge piping contains redundant check valves. Drain water from these sumps goes to the radwaste building. In this manner, each vault is not subject to back flooding caused by water accumulation in the other vault or from failure in nonsafety-related equipment.

All other safety-related equipment is located at higher elevations in the primary auxiliary building and at higher elevations in other buildings. The applicant has shown that drains in these areas assure that the worst-case hypothetical leakage including failure of nonseismic fluid systems in safety-related equipment areas does not result in a water level that will impair the functioning of safety-related systems before corrective action can be taken. Therefore, we conclude that the requirements of General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena," and 4, "Environmental and Missile Design Bases," and the guidelines of Positions C.1 and C.2 of Regulatory Guide 1.29, "Seismic Classification," are satisfied.

Based on the above, we conclude that the equipment and floor drainage system complies with the requirements of General Design Criteria 2, 4, and 60 with

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respect to protection against natural phenomena, environmental effects (flooding), and releases of radioactive material to the environment and the guidelines of Regulatory Guide 1.29 concerning seismic classification, and is, therefore, acceptable. The equipment and floor drains system meets the acceptance criteria of SRP Section 9.3.3.

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9.4.2 Spent Fuel Pool Area Ventilation System (Fuel Storage Building Heating and Ventilation)

The fuel storage building ventilation system was reviewed in accordance with Section 9.4.2 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP. Conformance with the acceptance criteria formed the basis for our evaluation of the fuel storage building ventilation system with respect to the applicable regulations of 10 CFR 50.

The fuel-handling storage building heating and ventilation system consists of a nonsafety-related normal heating and ventilation subsystem and a safety-related emergency air cleanup subsystem. The normal subsystem is designed to maintain a suitable temperature and environment during normal operation, maintenance, testing, and periods of general personnel access. The emergency air cleanup subsystem operates during fuel handling and is designed to maintain a negative pressure within the fuel storage building, to remove and retain airborne particulates and iodine, and to exhaust filtered air following a fuel-handling accident.

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The normal subsystem utilizes the primary auxiliary building ventilation system as its air supply. This air is filtered and heated before reaching the fuel-handling building (see Section 9.4.3 of this SER). Air is supplied to the fuel-handling building by a single train which feeds a duct system which distributes normal air to the spent fuel pool area and the spent fuel pool cooling pump room. Normally air is exhausted by sweeping across the spent fuel pool surface to an exhaust fan, isolation damper, and radiation monitor to the unit vent. Building pressure is maintained slightly negative. Isolation of the normal system from the emergency air cleanup system is achieved by seismic Category I parallel supply air dampers.

The emergency air cleanup subsystem is entirely seismic Category I, Quality Group C, and is utilized whenever irradiated fuel not in a sealed cask is handled in the building or during an emergency condition. In the fuel handling or emergency mode, the normal subsystem exhaust dampers are closed, the exhaust fan stopped and intake air is modulated by the seismic Category I, redundant normal supply dampers to maintain a negative building pressure. Air is exhausted through the emergency cleanup air subsystem. The cleanup system consists of redundant filter trains, redundant fans, ductwork, dampers, and controls. Each filter train consists of demisters, heaters, medium efficiency filter, HEPA filters, and carbon filters. Active components in each train are powered from separate emergency (Class 1E) power supplies. During fuel handling, only one filter train and fan will normally be operating. In the event of an accident, the second filter train and fan can be manually started. In the event of failure of an operating fan, the ductwork cross-connection will ensure an adequate redundant air flow across the partially loaded or fully loaded filter bed. Exhaust of the filtered air is through the radiation-monitored unit vent.

The normal subsystem is not necessary for safe shutdown operations. Air for the entire fuel storage building heating and ventilation system is supplied as part of the primary auxiliary building heating and ventilating system (see Section 9.4.3 of this SER). The fuel storage building heating and ventilation system is located in the primary auxiliary building, containment enclosure ventilation area, and the fuel storage building which are seismic Category I, flood- and tornado-protected structures (refer to Sections 3.4.1 and 3.5.2 of

this SER). The nonessential normal subsystem is separated from the essential portions of the system such that its failure will not prevent essential safety functions. Essential portions of the system itself are seismic Category I, Quality Group C, and are physically separated from high-energy systems. Thus the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Positions C.1 and C.2, are met. Refer to Sections 3.5.1.1, 3.6.1, and 9.3.3 of this SER for discussion of protection of essential spent fuel pool area ventilation system components from internally generated missiles, postulated failures in piping systems, and internal flooding.

Each station unit has its own independent fuel-handling area ventilation system; thus the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are not applicable.

The design described above meets the requirements of General Design Criteria 60, "Control of Releases of Radioactive Materials to the Environment," and 61, "Fuel Storage and Handling and Radioactivity Control," and the guidelines of Regulatory Guides 1.13, "Spent Fuel Storage Facility Design Basis," Position C.4, 1.52, "Design, Testing and Maintenance Criteria for Post Accident Engineered-Safety-Features Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Position C.2, and 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Positions C.1 and C.2, are satsified.

Based on the above, we conclude that the fuel storage building ventilation system is in conformance with the requirements of General Design Criteria 2, 60, and 61 as they relate to protection against natural phenomena, control of releases of radioactive materials, and radioactivity control, and the guidelines of Regulatory Guides 1.13, Position C.4, 1.29, Positions C.1 and C.2, 1.52 Position C.2, and 1.140, Positions C.1 and C.2, relating to protection against radioactive releases, seismic classification, and system design for emergency and normal operation, and is therefore acceptable. The fuel storage building ventilation system meets the acceptance criteria of SRP Section 9.4.2.

9.4.3 Auxiliary and Radwaste Area Ventilation System (Primary Auxiliary Building Heating and Ventilating System - Waste Processing Building HVAC Systems)

The primary auxiliary building ventilation system and the waste processing building ventilation system were reviewed in accordance with Section 9.4.3 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation that the above mentioned ventilation systems with respect to the applicable regulations of 10 CFR 50.

The nonsafety-related (nonseismic Category I, Quality Group D) primary auxiliary building (PAB) heating and ventilating system and waste processing building (WPB) HVAC systems are designed to maintain a suitable environment for equipment operation and personnel access and to limit potential radioactive releases to the environment during all modes of operation. The systems include the following: PAB supply and unfiltered exhaust system, PAB filtered exhaust system, and WPB ventilation and heating system. The only portion of any of these systems required for accident mitigation and safe plant shutdown is the unitized equipment associated with cooling the primary component cooling and boron injection pump areas. This equipment is discussed in Section 9.4.5 of this SER. where it and other engineered safety feature ventilation systems are discussed. Some ductwork and isolation dampers required for containment isolation are seismic Category I, Quality Group B. No other part of these systems is safety related.

The nonsafety-related PAB heating and ventilating system serves the normal heating and ventilating requirements of the following areas: mechanical equipment room, chemical and volume control tank area, boric acid tank area, degasifier areas, primary component cooling heat exchanger area, chiller pump area, heat exchanger and filter rooms, demineralizer rooms, pipe tunnel area, fume hood, and charging pump rooms. Ventilation is provided by PAB supply fans furnishing filtered and heated 100% outside air to all areas of the

PAB. Air is exhausted by either the filtered or unfiltered exhaust subsystems depending on the potential for radioactive contamination. DRAFT

The unfiltered subsystem normally operates with two of three exhaust fans on and discharges to the plant vent. Each exhaust fan will deliver one half of the total required exhaust air capacity. The third fan remains on standby, and is controlled by the operator from the main control panel. Seismic Category I, Quality Group B, redundant dampers are installed in the supply and exhaust ducts between the PAB and the containment enclosure area to permit isolation of the containment enclosure area in the event of a loss-of-coolant accident (LOCA) or failure of the PAB supply/exhaust system.

The nonsafety-related PAB filtered exhaust subsystem draws air through a filter train from the charging pump rooms, valve aisle, chemical and volume control tank area, sample heat for filtration prior to exhausting to the unit plant vent. The plant vent contains radiation monitors. The system, including exchanger room, fume hood, letdown heat exchangers, pipe tunnel area and degasifier area. The subsystem is manually controlled from the main control room and consists of return ducting to a filter room containing a prefilter, medium efficiency filter, HEPA filter, and a carbon adsorber section. The subsystem is used to collect air from potentially contaminated areas in the PAB and containment enclosure area while maintaining these areas at a slight negative pressure. In the event of a LOCA or a loss of the PAB filtered exhaust subsystem, seismic Category I, Quality Group B, redundant dampers will isolate the containment enclosure area from the PAB. The filter train and associated fans are housed within a room which has its own ventilation and heating systems. Ventilation is provided through redundant power roof ventilators and redundant operable outside air louvers.

Control of the power roof ventilators and louvers is manually from the main control panel or automatically by thermostats located in the filter room. Since both louvers and power roof ventilators are redundant, a single failure will not prevent operation of the filtered exhaust subsystem.

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The entire PAB HVAC system is designed to control air flow from areas of low potential airborne radioactivity toward areas of higher potential airborne radioactivity safety-related isolation dampers and ductwork is housed in seismic Category I structures which are designed to withstand the effects of flooding and tornado missiles. Each plant unit has a separate PAB heating, ventilating filtered exhaust system. A failure of any portion of the system will not compromise plant safety. Thus the requirements of General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena," and 60, "Control of Releases of Radioactive Materials to the Environment," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Positions C.1 and C.2, and 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," Positions C.1 and C.2, are satisfied.

The waste processing building (WPB) HVAC system provides filtered outside air to ventilate or heat the WPB. This building is common for Units 1 and 2. The system is nonsafety related and nonseismic Category I with the exception of certain ductwork over essential equipment which is seismically supported.

The WPB ventilation and heating system consists of two 50%-capacity supply fans which draw outside air through filters and heating coils and into ductwork distributing the air to all areas of the building. Air is supplied to the waste gas compressor areas and the hydrogen surge tank cubicles to maintain acceptable limits for radioactivity and hydrogen concentration. A larger quantity of air is exhausted from these areas than is supplied, resulting in their being maintained at a negative pressure.

Three exhaust systems are used as part of the building normal exhaust system. Two of the normal exhaust systems are similar in that they do not filter the exhaust air before discharging to the plant vent. The third exhaust system collects air from areas which because of possible airborne contamination require filtration before releasing the exhaust air to the plant vent. Booster fans in these three exhaust systems provide negative pressure in a controlled manner.

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In conjunction with the main system, several areas are furnished additional nonsafety-related cooling or ventilation by air conditioning units (carbon delay bed units and polymer storage tank area) or by room ventilators (refueling water storage tank area, reactor makeup water storage tank area, boron waste storage tank areas and the elevator equipment room).

The ambient carbon delay bed areas are safety related, and the ductwork in those areas is supported in such a manner so as to prevent its falling during an SSE. Failure of the nonsafety-related air conditioning systems servicing the carbon delay bed areas may result in a reduction in the adsorbent quality of the carbon delay beds. However, this will not result in an increase in the release level at the plant vent since the radioactive gases will be recirculated, or the system shut down, if necessary. Thus, the requirements of General Design Criteria 2 and 60 and the guidelines of Regulatory Guides 1.29, Position C.2, and 1.140, Positions C.1 and C.2, are satisfied.

Those features of the auxiliary and radwaste area ventilation system which are shared between Units 1 and 2 do not perform a safety function, and thus the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are not applicable.

Based on the above, we conclude that the auxiliary and radwaste area ventilation systems are in conformance with the requirements of General Design Criteria 2, 5, and 60 as they relate to protection against natural phenomena, shared systems, and control of release of radioactive materials to the environment, and the guidelines of Regulatory Guides 1.29, Positions C.1 and C.2, and 1.140, Positions C.1 and C.2, relating to seismic classification and system design for emergency and normal operation and are, therefore, acceptable. The PAB and WAB ventilation systems meet the acceptance criteria of SRP Section 9.4.3.

9.4.4 Turbine Area Ventilation System (Turbine Building Heating, Ventilation, and Air Conditioning Systems)

The turbine building heating, ventilation, and air conditioning systems were reviewed in accordance with Section 9.4.4 of NUREG-0800 (SRP). An audit

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review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the turbine building heating, ventilation, and air conditioning systems with respect to the applicable regulations of 10 CFR 50.

The turbine building heating, ventilation, and air conditioning systems consist of independent systems for each unit. Each consists of air intake louvers in the lower outside walls and roof exhaust fans discharging to the environment. Individual rooms within the turbine building have unitized air conditioning and heating equipment where necessary. The main turbine and heater bay areas do not have heaters.

The system is classified as nonsafety related (nonseismic Category I, Quality Group D). The system maintains an acceptable environment for personnel and the nonessential equipment served during normal plant operation. The system has no safety functions. The system is separated from safety-related plant systems and potentially radioactive areas; therefore, failure of the system will not compromise the operation of any essential plant systems or result in an unacceptable release of radioactivity, and thus it meets the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Position C.2. Conversely, the requirements of General Design Criterion 60, "Control of Releases of Radioactive Materials to the Environment," and the guidelines of Regulatory Guide 1.140, "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," are not applicable.

Based on our review, we conclude that the turbine building ventilation system meets the requirements of General Design Criterion 2 with respect to the need for protection against natural phenomena as its failure does not affect safety system functions, or result in release of radioactive material, and the guidelines of Regulatory Guide 1.29, Position C.2, concerning its seismic classification, and is, therefore, acceptable. The turbine building ventilation system meets the acceptance criteria of SRP Section 9.4.4.

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9.4.5 Engineered Safety Feature Ventilation System (Containment Enclosure Area Cooling and Ventilation System, Diesel Generator Building Heating and Ventilating System, 4-kV Switchgear Area, Battery Rooms and Electrical Tunnels Heating and Ventilation Systems, Emergency Feedwater Pump House Heating and Ventilation System, Service Water Pump House Heating and Ventilation System and Service Water Cooling Tower Heating and Ventilation System)

The engineered safety feature ventilation systems were reviewed in accordance with Section 9.4.5 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the engineered safety feature ventilation systems with respect to the applicable regulations of 10 CFR 50.

Engineered safety features ventilation system provides ventilation for the containment enclosure area, diesel-generator building, 4-kV switchgear area, battery rooms, electrical tunnels, emergency feedwater pump house, service water pump house, and service water cooling tower.

The containment enclosure area cooling and ventilation system is designed to remove equipment heat from the charging pump, safety injection pump, residual heat removal equipment, containment spray heat exchanger equipment, and pipe penetration areas. During normal operation, makeup air is introduced into the containment enclosure area from the primary auxiliary building ventilation system through isolation dampers. Redundant fans, drawing air from the containment enclosure, supply the above areas through redundant ducting and isolation dampers. Air from the areas supplied is exhausted through the normal cleanup exhaust system to the atmosphere via the unit plant vent. Following a LOCA, the emergency mode is actuated and air is returned to the containment enclosure cooling equipment area for recirculation, with no makeup air introduced. This is accomplished by closing the makeup air isolation dampers and the exhaust air isolation dampers, and opening the return dampers from the cooled areas. Recirculation is through the containment enclosure

emergency air cleanup system which consists of redundant filter trains, fans, dampers and controls and a common ductwork system. The air flow required to maintain a negative pressure in the containment enclosure building and is passed through demisters, HEPA filters, and carbon filters prior to exhausting through the plant vent.

The containment enclosure cooling units, return fans, return and isolation dampers, ductwork supports, and emergency exhaust air cleaning units are classified as seismic Category I, Quality Group C, and are located in seismic Category I, flood- and tornado-protected buildings (refer to Sections 3.4.1 and 3.5.2 of this SER).

The containment enclosure cooling units and return fans are redundant and, thus, a single active failure will not prevent system function. Each cooling unit is connected to a separate train of the emergency (Class 1E) power supplies, thereby ensuring availability of power in the event of loss of offsite power, and each is supplied cooling water from a separate loop of the primary component cooling water system. The makeup air and normal exhaust systems, except for the isolation and return dampers, are not required to operate following a LOCA. Nonseismic Category I portions of the system located in the vicinity of safety-related systems are separated or supported such that their failure due to a seismic event will not result in damage to essential equipment. The containment enclosure area cooling and ventilation system is independent for each plant unit; hence, the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are not applicable, and the requirements of General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena," 4, "Environmental and Missile Design Bases," and 60, "Control of Releases of Radioactive Materials to the Environment," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Positions C.1 and C.2, and 1.52, "Design, Testing and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power PLants," Position C.2, and 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plant," Positions C.1 and C.2, are met.

The diesel-generator building (DGB) heating and ventilating system is a once-through open system using ambient outside filtered air to provide all cooling and heating during normal and emergency conditions to the diesel fuel oil storage area, starting equipment, water cooling equipment, and lubrication equipment. The system contains a full capacity supply and exhaust fan for each diesel-generator room of each plant unit. Outside air enters the building through tornado-missile-protected wall vents at an elevation over 20 feet above grade and is filtered. A supply fan, drawing air from the diesel room, delivers air to essential areas. A tornado-missile-protected roof exhaust fan returns the heated air to the environment. Heating for the DGB is provided by four hot water unit heaters, unassociated with the cooling system. Ventilation to the basement oil storage area is through open grating. Combustion air is drawn from the open building area separately from the ventilation air.

All fans and dampers of the ventilation system are seismic Category I, Quality Group C. The dampers fail open on loss of power and the fans are powered from separate emergency (Class 1E) power supplies. The heating system, while not safety related, has its hot water heating piping, unit heater, and ductwork supports designed as seismic Category I, so that they will not fail in such a manner that may damage safety-related equipment in the event of an SSE.

The DGB for each unit is separate; thus, the requirements of General Design Criterion 5, "Sharing of Structures, Systems, and Components," are not applicable. Further, the DGB is not a source of radioactivity; thus, the requirements of General Design Criteria 60, "Control of Releases of Radioactive Materials to the Environment," and the guidelines of Regulatory Guides 1.52, "Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," and 1.140, "Design, Testing and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants," are not applicable.

The DGB ventilation system is housed in the seismic Category I, flood- and tornado-protected DGB (refer to Sections 3.4.1 and 3.5.2 of this SER). Fire dampers in the system ducts close and isolate the system upon activation of

the associated diesel-generator fire protection system. Thus, the requirements of General Design Criteria 2, "Design Bases for Protection Against Natural Phenomena," 4, "Environmental and Missile Design Bases," and 17, "Electric Power Systems," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Positions C.1 and C.2, and item 2, subsection A of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability," concerning protection of essential electrical components from failure due to dust accumulation are met.

The 4-kV switchgear area, battery rooms, and electrical tunnels heating and ventilation system supplies outside ambient air to cool the 4-kV switchgear rooms, battery rooms and electrical cable tunnel areas to the containment building. Heating of the air is provided by water or electrical heaters to the cable spreading room and battery rooms only. Outside air is drawn from each room of the diesel-generator building, through supply fans and ducting into the various areas. Exhaust ducting removes the heated air through exhaust fans directly to the outside environment through exhaust louvers. A separate battery room exhaust fan is also provided for each train of battery rooms. Failure of the fan is alarmed in the control room to provide for protection against unacceptable hydrogen accumulators.

Each unit has two redundant ventilation systems with full capacity supply and exhaust fans cooling the three areas. All four systems (two for each plant) are separate and independent; thus the requirements of General Design Criterion 5 are not applicable. The switchgear, battery room and electrical cable tunnel areas are not sources of radioactivity; thus the requirements of General Design Criterion 60 are not applicable. Each system is powered from the separate emergency (Class 1E) power supply serving its associated equipment room. Thus, operation of at least one division of miscellaneous electrical equipment and its associated battery is assured in the event of a single failure in any system component.

The 4-kV switchgear, battery rooms and electrical cable tunnel ventilation system is classified as seismic Category I, Quality Group C, and all ducting is seismically supported. The temperature of the battery rooms and switchgear

rooms is controlled. Off-normal conditions are alarmed in the control room. Fire dampers are provided in duct penetrations and/or ventilation openings in fire walls. The system is housed in the seismic Category I, flood- and tornado-protected control building (refer to Sections 3.4.1 and 3.5.2 of this SER). The outside air intakes and exhausts are tornado missile protected. Thus, the requirements of General Design Criteria 2 and 4 with respect to protection against natural phenomena and assurance of proper operating environment for essential equipment and the guidelines of Regulatory Guide 1.29 Positions C.1 and C.2, with respect to seismic classification are met.

The emergency feedwater pump house heating and ventilation system consists of two redundant supply fans which take outside air directly through the pump house walls by means of tornado-missile-protected and louvered intakes. The air is forced throughout the pump house without ducting and is exhausted through two redundant tornado-missile-protected exhaust openings in the pump house walls at the opposite end of the building from the supply fans. The heating system consists of two 100%-capacity hot water systems with four unit heaters distributed throughout the pump house. The ventilation system is safety related and its associated heating system is nonsafety related.

The emergency feedwater pump house is independent for each unit and is not a source of radioactivity; thus the requirements of General Design Criteria 5 and 60 are not applicable.

The redundant, seismic Category I, Quality Group C, pump room supply fans, and supply and exhaust dampers are supplied from separate emergency (Class 1E) power sources. Thus, ventilation is available in the event of an SSE, loss of offsite power or a single failure occur. Loss of air or electrical power to the pneumatically operated supply and exhaust dampers will cause them to fail open. Fan trip and pump house high temperature are alarmed. Heating system operation is not required to assure proper operation of the pumping equipment or the electrical equipment in the emergency feedwater pump house. If an unlikely loss of the heating system occurs, the low temperature alarm will alert the operator to a potential freezing situation, and corrective action can be taken to prevent freezing.

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The system is housed in a seismic Category I, flood and tornado-protected building (refer to Sections 3.4.1 and 3.5.2 of this SER). The outside air intakes and exhausts are tornado missile protected. Thus, the requirements of General Design Criteria 2 and 4 with respect to protection against natural phenomena and assurance of proper operating environment for essential equipment and the guidelines of Regulatory Guide 1.29, Positions C.1 and C.2, with respect to seismic calassification are met.

The service water pump house heating and ventilation system consists of a pump room area heating and ventilating system and a switchgear area heating and ventilating system. The pump house and pump room and switchgear room are common for both units. The pump room area system draws outside air through four tornado-missile-protected intake louvers at one end of the building and exhausts the air through four exhaust fans through louvers into a tornado-missile-protected discharge plenum. Each of the four one-third capacity pump room exhaust fans is powered from a separate emergency (Class 1E) power source from each plant unit. The switchgear area has two trains of electrical equipment each cooled by a separate ventilating system. Filtered outside air is supplied through fans and ducting and fire dampers located outside in tornado missile enclosures to the general area of each switchgear room. Air is exhausted to the same discharge plenum used by the pump room area system. The two 100%-capacity supply fans for the switchgear room cool both trains of both units and are powered from redundant Unit 1 emergency (Class 1E) power supplies. The pump room exhaust fans, switchgear supply fans, and dampers are safety-related and seismic Category I, Quality Group C. The above design assures that the system can maintain the temperature below limits during both normal and emergency plant operation including an SSE, loss of offsite power, and any single active failure. Thus, the requirements of General Design Criterion 5 with respect to sharing of systems between units are met.

The pump room area is heated by nonsafety-related hot water unit heaters. A hot water heating line break or heating system failure will not affect the operation of the service water pumping equipment. The switchgear areas are heated with nonsafety-related electric unit heaters. Each area has two half-size heaters. A failure of the heating system will not affect the operation of the

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switchgear. Instrumentation is provided to alarm on the main control board high/low temperature conditions in the service water pump and switchgear rooms and operator action can be taken to restore heating as necessary.

Since the service water pump house is not a source of radioactivity, the requirements of General Design Criterion 60 is not applicable.

The system is seismic Category I, Quality Group C and is housed in a seismic Category I, flood- and tornado-protected building (refer to Sections 3.4.1 and 3.5.2 of this SER). Thus, the requirements of General Design Criteria 2 and 4 with respect to protection against natural phenomena and assurance of proper operating environment for essential equipment, sharing of systems between units, and the guidelines of Regulatory Guide 1.29, Positions C.1 and C.2, with respect to seismic classification are met.

The service water cooling tower heating and ventilation system consists of a pump room area ventilating system and a switchgear room area heating and ventilating system. The pump rooms are not provided with a heating system because the water lines are drained in winter to preclude freezing. The pump rooms are cooled by exhausting air through two redundant roof fans for each unit. Air enters through tornado-missile-protected intakes at opposite ends of the cooling tower. The redundant switchgear room areas are cooled by outside air drawn from the building and ducted into the switchgear room areas by separate supply fans. Air is exhausted directly through discharge wall louvers. The nonsafety-related heating system for the switchgear rooms is electrical and may not be available under emergency conditions but is not required to assure a proper equipment operating environment. Pump and switchgear room fans are powered from redundant emergency (Class 1E) power supplies. Although the cooling tower structure is common for both units, the individual halves for each unit are ventilated by independent systems. Thus, the requirements of General Design Criterion 5 are not applicable. Since the service water cooling tower is not a source of radioactivity, General Design Criterion 60 is not applicable.

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The system is seismic Category I, Quality Group C and is housed in a seismic Category I, flood- and tornado-protected building (refer to Sections 3.4.1 and 3.5.2 of this SER). Thus, the requirements of General Design Criteria 2 and 4 with respect to protection against natural phenomena and assurance of proper operating environment for essential equipment and sharing of systems between units and the guidelines of Regulatory Guide 1.29, Positions C.1 and C.2, with respect to seismic classification are met.

The primary component cooling water (PCCW) pump and boron equipment areas ventilation system is the only ventilation system in the primary auxiliary building necessary for accident mitigation and safe plant shutdown. Normally, air is supplied to and exhausted from the PCCW pump and boron equipment areas by the main primary auxiliary building system (refer to Section 9.4.3 of this SER). Should this system fail or the temperature reach 105°F, one each of the two redundant supply and exhaust dampers will open and a separate PCCW and boron equipment room fan will start. Should the temperature reach 110°F, the second system fan will operate automatically in the same manner as the first. These fans are powered from redundant emergency (Class 1E) power supplies. The fans and associated automatic dampers are seismic Category I, Quality Group C. The fans may be controlled manually from the main control board or automatically.

In the above emergency condition, air is supplied from and discharged to the outside through tornado-missile-protected dampers. Outside air without additional cooling is adequate to maintain the temperature below the room limit (117°F). In the event of a radioactive release in the area, the dampers can be closed and the area air filtered through the normal primary auxiliary building filtered exhaust system.

The above design assumes that system function is maintained in the event of a failure of a single active component. The PCCW pump and boron injection equipment area ventilation system is not shared between units; therefore, the requirements of General Design Criterion 5 are not applicable.

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The system is housed in the seismic Category I, flood and tornado-protected primary auxiliary building (refer to Sections 3.4.1 and 3.5.2 of this SER). The outside air intakes are tornado missile protected. Thus, the requirements of General Design Criteria 2, 4, and 60 with respect to protection against natural phenomena, assurance of proper operating environment for essential equipment, control of radioactive releases and the guidelines of Regulatory Guide 1.29, Positions C.1 and C.2, with respect to seismic classification are met.

Based on the above, we conclude that the engineered safety features ventilation system is in conformance with the requirements of General Design Criteria 2, 4, 5, 17 (diesel-generator room ventilation system only), and 60 as they relate to protection against natural phenomena, assurance of proper operating environment for essential equipment including the diesel-generators and shared systems and protection against unacceptable radioactive release and the guidelines of Regulatory Guide 1.29, Positions C.1 and C.2, concerning seismic classification and NUREG-CR/0660, item 2, subsection A concerning protection of the emergency diesel generation from unacceptable dust accumulation and is, therefore, acceptable. The engineered safety features ventilation systems meet the acceptance criteria of SRP Section 9.4.5.

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10.4.5 Circulating Water System

The circulating water system was reviewed in accordance with Section 10.4.5 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" portion of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the circulating water system with respect to the applicable regulations of 10 CFR 50.

The nonsafety-related (Quality Group D, nonseismic Category I) circulating water system provides cooling water to the main condensers to remove the heat rejected by the turbine cycle. Three circulating water pumps for each unit are located in the circulating water pump house. The pumps take suction on the Atlantic Ocean through an undersea tunnel, pump the seawater through the condensers, and return the water to the sea through another undersea tunnel. The circulating water system is not required to maintain the reactor in a safe shutdown condition or mitigate the consequences of accidents.

The applicant has provided an analysis of the effects of possible flooding of safety-related equipment as a result of a postulated failure in the circulating water system. The circulating water system has the potential for flooding the turbine building to grade level if a line ruptured and the pumps were not shut off. No safety-related equipment is located in the turbine building, and no below-grade openings connect directly with any building containing safety-related equipment.

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Continued operation of the pumps would cause water to flow out of the turbine building through scuppers and doors to the yard and away from plant buildings. Shutdown of the pumps would eventually stop the flow. Thus, a total failure in the circulating water system will not result in flooding which would compromise plant safety.

Since no safety-related equipment is affected by a postulated failure in the circulating water system, the requirements of General Design Criterion 4, "Environmental and Misile Design Bases," with respect to protection of safety-related systems from failure of nonsafety-related systems, are satisfied.

Indication of leakage or potential failure in circulating water system components is provided to operators in the control room. Water level alarms are installed in pits in the turbine building and circulating water pump house. Pump shutoff is accomplished manually by the operator. Condenser performance is monitored by pressure and temperature indicators in the control room.

Based on our review, we conclude that the circulating water system meets the requirements of General Design Criterion 4 with respect to protection of safety-related systems from failures in nonsafety-related systems. We therefore conclude that the circulating water system is acceptable. The circulating water system meets the acceptance criteria of SRP Section 10.4.5.

10.4.7 Condensate and Feedwater System

The condensate and feedwater system was reviewed in accordance with Section 10.4.7 of NUREG-0800 (SRP). An audit review of each of the areas listed in the "Areas of Review" portion of the SRP section was performed according to the guidelines provided in the "Review Procedures" of the SRP section. Conformance with the acceptance criteria formed the basis for our evaluation of the condensate and feedwater system with respect to the applicable regulations of 10 CFR 50.

The condensate and feedwater system provides feedwater from the condenser to the steam generators and includes the piping and components from the condenser

hot well, through the condensate pumps, low pressure feedwater heaters, feedwater pumps, high-pressure feedwater heaters, and containment isolation valves to the four steam generators.

The system serves no safety function (with the exception of containment isolation integrity) and is therefore classified as nonsafety related (Quality Group D, nonseismic Category I). Adequate isolation is provided at connections between seismic and nonseismic Category I systems, and therefore failure of nonsafety-related portions of the condensate and feedwater system will not affect safe plant shutdown.

The portions of the system classified as safety related are: (1) the main feedwater piping from the containment isolation and check valves to the steam generators, (2) the piping in the main steam and feedwater pipe chases, and (3) the interconnecting piping between the auxiliary feedwater system and the feedwater lines. These portions of the system are designed to seismic Category I, Quality Group B requirements in order to assure feedwater isolation in accident situations and are located in seismic Category I, flood- and tornado-protected structures (refer to Sections 3.4.1 and 3.5.2 of this SER). Thus, the requirements of General Design Criterion 2, "Design Bases for Protection Against Natural Phenomena," and the guidelines of Regulatory Guide 1.29, "Seismic Design Classification," Positions C.1 and C.2 are satisfied. The structure also provides protection against tornado missiles. The essential equipment is separated from the effects of internally generated missiles and is not affected by failures in high-energy piping (refer to Sections 3.5.1.1, 3.6.1, and 3.11 of this SER). Thus, the requirements of General Design Criterion 4, "Environmental and Missile Design Bases," are satisfied. None of the condensate and feedwater system is shared between units so that the requirements of General Design Criterion 5, "Sharing of Structures, Systems and Components," are not applicable.

Automatic isolation of the main feedwater system is provided when required to mitigate the consequences of a steam or feedwater line break. The pneumatically operated main feedwater isolation valves (one per steam generator) close within 5 seconds on receipt of an ESF actuation signal. Redundant feedwater line isolation is provided by the fail-closed main feedwater regulating valves and bypass valves which serve as an acceptable backup. The nonsafety-related

startup feedwater train supplying the main feedwater line is the normal means for starting up and shutting down the plant. The safety-related auxiliary feedwater trains automatically provide flow to the steam generators via the main feedwater lines for decay heat removal upon failure of the condensate and feedwater system. Refer to Section 10.4.9 of this SER for further discussion of the auxiliary feedwater system. Thus, the requirements of General Design Criterion 44, "Cooling Water," are satisfied. The safety-related portions of the system are located in accessible areas and receive periodic inspection and testing in accordance with Plant Technical Specifications. Thus the requirements of General Design Criteria 45, "Inspection of Cooling Water System," and 46, "Testing of Cooling Water System," are satisfied.

The condensate and feedwater system is designed with features to preclude the potential for damaging flow instabilities (water hammer). These features include:

1. Providing the Westinghouse Model F steam generators with top discharge feedrings fitted with J tubes to permit uncovering of the feedring without subsequent drainage.

2. Installation of the feedwater piping from the feedwater isolation valve to the steam generator in such a manner as to be self-venting, with the steam generator being the high point of the system.

In addition, the applicant has agreed to perform tests to verify that unacceptable feedwater hammer will not occur using the plant operating procedures for normal and emergency restoration of steam generator water level following loss of normal feedwater. Thus, the guidance in Branch Technical Position ASB 10-2, "Design Guidelines for Water Hammers in Steam Generators with Top Feedring Design," has been followed.

Based upon the above, we conclude that the safety-related portion of the condensate and feedwater system meets the requirements of General Design Criteria 2, 4, 44, 45, and 46 with respect to its protection against natural phenomena, missiles and environmental effects, decay heat removal function, inservice

inspection and testing, and meets the guidelines of Regulatory Guide 1.29, Positions C.1 and C.2, and Branch Technical Position ASB 10-2 with respect to its seismic classification and design and testing for prevention of damaging water hammer, and is therefore acceptable. The condensate and feedwater system meets the acceptance criteria of SRP Section 10.4.7.

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11.0 Radioactive Waste Management

11.1 Source Terms

The applicant calculated the liquid and gaseous effluents from the Seabrook Station utilizing the PWR GALE computer program. The applicant utilized the source assumptions of Regulatory Guide 1.112, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid-Effluents from Light-Water-Cooled Power Reactors", and

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11.1 NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWRs)". Gaseous effluents were calculated from such sources as offgases from the main condenser evacuation system; leakage to containment, primary auxiliary building, and turbine building; noble gases stripped from the primary coolant during normal operation and at shutdown; and cover and vent gases from tanks and equipment containing radioactive material. Liquid effluents were calculated from such sources as shim bleed, leakage collected in equipment and floor drains of the turbine and primary auxiliary buildings, steam generator blowdown, contaminated liquids from anticipated plant operations such as resin sluices, filter backwash, decontamination solutions, sample station drains, and detergent wastes.

The staff has performed an independent calculation of the primary and secondary coolant concentrations and of the release rates of radioactive materials using the information supplied in the applicant's FSAR, the GALE computer program, and the methodology presented in NUREG-0017. Table 11.1-1 presents the principal parameters which were used in this independent calculation of the source terms. These source terms were utilized in Sections 11.2 and 11.3 to calculate individual doses in accordance with the mathematical models and guidance contained in Regulatory Guide 1.109,

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11.1 "Calculation of Annual Average Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I".

Liquid effluents occur from the boron recovery system, the steam generator blowdown system and the liquid waste system. The boron recovery system was assumed to treat letdown from the primary coolant system, valve leakoff and liquid collected in the primary drain tank. Additional information with respect to flow rates, DF's, fraction of primary coolant activity, etc., is contained in Tables 11.1-1 and 11.1-2 under shim bleed and equipment drains and the boron recovery system. The liquid waste system treats liquid wastes collected in the floor drain tanks. Additional information on the liquid waste system can also be found in Tables 11.1-1 and 11.1-2 under clean wastes. Information on the steam generator blowdown system is also contained in the same Tables.

Airborne effluents occur from the normal ventilation system, from the radioactive gaseous waste system (RGWS), the main condenser air ejector, and the turbine gland steam condenser.

All airborne effluents except those released from the turbine gland steam condenser are passed through HEPA filters and charcoal adsorbers prior to discharge.

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Table 11.1-1 Principal parameters and conditions used in calculating releases of radioactive material in liquid and gaseous effluents from Seabrook, Units 1 and 2

Reactor power level (MWt)	3654
Plant capacity factor	0.80
Failed fuel	0.12%[a]
Primary system	
Mass of coolant (lb)	5.05×10^5
Letdown rate (gal/min)	75
Shim bleed rate (gal/day)	1.44×10^3
Leakage to secondary system (lb/day)	100
Leakage to containment building (lb/day)	b
Leakage to auxiliary building (lb/day)	160
Frequency of degassing for cold shutdowns (times/yr)	2
Letdown cation demineralizer flow (gal/min)	0
Secondary system	
Steam flow rate (lb/hr)	1.5×10^7
Mass of liquid/steam generator (lb)	9.55×10^4
Mass of steam/steam generator (lb)	9.73×10^3
Secondary coolant mass (lb)	1.80×10^6
Rate of steam leakage to turbine area (lb/hr)	1.7×10^3
Containment building volume (ft^3)	2.7×10^6
Frequency of containment purges (times/yr)	24
Containment low volume purge rate (ft^3/min)	0
Containment atmosphere cleanup rate (ft^3/min)	4.0×10^3
Pre-purge cleanup time duration (hr)	16
Iodine partition factors (gas/liquid)	
Leakage to auxiliary building	0.0075
Leakage to turbine area	1.0
Main condenser/air ejector (volatile species)	0.15

Liquid radwaste system decontamination factors

Material	Boron Recovery System	Liquid Waste System	Steam Generator Blowdown System
Iodine	1×10^4	1×10^3	1×10^3
Cesium, rubidium	4×10^3	1×10^4	1×10^4
Other	1×10^5	1×10^4	1×10^4

[a]This value is constant and corresponds to 0.12% of the operating power product source term as given in NUREG-0017 (April 1976).

[b]1%/day of the primary coolant noble gas inventory and 0.001%/day of the primary coolant iodine inventory.

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Table 11.1-1
(continued)

Liquid Waste Inputs

Stream	Flow Rate (gal/day)	Fraction of PCA	Fraction Discharged	Collection time (days)	Decay Time (days)
Shimbleed Rate	1440	1.0	0.1	51.7	2.50
Equipment Drains	300	1.0	0.1	51.7	2.50
Clean Wastes	1380	0.3	1.0	2.91	0.22
Blowdown	173000	-	1.0	0	0

Gaseous Waste Inputs

There is continuous stripping of full letdown flow

Holdup time for xenon (days)	68.0
Holdup time for krypton (days)	4.1
Fill time of decay tanks (days)	0

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Table 11.1-2

Individual equipment decontamination factors

1. Evaporator

System	All nuclides except iodine	Iodine
Steam Generator Blowdown and Liquid waste system, radwaste evaporator	10^4	10^3
Boron recovery system, recovery evaporator	10^3	10^2

2. Demineralizers

System	Anions	Cesium, rubidium	Other nuclides
Cesium removal ion exchanger (H_3BO_3)	10	2	10
Primary coolant letdown cation demineralizer	1	10	10
Primary coolant letdown mixed bed demineralizer (Li_3BO_3)	10	2	10
Recovery demineralizer	10	2	10

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12 RADIATION PROTECTION

Summary Description

The staff has evaluated the proposed radiation protection program presented in Chapter 12 of the Seabrook FSAR against the criteria set forth in the Standard Review Plan (SRP), NUREG-0800, Chapter 12. The radiation protection measures at Seabrook are intended to ensure that internal and external radiation dose to plant personnel, contractors, and the general population due to plant conditions, including anticipated operational occurrences, will be within applicable limits of 10 CFR Part 20, and will be as low as is reasonably achievable (ALARA).

The basis of the staff's acceptance of Seabrook's radiation protection program is that doses to personnel will be maintained within the limits of 10 CFR 20, "Standard for Protection Against Radiation." The applicant's radiation protection design features and program are consistent with the guidelines of Regulatory Guide 8.8, "Information Relevant to Ensuring That Occupational Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable " (Rev. 3). ~~Some~~ The applicant's radiation protection features will help to ensure that occupational radiation exposures are maintained as low as reasonably achievable, both during plant operation and during decommissioning.

On the basis of our review of the Seabrook FSAR, we have concluded that the radiation protection measures incorporated in the design and the proposed radiation protection program will provide a reasonable assurance that occupation doses will be maintained as low as is reasonably achievable and below the limits of 10 CFR Part 20. These radiation design

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features and program are consistent with the guidelines of Regulatory Guide 8.8 (Rev. 3).

12.1 ASSURING THAT OCCUPATIONAL RADIATION DOSES ARE AS LOW AS IS REASONABLY ACHIEVABLE (ALARA)

The staff has audited the policy considerations, design considerations and operational considerations contained in the Seabrook Nuclear Power Plant's FSAR against the criteria set forth in NUREG-0800 (SRP), Section 12.1. The staff review consisted of ensuring that the applicant had either committed to following the criteria of the regulatory guides and staff positions referenced in Section 12.1 of the SRP or provided acceptable alternatives. In addition, the staff selectively reviewed the applicant's FSAR against the specific areas of review and review procedures identified in the SRP. This selective review found the plant acceptable in these areas. Details of the review follow.

12.1.1 POLICY CONSIDERATION

The applicant provides a management commitment to assure that Seabrook will be designed, constructed and operated in a manner consistent with Regulatory Guides 8.8, 8.10 and 1.8. The applicant has identified the specific corporate plan to implement that policy, and specified in detail facility and equipment design considerations to assure its accomplishment. This objective is reached through administrative dose control procedures, adequate work planning, and safe practices in all activities related to unit operation. The Station Superintendent has the overall responsibility for implementing the ALARA program. He

delegates this responsibility to the station Health Physics Supervisor, who is responsible for maintaining the health physics program and has the specific responsibility and authority for ensuring that the radiation protection program maintains doses ALARA. The Health Physics Supervisor is responsible for ensuring that the ALARA policy is implemented and monitored. He will review dose records and will compare results from past experience to assess the effectiveness of the ALARA effort. Station management will also review these records and seek to identify exposure area that indicate dose trends and the need for improvement in plant procedures or equipment. These policy considerations meet the criteria of Regulatory Guide 8.8 and NUREG-0800 and are therefore acceptable.

12.1.2 DESIGN CONSIDERATIONS

The objective of the plant's radiation protection design is to maintain individual and collective doses to plant workers, including construction workers, and to members of the general public, ALARA and to maintain individual doses within the limits of 10 CFR 20. The general arrangements and shielding provisions of Seabrook are in accordance with Regulatory Guide 8.8, and are designed to provide levels of dose to operating personnel that are ALARA. Radiation protection experience from the Yankee Nuclear Power operating plants (i.e., Maine Yankee, Vermont Yankee and Haddam Neck) has been applied by the Yankee Nuclear Services Division who carries the responsibility and authority required on all engineering related matters.

The applicant has used the following design features for assuring that occupational radiation dose is ALARA. Whenever possible, piping containing radioactive material is either run in shield pipe chases or shielded cubicles to minimize exposure to plant personnel. High radiation level equipment is located away from high frequency personnel traffic-ways; labyrinths and/or shielding doors are used to eliminate radiation streaming; penetrations are made with as small as possible diameters and are not in direct line with major radioactive sources; radioactive components are located in separate shielded cubicles to minimize exposure during maintenance and inspection activities; use is made of low nickle and cobalt alloys to reduce crud buildup; provisions are made for flushing and purging of contaminated systems; and valves with improved design seals to allow less and faster maintenance are located so that operation and maintenance is conveniently performed in low radiation areas by use of extension stems offset from sources of radiation. The radiation protection design review is an ongoing review throughout all phases of the design with formal reviews conducted at regular intervals by the applicant, Yankee Atomic Electric Company, Westinghouse, and United Engineering and Construction Inc. These reviews consider source calculations, shield thickness calculations, activation calculations and mapping of dose rates for a particular layout. These design reviews and criteria conform with the guidelines of Regulatory Guide 8.8 (Revision 3) and the criteria of NUREG-0800 and are acceptable. Additional radiation protection criteria relevant to design features are discussed in Section 12.3

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12.1.3 OPERATIONAL CONSIDERATIONS

Operational considerations were factored into the design considerations previously described and were derived from experience from operating plants. Procedures pertaining to radiation safety for routine and non-routine activities are developed and recommended by health physics personnel to assure that occupational dose is kept ALARA. Consequently, the complexity for the performance of maintenance, repair, surveillance and refueling tasks will be factored into the radiation protection and control procedures to minimize radiation dose in accordance with Regulatory Guide 8.8. Some of these procedures include the use of portable shielding; localized ventilation; training on a mock-up of a component, equipment or structure; pre-operational briefing by health physics personnel; development and use of special tools; prefabrication of complicated equipment and structures and radiological surveillance during jobs to identify changing conditions. The staff, concludes that the policy considerations, design considerations and operational considerations are in accordance with Regulatory Guides 8.8 and 8.10 and NUREG-0800 and are therefore acceptable.

12.2 RADIATION SOURCES

The staff has audited the contained and airborne radioactive source terms provided in Section 12.2 and Chapter 11 of the Seabrook FSAR against the criteria set forth in NUREG-0800, Section 12.2. These source terms are used as inputs for dose assessment and for the design of the shielding and ventilation systems. The staff review consisted of ensuring that the applicant had either committed to [illegible] the

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criteria of the regulatory guides and staff positions referenced in Section 12.2 of NUREG-0800 or provided acceptable alternatives. In addition, the staff selectively compared source terms for specific systems used by the applicant against those used for plant's of similar design. This selective review found the plant's source terms equivalent to those used at other plants. Details of the review follow:

12.2.1 Contained and Airborne Radioactive Sources

The applicant has used radiation source terms for normal operations as inputs to shield design calculations, to determine personnel protective measures and to perform dose assessments, and determine access controls. Source terms used to perform a radiation and shielding review following an accident, in accordance with Section II.B.2 of NUREG-0737, has not been addressed in the FSAR and is an open item. Sources for normal operations include neutron and gamma fluxes outside the reactor vessel, coolant activities, and fission and corrosion products. During power operation, ^{16}N determines the shielding requirement of the secondary shield wall and portions of the chemical and volume control system. Source terms used for normal operation and anticipated operational occurrences are based on ANSI Standard N237 "Radioactive Materials in Principal Fluids Streams of Light-Water Cooled Nuclear Power Plants," and Regulatory Guide 1.112. We therefore find the contained sources used by the plant acceptable.

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Also sources of the maximum expected airborne concentrations during reactor operations and at shut down have been tabulated inside major plant buildups in frequently occupied areas due to equipment leakage based on typical data from operating plants. The assumptions and parameters used in determining these leakage calculations are also provided and found to be appropriate. Health physics and plant operating experience will be implemented to ensure that plant personnel will not be exposed to concentrations of airborne radioactive material exceeding those specified in 10 CFR 20.103 and will be maintained at levels that are as low as is reasonably achievable. In accordance with NUREG-0800, the source terms used to develop airborne concentration values are comparable to estimates by other applicants with similar design and are acceptable. With respect to radiation sources due to accident conditions, which is an area of review identified in the Standard Review Plan, the applicant has referenced Chapter 15 for accident sources such as Design Basis Accidents, Fuel Handling, and Radwaste System Failure. However, these accidents should not affect the design characteristics of the shield nor the ventilation system.

12.3 Radiation Protection Design Features

The staff has audited the facility design features, shielding, ventilation, and radiation and airborne monitoring instrumentation contained in the Seabrook FSAR against the criteria set forth in NUREG-0800 Section 12.3. The staff review consisted of ensuring that the applicant had either committed to following the criteria of the

regulatory guides and staff positions referenced in Section 12.3 of NUREG-0800 or provided acceptable alternatives. In addition, the staff selectively reviewed the applicant's FSAR against the specific areas of review and review procedures identified in NUREG-0800. This review found the plant acceptable in these areas. Details of the review follow.

12.3.1 Facility Design Features

The applicant has addressed facility and equipment design considerations, planning and procedure programs, and techniques and practices employed in the overall design for maintaining doses ALARA. The FSAR was reviewed with respect to:

(1) The description of equipment design to be used for assuring that occupational exposure will be ALARA.

(2) Information concerning implementation of Regulatory Guide 8.8 Section C.2.

(3) The description of any special protection features that use shielding, geometric arrangement or remote handling to reduce occupational exposure.

The applicant describes features including design for filters, demineralizers, adsorber beds, recombiners, tanks, pumps and associated piping with respect to shielding. All field run process piping potentially containing radioactive materials are positioned to limit

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exposure to plant personnel. Where necessary and practicable, interior surfaces of piping and ductwork are designed to minimize contamination buildup.

Equipment is placed so as to enhance removal operations to minimize exposure; flush and drain connections will enable chemical decontamination prior to maintenance of equipment; and sample stations are located in shielded accessible areas to minimize personnel exposure during sampling. Whenever permanent shielding is not feasible to install, and where shielding may be required, the design philosophy of the applicant, which emphasizes adequate space for ease of motion, would allow portable shielding to be used.

12.3.2 Shielding

The objective of the plant's radiation shielding is to provide protection against radiation for operating personnel both inside and outside the plant, and for the general public, during normal operation, anticipated operational occurrences and accidents. The shielding was designed to meet the criteria of a radiation dose zone system that is based on expected frequency and duration of occupancy. The design of the radition shielding will consider the dose rate criterion for each zone based on maximum access time estimates in each compartment within the zone. The health physics staff will update entry requirements in accordance with 10 CFR 20.203. Shielding analysis were made using accepted codes, models and assumptions. The basic shielding analysis was performed using computer codes accepted by the staff such as QAD,

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Morse, CG and SHIELD. Resides limiting exposure to plant personnel, contractors, visitors, etc., the plant shielding also functions to reduce neutron activation of equipment, piping supports etc. and to limit radiation damage to equipment and materials to below the specified integrated life dose limits. All concrete shielding in the plant is based on the criteria of Regulatory Guide 1.69 which provides the guidance on the fabrication and installation of concrete radiation shields. The applicant has not provided a copy of the plant shielding design review as specified in NUREG-0737, Item II.B.2 The staff has concluded that the applicant has performed a shielding design review in accordance with the criteria of NUREG-0800, except for that review required by NUREG-0737, Item II.B.2, which is an open item.

12.3.3 Ventilation

The applicant's ventilation systems are designed to provide ventilation air suitable to ensure that ~~airborne~~ plant personnel are not exposed to airborne concentrations exceeding those in 10 CFR 20.103, concentrations to which personnel may be exposed meet the requirements of 10 CFR Part 20. In the design of all ventilation systems, the applicant intends to meet this objective and maintain exposures ALARA by (1) directing the airflow from areas of lesser potential contamination to areas of greater potential contamination, (2) providing airborne radiation monitoring, (3) allowing adequate space around units for servicing and replacement, (4) providing for ease in maintaing filters to preclude additional radiation exposure. After initial operations, periodic testing for filters and adsorbers will be performed and

frequency of changeout determined as a result of these tests.

The design criteria are in accordance with the guidelines of Regulatory Guide 8.8 and the atmospheric cleanup units conforms to Regulatory Guide 1.52 with respect to occupational exposure. The staff concludes that the applicant's ventilation system is designed to maintain personnel exposures at a small fraction of 10 CFR Part 20 values, meets the criteria of NUREG-0800 and, therefore, is acceptable with the following exception. Location of ventilation system intakes have not been identified so that intake of potentially contaminated air from other building exhausts can be shown to be minimized. This is an open item.

ENCLOSURE 2

OPEN ITEMS RELATED TO ENCLOSURE 1

2.3.1 Justify design basis temperatures for safety-related auxiliary systems and components.

2.3.3 (1) Applicant response to RAI 451.14 is under evaluation by the NRC staff. (2) Upgrade the Meteorological Measurement Program to comply with NUREG-0737, item III.A.2 and NUREG-0654.

2.4 Respond to RAIs to 240.38 through 240.41.

2.5 (1) Evaluate strong motion records of recent seismic activity (New Brunswick and New Hampshire events) and update related documents, particularly the FSAR, as appropriate. (2) Commit to developing an inservice inspection program for revetments.

3.5.2 Clarify discrepency between FSAR Section 3.5.2 and 6.4 regarding protection of control room air intakes.

3.7.1 Provide information on soil amplification for ductbank analysis.

3.9.6 (1) Submit a program for preservice and inservice testing of pumps and valves. (2) Provide additional information on the periodic leak testing of pressure isolation valves.

4.2.3 (1) Confirm that the predicted cladding collapse time exceeds the expected lifetime of the fuel. (2) Provide supplemental ECCS calculations. (3) Verify fuel assembly mechanical responses meet the requirements of NUREG-0609, Appendix E.

Enclosure 3

Status of Review for the Remainder of the SER

Section 1 Incomplete sections require further progress for the entire review (1.7 Outstanding Issues, 1.8 Confirmatory Issues, 1.9 License Conditions).

Section 2 2.1 (Geography) and 2.2 (Nearby Facilities) are being prepared by the staff. There may be a small number of open items.

Section 3 3.1 (General) and 3.2 (Classification): the staff is evaluating applicant responses. 3.9.2 (Dynamic Testing), 3.10 (Seismic Qualification), and 3.11 (Environmental Qualification) applicant responses due.

Section 4 4.2.4 (Surveillance) and 4.5 (Reactor Materials): applicant responses due.

Section 5 5.2 (RCS Boundary Integrity (RSB, ASB. MTEB)), 5.3 (Reactor Vessel (MTEB)), 5.4 (Component Analysis): applicant responses due.

Section 6 6.1 (ESF Materials (MTEB)): applicant response due. 6.2 (Containment Systems): staff is evaluating applicant responses. 6.3 (ECCS): applicant responses due (RSB RAIs). 6.5 (ESF Atmospheric Cleanup): applicant responses due (ETSB concerns), staff is preparing AEB portion. 6.6 (Inservice Inspection): applicant needs to submit ISI program.

Section 7 Several applicant responses due.

Section 8 (Electric Power Systems) Several applicant responses received; several others due. Staff needs to develop a position on associated circuits proposal by the applicant.

Section 9 9.1 (Fuel Storage and Handling): applicant responses due. 9.2 (Water Supply): applicant responses received. 9.3 (Process Auxiliaries): applicant responses due for 9.3.1 (Compressed Air) and 9.3.4 (CVCS). 9.4 (HVAC): applicant responses received, 9.5 (Other Systems): three fire protection responses due; one fire protection response received. Several PSB responses received, several others due.

Section 10 10.2 (Turbine Generator): Several PSB responses received, others due. FSAR 10.2.3 (Turbine Disc Integrity) requires updating. 10.3 (Main Steam): applicant responses received. Small number of PSB responses due.

Section 11 (Radwaste Treatment): applicant responses due.

Section 12 (Radiation Protection): applicant responses due.

Section 13 13.2 (Training Program): applicant STA proposal is under evaluation. 13.3 (Emergency Planning): applicant responses received. Several milestones required. This is anticipated to be an open item in the initial SER. 13.4 (Review and Audit): Technical Specification due for NSARC; ISEG documents required. 13.5 (Plant Procedures): Technical Specification required; additional work required on O&M procedures; related Appendix 17A review is in progress. 13.6 (Physical Security): applicant responses due.

Section 14 (Initial Test Program): Several applicant responses received; other responses are due.

Section 15 (Accident Evaluation): Several responses received; others are due.

Section 16 (Technical Specifications): Specific requirements for Technical Specifications are incorporated into the related SER section.

Section 17 (Quality Assurance): Q-list response is being evaluated by the staff.

Section 18 (Control Room Design Review): Staff and applicant preparations are in progress.