Enclosure 6

UFSAR – Selected Sections

2.3 METEOROLOGY

2.3.1 Regional Climatology

2.3.1.1 General Climate (Reference 1)

The Seabrook site is located along the coast of New Hampshire about 2 miles inland from the open Atlantic Ocean. The site topography is generally flat and causes no special climatic phenomena.

New Hampshire lies in the prevailing westerlies, the band of winds aloft that blow from west to east. A large number of air mass fronts and storm systems pass through New Hampshire each year. There are three distinct types of air masses that affect the site area.

a. Cold, dry air originating in sub-arctic North America

b. Warm, moist air from the Gulf of Mexico or the subtropical Atlantic

c. Cool, damp air moving in from the North Atlantic.

As the prevailing flow aloft over New Hampshire is usually offshore, the first two types of air masses influence the site area more than the third. The climate of the site is thus continental in character, but with an important maritime influence.

The prevailing surface wind comes from a westerly direction, predominantly northwesterly during the winter and southwesterly in the summer. In spring and summer a sea breeze is usually established along coastal New Hampshire, often penetrating inland, well past the site.

Winter temperatures at the site are modified because of the proximity of the ocean water, which is relatively warm compared to winter air temperatures. For this reason, a good proportion of winter storm precipitation falls in the form of rain or wet snow. As an onshore breeze is often present on summer days, lower summer maximum temperatures are observed along the New Hampshire coast than are observed farther inland. Relative humidity is generally moderate at the site, and is lowest in late winter or early spring and highest in late summer or early fall.

Precipitation is uniformly distributed throughout the year. Low pressure, or frontal, storm systems are the principal year-round moisture producers. New Hampshire is subjected not only to storms that track across the continental United States, but also to intense winter storms, "northeasters," that move northeastward along the U.S. east coast. During the winter months northeasters can produce heavy rain or snowfall, and occasionally bring ice storm conditions to the area. During the summer, thunderstorms produce locally heavy rainfall amounts.

Occasionally during the summer or fall months, a storm of tropical origin will affect New Hampshire, but only a very few will retain near or full hurricane force. The site, therefore, may be affected by a hurricane, including associated heavy rainfall, high winds and high tides.

2.3.1.2 Regional Meteorological Conditions for Design and Operating Bases

a. Regional Climatological Data Stations

Figure 2.3-1 shows the locations of the Seabrook site and weather stations in the general area from which climatological data were obtained. The general location and type of data available from these weather stations are as follows:

Portland International Jetport National Weather Service Office (Portland NWS) This station is located about 59 miles north-northeast of the site just inland from the Atlantic Ocean and is a primary source of regional meteorological data for the site. The Portland NWS collects complete meteorological data on a continuous basis.

Boston Logan International Airport National Weather Service Office (Boston NWS) This station is located about 38 miles south-southwest of site on a landfill that extends into Boston Harbor, which is part of the Atlantic Ocean. It is a primary source of regional meteorological data on a continuous basis.

Concord Municipal Airport National Weather Service Office (Concord NWS This station is located 40 miles west-northwest of the site. The Concord NWS collects complete meteorological data on a continuous basis.

Pease Air Force Base Air Weather Service Station (Pease AFB) This station is located about 13 miles north-northeast of the site in Portsmouth.

Instrumentation information regarding the above offsite NWS and military weather stations is presented in Table 2.3-1.

Data from the following cooperative weather stations was also used:

Portsmouth, New Hampshire This station is located about 13 miles north-northeast of the site and is maintained by the Department of Public Works, a cooperative weather observer.

Rockport Massachusetts National Weather Service Climatological Station This station is located about 27 miles southeast of the site. This station collects daily maximum and minimum temperature and precipitation data.

Sanford Maine National Weather Service Climatological Station This station is located approximately 35 miles north of the site. Daily maximum and minimum temperature and precipitation data are recorded at this station.

Greenland New Hampshire National Weather Service Climatological Station This station is located about 7 miles north of the site and collects daily maximum and minimum temperature and precipitation data.

b. Regional Severe Weather Climatology

1. Hurricanes

Atlantic hurricanes are most common during late summer and early fall. During the period 1871-1977, approximately 43 tropical cyclones passed within 100 nautical miles (115 statute miles) of the site. Of these, 22 storms were classified as hurricanes, and only 3 retained full hurricane state within 100 nautical miles of the site (Reference 2).

Tropical storms or hurricanes that reach the New England area usually pass northward west of the site or on a northeast track south of the site. Since, to date, the only hurricanes or tropical storms to reach the Seabrook area have had to travel a substantial distance overland, the potential impact of such storms is significantly reduced. Potential impact is usually confined to the effects of high tides and heavy rainfall (Reference 1).

2. Tornadoes and Waterspouts

Tornadoes have occurred in all the New England States. The mean annual number of tornadoes per 10,000 square miles for the period 1953-1976 in New Hampshire, Maine and Massachusetts are 2.5, 0.8 and 5.2, respectively (Reference 3).

A National Severe Storms Forecast Center (NSSFC) listing of tornadoes within a 50-nautical mile radius of the site indicates that 69 tornadoes occurred during the period 1950 through 1977, with a mean path area of 0.124 square miles (Reference 4).

Thom (Reference 5) has developed a procedure for estimating the probability of a tornado striking any point from an analysis of mean path length and width and the frequency of tornado occurrence in the area. Applying Thom's procedure to the NSSFC data gives an annual probability of a tornado striking any point within 50 nautical miles (57.6 miles) of the site of less than 7.8×10^{-5} with a mean recurrence interval of greater than 12,900 years. The calculation excluded the water area within the area of interest.

In spite of the low probability of a tornado occurrence, seismic Category I structures at the Seabrook site, except for the refueling water tank spray additive tank enclosure and cooling tower, are designed to withstand the "Standard Tornado" as described in NRC Regulatory Guide 1.76 (Reference 6). This design basis tornado has the following characteristics:

(a) A maximum wind speed of 360 miles per hour

(b) A rotational speed of 290 miles per hour

(c) A maximum translational speed of 70 miles per hour

(d) A minimum translational speed of 5 miles per hour

(e) A radius of maximum rotational speed of 150 feet

(f) A pressure drop of 3.0 pounds per square inch

(g) A rate of pressure drop of 2.0 pounds per square inch per second.

In an analysis of waterspout occurrences using Storm Data Reports (1959-1973) and ship log reports (1850-1940), a total of 14 waterspouts was reported off the coast between Boston and Portsmouth of which 3 were considered to have caused coastal damage (Reference 7). A waterspout coming ashore and striking the site would not have a destructive effect greater than that of a tornado. This is based on the wind speed of a waterspout not being greater than the design basis tornado of Regulatory Guide 1.76. With exactly the same wind speeds, it is concluded that a waterspout would be less destructive than a tornado as it would contain less solid debris than a tornado that had been traveling overland.

3. Thunderstorms, Lightning and Hail

Table 2.3-2 shows the mean number of days with thunderstorms for various weather stations in the general Seabrook area. Thunderstorms have occurred during every month of the year, with the maximum during the summer. Pease AFB data can be considered most representative of the Seabrook site, showing a thunderstorm frequency of about 19 per year with a maximum monthly mean of about 5 in July (Reference 11).

Using the thunderstorm frequencies shown in Table 2.3-2 for Pease AFB and statistics relating to thunderstorm occurrence and to the probability of cloud-to-ground lightning as presented by Viemeister (Reference 12), estimates of the frequency of occurrence of cloud-to-ground lightning were derived for the site on a seasonal and annual basis for objects extending to heights of 50, 100, 200 and 500 feet above grade. These results are provided in Table 2.3-3.

Marshall (Reference 13) presents an alternative methodology for estimating lightning strike frequencies which includes consideration of the attractive area of structures. Marshall's method consists of determining the number of lightning flashes to earth per year per km^2 and then defining an area over which the structure can be expected to attract a lightning strike. Assuming that there are 0.135 flashes to earth per thunderstorm days per km^2 near the Seabrook site (Reference 13) and that the Seabrook site experiences 19 thunderstorm days per year (Pease AFB data, Table 2.3-2), there are approximately 2.57 flashes to earth per year per km^2 around the Seabrook site area. If the length of a structure is L, its width W, and its height H, Marshall defines the total attractive area A of that structure for lightning flashes with a current magnitude of 50 percent of all lightning flashes as:

$$A = LW + 4H(L + W) + 12.57\,H^2$$

The following building complex dimensions were used to conservatively estimate the attractive area for the Unit #1 building complex:

$$L = 200m,\ W = 120m$$

Defined roughly by a rectangle outlined by the Turbine Building, Administrative Building, Fuel Storage Building, and containment structure.

$$H = 56m$$

Defined by the height of the primary vent stack.

$$A = 0.135\ km^2$$

Consequently, the lightning strike frequency computed using Marshall's methodology is given as 0.35 flashes/year, for the Unit #1 building complex.

Table 2.3-4 lists the total number of days with hail over a 40-year period for Boston, Portland and Concord. The data indicate that, on the average, the site should expect less than one day per year with hail (Reference 14). Hailstorms in the Seabrook area are seldom severe, although large hail has been reported. During the 13-year period between 1955 and 1967, an average of 0.2, 0.6 and 1.3 storms per year with hailstones 1.5 inches in diameter or larger have been reported for New Hampshire, Maine and Massachusetts, respectively (Reference 15).

4. Strong Winds

Table 2.3-5 lists the fastest mile wind speeds recorded at Boston, Portland and Concord. The data indicate that wind speeds over 40 mph can occur during any month of the year. During the winter these speeds are normally caused by northeasters that move up along the coast. During the warmer months, high winds are normally associated with thunderstorms and squall lines that pass through the area. Hurricanes could produce high wind speeds during the late summer and early fall.

Thom (Reference 16) plotted isotachs of annual extreme-mile wind speeds at 30 feet above ground for several recurrence intervals across the United States. The annual extreme-mile wind speed derived from Reference 16 for the Seabrook site indicates a sustained 95 mph wind speed can be expected with a 100-year recurrence interval. Other studies (References 17, 21), which also plotted isotachs for fastest mile of wind at 30 feet above ground across the United States, indicate a fastest mile of wind for a 100-year probable period of recurrence of 110 mph and 100 mph for the Seabrook site. These studies (References 16, 17, 21) were published in 1968, 1961 and 1972, respectively.

The more conservative value of 110 mph was used as the 100-year period of occurrence design wind velocity for seismic Category I structures at 30 feet above ground. The vertical wind velocity profile and the appropriate gust factor used for seismic Category I structure wind loading analyses are discussed in Subsection 3.3.1.

More recently published extreme wind speed-probability studies (References 50, 51) provide more reliable information since they are based on statistical analyses of the longer available data bases and Monte Carlo simulations of hurricanes. Table 2.3-6 shows the fastest-mile extreme wind speed derived from References 50 and 51 for various recurrence intervals at 10-meters and 30-meters above grade for the Seabrook site.

5. Snowload

The American National Standards Institute, Inc., (ANSI) gives the 100-year recurrence interval snow load on the ground in the Seabrook area as 42 pounds per square foot (Reference 21). The maximum 24-hour precipitation amount observed in the site during the snow season (November through April) is 5.4 inches of water, as shown in Table 2.3-23. From this value, a conservative 48-hour probable maximum snowfall is defined as having twice the water content of the maximum 24-hour storm, or 10.8 inches. As required by Regulatory Guide 1.70 (Reference 22), the Probable Maximum Winter Precipitation was determined, which resulted in a 48-hour precipitation of 16.1 inches (Reference 23). Assuming this amount of precipitation fell on top of the 100-year recurrence interval snowpack of 42 psf, as given by ANSI, it would result in a compacted snow load of 125.7 psf. This is considered an "unusual" load condition as described in Chapter 3. Roof loading for safety-related structures due to precipitation, including ice, snow and rain, are discussed in Subsection 2.4.2.3.

The February 6-8, 1978 snowstorm which struck New England was one of the most intense, persistent, severe winter storms on record (Reference 24). The highest melted precipitation associated with the storm was 4.55 inches reported at Pembroke, Massachusetts (Reference 18). The New England climatologist, Robert E. Lautzenheiser, had previously stated that the February 23-28, 1969 snowstorm was probably the worst storm in 100 years. The highest melted precipitation associated with that storm was 4.62 inches reported at Rockport, Massachusetts. The 4.62 inch value is equivalent to a snowfall load of 24 psf. When this is combined with the 100-year probable maximum snowpack of 42 psf, it results in a total snow load on the ground of 66 psf.

6. Ice Storms

Freezing precipitation, or glaze ice, does occur in the Seabrook area. Data for freezing rain at Portsmouth (Reference 25) are presented in Table 2.3-7. Mapped data for the period 1928 to 1937 indicate that the site averages 2-3 ice storms per year. For the nine-year period of study, about 12 storms occurred resulting in ice with a thickness of 0.25 inch or more, of which about 6 storms had ice of 0.5 inch or more (Reference 26). More recent mapped data for the period of 1950 to 1969 (Reference 27), indicates that the site averages about 8 ice storms per year.

7. High Air Pollution Potential and Mixing Heights

The Seabrook site is not in an area of frequent air pollution episodes or alerts. A study of synoptic weather map analysis for 1936 through 1975 shows high pressure stagnation conditions lasting four days or more over the site occurring 12 times with an average of 4.4 stagnation days per case (Reference 28).

Holzworth (Reference 29) analyzed five years of data to determine occurrences in the United States of episodes of meteorological conditions unfavorable for atmospheric dispersion. Holzworth indicated episodes of high air pollution potential as periods with low mixing depth and light winds. A summary of the Holzworth data as it applies to the site appears in Table 2.3-8. The data indicate that prolonged periods with a combination of low wind speed and low mixing height are uncommon in the site area.

Holzworth (Reference 29) also plotted isopleths of mean seasonal and annual morning and afternoon mixing heights across the United States from the same five years of data. For the Seabrook site, the seasonal and annual values of the mean daily mixing heights occurred as follows:

Mean Daily Mixing Heights

Season	Morning	Afternoon
Spring	710 m	1400 m
Summer	450 m	1400 m
Autumn	590 m	1100 m
Winter	700 m	900 m
Annual	600 m	1200 m

The above data represent estimates of the average depth of vigorous vertical mixing, which give an indication of the vertical depth of atmosphere available for mixing and dispersion of effluents.

8. Ultimate Heat Sink

Data collected at Pease AFB for the 25-year period (1956-1974 and 1976-1981) were used to evaluate the performance of the Ultimate Heat Sink with respect to maximum evaporation and drift loss, water cooling capacity, and basin water freezing during extreme winter weather.

Maximum evaporation and drift loss will occur during periods of large differences between the ambient dry bulb and wet bulb temperatures with accompanying high wet bulb temperatures. A seven-day (168 hours) averaging period is appropriate for analysis of water inventory. The maximum 168-hour average wet bulb temperature calculated from the Pease AFB data base was found to be 73.93°F and was accompanied by a concurrent dry bulb average temperature of 77.87°F. This condition occurred during the seven-day period ending on July 18, 1972 at 1500 LST. To assure that evaporative water losses would not be underpredicted, they were computed utilizing the seven-day average wet bulb temperature of 74°F while conservatively assuming a concurrent average dry bulb temperature of 85°F (see Subsection 9.2.5).

Minimum heat transfer to the atmosphere occurs during periods of high wet bulb temperature. Regulatory Guide 1.27, Revision 2, suggests that the worst-case meteorological conditions for the parameters of concern be used. In the post-LOCA plus seismic event case where the tower is used, primary component cooling water temperature undergoes a four-hour transient rise to a peak, then decreases. This case is the limiting parameter for the tower and requires the analysis of four hour average wet bulb temperatures. Four-hour average wet bulb temperatures were computed on a running average basis from the Pease AFB data base and occurrence frequencies were tabulated. The results of this analysis for average wet bulb temperatures greater than 65°F occurring during the "summer" period (beginning June 1 and ending September 15) are presented in Table 2.3-9. The frequency of occurrence of four-hour average wet bulb temperatures greater than or equal to 75°F during the "summer" period of the data base may be seen to be 1,275 occurrences. This corresponds to a probability of 2.012×10^{-2} of experiencing a four-hour average wet bulb temperature equalling or exceeding 75°F during any given four-hour period of the summer. This probability is used in evaluating the acceptability of cooling tower performance (see Subsection 9.2.5).

Additional meteorological information was used in establishing operating restrictions on the cooling tower as described in Subsection 9.2.5. This information consisted of frequency distributions of nine-hour average wet bulb temperatures for the second half of the month of July occurring during the hours of 7:00 PM to 9:00 AM (corresponding to ending hours of nine-hour averages from 3:00 AM to 9:00 AM). These frequencies are presented in Table 2.3-10.

Basin water freezing, as well as measures to mitigate this wintertime meteorological effect, were evaluated with respect to the occurrence of extreme low dry bulb temperatures (see Subsection 9.2.5). The meteorological parameters of most concern in this analysis are dry bulb temperature and wind speed. The number and lengths of contiguous periods of extreme loss (<15°F) 24-hour average dry bulb temperatures were identified and occurrence frequencies computed. These data are presented in Table 2.3-11. A conservative wind speed assumption of 20 mph was used in conjunction with this analysis.

2.3.2 Local Meteorology

2.3.2.1 Normal and Extreme Values of Meteorological Parameters

Monthly and annual summaries of meteorological parameters from long-term data stations representative of the area are presented in this section. Summaries of onsite meteorological data collected at Seabrook from November 1971 through March 1973 are also provided in this Section and in Appendix 2A. A new onsite meteorological tower has been erected at the same location as the old tower and became fully operational in April 1979. Data summaries from this new tower for the time period April 1979 through March 1980 are presented in Appendix 2B. Severe weather data (snowfall, tornadoes and wind velocities 75 to 125 mph) and extremely severe weather data (wind velocities greater than or equal to 125 mph) were used to determine an estimated frequency of loss of offsite power due to weather as input into the Station Blackout analysis (see Section 8.4.2).

a. Wind

Wind roses for the four seasons and 12-month period (November 1971-October 1972) of collected onsite data are provided in Figure 2.3-2, Figure 2.3-3, Figure 2.3-4, Figure 2.3-5 and Figure 2.3-6, respectively. The data indicate that westerly through northwesterly winds predominate during most of the year. During the summer months, southwesterly through west-northwesterly, and east-southeasterly through south-southeasterly winds are prevalent. Wind direction persistence summaries for 22.5 and 45.0 degree sectors are presented in Appendix 2A.

Seasonal and 12-month period wind roses collected onsite from the new onsite meteorological tower (April 1979-March 1980) are provided in Appendix 2B. Wind direction persistence summaries for this same period are also provided in Appendix 2B.

b. Temperature

Table 2.3-12, Table 2.3-13, Table 2.3-14, Table 2.3-15, Table 2.3-16 and Table 2.3-17 present long-term mean and extreme temperature values for a number of stations in the Seabrook area. Portsmouth data can be considered representative of long-term Seabrook temperatures. Monthly onsite mean and extreme temperature values for the time period April 1979 through March 1980 are presented in Appendix 2B.

Extremes of temperature are uncommon due to the proximity of the site to the Atlantic Ocean. During the winter, arctic air masses passing through New England can produce low minimum temperatures, but the frequency and persistence of such extreme values along the coast is less than for stations located farther inland. During the spring and summer a seabreeze usually moderates temperatures from reaching high extremes at the site.

Detailed analyses have determined that the highest hourly temperature recorded during the period 1957 through 1981 at Pease AFB (Portsmouth, NH) was 101°F on July 1, 1964 (hour 13). The hottest contiguous 24-hour period containing this temperature extended from June 30 (hour 15) through July 1 (hour 14). The hourly temperature progression for this period is provided in Table 2.3-18. Hourly temperature data associated with the five hottest and five coldest 24-hour average temperatures recorded at Pease AFB in the period 1957 through 1981 are given in Table 2.3-19.

An additional statistical analysis (Reference 30) of extreme temperature data collected at nearby weather stations (Pease AFB), climatological stations (Rockport, MA, Sanford, ME, and Greenland, NH) and at the Seabrook site results in 100-year return period maximum and minimum hourly temperatures for the Seabrook site of 102°F and -21°F, respectively. (These values were computed following the methodology found in NUREG/CR-1390.)

Since the design of certain equipment is dependent upon the maximum and minimum temperatures averaged over time periods greater than one hour, 100-year return period extreme temperatures for 2, 4, 8, 12, and 24-hour averaging periods were also determined. These values are listed below:

	100 Year Return Period Temperature (°F)	
Averaging Period	Maximum	Minimum
2-Hour	102	-21
4-Hour	101	-21
8-Hour	99	-20
12-Hour	96	-19
24-Hour	89	-16

c. Atmospheric Water Vapor

Long-term mean monthly relative humidity statistics at Pease AFB are provided in Table 2.3-20. Onsite dew point statistics for the period April 1979 through March 1980 are provided in Appendix 2B.

Joint frequency distributions of the onsite moisture deficit have been prepared for each stability category and wind direction.

Those data from April 1972 through March 1973 are presented in detail in Appendix 2A.

Outdoor relative humidity extremes of 10 percent minimum and 100 percent maximum were used in the design of the HVAC systems for all safety-related buildings. The basis for the selection of this humidity range is the assumption that relative humidities at or near 100 percent occur during fog, dew formation and precipitation that are frequently observed in this climate. Relative humidities less that 10 percent are not observed under the climatic conditions affecting the site.

d. Precipitation

On the average, the Seabrook area has about 129 days per year with measurable (0.01 inch or more) precipitation, as indicated in Table 2.3-21. Table 2.3-22, that shows mean monthly and annual precipitation amounts, indicates monthly precipitation is equally distributed over the year, with mean monthly amounts generally between 2.7 to 4.6 inches. The site can expect an annual precipitation of about 43 inches.

Summer rainfall is caused primarily by thunderstorms and convective shower activity. Precipitation during the rest of the year generally results from the passage of low pressure systems. During the colder months of the year, intense coastal storms or northeasters move north-eastward along the New England coast, usually affecting coastal locations with heavy rain or snow and on occasion, ice storm conditions. Occasionally during the summer or fall, a storm of tropical origin will cause substantial rainfall and high winds in the vicinity of the site.

Precipitation extremes for area stations are presented in Table 2.3-23, Table 2.3-24, Table 2.3-25 and Table 2.3-26. Based on the Portsmouth data, a maximum monthly precipitation amount of about 14 inches and a maximum 24-hour precipitation amount of about 7 inches could be expected at the site.

While periods of prolonged drought are not common, dry spells do occasionally occur. March 1915 and October 1924 were particularly dry, as indicated in Table 2.3-26.

Snow falls in the site area as early as November and as late as April. Mean snowfall statistics for the area, Table 2.3-27, indicate that the site can expect an annual snowfall of about 72 inches. Maximum snowfall data are presented in Table 2.3-28 and Table 2.3-29, which suggest a maximum 24-hour snowfall of about 22 inches and a maximum monthly snowfall of about 54 inches, based on Portsmouth data.

The ground is normally covered with snow from late December until well into March, although it may remain bare for several weeks during this period in a milder winter. A continuous snow cover of at least one inch lasts 30 to 45 days in a usual winter, but continued for 87 days in the snowy winter of 1955-1956. The average maximum snow depth is about 18-24 inches (Reference 25).

e. Fog

The proximity of the ocean is an important factor in fog occurrence at the site. During the spring and summer months, fog forms offshore as warm, moist air flows over the relatively cold ocean water. With any persistent eastern component in the wind direction, the fog that often lies just offshore during the warmer months can reach the Seabrook site. This situation is supported during the summer by local heating and a resulting seabreeze.

Table 2.3-30 provides information on the mean number of days with heavy fog at surrounding stations. Based on Pease AFB data, Table 2.3-31, all months of the year have a fairly consistent frequency of occurrence of fog. Although fog at Pease AFB occurs about 15 percent of the time, it is dense enough to restrict visibility to 1 mile or less only about 3.5 percent of the time (Reference 25). Table 2.3-32 lists the mean number of hours with visibility less than 0.5 miles.

Statistics on fog persistence at Portland are presented in Table 2.3-33 for the 10-year period (1968-1977). This table indicates that durations of periods of fog lasting 48 hours or longer can occur several times a year.

f. Atmospheric Stability

Joint frequency distributions of Pasquill stability class by the temperature difference (delta T) method are presented in Appendix 2A and Appendix 2B. Summaries of atmospheric stability persistence are also provided in both Appendices. The onsite data from the new meteorological tower indicate that from April 1979 through March 1980 unstable, neutral, and stable conditions occurred as follows:

Frequency of Stability Classes

Stability Classification	150'-43' Delta-T	209'-43' Delta-T
Unstable (A,B,C)	21.1 percent	12.7 percent
Neutral (D)	41.5 percent	43.3 percent
Stable (E,F,G)	37.3 percent	44.0 percent

2.3.2.2 Potential Influence of the Plant and Its Facilities on Local Meteorology

A map is presented in Figure 2.3-7 which shows the topography within a 5-mile radius of the site. Maximum elevation with distance is plotted in Figure 2.3-8 for each of 16 sectors radiating from the plant site. The heights shown in these cross sections are for the highest representative terrain at that distance in the sector, and not necessarily the exact height at the precise bearing and distance shown.

The immediate site area is tidal marsh with short grass, reeds and tidal channels. Short trees begin at the edge of the marsh as the terrain becomes slightly irregular. A few short ridges and hills occur within the first 5 miles of the site.

A map showing detailed topographic features within a 50-mile radius of the site is presented in Figure 2.3-1. The first hills and ridges of the White Mountains of New Hampshire occur 20-25 miles northwest, west and southwest of the site. Hilly terrain with peaks between 200 and 500 feet are found 25 to 40 miles from the site.

The plant is not expected to cause any significant influence on the local meteorology as cooling towers or spray ponds are not planned for normal operations.

2.3.3 Onsite Meteorological Measurement Program

2.3.3.1 Pre-Construction Program

An instrumented meteorological data tower was erected at the Seabrook site and operated from November 1971 until June 1974 for plant construction licensing purposes.

The tower was 150 feet high with a base at approximately 10 feet MSL, and was located near the south edge of the Browns River, about 700 feet east of the railroad. The location of the tower relative to the site is shown in Figure 1.2-1.

There were no trees or other vertical obstructions in the immediate vicinity of the tower site. The nearest significant growth was 25-35 foot trees about 500 feet to the west and south-west of the tower. There was no significant vegetation between the tower site and Hampton Harbor. Grass was planted under the tower out to a radius of 50 feet to assure conservative delta T data.

The tower was instrumented as shown in Table 2.3-34. After one year of data accumulation, the original Aerovane wind system at 30 feet was replaced with a Bendix 3-cup anemometer and vane system. Another Bendix wind system was installed at the 130-foot level at the same time.

Wind data were recorded on Bendix Model 141-2 dual strip chart recorders. The temperature systems used Rosemont precision resistance bridges and recorded on an Esterline-Angus multi-channel recorder. One channel of the recorder was used to print a reference value of 0 volts from which all traces were calibrated.

The temperature sensors were installed in aspirated shields on the tower. The vertical temperature difference (delta T) was measured between 30 and 130 ft. The system was scaled for a range of from -10°F to +18°F, for a full span scale of 28°F. The Rosemount platinum resistance sensors and bridge system had an accuracy of 0.1 percent of span or ±0.02 ohms, whichever was greater; the maximum possible system error therefore was ±0.09°F. The recorder accuracy was +0.25 percent of scale, or +0.07°F. As a result, the maximum delta T system error could be ±0.16°F, with a probable system error of ±0.11°F.

All equipment was checked for normal operation before installation on the tower. At that time, the delta T system was calibrated to a 0.0°F value by means of a simultaneous ice bath of both sensors. All laboratory tests were made with each sensor permanently connected to the cable to be used with the sensor on the tower.

The dew point sensor was installed on the tower in March 1972. Recorded dew point data had been verified by bi-weekly multiple sling psychrometer readings taken at the 30-foot level on the tower.

Occasional minor adjustments to the recorded dew point data had been made to maintain the data within an accuracy of ±0.5°C.

Data recovery rates for individual tower parameters are given in Table 2.3-35. This table shows that the Seabrook meteorological program satisfied the 90 percent data recovery specified in Regulatory Guide 1.23 (Reference 32).

In addition to bimonthly meteorological strip chart review, every three months recorded temperatures were checked against tower value obtained with ASTM precision thermometers. Wind systems were checked for trouble-free operation every three months. Wind direction and speed transmitters were removed from the tower and given a complete laboratory check at least every six months to assure they were working within the manufacturer's specifications.

Processing of the onsite meteorological strip charts was as follows. For hourly data values, a mean value for the 30 minutes preceding the hour was determined directly from the strip charts. This value was transferred to a punched card by means of a Gerber semi-automatic analog-to-digital converter. The punched cards were checked by computer for consistent values from one hour to the next. After all checks were verified, a punched card was prepared that contained the date, time and hourly values for all the parameters measured on the tower. These cards were used to prepare the data summaries in Appendix 2A.

2.3.3.2 Preoperational Program

A new 210-foot high instrumented meteorological data tower was erected at the same location as the old tower to collect data for plant operating licensing purposes. This new tower became operational in April 1979.

The meteorological tower was instrumented for wind measurements at heights of 13.1 meters (43 feet), and 209 feet above the base. The tower is located at an elevation of approximately 10 feet MSL, and as such, the low-level wind and temperature sensors were approximately 53 feet MSL. Since plant grade is 20 feet MSL, the low-level sensors were located at an elevation of approximately 10m above plant grade rather than 10m AGL. The difference in values measured at 33 feet (10m) AGL versus 43 feet AGL on the meteorological tower should not be significant. Wind speed and direction were observed by Climatronics F460 wind systems which had a starting speed of less than 1.0 miles per hour. Wind direction and speed were recorded on Esterline-Angus Model LllS2S strip chart recorders.

The ambient temperature difference was measured on the tower between 150 and 43 feet and between 209 and 43 feet. These data were obtained by Rosemount platinum temperature sensors and precision resistance bridges and recorded on an Esterline-Angus Model E1124E multi-channel recorder. Ambient temperature was also measured by this system for the 43-foot level. The temperature and delta T sensors were installed in Teledyne Geotech aspirated shields.

Dew point was initially measured at the 43-foot level on the tower by a General Eastern Model 1200 APS dew point system. The General Eastern dew point system was replaced in May 1981 with a Climatronics lithium chloride dew point system. Dew point data was also recorded on the Esterline-Angus E1124E multi-channel recorder.

A heated tipping bucket precipitation gauge and an Eppley pyranometer were also installed.

A digital recording system was the primary data collection mechanism for the Seabrook Meteorological System. A MODCOMP minicomputer scanned the wind parameters at 1-second intervals and all other parameters at 5-second intervals. The data were compiled as 15-minute averages. Four 15-minute averages per hour were recorded. The first 15-minute average for each hour was used to represent that hour's data in analytical computer programs. The analog strip charts were used as a backup source of data and for quality control analysis.

Table 2.3-36 presents the equipment components, performance specifications, and system accuracies for both the analog and digital data systems. Presented values are summaries from manufacturer's specification sheets.

All equipment was checked and calibrated before installation. The delta T and temperature systems were calibrated by means of constant-temperature baths.

All equipment calibration was performed with the sensor connected to the cable to be used with the sensor on the tower.

Preventive maintenance and complete calibration checks of the temperature, delta T and dew point systems were performed per proposed Technical Specification requirements. Wind transmitters were removed from the tower every six months and tested in a low-speed wind tunnel for normal operation and a starting speed of less than one mile per hour.

The sensors and data processing procedures met the requirements for time averaged values as specified in NRC Regulatory Guide 1.23.

2.3.3.3 Operational Program

a. Primary Meteorological Monitoring System

The 210-foot meteorological tower structure used for plant operating licensing purposes continues to be used during plant operation. The monitored parameters include those required by plant Technical Requirements and those used to perform dose projections for both routine and accidental atmospheric releases.

Wind speed and direction are monitored with cup anemometers and wind vanes mounted on booms located at 43 feet and 209 feet above ground level. The ambient temperature difference between the 150- and 43-foot levels and the 209- and 43-foot levels are measured with temperature probes housed in motor-aspirated temperature shields. Ambient temperature is also measured at the 43-foot level. A heated tipping bucket precipitation gauge and a pyranometer collect data near the base of the tower.

The Main Plant Computer System (MPCS) monitors the meteorological tower parameters. The wind speed, wind direction, and delta-temperature signals are sampled, averaged, and recorded. These data are available for on-demand display on MPCS terminals located in the control room, Technical Support Center (TSC), and Emergency Operations Facility (EOF).

All parameters are also recorded on recorders located in the Instrument Building near the tower's base. These recorders are used as a backup source of data and for quality control analysis. Backup power is supplied from the station's Train A diesel generator to the equipment at the meteorological tower and the main plant computer system. Table 2.3-37 presents the equipment ranges and performance requirements. A daily channel check of the wind and delta-temperature instrumentation is performed in accordance with station Technical Requirements to demonstrate channel operability. Corrective action is initiated if any of the meteorological instrumentation is determined to be malfunctioning.

The equipment preventative maintenance and calibration activities conducted during the preoperational monitoring program continue during the operational monitoring program. These activities occur at a frequency compatible with the station's Technical Requirements.

b. Backup Meteorological Monitoring System

A backup to the Primary Meteorological Monitoring System instrumentation is provided by an independent tower located approximately 200 feet southeast of the primary meteorological tower. The backup tower has wind speed and wind direction sensors mounted on a crossarm located at Elevation 53 ft MSL, the same elevation as the lower sensors on the primary tower. System accuracy is adequate to support the use of these data in the Radiological Emergency Plan.

The following are calculated and displayed in the main control room: average wind speed, average wind direction, wind direction standard deviation and atmospheric stability (determined as a function of wind speed, wind direction standard deviation, and time-of-day.) In addition, these four parameters are provided as input into the MPCS. The backup meteorological monitoring system is provided with power sources that are different than those used on the primary meteorological monitoring system. No one failure of the power sources or instrumentation associated with the primary or backup meteorological monitoring systems will prevent the availability in the control room of the meteorological information required to support the Radiological Emergency Plan.

Preventative maintenance and calibration are performed at the same frequency as the primary meteorological monitoring system.

2.3.4 Short-Term (Accident) Diffusion Estimates

2.3.4.1 Objective

Conservative and realistic estimates of atmospheric diffusion at the site boundary and the outer boundary of the low population zone (LPZ) were culated for appropriate time periods using meteorological data collected onsite during the time period April 1979 through March 1980.

2.3.4.2 Calculations

Gaussian diffusion models were used to compute estimates of the local atmospheric dilution factors for the exclusion area boundary and the low population zone using hourly meteorological data collected at the Seabrook site. Two sets of dilution factors (CHI/Q values) were calculated: (1) a concentration dilution factor for evaluating ground level concentrations of noble gases, tritium, carbon 14 and nonelemental iodines, and (2) an effective gamma dilution factor for evaluating gamma dose rates. Cumulative probability distributions were prepared for both types of dilution factors to determine the appropriate dilution factors for a number of time intervals ranging from 1 to 720 hours.

a. Concentration CHI/Q Equations

Hourly concentration dilution factors were computed for ground-level releases using both the plume centerline model and the sector average model. Plume centerline values are for estimating short-term atmospheric dispersion (up to 8 hours) and sector average values are for dispersion during longer periods of time. The formulae and assumptions are as follows:

1. Plume Centerline Model

During neutral and stable atmospheric conditions (i.e., for Pasquill stabilities D, E, F and G) and low wind speeds (less than 6 mps, atmospheric dispersion is computed by the following equations (Reference 34):

$$CHI/Q = \frac{\xi_M}{\pi \bar{u} M \sigma_y \sigma_z} \qquad (1)$$

$$CHI/Q = \frac{\xi_G}{\pi \bar{u} \Sigma_y \Sigma_z} \qquad (2)$$

and the lesser value is used. In these equations,

CHI/Q = the relative concentration (sec/m^3)

$\bar{u}$ = the wind speed at 13.1 meters above grade (m/sec)

σ_y = the lateral plume standard deviation (m)

σ_z = the vertical plume standard deviation (m)

Σ_y = the lateral plume standard deviation corrected for building-wake effects (see Eq. 3)

Σ_z = the vertical plume standard deviation corrected for building-wake effects (see Eq. 4)

M = the meander factor (a function of atmospheric stability and wind speed), and

ξ_G, ξ_M = correction factors that account for increases in ground-level concentration due to multiple eddy reflections from the ground and from the stable atmospheric layers aloft (see Eq. 6).

The adjusted standard deviations Σ_y and Σ_z, which account for the additional dispersion of the effluent plume within the wake caused by buildings adjacent to the release point, are defined as

$$\Sigma_y = \left(\sigma_y^{\,2} + 0.5\ A/\pi\right)^{1/2} \tag{3}$$

and

$$\Sigma_z = \left(\sigma_z^{\,2} + 0.5\ h_B^{\,2}/\pi\right)^{1/2} \tag{4}$$

where

h_B = the height of the building causing the additional dispersion (m), and

A = the building's smallest vertical cross-sectional area (m^2).

The maximum values of Σ_y and Σ_z are restricted by the conditions:

$$\left(\Sigma_y\right)_{max} = \sqrt{3}\sigma_y$$

and

$$(\Sigma_z)_{\max} = \sqrt{3}\sigma_z$$

Dependence of the meander factor on wind speed and atmospheric stability is shown in Figure 2.3-10 for distances up to 800 meters. Beyond this distance use is made of the adjusted meander factor defined as:

$$M' = (M-1)\left[\frac{\sigma_y(800)}{\sigma_y}\right] + 1 \tag{5}$$

where σ_y (800) is the lateral standard deviation of the plume at 800 meters. Note that for unstable conditions (stabilities A, B and C) and for wind speeds greater than 6 mps (independent of stability) the meander factor is equal to unity. For these conditions, atmospheric dispersion is based entirely on Eq. (2).

Definition of the reflection correction factors ξ_G and ξ_M was based on the plume trapping equation in the U.S. EPA workbook on atmospheric dispersion (Reference 35). The EPA equation was reduced to the form

$$\xi_G = \sum_{j=-n}^{n} e^{-(\gamma j)^2} \tag{6}$$

and

$$\xi_M = \sum_{j=-n}^{n} e^{-(\beta j)^2} \tag{7}$$

where

$$\gamma = \frac{\sqrt{2}\,L}{\Sigma_z} \tag{8}$$

and

$$\beta = \frac{\sqrt{2}\,L}{\sigma_z} \tag{8A}$$

L is the height of the reflection layer and 2n is the total number of reflections. The value of n was conservatively set equal to 6; n = 3 or 4 is normally sufficient to include the important reflections.

In the case of large plume standard deviations, where multiple reflections occur and uniform vertical mixing has taken place, the equations reduce to simpler forms. For instance,

$$\lim_{\Sigma_z \to \text{large}} (\xi_G) = \int_{-\infty}^{\infty} e^{-\tau 2q^2}\, dq = \frac{\Sigma_z}{L}\sqrt{\pi/2} \tag{8B}$$

and is achieved for Σ_z greater than approximately 2L. In most cases, where $\Sigma_z << L$, ξ_G and ξ_M are equal to unity.

2. Sector-Average Model

Atmospheric dispersion during intervals greater than 8 hours was based on the sector average model,

$$(\text{CHI/Q})_{sa} = \frac{2.032\,\xi_G}{X\bar{u}\Sigma_z} \tag{8C}$$

where X is the distance from the release point to the receptor. Note that when Eq. (8B) is substituted into Eq. (8C) the latter reduces to

$$(\text{CHI/Q})_{sa} = \frac{2.55}{X\bar{u}L} \tag{8D}$$

which is the familiar form of the sector-average dilution equation with uniform vertical mixing.

b. Effective Gamma CHI/Q Equations

Hourly effective gamma dilution factors were computed for ground level releases according to a sector average finite cloud model. The assumptions and formula used are described in Subsection 2.3.5.2.

c. Dilution Factors

Both concentration CHI/Q values and effective gamma CHI/Q values were computed by the above models for each sequential hour of measured meteorological data.

Evaluation of each hourly dilution factor was based on the average wind speed and the vertical temperature gradient indicated in the meteorological data.

A limited mixing layer depth L of about 900 meters was determined by calculating the average of the mean annual morning mixing height and the mean annual afternoon mixing height for the Seabrook site area (Reference 29).

Values for σ_y and σ_z were computed by applying parabolic interpolation (on a log-log basis) to tabular data of these parameters versus distance. The data were extracted from the Pasquill-Gifford curves for atmospheric stabilities A through G (Reference 34). σ_z values were restricted to a maximum of 1000 meters.

Building-wake effects were computed using a building cross-sectional area of 2090 sq. meters and a building height (h_B) of 54.8 meters.

The hourly dilution factors obtained as described and the corresponding direction in which the wind was blowing during each hour were then stored in sector-dependent arrays for sequential processing. This involved the averaging of selected hourly CHI/Q values over successive, overlapping time intervals of 1, 2, 8, 24, 96 or 720 hours, (the last five intervals correspond to the time periods: 1 to 2 hours, 2 to 8 hours, 8 to 24 hours, 1 to 4 days, and 4 to 30 days, as specified in Regulatory Guide 1.4).

For each selected interval size, the processings began with the first hourly CHI/Q value on record, and was then repeated for the same interval size starting with each subsequent hour of CHI/Q data. In the averaging process, the only CHI/Q values within a given time interval that were considered in evaluating the mean dilution factor for the interval were those for the specific wind direction being analyzed. Missing data were handled by imposing the condition that at least half of the entries within an averaging interval corresponded to valid observations. Missing data points were not included in the averaging.

As an illustrative example, consider a 4-hour interval and the following sequence of hourly wind directions: WWWWSMWMMWSSSSS. In this sequence, W is for the west sector, S is for the south sector and M represents missing data. Assuming that each hourly dilution factor is equal to unity, and limiting the total number of valid observations per averaging interval to at least 2 as described above, the sequence of average dilution factors for the west sector are as follows: 4/4, 3/4, 2/3, 2/3, 1/2, blank, 2/2, 1/2, 1/3, 1/4, 0 and 0.

The average dilution factors, computed as described, were subsequently classified for each twenty-two ½ degree sector into groups, and corresponding cumulative frequency distributions were prepared.

The CHI/Q value which is exceeded 0.5 percent of the total time was then determined for each sector, and the maximum value chosen as the maximum sector conservative CHI/Q.

Overall site accident dilution factors were also computed for each of the time intervals of interest. These parameters were determined by first developing arrays of sector-dependent CHI/Qs (averaged over selected time intervals) and then forming an equivalent sector-independent array consisting of the maximum CHI/Qs at equivalent locations in the sector-dependent arrays. These maximum CHI/Qs were then used to form an overall-site cumulative distribution, from which the values at desired percentile points were determined.

The CHI/Q value which was exceeded no more than 5 percent of the total time was then determined and classified as the overall-site conservative CHI/Q.

Sector-dependent concentration CHI/Q values and effective gamma CHI/Q values for the various time intervals are presented in Table 2.3-38 for the exclusion radius (914.4 meters) and in Table 2.3-39 for the outer boundary of the low population zone (2012 meters). Cumulative frequency distributions for the overall site and for the maximum sector CHI/Q values are provided in Table 2.3-40, Table 2.3-41, Table 2.3-42, Table 2.3-43, Table 2.3-44, Table 2.3-45, Table 2.3-46 and Table 2.3-47, and summaries of CHI/Q values for appropriate percentiles are given in Table 2.3-48 and Table 2.3-49.

Note: The short-term accident CHI/Q values for the low population zone, exclusion area boundary, and control room were recalculated as part of a dose reanalysis project for implementing alternative source term (AST). The revised CHI/Q values used for AST are based on guidance provided in Regulatory Guide 1.145, NUREG/CR-2858, NUREG/CR-6331, and

Regulatory Guide 1.194. Appendix 2Q discusses the determination of CHI/Qs for offsite locations and Appendix 2R discusses the determination of CHI/Qs for the control room.

2.3.5 Long-Term (Routine) Diffusion Estimates

2.3.5.1 Objective

Realistic estimates of annual average atmospheric transport and diffusion characteristics to a distance of 50 miles from the plant were calculated using meteorological data collected onsite during the period April 1979 through March 1980.

2.3.5.2 Calculations

a. Ground Level Concentration (CHI/Q) Deposition (D/Q) Values

Estimates of relative ground level concentration (CHI/Q), both undepleted by ground deposition and depleted by deposition, and deposition values (D/Q) have been calculated for all points of interest. Table 2.3-50, Table 2.3-51, Table 2.3-52, Table 2.3-53, Table 2.3-54, Table 2.3-55, Table 2.3-56 and Table 2.3-57 indicate the calculated CHI/Q and D/Q values. These calculations were performed in accordance with the model described below.

b. Diffusion Model

1. Background

Atmospheric dilution factors (CHI/Q) and deposition rates (D/Q) were calculated using a dispersion model which makes use of the following:

- hourly meteorological data
- straight-line trajectory with sector-averaged Gaussian dispersion
- fumigation and trapping
- part-time ground-level and part-time elevated releases (mixed mode release model)
- momentum plume rise
- terrain elevation

- depletion in transit, and
- multiple eddy reflections from both ground and stable inversion layers aloft.

The method of analysis involves computation of the following parameters on an hourly basis:

(CHI/Q) the nondepleted dilution factor for evaluating ground level concentrations of noble gases, tritium, carbon 14 and nonelemental iodines,

$(CHI/Q)_D$ the depleted dilution factor for evaluating ground level concentrations of elemental radioiodines and other particulates,

$(CHI/Q)\gamma$ an effective gamma dilution factor for evaluating gamma dose rates from a sector-averaged finite cloud (multiple-energy undepleted source), and

(D/Q) the deposition factor for evaluating dry deposition of elemental radioiodines and other particulates.

Average dilution and deposition factors were determined from:

$$\left(\overline{F}\right)_\ell = \frac{1}{N} \sum_{j=1}^{m} (F)_{\ell j} \qquad (9)$$

where F is any one of the four factors listed above, l is the sector identification number, m is the number of hourly values computed for the sector, and N is the total number of values for all sectors.

The fundamental equations used were based on Regulatory Guide 1.111 (Reference 36), and are described below.

2. Nondepleted Dilution Factors

(a) Diffusion Model

Atmospheric transport and diffusion was based on the straight-line flow model with Gaussian diffusion as presented by Sagendorf (Reference 37). The equation for this model, for use in computing sector-averaged hourly dilution factors, is:

$$(\mathrm{CHI/Q}) = \frac{2.032}{X\bar{u}} \left\{ \frac{E_\tau}{\Sigma_z} \xi_G + \frac{(1-E_\tau)}{\sigma_z} \xi_E \right\} \qquad (10)$$

where,

X = the distance from the release point to the receptor (m),

$\bar{u}$ = the hourly average wind speed (m/sec),

E_τ = the entrainment coefficient (equal to unity for ground-level releases and to zero for elevated releases),

σ_z = the vertical plume standard deviation at distance X for the atmospheric stability prevailing during the hour of interest (m),

Σ_z = the vertical plume standard deviation corrected for building wake effects (m),

ξ_G = the reflection correction for ground level releases, and

ξ_E = the vertical attenuation and reflection correction for elevated releases.

Note that Eq. 10 applies during normal atmospheric conditions. The effects of fumigation and trapping caused by seabreezes and onshore gradient flows are described later. The terms within the brackets represent, respectively, the contributions to the dilution factor from the entrained and the nonentrained portions of the release during the hour of interest. The effluent is considered to occur as an elevated release $(1-E_\tau)$ x 100 percent of the time (1 hour in this case) and as a ground-level release (E_τ) x 100 percent of the time. Details on the definitions of the various parameters are given in the sections that follow.

(b) Entrainment

As outlined in Regulatory Guide 1.111, effluents are considered as ground-level releases ($E_\tau = 1.0$), elevated releases ($E_\tau = 0.0$), or mixed mode releases ($0.0 < E_\tau < 1.0$) depending on (a) the elevation (h_S) of the release point above grade relative to the height (h_B) of adjacent solid structures and (b) the effluent-exit velocity (w_o) relative to the speed of the prevailing wind ($\bar{u}$) during the time period of interest. The various cases are as follows:

for $h_S \leq h_B$

$E_\tau = 1.0$

for $h_S \geq 2h_B$

$E_\tau = 0.0$

for $h_B < h_S < 2h_B$

$$E_\tau = 1.0 \text{ when } W_o/\bar{u} \leq 1 \quad (11)$$

$$E_\tau = 2.58 - 1.58\,(W_o/\bar{u}) \text{ when } 1 \leq W_o/\bar{u} \leq 1.5 \quad (12)$$

$$E_\tau = 0.3 - 0.06\,(W_o/\bar{u}) \text{ when } 1.5 \leq W_o/\bar{u} \leq 5 \quad (13)$$

$$E_\tau = 0.0 \text{ when } W_o/\bar{u} \geq 5 \quad (14)$$

(c) Vertical Standard Deviations and Building Wake

Values of σ_z, the plume vertical standard deviation, were computed by applying parabolic interpolation (on a log-log basis) to tabular data of σ_z versus distance. These data were extracted from Pasquill-Gifford curves in Regulatory Guide 1.111 for atmospheric stabilities A through G.

For ground-level releases, consideration was also given to the additional dispersion of the effluent plume within the wake caused by buildings adjacent to the release point. In such cases, use was made of an adjusted vertical standard deviation defined as:

$$\Sigma_z = (\sigma_z^2 + 0.5\, h_B^2/\pi)^{1/2} \tag{15}$$

The maximum value of Σ_z was restricted by the condition

$$(\Sigma_Z)_{max} = \sqrt{3}\,\sigma_z \tag{16}$$

(d) Vertical Dispersion and Reflection

The ξ_G and ξ_E parameters in Eq. (10) represent the exponential decrease in ground-level concentrations with increasing plume height, and the increase in concentration from multiple eddy reflections from the ground and stable atmospheric layers aloft. Definition of these parameters was based on the plume trapping equation in the U.S. EPA workbook on atmospheric dispersion (Reference 35):

$$\xi_G = \sum_{j=-n}^{n} e^{-(\beta' j)^2} \tag{17}$$

and

$$\xi_E = \sum_{j=-n}^{n} e^{-(\alpha+\beta j)^2} \tag{18}$$

where

$$\alpha = \frac{h_e}{\sigma_Z \sqrt{2}} \tag{19}$$

$$\beta = \frac{\sqrt{2}\ L}{\sigma_z} \tag{20}$$

and

$$\beta' = \frac{\sqrt{2}\ L}{\Sigma_z} \tag{21}$$

h_e is the effective plume height above ground, L is the height of the reflection layer and 2n is the total number of reflections. Parameters h_e and L are discussed later.

In the case of large plume standard deviations, where multiple reflections occur and uniform vertical mixing has taken place, the equations reduce to simpler forms. For instance,

$$\lim_{\sigma_z \to \text{large}} \left(\xi_E\right) = \int_{-\infty}^{\infty} e^{-(\alpha+\beta j)^2}\ dj = \frac{1}{\beta}\sqrt{\pi} \tag{22}$$

$$= \frac{\sigma_z}{L}\sqrt{\frac{\pi}{2}} \tag{23}$$

and is achieved for σ_z greater than approximately 2L.

Similarly, for ground level releases,

$$\lim_{\Sigma_z \to \text{large}} \left(\xi_G\right) = \frac{\Sigma_z}{L}\sqrt{\frac{\pi}{2}} \tag{24}$$

When Eqs. (23) and (24) are substituted into Eq. (10), the latter reduces to

$$(CHI/Q) = \frac{2.032}{X\bar{u}L}\sqrt{\frac{\pi}{2}} = \frac{2.55}{X\bar{u}L} \qquad (25)$$

which is the familiar form of the sector-averaged dilution equation with uniform vertical mixing.

In this work, n was conservatively set equal to 6; n = 3 or 4 is normally sufficient to include the important reflections.

(e) Effective Release Height

The effective release height, h_e, was determined in accordance with Regulatory Guide 1.111 from:

$$h_e = h_s + h_{pr} - h_t - c \qquad (26)$$

where

c	=	the downwash correction for low relative exit velocity (see below),
h_{pr}	=	the rise of the plume above the release point according to Sagendorf (Reference 37)(m),
h_s	=	the physical height of the release point (m), and
h_t	=	the maximum terrain height (above the release point grade elevation) between the release point and the receptor ($h_t > 0$)(m).

The downwash correction factor is defined as (Gifford, Reference 38)

$$c = 3(1.5 - W_o/\bar{u})d \text{ when } W_o/\bar{u} < 1.5 \qquad (27)$$

and

$$c = 0 \text{ when } W_o/\bar{u} \geq 1.5 \quad (28)$$

where

d = the inside diameter of the release vent (e.g., stack) (m),

W_o = the vertical exit velocity of the plume (m/sec)

$\bar{u}$ = the mean wind speed (m/sec)

(f) Plume Rise

Only nonbuoyant plumes were assumed to emanate from the vents of the power station. The momentum jet equations reported by Briggs (Reference 39) and Sagendorf (Reference 37) were utilized as outlined below.

For neutral or unstable conditions the transitional rise was computed by the equation

$$h_{pr} = 1.44 \left(\frac{W_o}{\bar{u}}\right)^{\frac{2}{3}} \left(\frac{X}{d}\right)^{\frac{1}{3}} d \quad (29)$$

and

$$h_{pr} = 3 \left(\frac{W_o}{\bar{u}}\right) d \quad (30)$$

and the lesser, more conservative value was used.

For stable conditions, the results from Eqs. (29) and (30) were compared with the results from the following two equations

$$h_{pr} = 4 \left(\frac{F_m}{S}\right)^{1/4} \quad (31)$$

$$h_{pr} = 1.5 \left(\frac{F_m}{\bar{u}} \right)^{\frac{1}{3}} S^{-\frac{1}{6}} \tag{32}$$

and the smallest value of hpr was used. In the last two equations, F_m is the momentum flux parameter and S is a stability parameter. They are defined as

$$F_m = W_o^2 \left(\frac{d}{2} \right)^2$$
$$S = \frac{g}{T} \frac{\partial \theta}{\partial z} \tag{33}$$

where

g = the acceleration of gravity (m/sec^2),

T = the ambient air temperature (deg K), and

$\partial\theta$ = the vertical potential temperature gradient

∂z (deg K/m) defined as (Reference 39):

$$\frac{\partial \theta}{\partial z} = \frac{\partial T}{\partial z} + 0.0098 \tag{34}$$

(g) Mixing Depths

Vertical diffusion of the plume can be inhibited by the existence of a stable atmospheric layer (an elevated inversion) aloft. The rate of vertical mixing is reduced in such cases and the stable layer can be considered as an effective lid on vertical transport of pollutants.

Although there are many hours in the year characterized by unlimited mixing depths, in this analysis it was conservatively assumed that all hours were characterized by the average of the mean morning and mid-afternoon mixing depths for the year.

The effect of plume trapping is included in the terms in Eq. (10).

3. Seabreeze and Onshore Gradient Flow

(a) General

Formulation of a thermal internal boundary layer (TIBL) under conditions of seabreeze or onshore gradient flow can lead to ground-level concentrations which are higher than those calculated for a simple straight-line model. Figure 2.3-11 illustrates the importance of the TIBL on the behavior of effluents.

During an onshore wind, the cool and stable marine air is heated from below by the land surface and becomes unstable in the lower levels. When the plume begins to intercept the top of the TIBL the material in the plume is mixed rapidly downward in the unstable air within the TIBL. This rapid downward mixing is referred to as fumigation (References 40, 41 and 42).

For releases which occur at ground level, the material is trapped within the TIBL. This lid limits the mixing volume to a greater extent than the average mixing depth for the area around the site and can also result in higher ground-level concentrations.

(b) Criteria for and Occurrence of TIBL Formation

The following criteria have been established.

(1) TIBLs Can Occur Only during Spring and Summer

The examination is restricted to those periods when the land-water temperature difference can result in the formation of a TIBL. A seabreeze season extending from April through September is appropriate for the Seabrook site.

(2) TIBLs Can Occur Only during Daytime

The examination is restricted to those times when there is sufficient solar intensity to generate a TIBL. TIBLs were conservatively assumed to occur between 0800 EST and 1800 EST during the seabreeze season.

(3) The Wind Direction Must Be Onshore

The overwater fetch must be sufficiently long to stabilize the air mass. Based on observations reported in the literature (Reference 43 and 44), an overwater fetch of five to ten miles will result in a marine inversion several hundred feet deep. Wind trajectories from the northeast clockwise through the south-southeast would have a sufficient overwater fetch to result in a stable air mass moving over the Seabrook site.

(4) The Wind Speed Must Be in an Appropriate Range

Too low a wind speed will not a support a TIBL. If the wind speed is too great, mechanical turbulence will overcome any thermal effects and a TIBL will not be formed. A range of wind speeds between about 4.5 and 22 miles/hour characterizes the conditions of interest. This range is consistent with the onsite data and with the fundamental nature of the seabreeze.

(5) Solar Radiation Must Be Sufficiently Strong

Since it is the heating of the land which causes the development of a TIBL, the intensity of solar radiation is an important parameter. A minimum value of 0.35 Langleys/minute has been assumed for solar radiation. This magnitude of solar intensity occurs early and late on bright days. This value compares with peaks of about 1.2-1.3 Langleys/minute at mid-day during clear summer days.

As shown in Table 2.3-58, TIBLs were estimated to form during approximately 100 days and lasted on average 5.3 hours per day during the period April 1979 - September 1979. The highest monthly TIBL frequency, 133 hours per month, occurred during June.

(c) Criteria for and Occurrence of Sea Breeze Conditions

Estimated frequency of sea breeze conditions for the Seabrook site during the same time period are also provided in Table 2.3-58. A sea breeze condition was defined as an hour where a localized daytime onshore flow occurred simultaneously with an opposing larger scale (but weaker) inland geostrophic wind directed offshore. Unlike gradient onshore flows, localized sea breeze onshore flows result in the development of sea breeze fronts/convergence zones and recirculation cells.

A localized sea breeze onshore flow would develop under the same conditions under which the TIBL forms; i.e., strong solar radiation and daytime land-surface temperatures rising above the ocean-surface temperatures. In order to differentiate between a true localized sea breeze and a gradient onshore flow, Worcester NWS wind data was used to determine if a larger scale offshore geostrophic wind existed further inland. Data from Worcester (located approximately 70 miles SW of the Seabrook site) were chosen because the observations are obtained on an elevated plateau free from localized terrain effects and are taken far enough inland (45 miles from the coast) to preclude influence by any sea breeze fronts.

The criteria used to determine frequency of sea breeze conditions at the Seabrook site are as follows:

(1) Seabreezes Occur Only under the Same Conditions Conductive to TIBL Formation

The criteria used above to identify TIBL formation are also used to identify seabreeze conditions.

(2) There is an Opposing Offshore Pressure Gradient

Winds observed simultaneously at Worcester are offshore (from the S clockwise to NNE).

As shown in Table 2.3-58, sea breeze conditions were estimated to occur during approximately 89 days and lasted an average of 4 hours per day during the period April 1979 - September 1979. The highest monthly seabreeze frequency, 92 hours per month, occurred during July.

(d) TIBL Geometry

No onsite measurements correlating the shape of the TIBL with the governing parameters are available. In the absence of site measurements, the following relation was selected (Reference 40, 41 and 43).

$$h_{TIBL}(X) = 8.8 \left(\frac{X}{\bar{u}\Delta\theta} \right)^{1/2} \qquad (35)$$

where

$h_{TIBL}(X)$ = the depth of the boundary layer above the ground surface (m),

X = the distance from the shore along the wind trajectory (m),

$\bar{u}$ = the hourly average wind speed (m/sec), and

$\Delta\theta$ = the potential temperature difference between the top and bottom of the marine inversion.

The TIBL was assumed to follow the terrain. The maximum depth of the TIBL was limited to the climatological mixing height. A reasonable value of 4°C was assumed for $\Delta\theta$ in Eq. (35).

(e) Ground-Level Concentrations

A model was developed to calculate ground-level concentrations from reactor effluents during conditions of seabreeze and onshore gradient flow. The calculation is based on the straight-line air flow equation (Eq. 10), with the following conditions:

- Pasquill stability class F is assumed above the TIBL (effects both dispersion and plume rise for releases above the TIBL).

- Tower-indicated stabilities, classes A through D (with E, F and G defaulting to D), are assumed below the TIBL whenever the TIBL is above the levels of measurements; stability class B is assumed for TIBLs lower than the levels of tower measurements.

- Effluents released beneath the TIBL are trapped; mixing depth (L) in Eqs. (20) and (22) is replaced by the height of the TIBL; the plume is not permitted to penetrate the TIBL even if the plume rise equation predicts so.

- Effluents released above the TIBL travel downwind until the plume and TIBL intersect (see below); beneath the TIBL, σ_z and Σ_z are set equal to twice the TIBL height to insure Eqs. (17) and (18) predict uniform mixing (applies also to the entrained portion of the elevated release); a correction factor is applied for partial mixing within the mixing zone.

Intersection between the plume and the TIBL is defined to occur when the turbulence just begins to disturb the lower portion of the plume, that is,

when

$$h_{TIBL}(X') = (h_s + h_{pr} - c) - 2.15\ \sigma_z \qquad (36)$$

where

$\sigma_z =$ for Pasquill stability class F. Total envelopment of the plume by the TIBL occurs

when

$$h_{TIBL}(X'') = (h_s + h_{pr} - c) + 2.15\ \sigma_z \qquad (37)$$

Before the plume-TIBL intercept, the standard elevated plume equation applies. After total envelopment, uniform vertical mixing is attained. For the intermediate zone, a correction factor is applied to the second term in Eq. (10) to account for partial mixing. This factor is equal to (Reference 42).

$$\frac{1}{\sqrt{2\pi}} \int_{-\infty}^{P} \exp\left(-0.5\, p^2\right) dp \tag{38}$$

where

$$P = (h_{TIBL} - h_e)/\sigma_z \tag{39}$$

h_e = the effective plume height defined by Eq. (26) and h_{TIBL} is the height of the TIBL at the receptor.

For the 527 hours of TIBL formation estimated to occur during the period April 1979 - March 1980, the lowest TIBL height predicted to occur over the plant primary vent stacks is 93 meters above ground level (AGL). Since the primary vent stack release height is approximately 56 meters AGL, all releases during TIBL formation occur below the TIBL. The closest the effective stack height ever approaches the TIBL during all hours of TIBL formation is 28 meters. As such, the effect the TIBL has on the annual average atmospheric dispersion estimates is to limit the vertical mixing depth during hours of TIBL formation.

TIBL terrain correction factors (defined as ratios of annual average relative concentrations (CHI/Q) and deposition rates calculated considering trapping versus annual average CHI/Q and deposition rates calculated without considering trapping) are presented in Table 2.3-59, Table 2.3-60, Table 2.3-61 and Table 2.3-62. These annual average TIBL terrain correction factors were calculated for primary vent stack releases and were compiled using April 1979 - March 1980 onsite meteorology. Distances beyond 20 miles are not presented because the annual average transport and diffusion model assumes the TIBL does not extend beyond 20 miles. In addition, because one of the criterion for TIBL formation is an onshore flow, only the SW clockwise through NNW downwind sectors show TIBL terrain correction factors other than one.

Table 2.3-59 shows that TIBL terrain correction factors for the undepleted CHI/Q average approximately 1.03 for the affected downwind sectors and range from a minimum value of 0.97 (0.25 miles W) to a maximum value of 1.19 (0.75 miles NW). TIBL terrain correction factors for the depleted CHI/Q, deposition rates, and effective gamma CHI/Q show similar patterns as indicated in Table 2.3-60, Table 2.3-61 and Table 2.3-62. Note that a few of the TIBL terrain correction factors are slightly less than one due to the stipulation that all E, F and G stability classes measured below the TIBL during TIBL occurrences default to D stability.

4. Depleted Dilution Factors and Deposition Rates

Only dry deposition of elemental radioiodines and other particulates and attendant plume depletion were considered for all releases. The effects of wet deposition and radioactive decay in transit were not included. The models employed are described below.

(a) Depleted CHI/Q

Due to impaction of particles on surfaces, such as walls, vegetation and ground, depletion of plumes containing radioiodine and other particulates occurs in the downwind direction. Ground-level concentrations corrected for depletion in transit were computed for each hour according to the formula

$$(\mathrm{CHI/Q})_{\mathrm{D}} = \frac{2.032}{\mathrm{Xu}} \left\{ \frac{\mathrm{E}\tau \mathrm{D}_{\mathrm{G}}}{\Sigma_z} \xi_G + \frac{(1-\mathrm{E}_\tau)\mathrm{D}_{\mathrm{E}}}{\sigma_z} \xi_{\mathrm{E}} \right\} \qquad (40)$$

where D_G and D_E are the depletion factors for ground-level and elevated releases and the remaining parameters are as defined in Subsection 2.3.5.2b.2.

(b) Deposition Rates

The depletion model employed was based on a deposition velocity v_d defined as

$$v_d = \frac{\text{rate of deposition}}{\text{concentration near the surface}} \tag{41}$$

Combining Eqs. (40) and (41), one obtains the deposition rate equation

$$(D/Q) = v_d\,(CHI/Q)_D \tag{42}$$

The relationship of average wind speed and areal grass density to deposition velocity was examined in detail by Pelletier and Zimbrick (Reference 45) and is as follows:

$$\frac{v_d}{\rho}\left(\text{cm}^3/\text{g}-\text{sec}\right) = 19.4\,\bar{u}(\text{m/sec}) \tag{43}$$

where

ρ = the areal grass density (dry weight, g/cm^2)

(c) Depletion Factors

The depletion factors D_G and D_E which account for dry deposition in transit, were computed according to (Reference 46)

$$D_G = \exp\left(-\delta \int_o^X \frac{dX}{\Sigma_z}\right) \tag{44}$$

$$D_G = \exp\left(-\delta \int_o^X \frac{dX}{\sigma_z \exp\left(h^2/2\sigma_z^2\right)}\right) \tag{45}$$

where

$$\delta = \sqrt{\frac{2}{\pi}}\ \frac{\nu d}{\bar{u}} \tag{46}$$

and

$$h = h_s + h_{pr} - 0.5h_t - c \tag{47}$$

Note that

$$h = h_e + 0.5h_t \tag{48}$$

and indicates the conservative use of an average terrain height between release and receptor points for depletion calculations.

Equations (44) and (45) were used to compute, by numerical integration, tables of depletion factors for all atmospheric stabilities and a number of effective plume heights. Depletion factors for use in Eq. (40) were then obtained by applying parabolic interpolation to the tabular data for the prevailing stability and effective h.

Depletion during trapping and fumigation conditions was calculated using the conservative assumption that the TIBL does not exist. For releases above the TIBL, an elevated plume was assumed if the receptor point was before the totally enveloped zone. Depletion based on a ground-level plume was assumed for receptors within this zone.

5. Gamma Dilution Factors

Evaluation of long-term external gamma doses from routine atmospheric releases is normally based on extensive simplifying assumptions that are essential in reducing the complexity of the problem. An evaluation model that is of interest to the nuclear power industry is described by D.H. Slade (Reference 46, Subsection 7-5.2.5). The model, which is characterized by a finite sector-averaged plume with Gaussian activity distribution in the vertical plane, is of interest because it provides a means of establishing effluent emission limits and making dosage estimates that are more realistic than those based on uniform semi-infinite clouds. The model has been included in Regulatory Guide 1.109 (Reference 47) as guidance toward evaluating compliance with 10 CFR Part 50, Appendix I.

This section presents the method by which the finite-cloud dose rate equation in Regulatory Guide 1.109 has been reduced to that for a semi-infinite cloud via the definition of an effective gamma dilution factor.

(a) Finite-Cloud Gamma Dose Rate Equation

According to Regulatory Guide 1.109, gamma air doses from elevated releases (and ground-level releases as well) may be computed using the equation

$$D^{\gamma}_{finite} = \frac{260}{X\Delta\phi} \sum_{n,s,k,i} \frac{Q_i' f_{ns} \mu_a^k \bar{E}_K \bar{I} A_{ki}}{\bar{u}_n} \qquad (49)$$

where

A_{ki} = the photon yield for gamma-ray photons in energy group k from the decay of radionuclide i, (photons/disintegration),

D^{γ}_{finite} = the annual total gamma air dose at a distance X in the sector at angle ϕ (mrad/year),

X = the distance from the release point to the receptor (m),

$\bar{E}_k$ = the energy of the kth photon energy group (MeV/photon),

f_{ns} = the fraction of the time that stability class s and wind speed n occur for sector φ, (dimensionless),

$\bar{I}$ = the result of the numerical integration accounting for the distribution of radioactivity according to meteorological conditions of wind speed and atmospheric stability. In addition, I is a function of the photon energy E_k and is $= {}_1 + K_2$ as formulated in Slade (Reference 46); K is the air buildup factor,

Q_i' = the release rate of radionuclide i (Ci/yr),

$\bar{u}_n$ = the mean wind speed of wind speed class n (m/sec),

$\Delta\phi$ = the sector width over which atmospheric conditions are averaged (radians), and

μ_a^k = the air energy absorption coefficient for the kth photon energy group (m^{-1}).

From Eq. (49) it is possible to define an energy dependent gamma dilution factor for use in computing hourly air dose rates as follows:

$$(\mathrm{CHI/Q})_k^{\gamma} = \frac{2\mu_a^k\left(\bar{I}_1 + K\bar{I}_2\right)}{\sqrt{\pi}\,\bar{u}X\Delta\phi} \tag{50}$$

As shown in Reference 48, the integrals I_1 and I_2 in this equation reduce to the following limits when the size of the finite cloud becomes large:

$$\bar{I}_1 = \frac{1}{\sqrt{2}\mu^k\sigma_z}\exp\left[\frac{-h_e^2}{2\sigma_z^2}\right] \tag{51}$$

and

$$\bar{I}_2 = \bar{I}_1 \tag{52}$$

μ^k is the air total linear attenuation coefficient for the kth photon energy group. For this condition, since

$$K = \left(\mu^k - \mu_a^k\right) / \mu_a^k \tag{53}$$

Eq. (50) reduces to

$$(\text{CHI/Q}) = \frac{2.032}{\bar{X} u \sigma_z} \exp\left(-h_e^2 / 2\sigma_z^2\right) \tag{54}$$

and is dependent of gamma energy. Eq. (54) is the basic equation for sector-averaged semi-infinite clouds.

To reduce Eq. (49) to a form similar to that for a semi-infinite cloud, it is necessary to define a few more parameters as follows:

$$\bar{A}_{K\sum_i} = \frac{Q_i' \; A_{ki}}{\sum_i Q_i'} \tag{55}$$

$$\bar{E}_k = \frac{\sum_i E_{ki} \; Q_i' \; A_{ki}}{\sum_i Q_i' \; A_{ki}} \tag{56}$$

$$\overline{CHI/Q}^{\gamma} = \frac{\sum_k E_{ki} \; Q_i' A_{ki}}{\sum_k E_k A_{ki}} \tag{57}$$

and (using Eq. (9) without the sector identification subscript)

$$\overline{\text{CHI/Q}}^{\gamma} = \frac{1}{N} \sum_{j=1}^{m} \left\{(\text{CHI/Q})^{\gamma}\right\}_j \tag{58}$$

In these equations, E_{ki} is the actual energy of a gamma ray in group k emitted by isotope i. The other parameters are as previously defined.

Substitution of these equations in Eq. (49), and replacing f_{ns} by () for the hourly factors, results in:

$$D^{\gamma}_{finite} = 3.17x10^{4}\left(\overline{CHI/Q}\right)^{\gamma}\sum_{i} Q_i^{'}\ (DF)_i^{\gamma} \tag{59}$$

where

$$(DF)_i^{\gamma} = \sum_{K} \frac{260\sqrt{\pi}}{(2)\ (3.17)x10^{4}} A_{ki}\ E_{ki} \tag{60}$$

is the gamma air dose factor for radionuclide i as used in Regulatory Guide 1.109. 3.17x104 is the number of pCI per Ci divided by the number of seconds per year. Note that Eq. (59) is similar in form to the dose equation for semi-infinite clouds.

(b) Application

Equations (50), (55), (56), (57) and (58) were used in computing effective gamma dilution factors for evaluating air gamma doses from finite clouds for both ground-level and elevated releases.

Equation (50) was computed on an hourly basis for 10 to 15 gamma energy groups covering the range 0.01 to 4 MeV. Energy-averaged factors were then computed by Eq. (57) for each hour of interest. The integrals $_1$ and $_2$ were computed by numerical integration using the model described in References 47 and 48. Data for the total linear energy absorption and attenuation coefficients were taken from Reference 49.

Gamma dilution factors during seabreeze conditions were calculated by ignoring the presence of the TIBL and applying the following modification to account for the increased concentrations near the ground due to the presence of the TIBL:

- For ground-level releases (trapping conditions) Σ_z was limited to the maximum value of 0.47 h_{TIBL}, where h_{TIBL} is the height of the h_{TIBL} at the receptor location.

- For trapped elevated releases, the plume effective height and standard deviation were adjusted to ensure that the entire plume is within the TIBL while maintaining the ground-level concentration:

$$\sigma z \rightarrow \sigma z^{\gamma} = \frac{h_{TIBL}}{(h_e / \sigma_z + 2.15)} \tag{61}$$

and

$$h_e \rightarrow h_e^{\gamma} = \frac{\sigma_z^{\gamma}}{\sigma_z} \tag{62}$$

- For releases above the TIBL that become entrained, E_z was set equal to 0.47 H_{TIBL}

- For releases above the TIBL that are not entrained, σ_z was retained for distances less than the plume-TIBL intercept and was set to 0.47 H_{TIBL} for receptors beyond that point; receptors within the fumigation zone were assumed to be exposed to both an elevated plume and a fully entrained plume, the contribution from each plume being based on Eq. (38).

6. Model Applicability

The Gaussian dispersion model, (straight-line trajectory airflow model described in Regulatory Guide 1.111) used for the CHI/Q and D/Q calculations has been widely accepted by the scientific and engineering community, including industrial users and regulatory agencies. The results of calculations are generally considered to represent reasonable, and often conservative, estimates of pollutant concentrations.

For the Seabrook site, the feature which has been specifically evaluated with respect to atmospheric dispersion is the effect of the land-water interaction. The model used, as well as similar models, to calculate the ground level concentrations during conditions of seabreeze and onshore gradient flow have undergone validation analysis at other sites.

A smoke tracer experiment was conducted at a coastal nuclear power station site as reported in Reference 40. The application of the model described above provided generally good agreement with observation of the location of the fumigation point.

The results of related field experiments are reported in References 41 and 42. These experiments show fairly successful application of similar models in predicting concentrations from SO_2 sources at Great Lakes locations.

2.3.6 References

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3.7(B) SEISMIC DESIGN

3.7(B).1 Seismic Input

This subsection contains a discussion of the input criteria to be used for seismic design of the plant. Items included in the discussion are design response spectra and the basis for their selection, earthquake time-motion records and the basis for their selection, response spectra obtained from time-motion records, and recommended percentages of critical damping to be used for seismic analysis.

3.7(B).1.1 Design Response Spectra

The design response spectra for horizontal and vertical motion, corresponding to the SSE applicable to the site, are presented in Subsection 2.5.2. The spectra amplification ratios for various levels of damping used to establish the design response spectra are based upon values presented in Regulatory Guide 1.60. The OBE response spectra are obtained by multiplying the SSE spectra ordinates by one-half. The duration of the earthquake is estimated at 10 to 15 seconds.

There are no existing earthquake records pertinent to the Seabrook Station site.

3.7(B).1.2 Design Time History

The three components of artificial time-history motion corresponding to the SSE are shown on Figure 3.7(B)-1, Figure 3.7(B)-2, and Figure 3.7(B)-3. These figures show the two horizontal motions and the one vertical motion, respectively. The components of the artificial time-history corresponding to the OBE are obtained by multiplying the ordinates of Figure 3.7(B)-1, Figure 3.7(B)-2, and Figure 3.7(B)-3 by one-half (1/2). The earthquake motion was generated by super-imposing sinusoidal waves of many frequencies. Phase angles of the sinusoidal waves are randomly chosen. The resulting wave form is then multiplied by a trapezoidal intensity function to cause the time variation of intensity. All artificial time histories used in the analyses are base-line corrected.

Response spectra obtained from earthquake time history are shown on Figure 3.7(B)-4, Figure 3.7(B)-5, Figure 3.7(B)-6, Figure 3.7(B)-7, Figure 3.7(B)-8, Figure 3.7(B)-9, Figure 3.7(B)-10, Figure 3.7(B)-11, Figure 3.7(B)-12, Figure 3.7(B)-13, Figure 3.7(B)-14, Figure 3.7(B)-15, Figure 3.7(B)-16, Figure 3.7(B)-17, Figure 3.7(B)-18, Figure 3.7(B)-19 and Figure 3.7(B)-20. Included on each figure is the corresponding design response spectra discussed in Subsection 3.7(B).1.1. The response spectra obtained from the time-motion record envelope the design response spectra for the period range of 0.03 seconds to 4.0 seconds. The response spectra have been computed using a method based on the exact solution of the governing differential equation for a single degree of freedom oscillator with viscous damping. To ensure that the response spectra are sufficiently accurate, they were calculated at a set of discrete values for period, T, forming a geometric progression, i.e.,

$$T_o, T_o r, T_o r^2, \ldots T_o r^{n-1}$$

$$r = 1.02$$

$$T_o = 0.03 \text{ seconds}$$

This ratio corresponds to a period interval varying from 0.0006 seconds at a period of 0.03 seconds to a period interval of 0.01 seconds at a period of 0.50 seconds.

3.7(B).1.3 Critical Damping Values

The percentages of critical viscous damping used for the seismic analysis of Category I structures, systems, and components are based on recommendations presented in Regulatory Guide 1.61. These percentages, which account for stress level as well as type of construction or fabrication, are summarized in Table 3.7(B)-1.

For seismic piping analysis, an alternative to Regulatory Guide 1.61 may be used. These values are shown graphically in Figure 3.7(B)-38.

For the Cable Raceway System, an alternative to Regulatory Guide 1.61 may be used. Critical damping levels may be a maximum of 20 percent for input acceleration levels of 0.35g and greater for OBE and SSE conditions. In cases where input accelerations are between 0.1g and 0.35g, the critical damping values may be interpolated between 7 percent and 20 percent, respectively.

3.7(B).1.4 Supporting Media for Category I Structures

All seismic Category I structures are founded on sound bedrock or on engineered backfill extending to sound bedrock. Engineered backfill was also placed around all seismic Category I structures.

The bedrock at the site is uniform, competent, and nonfragmented. Engineering properties of the bedrock measured in both the field and the laboratory are presented in Subsection 2.5.4.2a.

The engineered backfill consists of either fill concrete, backfill concrete, offsite borrow, tunnel cuttings, or sand-cement. Properties of the engineered backfill materials are described in Subsection 2.5.4.5. The type of engineered backfill used beneath all seismic Category I structures was fill concrete, except for safety-related electrical duct banks, five electrical manholes, and the service water pipes, which were founded on offsite borrow or tunnel cuttings, as shown in Table 2.5-20.

Identification of the safety-related electrical manholes founded on offsite borrow or tunnel cuttings, the depths of offsite borrow or tunnel cuttings over the bedrock under these particular manholes, the widths of their structural foundations and the total structural height are summarized below:

Manhole Numbers	Depths of Soil over Bedrock (ft)	Widths of Structural Foundations (ft)	Total Structural Height (ft)	Supporting Material
W13/W14	6-12	18 x 18½	9½	Offsite Borrow
W15/W16	6-12	18 x 18½	9½	Offsite Borrow
W19/20	15	23½ x 23½	12	Tunnel Cuttings
W29/W30	14	19 x 22½	15	Offsite Borrow
W33/W34	18	18 x 18½	12	Offsite Borrow

All manholes are fully embedded.

The values of shear modulus, G, and shear wave velocity, v_s, for both the offsite borrow and tunnel cuttings used for the analyses of the manholes are discussed in Subsection 2.5.4.7.

3.7(B).2 Seismic System Analysis

This subsection contains a discussion of the seismic analyses performed for seismic Category I structures and systems. Included in the discussion are the methods of seismic analysis used, the criteria used for mathematically modelling the structures and systems, the assumptions made in the analyses, and the effects considered.

3.7(B).2.1 Seismic Analysis Methods

The seismic response of Category I structures, systems and components has been determined from suitable elastic dynamic analyses. The results of these analyses are used for the design of seismic Category I structures, systems and components, and are input for subsequent dynamic analyses.

Two methods of seismic system analysis were used for seismic Category I structures: (1) the modal analysis response-spectrum method and (2) the mode-superposition time-history method. The time-history method was used to determine the dynamic response necessary to obtain amplified response spectra for component design. The input forcing functions (the time history of ground motion) are shown graphically in Figure 3.7(B)-1, Figure 3.7(B)-2 and Figure 3.7(B)-3. The time history shown on Figure 3.7(B)-1 is used in both horizontal directions. The peak acceleration is 0.25g for the SSE and 0.125g for the OBE. Design response spectra for the response-spectrum method are shown in Section 2.5.

The mathematical models used for the seismic Category I structures are typically lumped masses connected by linear elastic springs. Each structure, then, is described by a finite number of degrees-of-freedom chosen to represent the principal overall behavior of the system. The modelling is described in Subsection 3.7(B).2.3 in more detail. The number of masses or degrees-of-freedom included in the analysis is determined by requiring the total degrees-of-freedom to exceed twice the number of significant modes with frequencies less than 33 Hz. Up to four degrees-of-freedom were considered for each mass point, three translation and one torsion. The three orthogonal directions were run separately, and results were combined by the grouping method in accordance with Regulatory Guide 1.92.

All significant modes with frequencies up to 50 Hz were used in analyses for both local and overall effects.

The effects due to inertial characteristics of fluid contained within a structural component were considered in the analysis by techniques described in Reference 1. No soil-structure interaction effects were involved because of the rock siting.

The lumped mass mathematical models for representative seismic Category I structures, including the containment, are shown in Figure 3.7(B)-21, Figure 3.7(B)-22, Figure 3.7(B)-23, Figure 3.7(B)-24 and Figure 3.7(B)-25. Subsection 3.7(B)2.3 describes these models in more detail. Table 3.7(B)-2, Table 3.7(B)-3, Table 3.7(B)-4, Table 3.7(B)-5, Table 3.7(B)-6, Table 3.7(B)-7 and Table 3.7(B)-8 are the property tables for the indicated models.

Relative displacements between supports of structures are zero due to the base fixity. Maximum displacements throughout the structure (relative to the fixed base of the model) were computed for use in component analysis, as described in Section 3.7(B).3.

The description of the analysis methods used for seismic Category I systems and components is provided in other sections.

3.7(B).2.2 Natural Frequencies and Response Loads

The seismic analyses of Category I structures are based upon the time history modal superposition and response spectra normal mode methods using idealized lumped mass models of the individual structures. These methods of analyses use the natural frequencies, mode shapes and appropriate damping coefficients of the system. The system frequencies, modes shapes and structural response are obtained using the STARDYNE computer program and the models described in Subsections 3.7(B).2.1 and 3.7(B).2.3. Numerical results are tabulated in this subsection for the following representative seismic Category I structures: Containment Building, Primary Auxiliary Building, Control and Diesel Generator Building, and Fuel Storage Building. Table 3.7(B)-5, Table 3.7(B)-6, Table 3.7(B)-9 Table 3.7(B)-10, Table 3.7(B)-11, Table 3.7(B)-12, Table 3.7(B)-13, Table 3.7(B)-14, Table 3.7(B)-15, Table 3.7(B)-16, Table 3.7(B)-17, Table 3.7(B)-18, Table 3.7(B)-19, and Figure 3.7(B)-20 list natural frequencies, periods, mode classification, nodal accelerations, nodal displacements and peak forces for these buildings.

3.7(B).2.3 Procedures Used for Analytical Modeling

The seismic analyses of Category I structures are based upon dynamic analyses using idealized three-dimensional lumped mass models of the physical structures. The inertial properties of the models are characterized by the lumped mass, eccentricity of the mass, and the torsional mass moment of inertia at each mass point of the model. The locations of lumped masses or nodes are selected at floor slabs, at changes of cross-sectional area and at intermediate points, such as locations of equipment, etc. The number of dynamic degrees-of-freedom are chosen to exceed twice the number of significant modes with frequencies less than 33Hz as a minimum. The concentrated masses are connected by weightless elastic beams representing the resisting structural members between mass points. Torsion is accounted for by rigidly offsetting the centers of mass from the centers of rigidity. Floor slabs are assumed to be rigid in their own plane, and the stiffness of shear walls is appropriately considered taking openings into account. The stiffness properties of the models are characterized by the cross-sectional area, moment of inertia, shear shape factor, torsional constant, and Young's Modulus of Elasticity and Poisson's Ratio. The torsional stiffness of the structure is evaluated using the lateral stiffness of the resisting elements between mass points and the square of their distance from the center of resistance for a particular story. The stiffnesses of a particular story in the model are combined into a pseudo-elastic beam located at the center of resistance of the resisting elements.

All seismic Category I structures, with the exception of some electrical manholes and ductbanks (see Subsection 3.7(B).2.4), are supported on competent bedrock or concrete fill over bedrock. All mathematical models are, therefore, fixed against translation and rotation at their bases. The elevation of the point-of-fixity of the mathematical model is determined as follows:

a. Lowest elevation of upper surface of concrete backfill which bears directly against the structure (i.e., no seismic isolation material), or

b. Elevation of a continuous floor slab which does not amplify the ground response, whichever is higher.

The actual stiffness of resisting elements that extend below the assumed point-of-fixity is appropriately considered and reflected in the calculation of pseudo beam properties.

Equipment having relatively small mass/or high frequency is decoupled from the supporting structure, but its mass is included with the mass of the supporting system. The NSSS, a major equipment system whose dynamic behavior can have important interaction with the supporting containment concrete internals structure, was modeled as a coupled system, NSSS and structure, by Westinghouse in their dynamic analyses. The structural portion of the mode is represented by Figure 3.7(B)-22.

The hydrodynamic mass effects of the fluid inside the structure are considered in modeling the inertia properties. Equivalent weights of fluid that are effective in producing impulsive forces and the constrained weight of fluid are determined and included in the lumped mass model at the appropriate nodes. These weights were calculated using the method described in Reference 1. The combined effects of the two horizontal and one vertical motion are taken into account to obtain the design parameters. The three seismic responses or effects at a particular point caused by each of the three orthogonal components of seismic motion are combined by taking the square-root-of-the-sum-of-the-squares of the particular effect or response at that point, in accordance with Regulatory Guide 1.92. For symmetric structures, such as the containment, coupling between translational and torsional modes of vibration is negligible, and the dynamic degrees of freedom are therefore uncoupled for the two horizontal directions. The seismic analysis of structures in the vertical direction is performed separately considering different mathematical models of the structures. For this analysis, only the vertical dynamic degrees-of-freedom associated with each mass are retained.

Ground response spectra for the Seabrook site (Subsection 3.7(B).1) are in accordance with Regulatory Guide 1.60, factored by 0.25 to match the site maximum acceleration. These spectra are valid for structure and component design provided the system in question is supported on bedrock or concrete fill over bedrock, or is a portion of a structure which is subjected to ground response. The ground response spectra or artificial time-histories are applied as input at the structure's point-of-fixity. The lumped-mass models of Figure 3.7(B)-21, Figure 3.7(B)-22, Figure 3.7(B)-23, Figure 3.7(B)-24 and Figure 3.7(B)-25 have the properties listed in Table 3.7(B)-2, Table 3.7(B)-3, Table 3.7(B)-4, Table 3.7(B)-5, Table 3.7(B)-6, Table 3.7(B)-7 and Figure 3.7(B)-8, as noted previously in Subsection 3.7(B)2.1.

3.7(B).2.4 Soil/Structure Interaction

All major seismic Category I structures are founded on rock or concrete extending to rock, thus permitting a fixed base approach to be used. Static and dynamic earth pressures for Category I structures surrounded by offsite borrow are discussed in Subsection 2.5.4.10c, and the pressure diagrams are shown on Figure 2.5-66 and Figure 2.5-67. Analysis methods for computing static and dynamic earth pressures on seismic Category I structures are provided in Subsection 2.5.4.10c.

There are no hydrodynamic effects due to groundwater because there are no structures exposed to free groundwater. The only dynamic effect produced by the presence of groundwater is an increase in dynamic lateral soil pressure resulting from an increase in density of backfill material from that of the moist condition to that of the saturated condition.

Certain seismic Category I electrical manholes are founded on a thin stratum of soil (maximum depth of soil between the foundation and the bedrock is 18 feet), as described in Subsection 3.7(B).1.4. These manholes are analyzed using the multiple lumped mass-spring approach described by Whitman in Reference 2.

All seismic Category I manholes are completely embedded. The effect of embedment is to increase the soil spring stiffness thus increasing the natural frequency of the system resulting in reduced seismic design value. Hence, it was conservatively assumed to neglect embedment effect. The input design ground response spectra, discussed in Subsection 3.7(B).1.1, which were used for the analysis indicated that the increase in frequency would result in decrease in response.

To reduce amplification properties of the soil between the ground surface and the rock, the backfill material is controlled by controlling the placement requirement and material characteristics. The soil property given in Updated FSAR Subsection 3.7(B).1.4 is used for determining soil stiffness. The variation in soil parameters was not considered for the analysis. The lowest shear wave velocity was used to obtain lowest structural frequency which would give higher structural responses as stated above.

The above conservative assumption combined with lower system damping values of 7 percent for SSE and 4 percent for OBE causes seismic load in excess of the actual value.

The design of manholes is governed by dynamic soil pressure; the inertia loads had a minimum effect on the design.

3.7(B).2.5 Development of Floor Response Spectra

The analysis for overall dynamic response of the structural system is described in Subsections 3.7(B).2.1 through 3.7(B)2.3. Local amplification of overall response (generally of floor slabs in the vertical direction) is represented by amplified response spectra (ARS). A time-history seismic analysis is used to generate these spectra using either of two methods. In the first method, the local amplification of slabs, beam and columns is evaluated and an appropriate range of frequencies selected for all local frequencies below 33 Hz. Single-degree-of-freedom (SDOF) systems representing the computed range of local frequencies are connected to the stick model of the overall structure such that the total weight at the elevation in question (i.e., summation of the weight of the SDOF nodes plus the stick model node) equals the total weight, as if the slabs, beams, columns and walls were all rigid. The total stick model is then analyzed using the ground motion artificial time histories described in Subsection 3.7(B).1 as the input forcing function. The second method of evaluation of local amplification is to model the sub-structure with finite elements in sufficient detail to predict local modes of vibration. The response time-history from the overall stick model, at the elevation of the sub-structure, is then used as the input forcing function.

For symmetric structures, separate analyses are performed for two horizontal directions, and the individual time-history motions and amplified response spectra are obtained for the nodal point locations of the lumped-mass model. For unsymmetric structures, the floor response spectra values in both horizontal directions are initially obtained due to each horizontal component of motion. The resultant response spectra for each horizontal direction are then obtained by combining co-directional responses according to the square-root-of-the-sum-of-the-squares method. The floor response spectra in the vertical direction are obtained separately from the vertical response time-history motions. No coupling of vertical and horizontal motion exists, since rock foundation precludes rocking motion.

Structural damping used in the seismic analysis of all Seabrook structures is in conformance with Regulatory Guide 1.61. Amplified response spectra are generated at 1, 2 and 4 percent equipment damping associated with the OBE, and at 1, 2, 3, 4 and 7 percent equipment damping for the SSE.

To obtain complete and accurate response spectra with respect to peak values, the spectrum ordinates are calculated at natural frequencies of supporting structures where peaks are normally expected, and at other frequencies at sufficiently small frequency intervals. Frequencies listed in Table 3.7(B)-21 are used to compute floor response spectra. This table is based on Table N-1226-1 of the ASME Boiler and Pressure Vessel Code, Section III, Division I, Nuclear Power Plant Components, 1980 Edition, Appendix N, 'Dynamic Analysis Methods.' The floor response spectra generated based on Table 3.7(B)-21 including the natural frequencies of supporting structures will meet the intent of SRP Subsection 3.7.1 (and also R.G. 1.122). The response pectra developed at the subsystem supports as described above, and widened and enveloped as described in Subsection 3.7(B).2.9, are used directly, for the analysis of the subsystem. See Figure 3.7(B)-26, Figure 3.7(B)-27, Figure 3.7(B)-28 and Figure 3.7(B)-29 for typical amplified response spectra for representative seismic Category I structures. Figure 3.7(B)-35 and Figure 3.7(B)-36 show typical comparisons of enveloped floor response spectra for 1 and 4 percent damping generated using frequency intervals based on Table 3.7(B)-21 and R.G. 1.122. A single absolute maximum spectrum is generated from the appropriate response points if more than one set of spectra are generated for a particular elevation or portion of a floor area.

3.7(B).2.6 Three Components of Earthquake Motion

The seismic analyses of the Category I structures are performed using three-dimensional lumped mass models of the structures. The maximum response in principal directions is calculated using each of the three components of earthquake motion, and then the responses in each direction are obtained by taking the square-root-of-the-sum-of-the-squares of the maximum co-directional responses at a particular point in the structure. This procedure is in accordance with Standard Review Plan 3.7.2, Section II.6a and b(1).

3.7(B).2.7 Combination of Modal Responses

Modal responses are combined as per Regulatory Guide 1.92, paragraph 1.1 and 1.2.1. When the response spectrum method of analysis is used to determine seismic response of seismic Category I structures, the most probable response is obtained as the square-root-of-the-sum-of-the-squares of the responses from the individual modes.

For the closely spaced modes having frequencies within 10 percent of each other, the structural response is obtained by first obtaining the absolute sum of the responses of the closely spaced modes and then combining this sum with other remaining modal responses using the square-root-of-the-sum-of-the-squares method.

3.7(B).2.8 Interaction of Non-Category I Structures with Category I Structures

All nonseismic Category I structures which, due to their proximity to seismic Category I structures, could possibly compromise the safety function of the seismic Category I structures by their collapse, are either designed for SSE loading or are designed to collapse away from the adjacent seismic Category I structures (see Table 3.7(B)-22 for further discussion). The methods described in Subsection 3.7(B).2.1 are used for the analyses of these structures.

3.7(B).2.9 Effects of Parameter Variation on Floor Response Spectra

The location of the peak responses on amplified response spectra curves vary as a result of the variation in material properties. This impacts subsystem design as discussed in Subsection 3.7(B).3. The variability in structural materials' properties and modeling assumptions is accounted for by peak spreading when generating envelopes of the response spectra. All Category I structures for which in-structure response spectra are generated are supported on rock and, hence, variability of soil properties is not the consideration in broadening the peaks of floor response spectra. A value of at least ± 10 percent is used to broaden the peaks of the floor response spectra associated with the structural frequencies. Regulatory Guide 1.122 is complied with except as noted in Section 1.8.

3.7(B).2.10 Use of Constant Vertical Load Factors

Detailed vertical seismic system analyses are performed for seismic Category I structures; therefore, constant vertical static factors are not used.

3.7(B).2.11 Method Used to Account for Torsional Effects

Torsional degrees-of-freedom are included in the dynamic analysis to obtain the natural frequencies and mode shapes of seismic Category I structures. For structures in which the center of mass and center of rigidity is not coincident, torsional effects are automatically considered when the response to horizontal motion is obtained. The design story shears are computed by considering both the direct shear from the orthogonal earthquake components and the shear due to torque.

An accidental torsion, based on 5 percent eccentricity (SRP 3.7.2, Rev. 1, July 1981) was shown to have negligible effect on the design.

3.7(B).2.12 Comparison of Responses

The nodal displacement and acceleration responses obtained from response spectrum and time-history methods of analysis are compared for the Containment Building and Fuel Storage Building. Table 3.7(B)-10 and Table 3.7(B)-11 list the displacements and accelerations computed by both methods for horizontal and vertical directions and SSE and OBE conditions (as noted on the tables) for the Containment Building. Referenced elevations correspond to mass points as shown on Figure 3.7(B)-21. Table 3.7(B)-19 shows similar information for the Fuel Storage Building, whose model is shown on Figure 3.7(B)-25. Approximate equivalency can be seen in the results of the two methods.

3.7(B).2.13 Methods for Seismic Analysis of Category I Dams

This section is not applicable to Seabrook.

3.7(B).2.14 Determination of Seismic Category I Structure Overturning Moments

All seismic Category I structures are designed to resist overturning due to the combined effects of the vertical and two horizontal components of seismic ground motion. A structure's ability to resist overturning is calculated by either of two conservative approaches: moment equilibrium or a work-kinetic energy method.

a. Moment Equilibrium Method

In the moment equilibrium method, the response of a structure due to three directions of earthquake is obtained from the dynamic analyses. The maximum overturning moment, about the toe of the mat, is computed as:

$$M_O = M_{h1} \text{ or } M_{h2}$$

where: M_O = overturning moment

M_{h1} = maximum overturning moment in one horizontal direction due to effects of three directions of earthquake.

M_{h2} = maximum overturning moment in second horizontal direction due to effects of three directions of earthquake.

M_{h1} or M_{h2} include all dynamic effects on the structure including the dynamic effect of soil caused by seismic motions.

The resisting moment is computed as follows:

$$M_R = (W - V_S) X_1 - (V_h) X_2 + M_b$$

where: M_R = resisting moment

W = weight of the structure and any fill

V_s = maximum vertical seismic force of structure acting upward due to three directions of earthquake

V_h = maximum vertical hydrostatic force on the structure acting upward

X_1 = horizontal distance of the centroid of the structure from the toe of the mat

X_2 = horizontal distance to center of hydrostatic area from the toe of the mat

M_b = resisting moment due to key action of mat or passive resistance of the structure.

The factor of safety against overturning is:

$$FS = \frac{M_R}{M_O}$$

As long as the factor of safety is equal to or greater than 1.10 for SSE and 1.50 for OBE load conditions, the structure is considered stable against overturning.

b. Work-Kinetic Energy Method

In the work-kinetic energy method, the kinetic energy in a structure due to an OBE or SSE is estimated by:

$$KE = \frac{1}{2} \sum M_i \left[(V_H)_i^{\,2} + (V_V)_i^{\,2} \right]$$

M_i = Mass concentration at some point (i) in the structure

$(V_H)_i$ = Maximum total lateral velocity

$(V_v)_i$ = Maximum total vertical velocity

$(V_H)_i$ and $(V_v)_i$ are computed as follows:

$$(V_H)_i^2 = (V_x)_i^2 + (V_H)_g^2$$

$$(V_v)_i^2 = (V_z)_i^2 + (V_v)_g^2$$

where:

$(V_H)_g$ = Peak horizontal ground velocity

$(V_v)_g$ = Peak vertical ground velocity

$(V_x)_i$ = Maximum relative lateral velocity of mass, M_i, due to three directions of earthquake

$(V_z)_i$ = Maximum relative vertical velocity of mass, M_i, due to three directions of earthquake.

$(V_H)_g$ and $(V_v)_g$ are obtained from ground response spectra of the time history used in seismic analysis. $(V_x)_i$ and $(V_z)_i$ are obtained from seismic analysis of the structure.

The work, W, required to overturn the structure is computed as:

$$W = M_t\, g\, h + W_p - W_b$$

where:

M_t = Total mass of the structure and foundation mat

g = Gravitational acceleration

h = The vertical distance for which the center of mass of a structure must be lifted to reach the overturning position

W_p = The additional work required to displace the soil on the toe side of an embedded structure

W_b = The work done by the buoyancy force on the submerged portion of a structure.

Each collinear effect (as described in both a. and b. above) due to the effects of three directions of earthquake is obtained by combining the respective individual components using the SRSS method.

The structure is considered stable against overturning when the ratio W/KE exceeds the safety factors described in a. above.

3.7(B).2.15 Analysis Procedure for Damping

When the components of a seismic Category I structure or system are constructed of different materials, and these components cannot be uncoupled due to dynamic interaction effects, an average modal damping value is used for the dynamic analysis of the system. This average modal value is computed from the damping values of the various components, as listed in Table 3.7(B)-1, each weighted by the energy stored in these components in the various modes of vibration. This value is computed as:

$$D_n = \frac{\sum D_i E_{in}}{E_{Tn}}$$

where:

D_n = Average damping value of the nth mode

D_i = Damping value of the ith component

E_{in} = Energy stored in the ith component in the nth mode

E_{Tn} = Total energy stored in the structure or system in the nth mode.

The energy values are computed from the stiffness matrices of the various components and the mode shapes associated with undamped vibration.

By this procedure, a diagonal damping matrix is computed allowing for the uncoupling of the equations of motion and the use of modal super-position.

3.7(B).3 Seismic Subsystem Analysis

This section describes the seismic analysis performed on all subsystems, exclusive of those within the NSSS supplier's scope of responsibility. See Subsection 3.7(N).3 for the NSSS supplier's discussion.

3.7(B).3.1 Seismic Analysis Methods

The seismic analyses of all Category I subsystems and components use either a dynamic analysis method or an equivalent static load method. Selection of the particular method of analysis depends upon the suitability of the mathematical model to adequately represent the behavior of the system or component. All points of the systems or components where the deflections, loads and stresses are expected to be significant are included in the mathematical model. The use of the equivalent static load method is limited to systems which can be represented by simple mathematical models.

a. Dynamic Analysis

Dynamic analyses were performed on subsystems and components such as electrical cable tray supports, electrical conduit supports, instrument racks, battery racks, etc., using the modal response spectrum analysis technique. The mathematical model for the electrical cable tray supports, as represented by Figure 3.7(B)-30, is typically represented by Figure 3.7(B)-31. Modeling considerations are discussed in Subsection 3.7(B).3.3b.

The number of modes considered in determining the response of the equipment or components is such that inclusion of additional modes would not result in more than a 10 percent increase in the response.

b. Equivalent Static Analysis

The equivalent static analysis method consists of applying a load at the center-of-gravity of the component or equipment in the direction of seismic excitation, where the applied load is calculated by multiplying the total weight of the component or equipment with the applicable seismic acceleration level corresponding to the fundamental frequency of the component or equipment, using an appropriate static coefficient to account for the combined modal participation of the higher modes of vibration. When the equipment or component fundamental frequency is less than 33 Hertz, a static coefficient of 1.5 will be used, except where a reduced static coefficient can be justified. When the equipment or component fundamental frequency is equal to or greater than 33 Hertz, a static coefficient of 1.0 will be used when the equipment or component has a single predominant frequency. When there is more than one predominant frequency, a static coefficient of 1.3 will be used, except where a reduced static coefficient can be justified. The criteria for selecting the applicable seismic acceleration level are as follows:

1. When the equipment or component fundamental frequency is greater than 33 Hertz, the acceleration level corresponds to the 'g' value of the rigid range of the amplified floor response spectra.

2. When the equipment or component fundamental frequency is less than or equal to 33 Hertz, or is not evaluated, the acceleration level corresponds to the peak 'g' value of the amplified floor response spectra. However, if the equipment or component fundamental frequency is predominant over all other frequencies, the actual 'g' value corresponding to the fundamental frequency is used, except that such 'g' value must be equal to or greater than the maximum 'g' values corresponding to all higher frequencies.

3.7(B).3.2 Determination of Number of Earthquake Cycles

During the forty-year design life of the plant, the occurrence of one safe shutdown earthquake (SSE) and five operating basis earthquakes (OBE) are postulated. For fatigue analysis requirements of safety Class 1 components, a maximum of twenty stress cycles per earthquake is postulated, which results in a total of one hundred cycles for the five OBE events. For the SSE, no estimate of the number of stress cycles is made, since it is a faulted condition, and a fatigue analysis is not required.

3.7(B).3.3 Procedure Used for Modeling

a. Modeling of Piping Systems for Dynamic Analysis

The piping systems were analyzed using a three-dimensional structural model composed of concentrated lumped masses connected by massless spring elements having the same strength and stiffness properties as the pipe. The model accounts for the interaction effect between piping, equipment and supports. Supports were modeled as flexible members with the appropriate spring rate to represent the support stiffness. The piping model was terminated at equipment nozzles that were modeled as rigid anchors with consideration given to the seismic amplification of equipment, as follows:

1. For rigid equipment in which the fundamental frequency was equal or higher than 33 Hz, the amplified response spectra of the structure was used.

2. For equipment in which the fundamental frequency was lower than 33 Hz, the amplified response spectra and the seismic anchor displacement of the equipment at the pipe/nozzle interface point was used.

All in-line components were included in the model. The concentrated mass of in-line components such as valves, flanges, and strainers were represented as lumped masses. Valve operators were modeled as an offset lumped mass to account for the torsional and in-plane bending effects on the piping.

The following criteria were used for the decoupling of piping subsystems:

1. Piping was decoupled from the equipment, and the nozzle modeled as a full, six-degree-of-freedom restraint.

2. Branch connections were decoupled from the main runs when the ratio of the branch to run section moduli was equal to or less than 0.05.

3. Piping subsystems which were decoupled into separate analytical models satisfied one of the following criteria:

   (a) The boundary of the decoupling point is a full anchor for the piping of both separate models.

   (b) The boundary of each decoupled model contains a region of common overlap to both models which provides restraint(s) in each of the three orthogonal directions.

b. Modeling of Equipment

Seismic Category I equipment is modeled as lumped-mass spring systems which consist of a series of discrete mass points connected by massless elastic members. All significant concentrated weights are represented as lumped masses. Typical examples of concentrated weights are weights of motor rotor and pump impeller in the analysis of shafts. In general, masses are lumped at points where the maximum displacements are anticipated to occur. The number of lumped masses are such that the mathematical model represents the equipment response as closely as possible.

3.7(B).3.4 Basis for Selection of Frequencies

All frequencies below 33 Hz are considered in computing the total response of the equipment and components. Whenever possible, equipment and components are designed so that their fundamental frequencies are less than half or more than twice the dominant frequencies of the support structure. Where this is found to be impractical or impossible to achieve, the equipment and components are adequately designed for the amplified loading.

3.7(B).3.5 Use of Equivalent Static Load Method of Analysis

The criteria and procedure for the use of the equivalent static load method are described in Subsection 3.7(B).3.1b.

3.7(B).3.6 Three Components of Earthquake Motion

Seismic Category I subsystems and components are analyzed by considering the combined effects of seismic loads occurring in three mutually perpendicular directions, two in the horizontal direction and one in the vertical direction. The total combined response (displacements, stresses, and forces) due to the three components of earthquake motion is obtained by using the square root of the sum of the squares method as follows:

$$X_c = \sqrt{\sum_{j=1}^{3} X_j^2}$$

Where X_c = Total combined response of the parameter x (displacement, stress or force etc.)

X_j = Value of the combined response of the parameter x in the j-direction of the earthquake.

3.7(B).3.7 Procedures for Combining Modal Responses

The combined response of equipment and components in a given earthquake direction is obtained by combining the individual modal responses using the square-root-of-the-sum-of-the-squares method. For equipment and components having modes with closely spaced frequencies, this method is modified to include the possible effects of these modes. The groups of closely spaced modes are chosen such that the difference between the frequencies of the consecutive modes in a group does not exceed ten percent. No one frequency is in more than one group. For closely spaced modes, the modal response in a particular earthquake direction is obtained by using Equation (1) for piping and associated in-line components and by Equation (2) for equipment and other components.

For closely spaced modes, X_j is given by:

$$X_j = \left[\sum_{k=1}^{N'} X_k^2 + \sum_{q=1}^{P} \left\{ \sum_{m=1}^{nq} |X_{mq}| \right\}^2 \right]^{1/2}$$

Where, $N' = N - \sum_{q=1}^{P} n_q$

P	=	Number of groups of closely spaced modes
n_q	=	Number of closely spaced modes in group number q
N	-	Total number of modes
X_{mq}	=	Maximum value of the response of the element attributed to the mth mode of group number q
X_K	=	Maximum value of the response of the element due to Kth mode

An alternate expression for X_j is given by:

$$X_j = \left[\sum_{k=1}^{N'} X_k^{\,2} \right]^{1/2} + \sum_{m=1}^{R} |X_m|$$

Where the variables are as defined above except for the following:

N'	=	N-R
R	=	Modes with closely spaced frequencies.
X_m	=	Maximum value of the response of the element attributed to the mth mode.

The combined response of piping in a given earthquake direction is obtained by combining the individual modal responses using the square-root-of-the-sum-of-the-squares (SRSS) method. The closely spaced mode technique described in Subsection 3.7(B).3.7 may also be used.

3.7(B).3.8 Analytical Procedures for Piping

The procedure used for modeling of piping systems is described in Subsection 3.7(B).3.3a. The analytical procedures applicable to piping systems are as follows:

a. Dynamic Analysis of Piping Systems

Detailed seismic analysis of piping systems is performed by using finite element analysis programs to ensure that the stresses in the piping system meet the applicable ASME Section III Code requirements. Modal Response Spectrum Analysis technique is used for the dynamic analysis of piping systems. The seismic loading input consists of amplified floor response spectra obtained for discrete structural locations.

The piping system is represented by a three-dimensional lumped-mass model, and is analyzed by determining the response of the system to the three components of the earthquake motion. The procedure for determining the total combined response is described in Subsection 3.7(B).3.6.

b. Procedures to Account for Differential Piping Support Movement

The piping system is analyzed for differential support movement at different support points located within a building and between buildings by using the static analysis method, the maximum relative support displacements are obtained from structural response calculations, and the worst differential movements between support points is used for the piping analysis.

Pipe Stresses from support movement analysis are combined with the other stresses in appropriate Code equations. Loads on supports are added to other applicable loads.

c. Equivalent Static Analysis of Piping Systems

When seismic analysis of piping systems is performed using the static analysis approach, the total combined response (displacements, stresses, and forces) due to the three components of earthquake motion is obtained by using the square-root-of-the-sum-of-the-squares method. The response of the above parameters in each earthquake direction is obtained by applying a uniformly distributed 'g' load in that direction multiplied by a static coefficient of 1.5. The magnitude of the 'g' value corresponds to the fundamental frequency of the system, and is obtained from the applicable amplified response spectra curve.

3.7(B).3.9 Multiply-Supported Equipment Components with Distinct Inputs

When the response spectrum method is used to analyze equipment and components supported at different elevations within a building or between buildings, typically, an upper bound envelope of all the individual response spectra for these locations is used to calculate the inertial responses of the multiply-supported equipment and components. In addition, the relative displacements at the support points are determined according to the procedure outlined in Subsection 3.7(B).3.8b.

Where the response of selected items in the analytical model is under the direct influence of a specific support point amplified response spectra, then this analysis may be performed using this amplified response spectra. Additionally, where necessary, alternate methods are used which employ acceptable techniques which consider multiple amplified response spectra.

3.7(B).3.10 Use of Constant Vertical Static Factors

Constant vertical static factors are not used for the design of seismic Category I subsystems.

3.7(B).3.11 Torsional Effect of Eccentric Masses

The mathematical model generated to represent a piping system includes consideration of eccentric masses, such as valve operators.

The eccentric mass is represented in the model as a lumped mass at the end of a weightless beam cantilevered from the valves. The length of the beam is equal to the distance between the center of gravity of the operator and the axis of the pipe.

3.7(B).3.12 Buried Seismic Category I Piping Systems and Tunnels

Seismic Category I, ASME Section III, piping systems have been analyzed by utilizing the schemes and procedures developed by Newmark and Rosenblueth (Reference 3), Shah and Chu (Reference 4) and Iqbal and Goodling (Reference 5).

The criteria for analysis of buried piping systems is to demonstrate their capability to withstand seismic soil strain and internal pressure loads without exceeding code allowable stress levels. Analysis of these underground pipes subjected to seismic ground motion is based on the piping system configuration, boundary conditions, and the elastic properties of the soil and piping.

Seismic Category I, ASME Section III, buried piping which penetrates structures where seismic soil-structure differential movements are expected is protected by providing adequate clearance between pipe and sleeve at the penetrations or by additional bend for flexibility.

There are no seismic Category tunnels which are analyzed as subsystems. Electrical, piping and passage tunnels are analyzed as seismic Category I structures, and are discussed in Subsection 3.7.2.

3.7(B).3.13 Interaction of Other Piping with Seismic Category I Piping

The loading effects of nonseismic Category piping portions on seismic Category I piping is kept to a minimum by providing a restraint, or a group of restraints, at or beyond the defined boundaries, and extending the seismic analysis performed for the seismic Category I portion of the piping system to the boundary restraints. These restraints are designed to withstand the most severe loading combinations that could result from the nonseismic Category I piping.

More specifically, to assure that an earthquake of SSE intensity will not result in failure of seismic Category I piping, the mathematical models constructed for Category I piping include the nonseismic Category I piping up to a boundary restraint (defined as a set of one or more restraints or anchor). Typical functions and locations of these restraints are shown in Figure 3.7(B)-37. In Case 1, moments and forces introduced by the nonseismic piping are reacted through the six degree-of-freedom restraint as shown. In Case 2, moments are reacted by couples between pairs of restraints which limit motion in the indicated directions. These boundary restraints are designed to withstand the plastic capability of the contiguous piping.

The structural integrity of the nonseismic Category I piping is assured by compliance with the stress limits shown on Figure 3.7(B)-37. Class 2 or 3 piping must satisfy the limits of ASME III, NC-3600 or ND-3600. Pipes and supports in the shaded regions are included in the mathematical model of the seismic Category I piping and are subjected to SSE loads. Nonseismic piping beyond the boundary restraints satisfies the limits of B31.1. Loading conditions and stress limits for all pipe classifications are summarized in Subsection 3.9.3.

3.7(B).3.14 Seismic Analysis of Reactor Internals

See Subsection 3.7(N).3.14.

3.7(B).3.15 Analysis Procedure for Damping

Where the equipment or component consists of subcomponents with the same damping characteristics, the same critical damping value is used for the entire equipment or component. The corresponding critical damping value is chosen from Table 3.7(B)-23. For seismic Category I equipment or component consisting of subcomponents with different damping characteristics, the lowest critical damping value associated with the subcomponents in the equipment or component is used in the analysis for all modes.

Alternate critical damping values are used for a specific component or subsystem, where documented test data justifies such usage.

3.7(B).4 Seismic Instrumentation

In the event of an earthquake, the seismic instrumentation will provide information on the input ground motion and resultant vibratory responses of representative Category I structures and equipment so that an evaluation can be made as to:

- Whether input design response spectra were exceeded
- Whether the vibratory responses of the representative Category I structures and equipment were exceeded
- The need for shutdown of the plant
- The degree of validity of the mathematical models used in the seismic analysis of the buildings and equipment.

The design requirement of the seismic instrumentation is based on a Safe Shutdown Earthquake (SSE) of 0.25g (peak) and time duration of fifteen (15) seconds, and an Operating Basis Earthquake (OBE) of 0.13g (peak).

The SM instrumentation is not Class 1E qualified. It is not designed for a design basis event environment, except the SSE. It is ANS Safety Class NNS and Seismic Category I.

3.7(B).4.1 Comparison with Regulatory Guide 1.12

The SM system instrumentation type, number and locations described follow the guidance provided in USNRC Regulatory Guide 1.12, revision 1 (RG 1.12), 1974, with exceptions as discussed below. These exceptions are consistent with draft Regulatory Guide DG-1016, "Nuclear Power Plant Instrumentation for Earthquakes."

Triaxial peak accelerographs, recommended by RG 1.12, for attachment to reactor equipment and piping inside containment, and to Seismic Category I equipment or piping outside containment, are not installed as part of the SM system.

All locations where response spectrum recorders are recommended by Regulatory Guide 1.12 are monitored by triaxial time history accelerographs. Triaxial time history accelerograph XT-6701, linked to the SSA-3 digital recording system, provides the seismic response of the Containment foundation.

RG 1.12 recommends the installation of accelerometers on two elevations (not including the foundation) of the Containment building. Triaxial time history accelerographs are installed on the Containment foundation at elevation (-)26', and the operating deck at (+)25'.

RG 1.12 recommends the installation of accelerometers on two independent Seismic Category I foundations (not including the Containment foundation). One triaxial time history accelerograph is installed in the Service Water Pump House at elevation 22'.

One triaxial time history accelerograph is installed on elevation 53' in the PAB, not on two elevations of two independent Seismic Category I buildings as recommended by RG 1.12.

3.7(B).4.2 Location and Description of Instrumentation

The SM instrumentation described below is intended to provide timely and accurate technical information necessary for an informed assessment of the integrity of safety-related components, systems and structures immediately following an earthquake.

Electric power for seismic instrumentation is provided from a Class 1E, 120V, uninterruptable power supply.

a. Time History Accelerograph System SSA-3/SSP-1

Seismic instrumentation is provided, one each at the following locations:

1. At a free field position in the control room east air intake, on bedrock.

2. Between columns 16 and 17 in the containment building foundation at elevation (-) 26'-0".

3. Between columns 16 and 17 on the concrete operating floor in the Containment Building at elevation (+)25'-0".

The triaxial accelerometer (type FBA-3) has a frequency range from 0.0 to 50.0 Hz. Each triaxial accelerometer is connected to a digital recorder that is part of the SSA-3/SSP-1 system located in seismic control panel 1-SM-CP-58. The trigger method is a threshold exceedance type. Its threshold level is set at 0.01g to avoid actuation due to insignificant motion but to record a seismic disturbance which creates significantly lower ground accelerations than that of the OBE (0.13g).

The two seismic instrumentation packages located in the Containment building are oriented so that the axes of the accelerometers are pointing in the same direction and are aligned to the axis of the building.

The Kinemetrics solid state accelerograph system, SSA-3/SSP-1, is a central recording, time history accelerograph and playback system. The SSA-3 digital triggering system continuously monitors the signals sent by remote acceleration sensors. When motion exceeds the preprogrammed threshold, the recorders turn on and begin sampling concurrently on all channels. The data is automatically recorded with information such as sensor location, event time and sensor serial number. When the system detriggers, the SSP-1 interrogates the recorders and downloads the event files into a dedicated PC for automatic OBE analysis. The total capacity of the SSA-3 is 25 minutes.

b. Solid State Accelerographs, SSA-1

Two stand-alone solid state accelerographs, SSA-1, are installed in the following locations:

1. In the Primary Auxiliary Building, PAB, at elevation 53'.

2. On the southwest corner of the electrical control room of the Service Water (SW) Pumphouse at elevation 22'-0".

Each recorder can trigger and record a seismic event using a preset threshold level as described for the SSA-3 system. Each SSA-1 recorder has an internal triaxial accelerometer with a full scale of $\pm 2g$, a frequency response from 0.0 to 50.0 Hz, and a 70% damping coefficient.

c. Criteria for Instrument Location

The selection of the above locations is based on the guidance provided in USNRC Regulatory Guide 1.12, Revision 1, for an SSE acceleration of less than 0.3g with exceptions as provided in Subsection 3.7(B).4.1.

All instruments are accessible for inspection, test and service.

Table 3.7(B)-24 summarizes the location of each primary instrument.

3.7(B).4.3 Control Room Operator Notifications

Audio visual alarms are provided on the main control board for the following parameters:

a. "SEISMIC EVENT IN PROGRESS"

b. "SEISMIC MONITOR LOSS OF AC/DC"

These conditions apply to instruments at the following locations:

1. East air intake
2. Containment building foundation
3. Containment operating floor alarmed as "Seismic Event in Progress"

3.7(B).4.4 Comparison of Measured and Predicted Responses

The SSP-1 is an automatic data retrieval and analysis system based on a high-speed, rack-mounted computer. The system remains on at all times monitoring the triggering activity of the SSA-3. When the SSA-3 has recorded an event, the SSP-1 retrieves the time history data and automatically evaluates OBE exceedance criteria including a computation of response spectra, cumulative absolute velocity (CAV) and spectral velocity for each axis (X,Y,Z). A graphics printer allows immediate print and plot results for operator review.

Detailed comparisons are made between SM system measured responses and calculated responses based on plant dynamic models.

The time history records from the east air intake are used to calculate response spectra at the appropriate critical damping ratio. This is compared to the design response spectra.

The response spectra measured at the Service Water Pumphouse foundation and the Primary Auxiliary Building at elevation 53' are compared with those calculated by using the time history records from the east air intake as input ground motion to the containment and primary auxiliary building dynamic models. These comparisons indicate the validity of the dynamic model and form the basis for adjustment of the model.

Measured structural responses and response spectra are compared against the original design and analysis parameters to permit evaluation of the seismic effect on structures and equipment. These comparisons provide the basis for a detailed physical inspection of structures and equipment.

3.7(B).5 In-Service Surveillance

Calibration and alignment on three orthoginal axes are performed prior to fuel loading to assure proper operation. Periodic testing and calibration is performed in accordance with Technical Requirements.

3.7(B).6 References

1. "Nuclear Reactors and Earthquakes," TID-7024, Prepared by Lockheed Aircraft Corporation and Holmes & Narver, Inc., for the Division of Reactor Development, U. S. Atomic Energy Commission, Washington, D. C., August 1963.

2. Whitman, R.V., 1969: "Equivalent Lumped System for Structure Founded Upon Stratum of Soil," Proc. 4th World Conf. on Earthquake Engineering, Chile, Vol. III, Section A-6, pp. 133-142.

3. Newmark, N. M. and Rosenblueth, E., "Fundamentals of Earthquake Engineering," Prentice-Hall, 1971.

4. Shah, H. H. and Chu, S. L., "Seismic Analysis of Underground Structural Elements," Journal of the Power Division, Proceedings of the American Society of Civil Engineers, Vol. 100, No. P01, July 1974, PP. 53-62.

5. Iqbal, M. A. and Goodling, E. C., "Seismic Design of Buried Piping," and ASCE Specialty Conference on Structural Design of Nuclear Plant Facilities, December 1975.

3.8 DESIGN OF CATEGORY I STRUCTURES

3.8.1 Concrete Containment

The containment structure houses the major portion of a PWR Nuclear Steam Supply System (NSSS). During the operating life of the plant, it will also provide the following functions:

a. Limiting the leakage rate to the maximum allowable Type "A" test leakage rate, 0.15 percent by weight of the containment contained air mass per day at calculated peak pressure and associated temperature, resulting from any loss-of-coolant accident (LOCA) and other postulated accidents.

b. Providing continuing radiation shielding during normal plant operation in accordance with 10 CFR 20 and during accident conditions in accordance with 10 CFR 100.

c. Protecting the reactor vessel and all other safety-related systems, equipment and components located inside the containment against all postulated external environmental conditions and resulting loads.

3.8.1.1 Description of Containment

The containment, Figure 1.2-2, Figure 1.2-3, Figure 1.2-4, Figure 1.2-5 and Figure 1.2-6, is a seismic Category I reinforced concrete dry structure, which is designed to function at atmospheric conditions. It consists of an upright cylinder topped with a hemispherical dome, supported on a reinforced concrete foundation mat which is keyed into the bedrock by the depression for the reactor pit and by continuous bearing around the periphery of the foundation mat. The inside diameter of the cylinder is 140 feet and the inside height from the top of the base mat to the apex of the dome is approximately 219 feet; the net free volume is approximately 2,704,000 cubic feet.

A welded steel liner plate, anchored to the inside face of the containment, serves as a leaktight membrane. Although not a code requirement, welds that are embedded in concrete and not readily accessible are covered by a leak chase system which permits leak testing of those welds throughout the life of the plant. Exemptions to these inaccessible welds are the welds joining mechanical penetrations X-60 and X-61 to the steel liner plate. (The venting pipes which join the leak chase channels for these penetrations to the atmosphere were not provided; however, these welds underwent proper testing before they became inaccessible.) The liner on top of the foundation mat is protected by a four feet thick concrete fill mat which supports the containment internals and forms the floor of the containment.

The containment is designed to assure that the base mat, cylinder, and dome behave integrally to resist all loads.

Located outside the Containment Building and having a similar geometry is the Containment Enclosure Building. This structure provides leak protection for the containment and protects it from certain loads, as discussed in Subsection 3.8.1.3. The Containment Enclosure Building is described in Subsection 3.8.4.

a. Base Mat

The reinforced concrete base mat is 153 feet in diameter and 10 feet thick. It is designed to carry the loads from the shell of the containment and from the internal structures.

An orthogonal grid rebar arrangement is provided for the bottom face of the base mat to simplify fabrication and construction. A radial and hoop pattern is used at the top face to minimize interference with cylinder dowels. Vertical and inclined shear reinforcement are provided to resist the transverse shear forces caused by design accident pressure and seismic loads. Details of the base mat reinforcing steel are shown on Figure 3.8-1. The mat liner plate is 1/4" thick with joints welded to leveling angles which serve as welding backup strips.

Internal structures are supported on and anchored to the fill mat, as indicated above. The mat is not anchored to the base mat. Stability of the containment with internals is provided by the keying action of the base mat and reactor pit in the rock and by bearing against the foundation for the Enclosure Building, which in turn transfers all horizontal shears directly into the bedrock through fill concrete.

b. Cylinder

The cylinder has an inside diameter of 140 feet and is nominally 4'-6" thick. Also, it is thickened to provide room for additional reinforcing steel around the openings for the equipment hatch and the personnel air lock.

The reinforcing bars in the cylinder are arranged and oriented to resist hoop, meridional and shear forces, including hoop, meridional and radial shear forces produced by bending moments. Orthogonal layers of bars in the meridional and hoop directions are provided on each face to resist the membrane forces primarily from pressure and seismic loads.

An orthogonal set of bars inclined at 45 degrees to the horizontal is provided on the outside face to resist in-plane seismic shear forces and membrane tension from other loads. Near the base of the containment, additional meridional bars and radial inclined stirrups are provided to resist discontinuity moments and radial shears, respectively, caused by the restraint on the cylinder at the junction of the cylinder and the base mat. Stirrups are also provided at the springline to resist radial shear.

Where there are large openings for access hatchways and penetrations, the main reinforcing bars are continued without interruption around the openings. No main reinforcing bars are terminated at any opening. Furthermore, additional bars are provided to resist the local effects of these openings and, around large openings such as the equipment hatch (28'-0" inside diameter) and personnel air lock (7'-1¼" inside diameter), the concrete is thickened locally to resist the additional local forces and to accommodate the additional reinforcing.

Basic details of reinforcing steel in the cylinder are shown on Figure 3.8-2, which also includes reinforcing steel in the dome and junction of the wall and base mat. Details of the reinforcing steel at the equipment hatch and personnel air lock are shown on Figure 3.8-3 and Figure 3.8-4, respectively. These figures also show the transition of reinforcing steel between the openings and the membrane regions.

The liner plate in the cylinder is 3/8" thick in all areas except penetrations and the junction of the base mat and cylinder where it is 3/4" thick. The liner is provided with an anchorage system to assure that it can withstand accident loadings while maintaining leak tightness. In addition, the anchorage system assures that the liner, which is used as a form during construction, can resist the hydrostatic concrete loads while maintaining liner tolerances within allowable values. The anchorage system consists of vertical tees spaced every 20 inches around the circumference of the cylindrical wall. The webs of the tees are welded to the liner plate with two ¼" continuous fillet welds. Bent studs are attached to the flange of vertical tees as required to accommodate placement of rebar. Liner details are shown on Figure 3.8-5.

Containment penetrations, other than the equipment hatch and personnel air lock, are located in the lower portion of the cylindrical structure. In general, a penetration consists of a sleeve anchored in the concrete cylinder wall and welded to the locally thickened containment liner. The weld between the liner and the sleeve is covered by a leak chase system which can be pressurized to demonstrate the integrity of the penetration-to-liner weld joint. The piping, electrical cable and instrumentation cable pass through the embedded sleeves and the ends of the resulting annuli are closed off either by welded end plates or a flued head welded to the sleeve outside the containment. If the pipe carries hot fluid, the space between the pipe and the sleeve is insulated to maintain the concrete temperature adjoining the embedded sleeve at or below 200° F during normal plant operation. The fuel transfer tube passes through an embedded sleeve which has its ends closed off by an expansion bellows and an end plate. In the case of ventilation ducts, the sleeve forms the wall of the duct.

Sleeves for all penetrations, including the equipment hatch and personnel air lock, are embedded in the concrete wall by an engineered anchorage system that is welded to the penetration sleeve. Reinforcing steel, hoop, meridional and diagonal, is splayed around penetrations permitting all bars to be continuous. See Figure 3.8-6 for details of reinforcing at penetrations.

All brackets and attachments are welded to attachment plates which are welded to the liner plate and anchored into the concrete by studs welded on the opposite side of the liner, thereby transmitting the forces directly to the concrete. See Figure 3.8-5. In the reactor cavity pit, brackets are used, as a construction aid, on the concrete side of the liner to temporarily support the reinforcing steel. This is a permanent attachment which only functions until the concrete is placed. Each of these brackets is welded to an attachment plate; the attachment plate is welded to the liner with a continuous fillet weld and is not backed by a stud.

c. Dome

The dome is a reinforced concrete shell 3'-6" thick and 69'-11" in radius. Due to the change in concrete thickness, the discontinuity of concrete at the springline is on the outer surface.

Reinforcing steel in the dome consists of hoop, meridional, and diagonal bars, as in the cylinder. The meridional and diagonal bars are continuous with those in the cylinder.

One-half of the meridional bars, in an alternating fashion, are terminated at 60° above the springline with the remaining bars evenly spread and continued across the upper 30° of the dome.

The hoop bars in the dome are terminated where no longer needed at 75° above the springline.

One-half of the diagonal bars are terminated where no longer needed, at approximately 30° above the springline; the remaining bars are terminated at approximately 45° above the springline.

See Figure 3.8-7 for details of the reinforcing steel.

The dome liner is 1/2" thick and flush with the outside face of the cylindrical liner. The anchorage system consists of tees on a 5'-0" grid pattern. A bent stud is located in the center of each of the resulting 5'-0" x 5'-0" panels to provide some additional anchorage. See Figure 3.8-5.

d. Steel Components

Steel components that resist pressure and are not backed by structural concrete include the following:

1. Equipment hatch
2. Personnel air lock
3. High energy piping penetrations
4. Moderate energy piping penetrations
5. Electrical penetrations
6. Fuel transfer tube assembly
7. Instrumentation penetrations
8. Ventilation penetrations

These are discussed in Subsection 3.8.2.

3.8.1.2 Applicable Codes, Standards and Specifications

The design, materials, fabrication, construction, testing and inspection of the containment structure conform to the applicable sections of the following codes and specifications which are used to establish design bases and methods, analytical techniques, material properties and quality control provisions.

Dates and revision given for the listed codes are of the earliest version that were used. Subsequent issues were incorporated into the design where practicable or where the new issue directly affected the safety of the structure.

Any exceptions to the indicated issue of the ASME B&PV Code, Section III, Division 2, are indicated in the text, either in Subsection 3.8.1.4 or in Subsection 3.8.1.5, as appropriate.

Code or Specification	Title
ACI 211.1-74	Recommended Practice for Selecting Proportions For Normal Weight Concrete
ACI 214-65	Recommended Practice for Evaluation of Compression Test Results of Field Concrete
ACI 301-72	Structural Concrete for Building
ACI 304-73	Recommended Practice for Measuring, Mixing, Transporting, and Placing Concrete
ACI Committee Report 74-33	Recommended Practice for Hot Weather Concreting
ACI Report 306R-78	Recommended Practice for Cold Weather Concreting
ACI 308-71	Recommended Practice for Curing Concrete
ACI 309-72	Recommended Practice for Consolidating Concrete
ACI 311-64	Recommended Practice for Concrete Inspections
ACI 315-65	Manual of Standard Practice for Detailing Reinforced Concrete Structures

Code or Specification	Title
ACI 318-71	Building Code Requirements for Reinforced Concrete (with Commentary)
ACI 347-68	Recommended Practice for Concrete Formwork
ACI 614-59	Recommended Practice for Measuring, Mixing and Placing Concrete
ACI Committee Report 68-33	Placing Concrete by Pumping Methods
ACI Committee Report 72-33	Placing Concrete with Belt Conveyors
ACI SP2	Manual of Concrete Inspection, 1975 Edition
CRSI	Reinforced Concrete - Manual of Standard Practice, 22nd Edition, first printing, 1976
CRSI	Recommended Practice for Placing Reinforcing Bars, 1968 Edition
ASME	ASME Boiler and Pressure Vessel Code, Section III, Division 2, Code for Concrete Reactor Vessels and Containments:
	Concrete Containment Component - 1975 Edition
	Containment Liner - 1975 Edition (1976 Winter Addendum used for liner allowable stresses)
	Reinforced Concrete - 1975 Edition through Winter 1976 Addenda (also 1977 Winter Addendum, Article CC-3422 and 1979 Summer Addendum, Article CC-3422)

[1] From here on referred to as Division 2.

[2] Article CA-8000 of Division 2 applies, except that in lieu of code symbol application a New Hampshire State special waiver, dated March 18, 1976, regarding marking, stamping and recording has been granted.

Code or Specification	Title
	SIT - 1980 Edition, as referenced in Updated FSAR
	Subsection 3.8.1.7a, except as noted in applicable subsections of Updated FSAR Section 3.8.[1 2]
ASME	ASME Boiler and Pressure Vessel Code, Section II, Material Specification, Part C, Welding Rods, Electrodes and Filler Metals (up to and including Winter 1975 Addenda)
ASME	ASME Boiler and Pressure Vessel Code, Section IX, Welding and Brazing Qualifications (up to and including Winter 1974 Addenda)
ASME	ASME Boiler and Pressure Vessel Code, Section V, Nondestructive Examination (up to and including 1974 Summer Addendum)
ASME	ASME Boiler and Pressure Vessel Code, Section III, Division 2 Code Cases:
	Code Case N-218, Testing Lots of Carbon Steel Solid, Base Welding Electrode on Wire, August 14, 1981.
	Code Case N-219, Rules for Design of Peripheral Shear Reinforcing, January 8, 1979.
	Code Case N-287, Rules for Design of Radial Shear Reinforcing, July 14, 1980.
	Code Case N-232, Alternate Rules for Development Length of Reinforcing Steel Not Required to Carry Load, January 21, 1982.
ASME	ASME Boiler and Pressure Vessel Code, Section II, Material Specification, Part A, Ferrous Materials (up to and including Winter 1974 Addenda)
SA-36	Specification for Structural Steel
SA-240	Specification for Stainless and Heat Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet and Strip for Fusion-Welded Unfired Pressure Vessels

Code or Specification	Title
SA-300	Specification for Notch Toughness Requirements for Normalized Steel Plates for Pressure Vessels
SA-333	Specification for Seamless and Welded Steel Pipe for Low Temperature Service
SA-442	Specification for Carbon Steel Plates with Improved Transition Properties for Pressure Vessels
SA-516	Specifications for Carbon Steel Plates for Pressure Vessels for Moderate and Lower Temperature Service
ASTM A36-70	Specification for General Requirements for Delivery of Rolled Steel Plates, Shapes, Sheet Piling and Bars for Structural Use
ASTM A184-65	Specification for Fabricated Steel Bar or Rod Mats for Concrete Reinforcement
ASTM A185-70	Specification for Welded Steel Wire Fabric for Concrete Reinforcement
ASTM A325-71	Specification for High Strength Bolts for Structural Steel Joints, Including Suitable Nuts and Plain Hardened Washers
ASTM A370-75a	Methods and Definitions for Mechanical Testing of Steel Products
ASTM A449-68	Specification for Quenched and Tempered Steel Bolts for Structural Steel Joints
ASTM A490-71	Specification for Quenched and Tempered Alloy Steel Bolts for Structural Steel Joints
ASTM A501-71	Specification for Hot Formed Welded and Seamless Carbon Steel Structural Tubing
ASTM A514-70	Specification for High-Yield Strength, Quenched and Tempered Alloy Steel Plate, Suitable for Welding
ASTM A519-75	Standard Specification for Seamless and Alloy Mechanical Tubing

Code or Specification	Title
ASTM A615-75	Specification for Deformed Billet-Steel Bars for Concrete Reinforcement
ASTM C29-71	Standard Methods of Test for Unit Weight of Aggregate
ASTM C31-69	Standard Method of Making and Curing Concrete Compressive and Flexural Strength Test Specimen in the Field
ASTM C33-71a	Specification for Concrete Aggregates
ASTM C39-71	Standard Method of Test for Compressive Strength of Molded Concrete Cylinders
ASTM C40-73	Standard Method of Test for Organic Impurities in Sands of Concrete
ASTM C42-68	Standard Method of Obtaining and Testing Drilled Cores and Sawed Beams of Concrete
ASTM C70-73	Standard Method of Test for Surface Moisture in Fine Aggregate
ASTM C87-69	Standard Method of Test for Effect of Organic Impurities in Fine Aggregate on Strength of Mortar
ASTM C88-73	Standard Method of Test for Soundness of Aggregates by Use of Sodium Sulfate or Magnesium Sulfate
ASTM C109-73	Standard Method of Test for Compressive Strength of Hydraulic Cement Mortars (Using 2-inch Cube Specimens)
ASTM C114-69	Standard Methods of Chemical Analysis of Hydraulic Cement
ASTM C117-69	Standard Method of Test for Materials Finer than No. 200 (75-uM) Sieve in Mineral Aggregates by Washing
ASTM C123-69	Standard Method of Test for Light Weight Pieces in Aggregate
ASTM C125-74	Standard Definitions of Terms Relating to Concrete and Concrete Aggregates

Code or Specification	Title
ASTM C127-73	Standard Method of Test for Specific Gravity and Absorption of Coarse Aggregate
ASTM C128-73	Standard Method of Test for Specific Gravity and Absorption of Fine Aggregate
ASTM C131-69	Standard Method of Test for Resistance to Abrasion of Small Size Coarse Aggregate by the Use of the Los Angeles Machine
ASTM C136-71	Standard Method of Test for Sieve of Screen Analysis of Fine and Coarse Aggregates
ASTM C138-75	Standard Method of Test for Unit Weight, Yield, and Air Content (Gravimetric) of Concrete
ASTM C142-71	Standard Method of Test for Clay Lumps and Friable Particles in Aggregates
ASTM C143-71	Standard Method of Test for Slump of Portland Cement Concrete
ASTM C150-71	Specification for Portland Cement
ASTM C151-74a	Standard Method of Test for Autoclave Expansion of Portland Cement
ASTM C172-71	Standard Method of Sampling Fresh Concrete
ASTM C173-75	Standard Method of Test for Air Content of Freshly Mixed Concrete by the Volumetric Method
ASTM C192-69	Standard Method of Making and Curing Concrete Test Specimens in the Laboratory
ASTM C231-71T	Tentative Method of Test for Air Content of Freshly Mixed Concrete by the Pressure Method
ASTM C233-73	Standard Method of Testing Air-Entraining Admixtures for Concrete

Code or Specification	Title
ASTM C235-68	Standard Method of Test for Scratch Hardness of Coarse Aggregate Particles
ASTM C260-69	Specification for Air-Entraining Admixtures for Concrete
ASTM C295-65	Recommended Practice for Petrographic Examination of Aggregates for Concrete
ASTM C309-74	Standard Specification for Liquid Membrane-Forming Compounds for Curing Concrete
ASTM C494-71	Specification for Chemical Admixtures for Concrete
ASTM C496-71	Standard Method of Test for Splitting Tensile Strength of Molded Concrete Cylinders
ASTM C566-67	Standard Method of Test for Total Moisture Content of Aggregate by Drying
ASTM C666-75	Standard Method of Test for Resistance of Concrete to Rapid Freezing and Thawing
ASTM D75-71	Standard Method of Sampling Aggregates
ASTM D297-74	Methods of Chemical Analysis of Rubber Products
ASTM D412-68	Method of Tension Testing of Vulcanized Rubber
ASTM D573-67	Test for Accelerated Aging of Vulcanized Rubber by the Oven Method
ASTM D624-73	Tests for Tear Resistance of Vulcanized Rubber
ASTM D746-73	Test for Brittleness Temperature of Plastics and Elastomers by Impact
ASTM D816-55	Standard Method of Testing Rubber Cements
ASTM D1149-64	Test for Accelerated Ozone Cracking of Vulcanized Rubber

Code or Specification	Title
ASTM D2240-75	Test for Indentation Hardness of Rubber and Plastics by Means of a Durometer
ASTM E96-66	Standard Methods of Test for Water Vapor Transmission of Materials in Sheet Form
ASTM E329-72	Recommended Practice for Inspection and Testing Agencies for Concrete as Used in Construction (Articles 7, 8 and 9 of ASTM E329 shall not apply).
AISC	Code of Standard Practice for Steel Buildings and Bridges (July 1, 1970)
AISC	Specification for the Design, Fabrication and Erection of Structural Steel for Buildings 1969 Edition (including supplements 1, 2 and 3)
AISC	Specification for Design of Light Gage Cold-Formed Steel Structural Members, 1977
Research Council for Riveted and Bolted Structural Joints of the Engineering Foundation	Specification for Structural Joints Using ASTM A325 or A490 Bolts (September 1, 1966)
ANS 7.60-1972	Standard for Leakage Rate Testing of Containment Structures for Nuclear Reactors
ANSI N177	Plant Design Against Missiles (Draft Standard, April, 1974)
ANSI A58.1-1972	Building Code Requirements for Minimum Design Loads in Buildings and Other Structures
ANSI N45.2-1974	Quality Assurance Program Requirements for Nuclear Power Plants
ANSI N45.3-1973	Electrical Penetration Assemblies in Containment Structures for Nuclear Fueled Power Generating Stations

Code or Specification	Title
ANSI N45.4-1972	Leakage Rate Testing of Containment Structures for Nuclear Reactors
ANSI N101.2-1972	Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities
ANSI N101.4-1972	Quality Assurance for Protective Coating Applied to Nuclear Facilities
ANSI N512-1974	Protective Coatings (Paints) for the Nuclear Industry
AWS D1.0-69	Standard Code for Arc and Gas Welding in Building Construction
AWS B3.0-41	Standard Qualification for Procedures
AWS D12.1-75	Recommended Practice for Welding Reinforcing Steel, Metal Inserts and Connections in Reinforced Concrete Construction
Uniform Building Code	International Conference of Building Officials, Uniform Building Code, 1973 Edition
U.S. Dept. of Labor OSHA	Occupational Safety and Health Administration Standards, October, 1975 Edition
U.S. Dept. of Labor QQ-C-40	Caulking: Lead Wool and Lead Pig, 1963
NRC TID 7024	Nuclear Reactors and Earthquakes (Prepared by Lockheed Aircraft Corp. for NRC, August 1963)
PS-1-74	U.S. Dept. of Commerce, National Bureau of Standards, Construction and Industrial Plywood
Steel Structures Painting Manual Vol. 1	Steel Structures Painting Council, Good Painting Practice, 1966
Steel Structures Painting Manual Vol. 2	SSPC, Systems and Specifications, 1973

Code or Specification	Title
Federal Specifications and Standards Handbook 44	National Bureau of Standards Specifications, Tolerances and Other Technical Requirements for Commercial Weighing and Measuring Devices, 1971

U.S. Army Corps of Engineer Specifications	Title
CRD-C38-73	Method of Temperature Rise in Concrete
CRD-C39-55	Coefficient of Thermal Expansion
CRD-C44-63	Coefficient of Thermal Conductivity
CRD-C119-53	Method of Test for Flat and Elongated Particles in Coarse Aggregate
CRD-C-588-78A	Specification for Expansive Grouts
CRD-C-589-70	Methods of Sampling and Testing Expansion Grouts

NRC Regulatory Guides	Title
1.10, Rev. 1, 1/73	Mechanical (Cadweld) Splices in Reinforcing Bars of Concrete Containments
1.15, Rev. 1, 12/72	Testing of Reinforcing Bars for Concrete Structures
1.18, Rev. 1, 12/72	Structural Acceptance Test for Concrete Primary Reactor Containments
1.19, Rev. 0, 8/72	Nondestructive Examination of Primary Containment Liners
1.54, Rev. 0, 6/73	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants
1.55, Rev. 0, 6/73	Concrete Placement in Category I Structures

NRC Regulatory Guides	Title
1.57, Rev. 0, 6/73	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components
1.60, Rev. 1, 12/73	Design Response Spectra for Seismic Design of Nuclear Power Plants
1.61, Rev. 0, 10/73	Damping Values for Seismic Design of Nuclear Power Plants
1.76, Rev. 0, 4/74	Design Basis Tornado for Nuclear Power Plants
1.84, Rev. 15. 5/79	Code Case Acceptability ASME Section III Design and Fabrication
1.85, Rev. 15, 5/79	Code Case Acceptability ASME Section III Materials
1.92, Rev. 1, 2/76	Combining Modal Responses and Spatial Components in Seismic Response Analysis
1.94, Rev. 1, 4/76	Quality Assurance Requirements for Installation and Testing of Structural Concrete and Structural Steel During Construction Phase of Nuclear Power Plants
1.136, Rev. 2, 6/81	Materials, Construction, and Testing of Concrete Containments

Code of Federal Regulations	Title
10 CFR 20	Standards for Protection Against Radiation
10 CFR 50 App. A	General Design Criteria for Nuclear Power Plants
10 CFR 50 App. B	Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants
10 CFR 50 App. J	Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors
10 CFR 100 App. A	Reactor Site Criteria

The below listed UE&C design and construction specifications applicable to the containment structure were prepared in accordance with applicable codes, quality control requirements and NRC Regulatory Guides:

UE&C Specifications	Title
9763.006-1-1	General Concrete Construction, Steel Erection, and Circulating Water Pipe Installation
9763.006-5-1	Civil Testing Facility and Services
9763.006-5-4	Containment Initial Integrated Leakage Rate Test (ILRT)
9763.006-5-5	Containment Structural Integrity Test (SIT)
9763.006-10-1	Dewatering
9763.006-11-1	Foundation Waterproofing
9763.006-13-2	Containment Concrete Work
9763.006-14-1	Furnishing, Detailing, Fabricating, and Delivering Reinforcing Bars
9763.006-14-2	Installation of Reinforcing Bars in Containment Structure
9763.006-15-1	Containment Liner
9763.006-15-2	Containment Equipment Hatch and Personnel Locks
9763.006-18-2	Installation of Miscellaneous Embedded Steel and Weldments
9763-006-18-4	Furnishing and Installing Embedded Steel and Miscellaneous Steel
9763.006-41-4	Furnishing of Protective Coating (Paint) System Materials and Related Services
9763.006-41-7	Field Painting of Containment Structure Interior
9763.006-69-1	Concrete Batch Plant
9763.006-69-3	Concrete Mixes

UE&C Specifications	Title
9763.006-69-7	Standard Concrete Mixes
9763.006-80-1	Containment Design
9763.006-80-2	Construction of Containments
9763-MPS-1	Material and Processing Requirements for Nuclear Power Plant Components
9763-MPS-3	Material and Processing Requirements for Bending of Welded Studs, Reinforcing Bars and Anchor Bolts
9763-QAS-1	Quality Assurance Administrative and System Requirements (Nuclear)
9763-RM-1	Instructions for Site Records Management System
9763-SD-15-2	Seismic Requirements
9763-WS-1	Requirements for Welding and Nondestructive Examination for Nuclear Pressure Components and Nuclear Power Piping
9763-WS-1-NE	Requirements for Welding and Nondestructive Examination for Nuclear Pressure Class MC Components
9763-WS-4A	Requirements for Welding and Nondestructive Examination for Nuclear Containment Structure Liner
9763-WS-4B	Requirements for Stud Welding and Nondestructive Examination for Nuclear Containment Structure Liner
9763-WS-4C	Requirements for Mechanical Splicing and Nondestructive Examination of the Reinforcing Bars Spliced by the Cadweld Method

3.8.1.3 Loads and Loading Combinations

The containment is designed to withstand all credible conditions of loading, including preoperational test loads, normal loads, severe environmental loads, extreme environmental loads, and abnormal loads. These loads are determined in accordance with Article CC-3000 of Division 2 and are considered in the applicable service and factored load combinations to assure that the response of the structure will remain within the design limits prescribed in Subsection 3.8.1.5. Site-related loads are also considered.

a. Design Loads

The definitions of the loads used in the design of the containment include the following:

1. Preoperational Test Loads

Test loads are those which are applied during the initial and any subsequent structural integrity or leak rate testing of the containment. The definitions for dead loads and live loads are those given in the Normal Load section. The following loads are also considered:

(a) Test Pressure (P_t)

The containment is pressurized to 115 percent of the design pressure to test its structural integrity; i.e., test pressure is 60 psig with design pressure being 52 psig.

(b) Test Temperature (T_t)

The thermal loads occurring during the structural integrity test conditions of the containment are induced by the temperature gradient between the inside containment air temperature and the outside air temperature. The maximum and minimum temperatures considered inside the containment during the test are 100° F and 50° F, respectively. During this test, temperature outside the containment (within the Enclosure Building) will be monitored and controlled to maintain a minimum temperature of 30°F; also, the average differential temperature between the inside and outside of the containment does not exceed 65° F, except the equipment hatch and personnel air locks.

2. Normal Startup, Operational, and Shutdown Loads

Normal loads are those loads encountered during normal plant startup, operation, and shutdown. They include the following:

(a) Dead Loads (D)

Dead loads are all permanent gravity loads including, but not limited to, the weight of the base mat, cylindrical wall, dome, internal concrete structures including fill mat, steel liner plate and structural framing, equipment, piping, cable and cable trays and miscellaneous building loads within the containment. Gravity loads from the internal structures are transmitted to the base mat through the fill mat. Included as a dead load is the buoyant force of the groundwater at El. +20'-0" (i.e., 50'-0" above the top of the base mat) as described in Subsection 2.4.13.5. Buoyancy from the hydrostatic loading is considered in the base mat design and containment stability analysis.

(b) Live Loads (L)

Live loads are those loads which vary in intensity and/or occurrence. During normal operation there are no significant live loads on the external containment surface since the Containment Enclosure Building resists snow loads and lateral loads from soil pressure and normal wind. Live loads from the internal structures are transmitted to the base mat through the fill mat and have no significant effect on the containment shell or base mat design.

(c) Operational Thermal Loads (T_o)

The thermal loads experienced during normal operating or shutdown conditions are generated by temperature gradients through the containment shell and by liner expansion. The gradient through the shell during these normal operating conditions varies between 120°F on the inside and -10°F on the outside. In the base mat the gradient varies between approximately 97°F on the inside to a constant 40°F at the interface with the bedrock; the top of the fill mat is at 120°F. These gradients are shown on Figure 3.8-8. Other temperature gradients, such as 50°F on the inside varying to 90°F on the outside may occur, but the loads produced by them do not control the design of the concrete containment. Note also that the effect of the Containment Enclosure Building in reducing the lower bound outside temperatures was not considered in the containment design; that is, -10°F was used in the evaluation of thermal loads.

(d) Operational Pipe Reactions (R_o)

Piping reactions transmitted to the containment during normal operation or shutdown conditions are based on the most critical transient or steady-state condition. The magnitudes of these loads are determined by the piping design and are included in the Design Report. (The Seabrook Station Containment Design Report is prepared in accordance with Subarticle CA-3240 of Division 2 and is retained by the Owner in accordance with the requirements of Subarticles CA-3100 and CA-4832.)

(e) Pressure Variation (P_V)

Differential pressure loads result from pressure variation either inside the containment or in the annulus between the containment and the Containment Enclosure Building. This pressure variation is produced either by atmospheric fluctuations or by HVAC equipment. The containment structure was designed to withstand a maximum external pressure of 3.5 psi (differential).

The stability of the shell was investigated under the effects of this external pressure load.

3. Severe Environmental Loads

Severe environmental loads are those loads that would result from external conditions which could infrequently be encountered during the plant life. The following loads are included in this category:

(a) Wind Load (W)

There is no wind load considered in the containment design due to the presence of the containment enclosure.

(b) Operating Basis Earthquake Loads (E_o)

These are the loads generated by the Operating Basis Earthquake, which is the earthquake that could reasonably be expected to occur at the plant site during the operating life of the plant. Only the actual dead load and weights of fixed equipment are considered in evaluating the seismic response forces. The horizontal and vertical design response spectra for the OBE are derived by applying a factor of 0.5 to the response spectra given for Safe Shutdown Earthquake (SSE), which is described below. The effects of two (2) orthogonal horizontal components and one (1) vertical component of earthquake are considered and combined by the square-root-of-the-sum-of-the-squares method. Due to the presence of the Containment Enclosure Building, there are no dynamic effects of soil on the containment.

4. Extreme Environmental Loads

Extreme environmental loads are those loads which result from postulated events which are credible, but highly improbable. The following loads are included in this category:

(a) Safe Shutdown Earthquake Loads (E_{ss})

These are the loads generated by the Safe Shutdown Earthquake, which is the maximum potential earthquake that could occur in the vicinity of the site, based on geological and historical investigations. Dead and fixed equipment loads are described under the Operating Basis Earthquake, above. The horizontal and vertical forces on the containment are developed from the response spectra given in Figure 2.5-38 and Figure 2.5-39, the development of which is described in Subsection 2.5.2.6.

The effects of two (2) orthogonal horizontal earthquakes and one (1) vertical earthquake are considered and combined by the square-root-of-the-sum-of-the-squares method. Due to the presence of the containment enclosure, there are no dynamic effects of soil on the containment.

(b) Tornado Loads (W_t)

Due to the presence of the Containment Enclosure Building, wind pressure and pressure variation are not considered in the containment design. In addition, the containment structure is 3'-6" thick at the minimum, and tornado-generated missiles are not deemed capable of penetrating a 24" thick reinforced concrete wall (see Subsection 3.5.1.4). Therefore, missile effects due to the tornado are also not considered in the containment design.

5. Abnormal Loads

Abnormal loads are those generated by postulated high energy ruptures, particularly a rupture in the Reactor Coolant System resulting in a loss-of-coolant accident (LOCA). Post LOCA containment flooding is also considered. The maximum level of flooding, however, is 5'-4" above the top of the fill mat (-26 feet); this depth of water causes negligible loading on the containment structure.

(a) Accident Pressure (P_a)

A transient pressure load is used for the design of the containment. The maximum calculated internal pressure associated with the DBA is 49.6 psig. This provides a margin of 4.8 percent for the design pressure which is 52.0 psig. The pressure-transient curve for the containment is shown on Figure 3.8-9.

(b) Accident Temperature (T_a)

The transient temperature increase of the liner was considered in the design of the containment. The maximum liner temperature is 268°F. However, a maximum liner temperature of 271°F has been used in the design. The temperature transient curves for the containment liner are shown on Figure 3.8-10 and Figure 3.8-11.

The time-dependent thermal gradient through the concrete of the dome, cylinder and base mat was also considered in the design of the containment. When the accident pressure, P_a, is considered, the coincident thermal gradient is equivalent to the normal operating gradient. Due to the high insulating properties of the concrete, the pressure peak occurs before the temperatures within the concrete are appreciably altered. For design of the cylinder and dome, the peak liner temperature and peak pressure were also considered to occur simultaneously. This produced the most conservative design condition where responses to the loads are additive. Where responses are not additive, peak pressure was considered without the thermal loads. The thermal gradients used for the design of the containment are shown on Figure 3.8-8.

For the design of the liner, the transient conditions of liner temperature and coincident accident pressure were considered in order to produce the most conservative design.

(c) Accident Pipe Reactions (R_a)

Pipe reaction loads due to thermal conditions generated by the DBA, including R_o, were considered in the design. The magnitude of these loads was determined by the piping design and is included in the Design Report.

(d) Pipe Break Loads (R_r)

These are local effects on the containment due to the DBA, as follows:

(1) R_{rr} = load on the containment generated by the reaction of a ruptured high energy pipe. The time dependent nature of the load and the ability of the containment to deform beyond yield are considered in establishing the structural capacity necessary to resist the effects of R_{rr}.

(2) R_{rj} = load on the containment generated by jet impingement from a ruptured high energy pipe. In general, direct impingement of steam on the containment does not produce significant design loadings due to the distance between the wall and the break location. Where a break is postulated to occur close enough to produce a critical loading, a shield or deflector is provided, and the loading is transferred to the embedment for the pipe whip restraint to which the shield is attached.

(3) R_{rm} = load on the containment resulting from the impact of a ruptured high energy pipe. Generally, all high energy lines are constrained by pipe restraints, and loading of this nature is prevented. However, in isolated cases where there are postulated pipe whip impact loads, liner adequacy was evaluated for such loads.

6. Site-Related Loads

Site-related loads are loads peculiar to the Seabrook site. These loads are not combined with abnormal loads but are considered on an individual basis.

(a) Aircraft Impact

The impact of an FB-111 type aircraft weighing 81,800 pounds and traveling at 200 mph was evaluated. The effects of this aircraft crash on the containment have been investigated and were found not to be a controlling design consideration. For a complete discussion of the aircraft impact analysis, see Appendix 2P, entitled "Seabrook Station Containment Aircraft Impact Analysis."

(b) Turbine Missiles

The probability that a turbine missile will strike the containment has been shown to be less than 1.0×10^{-7} per year. Consequently, this type of missile was not considered in the design. See Subsection 3.5.1.3 for a complete discussion.

c) Explosions and Delayed Ignition of Vapor Cloud

Structural loadings from explosions and delayed ignition of a vapor cloud were shown to be of a very low magnitude and were not considered in the design. See Subsection 2.2.3.1 for a complete discussion.

(d) Flood Loads (F)

The effect of the design basis flood to Elevation +20.6' (50.6' above the top of the base mat) is considered in the design of the containment. Due to the presence of the Containment Enclosure Building, the only effect of the flood is its buoyant effect on the containment of the structure.

7. Other Pipe Break Loads

All high energy pipe lines are postulated to break, and all moderate energy lines are postulated to crack. The loads produced by these accidents, however, have no effect on the containment design, either because of shielding around the pipes or because of the distance separating the pipes and the containment wall.

8. Cyclic Loading

The various cycles loads were considered in the design. The following design conditions were considered in the fatigue analysis:

120 cycles start and shutdown

500 OBE cycles

100 SSE cycles

1 accident cycle (LOCA)

160 pressure test cycles (equipment hatch and personnel air locks)

b. Load Combinations

Various combinations of loads were used to determine the maximum strength required of the containment at various locations. These combinations were divided into service and factored load combinations.

1. Service Load Combinations

Service load combinations include conditions encountered during testing, normal operation, shutdown and severe environmental conditions; these are listed in Table 3.8-1.

Under these conditions, the structural components are designed to remain within elastic limits and satisfy the stress limitations specified in Subsection 3.8.1.5.

2. Factored Load Combinations

Factored load combinations include those conditions resulting from severe environmental, extreme environmental, abnormal, abnormal/severe environmental and abnormal/extreme environmental loads, as defined in Division 2 and listed in Table 3.8-1.

For each of these loading categories the structure is designed so that the allowable stresses comply with Article CC-3000 of Division 2, and the overall structural behavior is predicted to remain in the elastic range when thermal effects are not included. Design assumptions are presented in Subsection 3.8.1.4.

3. Additional Notes on Load Combinations

The load combinations, with their appropriate load factors, which require investigation to assure that the maximum effects of all load combinations are considered, are in accordance with the requirements of Division 2 and are given in Table 3.8-1. These combinations are used in the overall design of the concrete containment and also in the design of localized areas, such as penetrations and shell discontinuities.

For loads that vary, those values which produce the most critical combination of loading were considered. The live load (L) is considered to vary from zero to full value for all load combinations.

Long-term conditions, such as operating thermal loads, creep and shrinkage, which produce compression in the reinforcing steel, do not have a significant effect on the structural integrity of the containment structure, since the accident loads, which are the most significant loads, are generally resisted in tension by the reinforcing steel. In addition, the accident loads are short once-occurring loads which will have negligible creep effects.

For the design of the liner, the load combinations in Table 3.8-1 are applicable with the exception that coefficients for all load cases are taken equal to 1.0.

Steel components of the containment shell that are pressure-resisting but unbacked by concrete are designed in accordance with the ASME Code, Section III, Division 1, using loads described in Subsection 3.8.2.

Earthquake effects are not assumed to occur simultaneously with flooding effects since the maximum flood is not associated with an earthquake.

Effects of a thermal gradient through the concrete section are not considered where the effects of the gradient reduce the effects caused by an abnormal loading condition.

Maximum values of time-dependent loads such as accident pressure, temperature and pipe break loads are considered.

The load combinations in Table 3.8-1 are applicable to the computations of factors of safety against overturning, sliding and flotation, with the exception that the coefficient for live load is zero. Buoyant forces are conservatively considered to decrease the dead loads for determination of overturning and sliding.

3.8.1.4 Design and Analysis Procedures

The containment structure is designed as a reinforced concrete thin shell structure in accordance with the requirements of Article CC-3000 of Division 2, as described in this Subsection and in other subsections of Subsection 3.8.1, and in accordance with the other applicable codes, standards and specifications defined in Subsection 3.8.1.2. The containment structure is designed to safely withstand the load combinations as defined in Subsection 3.8.1.3, and to provide biological shielding for normal and accident conditions. The critical areas for analysis were the base mat, the intersection between cylinder wall and base mat, the liner plate system, and the penetration openings.

The walls, dome and base mat of the containment are a reinforced concrete system with a leaktight steel liner attached to the inside surface, and were designed to behave as a single integrated system under the applied load combinations.

The containment structure as a whole behaves as a membrane structure, except in areas of discontinuity where there are local shear forces and moments. When subjected to internal pressure, the dome and walls displace outward with slight discontinuity effects at the dome and wall intersection. Discontinuity effects also exist near openings and at the intersection of the wall and base mat.

An iterative approach to design was taken, in which a proposed structural system was analyzed for stresses, strains and displacements using the finite element method, and then checked against design acceptance criteria, as defined in Subsection 3.8.1.5. The iteration process was repeated until an acceptable design was achieved.

The objective of the analysis of the reinforced concrete portion of the containment was the determination of maximum stresses in concrete and reinforcement over the range of boundary conditions, cracking assumptions and load combinations. To this end, the reinforced concrete was analyzed as a layered system, which allowed the investigation of each discrete layer of meridional, hoop and seismic reinforcing.

a. Design

The containment was designed to retain its functional capability during normal operation and emergency conditions. To meet this criterion, the leaktight integrity of the liner is maintained, and the structure was designed to respond elastically under all mechanical loading, except as noted in Subsection 3.8.1.5. The design criteria are based on the applicable codes discussed in Subsection 3.8.1.2.

Typical reinforcing details are shown on Figure 3.8-1, Figure 3.8-2, Figure 3.8-3 and Figure 3.8-4, Figure 3.8-6 and Figure 3.8-7.

b. Analytical Techniques

The analysis of the containment was performed by computer programs which utilize the finite element method. The finite element method was implemented by idealizing the containment with a system (model) of appropriately shaped elements which are interconnected at node points. These node points are located at the intersections of the lines which define the boundaries of each element. Generally, two types of analyses, axisymmetric and three-dimensional, were performed.

A finite element axisymmetric analysis, using solid and shell ring elements with orthotropic and elastic material properties, was used in the determination of displacements and forces in the containment structure with temperature, pressure and dead loads. The concrete, steel reinforcement, and the steel liner were represented by a system of ring elements which are triangular, linear, or quadrilateral in shape, as shown in Figure 3.8-12. The hoop, meridional and seismic reinforcement was modeled with shell elements located near the inside and outside surfaces of the wall and dome, as shown in Figure 3.8-13. Each layer of reinforcement was idealized as an independent uncoupled orthotropic steel plate.

Three-dimensional finite element analyses using elements of appropriately shaped sections were applied in the determination of displacements, stresses and strains for nonaxisymmetric loadings and local regions of the containment. Various types of elements were used for this analysis, such as shells and solids with isotropic and orthotropic material properties. The concrete, reinforcing steel and steel liner were idealized using layered elements. The three-dimensional finite element method was applied to the analysis of localized areas of the containment. A detailed discussion of the finite element programs used in the above mentioned analyses is given in Subsection 3.8.1.4g, and Appendix 3F.

c. Assumptions on Boundary Conditions

1. Containment Shell Analyses

The base mat was analytically treated as an integral part of the containment to realistically represent its effect on the containment shell behavior. The bottom of the base mat was assumed fixed in the vertical direction but free in the radial direction in all axisymmetric analyses for pressure, temperature and dead loads. For the seismic analyses described in Subsection 3.7.2, full fixity was assumed at the base of the containment wall.

Table 3.8-4 summarizes the load case and boundary condition assumptions for the containment shell analyses.

The containment mat with the reactor cavity pit and a portion of the shell was modeled in separate analyses. A discussion of boundary conditions is included in Subsection 3.8.1.41.

2. Large Opening Analyses

A quarter sector of the containment was modeled. This model extended from the base to the apex of the dome in the meridional directions and from a plane which passes through the opening to a plane at 90° to the centerline of the opening, as shown in Figure 3.8-14. The base was assumed fully fixed; the boundary conditions on the two sides of the model are defined as shown in Figure 3.8-14 and Table 3.8-5. These displacement boundary conditions correspond to the displacement degrees-of-freedom of the thick shell element discussed in Subsection 3.8.1.4j.

3. Aircraft Impact Analysis

The aircraft impact analysis described in Appendix 2P included two elastic dynamic analyses of the containment: an axisymmetric analysis (impact at apex of dome) and an asymmetric analysis (impact at springline). A shell model of the containment was used and full fixity was assumed at the base.

d. Axisymmetric and Nonaxisymmetric Loads

1. The primary behavior of the containment under axisymmetric loads is predominantly thin shell, membrane behavior except in regions near the base, springline and large openings where discontinuity moments and shears are significant. These latter effects are of localized nature and are discussed further in Subsections 3.8.1.4e and 3.8.1.4j. Thermal, pressure and dead loads were applied as axisymmetric loads to the axisymmetric containment model. These are the first four load cases listed in Table 3.8-4. The cracking patterns tabulated in the second column of the table were found by an iterative method. This is discussed further in Subsection 3.8.1.4f. The analyses were performed using the finite element computer code, Wilson I, "A Finite Element Analysis of Axisymmetric Solids Subjected to Symmetric Loads," by E. L. Wilson. This code is described in Subsection 3.8.1.4g and in Appendix 3F.

Figure 3.8-13 illustrates the reinforced concrete idealization used in the model, and Figure 3.8-15 shows the shell-type force and moment resultants which are calculated for sections along the shell meridian, and subsequently used in the design of the reinforcement. Figure 3.8-16 illustrates the axisymmetric containment model with integral mat, wall and dome used in the first four analyses of Table 3.8-4. Representative node points and section numbers are shown where the latter range from 1 at the apex of the dome to 98 at the base of the wall. At any section, the model is composed of shell and solid elements representing the steel liner and reinforcement and concrete, respectively. Material properties are assigned to each layer according to the orientation of reinforcement and cracked or uncracked state of concrete. Force and moment resultants were computed for each load case by integrating stresses through the cross section. These cross-sectional forces and moments are combined for all applicable load combinations of Subsection 3.8.1.3.

2. Nonaxisymmetric Loads

Nonaxisymmetric loads applied to the containment structure, other than localized loads, such as internal missiles, are the seismic and aircraft impact loads. The seismic input and seismic analysis are discussed in Subsections 3.7.1 and 3.7.2, respectively. The aircraft impact load and analysis are described in Appendix 2P. Wind and tornado loads are not considered because of the presence of the Enclosure Building. An elastic analysis of the aircraft impact loading was made using the Wilson II finite element code, "Dynamic Stress Analysis of Axisymmetric Structures Under Arbitrary Loading," by Ghosh and Wilson, as revised in September 1975. This code is further described in Subsection 3.8.1.4g and Appendix 3F. The loading was represented by Fourier series functions applied to the axisymmetric model of the containment.

The shear forces and overturning moments due to OBE and SSE seismic loads were obtained from the seismic analysis described in Subsection 3.7.2. These shear forces and moments were treated as effective static forces acting over the cross-section of the containment shell. Modeling assumptions for these two load cases are shown in Table 3.8-4.

3. Design of Reinforcement

The force and moment resultants for the individual load cases were combined for the load combinations of Subsection 3.8.1.3 and Article CC-3000 of the ASME Code. The provisions of Subarticle CC-3432 of Division 2 were used in the sizing of reinforcement steel.

e. Transient and Localized Loads

A "hot liner" transient load was analyzed and accounted for in the design of the reinforced concrete wall. A temperature spike was placed on the liner for two cases: (a) "hot liner" with no temperature gradient on concrete wall, and (b) "hot liner" with a temperature gradient through the wall (see Table 3.8-4). When combined with mechanical loads, the effect of thermal loadings is a redistribution of stresses and strains on the cross section. The stresses and strains in the liner are discussed in Subsection 3.8.1.4k, "Steel Liner Plate and Anchors."

The aircraft impact analysis is described in detail in Appendix 2P. Three analyses were performed: two of the overall structural behavior and one localized elasto-plastic dynamic analysis. The analyses of overall behavior considered the conditions of impact on the dome (axisymmetric structure with axisymmetric loading) and impact on the springline (axisymmetric structure with unsymmetric loading). Both analyses assumed linear behavior and used the Wilson II finite element code. The asymmetric loading of the second analysis was represented by a Fourier series. Both analyses showed that yielding would occur local to the point of impact. Accordingly, a localized, nonlinear analysis was used to determine the extent of damage to the containment shell. The details of this analysis are also found in Appendix 2P. In brief, effective mass and stiffness properties were determined for an assumed mode of collapse consisting of a circular fan yield-line configuration.

An equivalent single-degree-of-freedom nonlinear model was then subjected to an idealized force-time loading function and the maximum deformations determined. It was shown that the "as designed" containment structure with Enclosure Building can withstand impact of an FB-111 aircraft at 200 -mph impact speed without collapse or impairment of the leak-tight integrity of the liner.

f. Creep, Shrinkage and Cracking of Concrete

Due to low sustained concrete stress associated with conventionally reinforced concrete structures, the effects of concrete creep are negligible. Since the load combinations which control the rebar design involve accident pressure, which effectively cracks the concrete and places the reinforcement into tension, creep and shrinkage-induced stresses are not limiting factors in design. In addition, the structural integrity test cracks the concrete thereby relieving creep and shrinkage stresses that occur subsequent to construction. Therefore, the effects of creep and shrinkage were not considered in the analysis and design.

Since it was assumed that concrete has no tensile capacity, a cracked section, iterative approach was used for the analysis under the thermal, pressure and dead loads. The procedure for determining the crack pattern involved a check of the stresses in each concrete layer of the cross section. Where concrete stresses were in tension, the elastic properties in that direction (modulus of elasticity and associated Poisson's ratio) were set equal to zero. The procedure was repeated until there were no significant changes in moments and forces on the containment sections in two successive iterations. Since the finite element representation of the cross-section modeled both concrete and rebar using a layered model, as shown in Figure 3.8-13, an accurate portrayal of cross section stiffnesses under various configurations of cracking was possible. Table 3.8-4 defines the cracking pattern used for each load case. The analysis of the containment subjected to the nonaxisymmetric seismic loads considered the concrete to be uncracked. For this load case, the magnitudes of flexural moments are small compared to the flexural moments resulting from the axisymmetric loads. Hence, the redistribution of moments due to cracking under seismic loading does not control the design.

g. Description of the Computer Programs Utilized in the Design and Analyses

The computer programs used in the containment design and analysis are briefly described in this subsection. A summary of the comparisons of results used to validate them is given in Appendix 3F. The programs are either of two types: a recognized program in the public domain with sufficient history of use and documentation to justify its applicability and validity without further demonstration, or a program which gives solutions to a series of test problems that have been demonstrated to be substantially identical to those obtained from classical solutions and/or analytical results published in technical literature. Utility programs used to replace hand calculations are not discussed. These programs were validated by comparison to sample hand calculations whenever used in the analysis.

1. STARDYNE, Static and Dynamic Structural Analysis System, by Mechanics Research, Inc., 9841 Airport Boulevard, Los Angeles, CA., 90045. Documentation is available from Control Data Corporation (Publication No. 76079900). The STARDYNE system is designed to analyze linear elastic structural models for a wide range of static and dynamic problems.

2. MARC-CDC, Nonlinear Finite Element Analysis Program, by Dr. Pedro Marcel and Associates of the MARC Analysis Corporation, 260 Sheridan Ave., Palo Alto, CA., 94306. Documentation is available from Control Data Corporation. MARC-CDC provides elastic, elastic-plastic, creep, large displacement, buckling and heat transfer analysis capabilities. It also performs dynamic analysis by the modal or direct integration procedures.

3. WILSON 1, (SAG 001) Finite Element Analysis of Axisymmetric Solids Subjected to Axisymmetric Loads, by E. L. Wilson of the University of California, Berkeley, July 1967-Revised, November 1969. Documentation is available from the Earthquake Engineering Research Center of the University of California, Berkeley. The Wilson 1 computer program is based on the finite element direct stiffness method, and is applied to the determination of stresses and displacements in axisymmetric structures (solids and/or shells of revolution) subjected to axisymmetric mechanical loads or temperature gradients. The theoretical basis of the program is the work of E. L. Wilson, References 1 and 2.

4. WILSON 2, (SAG 010), Dynamic Stress Analysis of Axisymmetric Structures under Arbitrary Loading, by S. Ghosh and E. L. Wilson of the University of California, Berkeley, September, 1969-Revised September 1975. Documentation is available from the Earthquake Engineering Research Center of the University of California, Berkeley, Report, No. EERC 69-10. The Wilson 2 program is based on the finite element stiffness method and is applied to complex axisymmetric structures subjected to any arbitrary static or dynamic loading or base excitation. The three-dimensional axisymmetric continuum is represented as an axisymmetric thin shell, a solid of revolution, or as a combination of both.

5. LESCAL calculates the stresses and strains in rebars and/or concrete in accordance with the criteria set forth in Subarticle CC-3511 of Division 2. The section may be reinforced with horizontal, vertical and diagonal rebars. The applied loads are axial forces and moments in the vertical and horizontal faces and in-plane shear. When in-plane shear forces are included, a solution is obtained by solving Duchon's equations, Reference 3.

6. SAG 054, Amplified Floor Response Envelope. This program generates an envelope for amplified response spectra, spreading the peaks by a user-specified amount.

7. SAG 058, Response Spectra. This program calculates the response spectra of a single degree-of-freedom damped oscillator due to a transient base motion. The input base motion may be an arbitrary forcing function. The output consists of the maximum relative displacement, the maximum relative velocity and the maximum absolute acceleration for the various selected frequencies and the times when these values occur.

8. TAPAS, (SAG 008), "Transient Temperature Analysis of Plane and Axisymmetric Solids," Reference 4, was developed to determine the temperature distribution through a solid body as a function of time when subjected to temperature variation or heat flux inputs. A finite element technique coupled with a step-by-step time integration procedure is used. Both steady-state and transient heat flow can be treated.

9. SAG 017, Fourier Coefficient Expansion Program, was developed to be used in conjunction with the Wilson 2 program to compute Fourier series representation of general nonaxisymmetric load functions.

10. SAG 024, MMIC, calculates weight, weight moments of inertia and plan location of the center of weight of a segment of a structure given the dimensions, density and location of each structural component and the magnitude and location of all concentrated loads.

11. SAG 025, SECTION, calculates beam section properties of structures for use in lumped mass stick models for dynamic analysis.

h. Tangential Shear

The design of the containment for seismic loads incorporates diagonal shear reinforcing to resist tangential shear. The design for tangential shear is in accordance with the provisions of Article CC-3000 of the ASME Code, with the exception of the limitation on allowable concrete shear stress shown as Note 1 of Table 3.8-2.

The forces and moments due to seismic loading are discussed in Subsection 3.7.2. Three components of the seismic motion were assumed to exist simultaneously, and the resulting component forces and moments were combined by the square-root-of-the-sum-of-the squares (SRSS) method. The maximum tangential shear from the SRSS combination is assumed to act simultaneously at all points on the containment circumference at the given elevation. All forces and moments were combined per the specified load combinations. The LESCAL program, described in Subsection 3.8.1.4g, was used to calculate rebar stresses for all sections and elevations combining the stresses due to in-plane forces and moments. Duchon's equations (Reference 3) are incorporated into LESCAL for calculating rebar stresses (including inclined rebar) for the combined membrane forces.

i. Variation in Physical Material Properties

The effects of variations in material properties were considered in the design and analysis. Material properties which can strongly influence both analysis and design due to variability or uncertainty include: (1) dynamic modulus of soils, (2) the modulus of elasticity of concrete and, (3) material strengths.

As this containment is founded on rock, the first of these sources of variability is removed from consideration. The modulus of elasticity of concrete is a function of concrete compressive strength which in turn is typically substantially higher in the "as-built" structure than assumed for analysis and design. While variability in concrete modulus has no significant effect on structural design, it influences structural stiffness and natural frequency, and, subsequently, the amplified response spectra of the seismic analysis. This impacts equipment design as discussed in Subsection 3.7.3. The variability was accounted for by peak spreading when generating envelopes of the response spectra. Variability in material strength is taken into account in Division 2, Subarticle CC-3400, design allowables.

j. Thickened Penetrations

The equipment hatch and personnel air lock (27'-5" and 7'-0" diameters, respectively) are large openings with thickened bosses. Figure 3.8-3 and Figure 3.8-4 are drawings of typical wall sections with their reinforcement configurations. The wall thicknesses and reinforcement were designed for the stress concentration effects induced by the presence of the openings in the cylinder. The main hoop and meridional reinforcement is bent around the opening to provide continuity. Additional local reinforcement is provided, including stirrups and tiebacks for bent bars.

The analyses of the large openings were by the MARC finite element computer code using a three-dimensional model of a quadrant of the containment cylinder and dome, Figure 3.8-17. The two openings are sufficiently separated to allow independent analyses. Therefore, two planes of symmetry were assumed and the single quadrant model resulted. Appropriate symmetry and antisymmetry in loads and boundary conditions were defined. Figure 3.8-14 and Table 3.8-5 show the boundary conditions for the various load cases. The model extends from the base to the apex of the dome using a thick-shell superparametric element (element 22 in the MARC program element library). This element incorporates eleven through-thickness layers to accurately represent the liner, reinforcement and concrete. Figure 3.8-18 illustrates the partitioning of a typical cross-section into the 11 layers. The tapered-thickness transition region between the normal 54-inch wall and the thick boss is modeled by stepping the element thicknesses through this region. The thick-shell element gives all stresses, except the through-thickness (normal to wall) direct stress, at each of the eleven layers. The conventional shell-type force and moment resultants, including transverse shears, are calculated from these stresses and were used as the basis for design of the reinforcement.

Design of the reinforcement was accomplished in conformance with Division 2 provisions for load combinations and material stress allowables, as described in Subsections 3.8.1.3 and 3.8.1.5, respectively.

k. Steel Liner Plate and Anchors

1. Design

The liner is anchored to the reinforced concrete with embedded stiffeners and anchors. Typical liner details are shown in Figure 3.8-5. To maintain the leak-tight integrity of the liner under service loads and factored loads, the liner was designed to follow the major strain pattern of the surrounding concrete. The liner plate was thickened in areas around penetrations as required. The stresses and strains in the liner, determined by the analysis, are within the design allowables given in Table CC-3720-1 of Division 2.

The anchorage system was designed so that it can accommodate the in-plane loads or deformations exerted by the liner plate and/or loads applied normal to the liner surface. The anchorage system was designed so that a progressive failure of the anchorage system is precluded in the event of a defective or missing anchor. The design of the anchors considered the effects as indicated in Subarticle CC-3810 of Division 2.

The displacements and forces in the anchorage system obtained by analysis are not allowed to exceed the allowables given in Table CC-3730-1 of Division 2.

Penetrations are provided with an anchorage system capable of transferring thermal loads, pressure loads and other mechanical loads, such as piping reactions, to the concrete containment. The penetrations, brackets and attachments are designed in accordance with the provisions of the ASME Code, Section III, Division 2. Designing those portions of the penetrations which fall within the jurisdiction of Division 1, that is, those portions not backed by concrete, is described in Subsection 3.8.2.

2. Analysis

The stresses and strains in the liner due to dead load, internal pressure, a "hot" liner and a temperature gradient were obtained from the containment analyses using an axisymmetric finite element model, as discussed in Subsection 3.8.1.4d. These analyses considered the liner to be integral with the concrete structure. The liner anchors were analyzed assuming the unbuckled liner remained elastic under all conditions. Construction and material imperfections described in Division 2, Subarticle CC-3810, were considered in the liner anchor analysis including the possibility of a buckled liner panel. Liner stresses were assumed for the combined load cases. These stresses produced liner/anchor displacements in the direction of the buckled panel. An analytical model was developed in which the buckled panel was assumed to be elastic-perfectly plastic and was represented by an equivalent non-linear spring whose stiffness was determined by a separate nonlinear analysis using the MARC-CDC finite element code (Subsection 3.8.1.4g). The anchors were also represented by equivalent nonlinear springs whose force-displacement properties were determined by testing. The analytical model yielded forces and displacements of all anchors.

l. Containment Mat

The containment mat, the reactor cavity pit and 55' of the cylindrical containment shell above the mat were modeled with Element 22 in the MARC-CDC finite element program. A portion of containment shell in addition to the reactor cavity pit was included in the model to represent the stiffening effect of the containment shell on the mat. The internal structure and the fill mat were not modeled, but the loads which they transmit to the structural mat were considered. The stiffening effect of the dome and the cylindrical portion of the shell above the top of the model was considered by providing additional constraint equations which enforce the plane section remaining plane assumption. In effect, these equations insure that the top of the model remains circular and lies in a plane after deformation.

The rock foundation supporting the containment structure was modeled using the continuous foundation option which is available in the MARC-CDC program. This option requires as input foundation properties in both tension and compression at 9 integration points on the surface of each foundation element. To account for lift off, a zero stiffness was specified in tension. A very large stiffness is given to represent the rigid behavior of the rock foundation in compression. There were no vertical boundary conditions prescribed at node points on the mat. The consistent formulation in the element automatically accounts for this vertical restraint in compression. The stiffness of the internals and fill mat was not considered since there is no structural connection between these structures and the structural mat. Because of symmetry, only one-half of the structure was modeled, (Figure 3.8-19).

The following boundary conditions were applied to the model:

1. All node points in the global 134 xz plane have symmetry boundary conditions as given below:

   $u_y = \theta_x = \theta_z = 0$

   where:

   u_y = translation in the global y-direction

   θ_x = rotation about the global x-axis

   θ_z = rotation about the global z-axis

   The rotational boundary conditions are input into the program in a local coordinate system.

2. All node points at the outside edge of the mat within + 50° from the direction of the global x-axes have been restrained in the global x-direction to account for the lateral restraint provided by the rock below grade. (The earthquake forces are assumed in global x-direction.) Thus, all horizontal loads are resisted in bearing through the reactor cavity pit and a circular arc at the elevation of the centerline of the mat. The boundary conditions used in the mat analysis differ from those used in the axisymmetric analyses since the mat is represented by a three-dimensional model rather than the two-dimensional axisymmetric model. This permitted a more accurate representation of the boundary effects. The mat was modeled in the axisymmetric analyses only to simulate the mat restraining effect on the containment shell.

   The mat was analyzed for dead, live, pressure and seismic loads. Seismic loads were applied as inertia forces which were calculated from absolute accelerations computed in the seismic analysis, Subsection 3.7.2. At the top of the model, external forces were applied which represent the internal forces at that elevation for each load combination. Dead load, pressure, and vertical seismic loads were assumed uniformly distributed, and the moments and shears from the horizontal earthquake were distributed as normal and shear forces which vary as cosine and sine functions, respectively.

   The nonlinear effects resulting from uplift required an iterative analysis. Five iterations were required to reach the equilibrium state.

   When combining seismic loads with other loads, due to the inherent nonlinearity associated with mat uplift, the response due to the vertical component of the earthquake has been added algebraically to that of one horizontal component. Since the final stress state in a nonlinear problem can only be determined when all loads are applied simultaneously, it is not possible to separate out the individual contributions to the total response of two horizontal components and a vertical component of an earthquake in order to perform the SRSS. Consequently, only one horizontal component and the vertical component were applied and no SRSS process was used. The horizontal component was assumed to act along the longer direction of the reactor cavity pit (x-direction).

m. An ultimate capacity analysis of the containment structure for internal pressure loads was performed. The pressure-retaining capacity of the overall containment structure and the localized areas was determined. The analysis was based on actual material properties.

n. A Design Report (9763.102-CDR-1) of the containment structure was prepared in accordance with the requirements of Division 2. This report contains sufficient information to substantiate that the design of the containment structure is in accordance with the requirements of the containment design specification and the ASME codes.

3.8.1.5 Structural Acceptance Criteria

a. General

The containment structure, including liner and penetrations, was designed to remain within elastic limits under service load conditions and under the mechanical loads of the factored load conditions. With thermal loads included, the reinforcing steel yielded in some regions but was within the allowable strain limit of 2 x $\in_y$. Gross deformations of the containment were also checked to assure that there is no interaction with other structures or components.

The design limits imposed on the various parameters that serve to quantify the structural behavior and provide a margin of safety are in compliance with Article CC-3000 of Division 2. The allowable limits on these parameters, for service and factored loads, are given in Table 3.8-2.

b. Concrete

The allowable compressive stresses, including membrane, membrane plus bending and localized stresses, and shear stresses under service loads and factored loads are as specified in Article CC-3400 of Division 2, with the following exceptions:

1. The allowable shear stress, v_c, to be resisted by the concrete will not exceed 40 psi and 60 psi for load combinations 7 and 8, respectively, in Table 3.8-1.

2. The allowable concrete stresses or radial shear are based on Code Case No. N-287.

3. The allowable concrete stresses for peripheral shear are based on Code Case No. N-219.

c. Reinforcing Steel

The stress and strain limitations for reinforcing steel, under service loads and factored loads, are as specified in Subarticles CC-3432 and CC-3422 of Division 2, respectively.

Stress concentrations normally occurring around openings and penetrations were controlled by the use of additional reinforcing steel, which resulted in stresses and strains within the above indicated limits.

There is, however, some yielding of the seismic (diagonal) reinforcing under the mechanical loads of the abnormal/severe environmental and abnormal/extreme environmental loading conditions at the edges of the transition regions below the equipment hatch and personnel air lock adjacent to what may be considered the membrane region. The net strain at these locations, with thermal load included, is less than $2\epsilon_y$ (.00414 in/in), which is the limit established in Subarticle CC-3422.1 (d) of Division 2 (1977 Winter Addendum). The structural integrity of the containment is not affected, however. The strain limit of $2\epsilon_y$ insures that the yielding under thermal load does not result in concrete cracking which would cause deterioration of the containment.

d. Liner Plate and Liner Anchorage System

Tensile and compressive stress/strain limits in the liner plate, including membrane and membrane plus bending conditions, are in accordance with Subarticle CC-3700 of Division 2. (For the liner plate, the 1976 Winter Addendum of Division 2 is used, and for anchorage system, the January 1, 1975 Edition of Division 2 is used. All supporting documentation for material procurement and fabrication activities that commenced prior to July 1, 1975, included Certificates of Conformance to the technical requirement of this edition.)

The allowable values for forces and displacements of liner anchors embedded in concrete are based on test data, and are in accordance with the limits of Table CC-3730-1 of Division 2. Tests were performed to obtain the shear load - displacement relationship of liner anchors. The details of liner anchor load test procedure and results are included in Appendix 3G.

The stresses and strains in penetration assemblies, brackets and attachments are also in accordance with allowables given in Subarticle CC-3700 of Division 2.

e. Stability

Acceptance criteria for stability against overturning, sliding and flotation are given in Subsection 3.8.5.5.

3.8.1.6 Materials, Quality Control and Special Construction Techniques

Materials used for the containment include concrete, reinforcing steel, liner plate steel and attachments thereto, and coatings, the requirements of which are in compliance with Article CC-2000 of Division 2 and the applicable NRC Regulatory Guides listed in Subsection 3.8.1.2.

Quality control procedures employed for the fabrication and construction of the containment are in compliance with Articles CC-4000 and CC-5000 of Division 2 and with the applicable NRC Regulatory Guides listed in Subsection 3.8.1.2. For all materials, suppliers were required to furnish Certified Materials Test Reports (CMTR), prepared in accordance with the requirements of Subarticle CC-2130 of Division 2. A CMTR includes results of all required chemical analysis, physical tests, mechanical tests, examinations including radiographic film, repairs and heat treatments performed on the material.

Materials used in construction, as well as their respective quality control procedures, are further described in the sections that follow. Engineering properties are given in Table 3.8-3. Construction tolerances are in accordance with the criteria of Division 2.

There are no special construction techniques.

a. Concrete

The ready-mixed concrete which was used is a dense durable mixture of sound coarse aggregate, fine aggregate, cement, water and admixtures, in accordance with the material, proportioning, mixing, transporting, placing and testing requirements of Division 2.

Testing which was performed on the concrete and on its components is described in UE&C Specification 9763.006-69-1, "Concrete Batch Plant." Details of the placing procedures, including hot and cold weather precautions, are described in UE&C Specification 9763.006-13-2, "Containment Concrete Work." In the paragraphs which follow, however, a brief description of the material is presented.

Except as noted below, the containment is constructed of concrete which has a standard compressive strength at 28 days of at least 3000 psi. The base mat, reactor pit, bottom 10 feet of the cylinder, and the regions in the cylinder near the equipment hatch and personnel air lock, in which the reinforcing anchor plates are located, are constructed of concrete which has a standard compressive strength at 28 days of at least 4000 psi. To insure that these strengths were attained, verification testing was performed in accordance with the requirements of Division 2.

The nominal density of the reinforced concrete was considered as 150 lbs/ft^3. The mix proportions were established on the basis of laboratory trial batches which were designed by the testing laboratory per ACI 211.1-74. The concrete was produced under controlled conditions by a fully automatic central batch plant located on the site to assure that the proper ingredients and proportions determined by the design mix were achieved.

The cement conforms to the requirements of ASTM C150, "Specification for Portland Cement," Type II, having a low alkali content and a moderate heat of hydration.

The coarse and fine aggregates conform to the requirements of ASTM C33, "Specification for Concrete Aggregates" and the additional requirements of Subarticles CC-2222 and CC-3421.5.1 of Division 2.

The maximum coarse aggregate size in the containment is 1½", except in congested areas where a 3/4" (size #7, as specified in ASTM C33) maximum aggregate size mix was used to accommodate proper placement of the concrete.

The mixing water conforms to the requirements of Subarticle CC-2223 of Division 2.

To assure a plastic and workable mix, increase durability, and increase ease of placement in congested areas, admixtures were used in the concrete mix design. These admixtures consisted of air entraining agents, water reducing agents, and retarding agents. Their effects on the strength of the mix were considered in the mix design, such that the properties described previously were the properties which were obtained after the inclusion of admixtures.

Admixtures containing chloride ions were not used in the concrete for the containment.

The maximum slump permitted in mass concrete for the containment was 3", except in congested areas where a 4" slump was allowed to accommodate proper placement, with slumps greater than 4" but not more than 6" (Special High Slump Concrete) used in highly congested areas. The maximum slump for concrete utilizing a superplasticizer (high range water reducer) was 8" (9" on a case-by-case basis). The maximum slump permitted for all other concrete was 4".

No aluminum materials were used in the mixing, handling, storing, transporting, or placing of concrete materials or mixes, nor were any aluminum embedments used.

The maximum concrete mix temperature during placement was 80°F.

All concrete operations during cold weather conditions followed the practice defined in ACI 301 and 306R-78 except that concrete as placed shall not be lower than 45°F. Concrete was maintained at 50°F.

During cold weather curing of the concrete, concrete surfaces whose temperatures are below 50°F by accident for short periods of time, but remain 40°F or above, have had the 7-day curing period extended by the amount of time the concrete was below 50°F (rounded out to the nearest whole day).

b. Reinforcing Steel

Reinforcing steel consists of high-strength deformed billet steel bars conforming to ASTM A615, Grade 60. This steel has a minimum yield strength of 60,000 psi, a minimum tensile strength of 90,000 psi and a minimum elongation of 7 percent in an 8" gage length.

In addition to the Certified Material Test Reports, user tests, as required by Division 2 and Regulatory Guide 1.15, were performed by the Material Manufacturer on full-size diameter test specimens to further verify the physical properties of the rebar.

Arc welding of rebar was not permitted.

All reinforcing bars were detailed by the Manufacturer in accordance with the requirements of the Design Drawings. Detail drawings were reviewed by the Designer.

All reinforcing bars were spliced in accordance with Division 2 and UE&C Specification 9763-WS-4C. No. 14 and 18 bars were joined by mechanical butt splices (Cadweld splices). These splices met the requirements of Subarticle CC-4333.4.4. of Division 2. The splice sleeve material conformed to ASTM A519 (85 ksi - Minimum Ultimate Tensile Strength). Cadweld anchor plates used as mechanical anchorage for the terminated rebars, predominately in the equipment hatch and personnel air lock areas, conformed to SA537, Class 1.

The splices were tested in accordance with the requirements of UE&C Specification 9763-WS-4C to assure that they develop the specified strength. This test program is in accordance with the requirements of Division 2 and NRC Regulatory Guide 1.15, as discussed in Section 1.8, Regulatory Guide 1.136.

c. Containment Liner

The steel liner plate is carbon steel conforming to ASME SA 516, Grade 60. This steel has a minimum yield strength of 32,000 psi and a minimum tensile strength of 60,000 psi, with an elongation of 21 percent in an 8" gauge length, when tested to failure.

The materials, standards and quality control procedures used for the liner and penetrations are in accordance with Division 2 and NRC Regulatory Guide 1.19. Inspections are made in accordance with Article CC-5000 of Division 2.

The following materials were used for the penetration sleeves, equipment hatch and personnel locks:

1. Equipment Hatch-ASME SA-516, Gr. 60 Normalized

2. Personnel Locks-ASME SA-516, Gr. 70 Normalized

3. Fuel Transfer Sleeve-ASME SA-240, Type 304 Stainless Steel

4. Seamless Penetration Sleeves-ASME SA-333, Gr. 1

5. Rolled Penetration Sleeves-ASME SA-516, Gr. 60 Normalized.

Permanent attachments to the liner were fabricated from SA-36 or SA-516 Grade 60 or SA-516 Grade 70 plate. The anchorage system, tees, brackets and attachments to the liner plate were ASME SA-36.

All backing strips were of the same material specification as the item being welded.

All welding materials conformed to the requirements of UE&C Speci fication No. 9763-WS-4A and ASME Section II, Material Specification, Part C - Welding Rods, Electrodes and Filler Metals.

Studs and stud welding materials were in accordance with the requirements of Subsection CC-2620 of Division 2.

All liner materials are normalized, and sections in excess of 5/8" thickness were impact tested to 15 ft-lbs. at 20°F in accordance with the requirements of Division 2. For penetrations, the equipment hatch and personnel air lock, the test temperature was -10°F in the heat affected zone and -25°F in the parent metal.

When required, welds were post-weld heat treated in accordance with the requirements of Division 2.

The portions of the equipment hatch and personnel air lock within the jurisdiction of Division 1 were designed and detailed by the Fabricator. These portions are described in Subsection 3.8.2.

d. Steel and Concrete Coating System

The coatings used on steel, other than stainless steel, and concrete surfaces inside Containment, that might be exposed to spray have been tested and accepted in accordance with the requirements of ANSI Standard N101.2, except in the acceptance criteria used, and were applied in accordance with the manufacturer's printed instructions over properly prepared surfaces. The acceptance criterion for power tool cleaning methods, intended for limited use in the containment, is adherence of the coating (no solid debris generated) rather than specific ANSI blister size and frequency. (See also Section 6.1 for further discussion on BOP and NSSS equipment and structures coating systems inside Containment.)

Ferrous surfaces to which coatings were applied were abrasive blast cleaned in accordance with the Steel Structure Painting Council Specifications SP6 and SP10, except for the limited use of qualified power tool cleaning methods where blast cleaning became impractical. All horizontal and vertical concrete surfaces which were coated were washed and neutralized just prior to the initial coating application in order to produce a clean contamination-free surface. All concrete patching work, such as pointing of form tie holes and removal of sharp edges and fins, was completed before the washing was begun.

e. Grout

Sand Cement Grout used in general repair and patch work was in conformance with the requirements of ASME Section III, Division 2 Code, 1975 edition including Winter 1975 Addenda Subarticle CC-2240.

Prepackaged cement grout used in general repair and patch work met the requirements of ASME Section III, Division 2 Code, 1975 edition including Winter 1975 Addenda Subarticle CC-2240 except as modified herein. Aggregates met the requirements of ASTM C33 except that the gradations were adjusted as required to meet the Material Manufacturer's requirements for the product's applications.

The ASTM C88 Standard Test method for soundness of aggregates was not required to be performed provided alternate testing was performed by the Material Manufacturer to assure that the grout provided adequate resistance to weathering action.

The Material Manufacturer provided a notarized Certificate of Conformance (C of C) and Certified Material Test Reports (CMTRs) which contain the results of tests required in Corps of Engineers' CRD-C-589-70 and CRD-C588-78A. Compressive strength of grout was tested in accordance with CRD-C-588-78A, Section 10.3 "Compressive Strength" except that nine (9) cubes were made from each test mixture, and cubes were tested for compressive strength at 3, 7, and 28 days. CMTRs and Master Builders 713 Grout did not include chemical analysis.

f. Epoxy Bonding Materials

Sikadur High-Modulus Epoxy Bonding Adhesive (Sikastix 370) was used in general concrete repair and patching work when directed by the Construction Manager. The bonding adhesive was stored, mixed and applied in accordance with the Material Manufacturer's instructions.

The Material Manufacturer provided a notarized Certificate of Conformance for each batch of material supplied.

3.8.1.7 Testing and In-Service Inspection Requirements

The structural testing and in-service inspection program consists of the integrated leak rate test, in-service leak rate testing, the preoperational structural integrity test, and general visual inspection of structurally critical areas.

The in-service leak rate testing of the containment and visual inspection requirements are discussed in Subsection 6.2.6. The preoperational structural integrity test and visual examination are described below.

Since no new or previously untried design approaches are used for the containment structure, there are no special testing or in-service surveillance requirements.

a. Structural Integrity Test

To demonstrate that the concrete containment structure will respond satisfactorily to the postulated internal pressure loads, a preoperational structural integrity test (SIT) will be performed at 1.15 times the containment design pressure of 52 psig and at an average differential temperature between the inside and outside of the containment (within the Enclosure Building) not to exceed 65°F.

The SIT will be conducted in accordance with the nonprototype requirements of Article CC-6000 of Division 2. The Consolidated Edison's Indian Point Unit No. 2 and Washington Public Power Supply System Unit No. 1 are the prototypes for this test. Structural acceptance is based on gross deformations (diameter change, vertical growth and radial growth at the equipment hatch), concrete crack widths, deflection recovery and post test visual examination of concrete and liner. Prior to the test, a table of predicted gross deformations, crack widths, etc., will be provided as a guide for verifying satisfactory structural response during the test. These acceptance criteria will be as follows:

1. No yielding of reinforcement as determined by analysis of crack width and deflection data.

2. No visible signs of permanent damage to the structure or liner.

3. The measured maximum deflections at points of maximum predicted deflection shall not exceed predicted values by more than 30 percent. This requirement will be waived if the 24-hour recovery is greater than 80 percent.

4. The deflection recovery 24 hours after complete depressurization shall be a minimum of 70 percent.

   The instrumentation required to obtain the data needed to verify the structural response will include methods to measure radius and diameter deformation, vertical deformation, deformation around the large equipment hatch opening and breech-type personnel air lock opening, and crack measurements as required by ASME Section III, Division 2. The data will be obtained without embedding any devices in the cylinder or dome. A test report will be provided which will compare test results with predicted and allowable limits and evaluate any deviations.

   Before and after the test, the containment will be visually examined to assure that no distress has developed on the concrete or liner.

3.8.2 Steel Containment

The following sections contain the physical descriptions, applicable codes, standards and specifications, loads and load combinations, design and analysis procedures, allowable stresses, quality control and testing requirements for the major steel penetrations of the concrete containment structure that are pressure-resisting but unbacked by concrete. The containment structure itself is constructed of reinforced concrete and, as such, is discussed in Subsection 3.8.1.

3.8.2.1 Description of the Containment Penetrations

The containment penetrations described in this Subsection are the personnel air lock and equipment hatch, the fuel transfer tube assembly, the piping, electrical and instrumentation penetrations, and the ventilation penetrations. These components penetrate the containment shell to provide access, anchor piping, or furnish some other operational requirement. They also maintain leak tightness for the containment shell, and meet the maximum allowable leakage rate, as described in Subsection 3.8.1. Other functional characteristics are described below in the descriptions of individual components.

All penetrations are anchored to sleeves (or to barrels) which are embedded in the concrete containment wall. This embedment is accomplished by means of an engineered anchorage system that is welded to the sleeve (or barrel) which is, in turn, welded to the locally thickened liner (see Subsection 3.8.1).

a. Personal Air Lock (Breech Type)

The personnel air lock (Figure 3.8-20) consists of the air lock doors and the lock barrel. Its centerline is located at elevation 29'-6" and an azimuth of 315°, as shown on Figure 1.2-4.

Significant dimensions are as follows:

Parameter	Dimension
Clear Opening	7'-0"
O.D. of Flange on Door	7'-9 1/8"
Barrel Thickness	5/8"
Cover Thickness (Spherical Dished Head)	5/8"

The air lock barrel has a door on each end, each of which is designed to withstand the design pressure from inside the containment. The doors are hinged and swing away from the air lock barrel. Each door is fitted with two seals that are located so that the area between seals can be pressurized and tested to 52.0 psig. A leak chase system is provided over the barrel-liner joint of the personnel air lock for leak testing.

The locking device for the doors is a rotating, third ring, breach-type mechanism. These doors are interlocked so that only one door can be opened at a time. The capability exists for bypassing this interlock and relieving the internal pressure by use of special tools. The doors may be operated mechanically.

A sight glass is provided at the exterior of each door to permit observation of the status of the opposite door.

Separate limit switches are provided to allow remote indication of door position, and a signal is furnished for use by the control room.

The barrel, which is also the sleeve for the personnel air lock, is embedded and anchored in the shell of the concrete containment.

b. Equipment Hatch

The equipment hatch (Figure 3.8-21) consists of the barrel, the spherical dished cover plate with flange, and the air lock mounting sleeve. The centerline of the hatch is located at elevation 37' – 0 1/2" and an azimuth of 150°, as shown on Figure 1.2-4. The hatch opening has an inside diameter of 27' - 5".

A sleeve for a personnel air lock, the inside diameter of which is 9'-10", is provided at centerline elevation 30'-6". Thicknesses of the primary components are as follows:

Component	Thickness
Barrel	3 1/2"
Spherical Dished Cover Plate	1 3/8""
Flange	5 3/8"
Air Lock Mounting Sleeve	1 1/2"

The equipment hatch cover is fitted with two O-ring seals that enclose a space which can be pressurized and tested to 52.0 psig. The flange of the cover plate is attached to the hatch barrel with 32 swing bolts, 1 3/8" in diameter. A leak chase system is provided over the barrel-liner joint of the equipment hatch for leak testing.

The barrel, which is also the sleeve for the equipment hatch, is embedded in the shell of the concrete containment.

Provision has been made to lift the equipment hatch cover to the side to clear the opening and to store the cover in the storage saddles designed for this purpose.

A platform to allow access to the swing bolts at the top of the hatch has been permanently installed on the equipment hatch. Extensions have been added to the lower lifting lugs. These extension legs make the lifting device attachment points at approximately the same elevation for all four lug locations. The extension legs and the upper lifting lugs are used to support the hatch access platform. This platform is provided with fall protection chain railings and safety harness tie-off points.

Inserted into the mounting sleeve through the equipment hatch cover is a personnel air lock consisting of two air lock doors, two air lock bulkheads, and the air lock barrel.

Significant dimensions of the air lock are as follows:

Parameter	Dimension
Inside Diameter of Barrel	9'-6"
Barrel Thickness	1/2"
Door Opening	6'-8"x3'-6"
Door Thickness	3/4"
Bulkhead Thickness	1 1/8"

Each door is locked by a set of six latch pin assemblies, and is designed to withstand the design pressure from inside the containment. To resist the test pressure, each door is fitted with a set of test clamps. The doors are hinged and both swing into the containment. Each door is fitted with two seals that are located such that the area between seals can be pressurized and tested to 52.0 psig.

The doors are mechanically interlocked so that only one door can be opened at a time. The capability exists for bypassing this interlock and equalizing the pressure by use of special tools. The doors may be operated mechanically.

A sight glass is provided in each door to permit observation of the opposite door.

Separate limit switches are provided to allow remote indication of door position and to alarm in the control room.

c. Piping Penetrations

Piping penetrations are divided into two types, high energy and moderate energy. Moderate energy piping penetrations are used for process pipes in which both the pressure is less than or equal to 275 psi, and the temperature of the process fluid is less than or equal to 200°F. High energy piping penetrations are used for that piping in which the pressure or temperature exceeds these values.

High energy piping penetrations (Figure 3.8-22) consist of a section of process pipe with an integrally-forged flued head, a containment penetration sleeve and, where a pipe whip restraint is not provided, a penetration sliding support inside the containment. The sliding support provides shear restraint while permitting relative motion between the pipe and the support. The annular space between the process pipe and the sleeve is completely filled with fiberglass thermal insulation. The pipe and the flued head, are classified as ASME III, Safety Class 2 (NC), whereas the sleeve is classified as part of the concrete containment, ASME III (CC). The sliding support inside the containment is classified as an ASME Safety Class 2 component support (NF).

Moderate energy piping penetrations (Figure 3.8-23) consist of one or more process pipes, the containment penetration sleeve, and a flat circular end-plate. The pipe is classified as ASME III Safety Class 2 (NC). The sleeve is classified as ASME III Div. 2 (CC). The sleeve backed by concrete is ASME III (CC); the sleeve not backed by concrete is ASME III (Class 2). The end-plate material is classified as ASME Class 2.

Both kinds of assemblies, however, were analyzed to meet the intent of Class MC criteria (NE) through the use of a simplified, but conservative, approach described in Subsection 3.8.2.4.

d. Electrical Penetrations

Electrical penetrations (Figure 3.8-24) consist of a stainless steel header plate (bulkhead) with an attached terminal box, electrical modules which are clamped to the header plate, and a carbon steel weld ring which is welded to the header plate and to the sleeve. The metallic pressure resisting parts, the sleeve, stainless steel header plate and carbon steel weld ring were designed as ASME III Safety Class MC components (NE); that portion of the sleeve which is backed by concrete was designed as part of the concrete containment, ASME III (CC).

Double silicone and Hypalon O-rings provide a seal with a cavity for leakage monitoring between the header plate and the modules. The header plate is provided with a hole on the outside of the containment to allow for pressurization of the penetration assembly for leakage monitoring.

e. Instrumentation Penetrations

Instrumentation penetrations are of two types, electrical and fluid. The electrical type is similar in construction to the other electrical penetrations, and the discussion in Subsection 3.8.2.1d, as well as material appearing elsewhere in Subsection 3.8.2 which pertains to electrical penetrations, is applicable to these penetrations.

The second type of instrumentation penetrations, the fluid type, is similar in construction to the moderate energy piping penetrations. Consequently, the discussion in Subsection 3.8.2.1c on moderate energy piping penetration and other material in Subsection 3.8.2 relating to these penetrations are also applicable to this type of instrumentation penetration.

f. Fuel Transfer Tube Assembly

The fuel transfer tube assembly (Figure 3.8-25) consists of the fuel transfer tube, the penetration sleeve, the fixed saddle on the reactor side, and the sliding saddle in the Fuel Storage Building. The fuel transfer tube and its flanges were designed as ASME III Class 2 components (NC). The quick closure hatch on the refueling canal side of the tube was designed and fabricated to ASME Section III Class MC component (NE) requirements. That portion of the sleeve which is backed by concrete was designed as part of the concrete containment, ASME III (CC). The remaining pieces of the assembly were designed as ASME III Component Supports (NF).

g. Ventilation Penetration Assemblies

There are two types of ventilation penetrations, the containment air purge penetrations and the containment online purge penetrations. The containment air purge penetrations (Figure 3.8-26) each consist of a pipe sleeve (a rolled and welded pipe section, 36" O.D. by 1/2" wall thickness) which is flanged at each end with 36" weld neck flanges and, attached to these flanges, is a butterfly valve inside containment and a testable blind flange outside containment. Together with the pipe, the blind flanges form a part of the containment pressure boundary during plant Modes 1, 2, 3, and 4. During Modes 5 and 6, the blind flanges are replaced by spool pieces to configure the CAP system to perform its heating and ventilation functions. The valves are 36" diameter butterfly valves with fail-safe pneumatic operators. The weld between the pipe and the containment liner is equipped with a leak chase for pressure testing.

The containment online purge penetrations each consist of a pipe sleeve (a rolled and welded pipe section, 8" O.D. by 1/2" wall thickness). A short section of pipe with a nipple is welded to the sleeve on the outside of the containment, and a 3/4" valve and test connection is attached to it. The ends of this resulting assembly are welded to 8" weld neck flanges which are through-bolted to the inner and outer isolation valves. These valves are 8" diameter butterfly valves having fail-safe pneumatic operators. The weld between the pipe sleeve and the containment liner is equipped with a leak chase for pressure testing. Since the details, other than size, of these penetrations are essentially the same as those of the containment air purge penetrations, an additional figure is not included.

The portion of the penetration sleeve which is backed by concrete was designed as part of the concrete containment, ASME III (CC). That portion of the sleeve extending beyond the containment was designed to meet the requirements of ASME Section III, Division 1, Subsection NE, in addition to the fabrication and nondestructive examination requirements of Division 2. The valves and flanges were designed as ASME III Code Class 2 components (NC) and are further described in Subsection 3.9.3.

3.8.2.2 Applicable Codes, Standards, and Specifications

The design, materials, fabrication and inspection requirements for the major containment penetrations conform to, but are not necessarily limited to, the applicable sections of the following codes and specifications which are used to establish or implement design bases and methods, analytical techniques, material properties and quality control provisions.

Dates and revisions given for the listed codes are the earliest version that was used. Subsequent issues were incorporated into the design where practicable, or where the new issue directly affected the safety of the structure.

Code or Specification	Title
ASME Boiler & Pressure Vessel Code	Section II - Material Specification
	Section III, Division 1, Subsection NA, General Requirements
	Section III, Division 1, Subsection NB, Class 1 Components

Code or Specification	Title
	Section III, Division 1, Subsection NC, Class 2 Components
	Section III, Division 1, Subsection NE, Class MC Components
	Section III, Division 1, Subsection NF, Component Supports
	Section III, Division 2 Code for Concrete Reactor Vessels and Containments
	1975 Edition including Winter 1976 Addendum for Containment Liner
	Section IX - Welding Qualifications

Applicable Codes Dates for ASME B&PV Code, except Section III, Division 2, Components:

1971 Edition, through Summer 1973 Addenda (Equipment Hatch)

1974 Edition, through Winter 1975 Addenda (Personnel Air Lock in Equipment Hatch)

1974 Edition, no Addenda (Breech Type Personnel Air Lock)

1974 Edition, including Summer 1975 Addendum (High Energy Piping Penetrations)

1974 Edition, no Addenda (Moderate Energy Piping Penetrations)

1974 Edition, including Winter 1976 Addendum (Electrical Penetrations)

1971 Edition through Summer 1973 Addenda (Fuel Transfer Tube Assembly)

1971 Edition, including Winter 1973 Addendum (Ventilation Penetration Assemblies)

Code or Specification	Title
U.S. Department of Labor	Occupational Safety and Health Administration Standards, October 1975 Edition
UBC	International Conference of Building Officials, Uniform Building Code, 1973 Edition
AISC	Specification for the Design Fabrication and Erection of Structural Steel for Buildings, 1969 Edition, (including Supplements 1, 2 and 3)
ANSI B16.5-1968	Steel Pipe Flanges, Flanged Valves, and Fittings
ANSI N45.2-1974	Quality Assurance Program Requirements for Nuclear Power Plants
ANSI N45.4-1972	Leakage Rate Testing of Containment Structures for Nuclear Reactors
IEEE STD-317-1972	Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations
IEEE STD-323-1974	Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations
IEEE STD-344-1971	Guide for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
SP-44	Manufacturer's Standardization Society Steel Pipe Line Flanges, 1975 Edition
10 CFR 50, App. J	Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors

NRC Regulatory Guides	Title

Code or Specification	Title
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components (Rev. 0, 6/73)
1.63	Electrical Penetration Assemblies in Containment Structures for Water Cooled Nuclear Power Plants (Rev. 2, 7/78)
1.84	Code Case Acceptability, ASME Section III Design and Fabrication (Rev. 15, 5/79)
1.85	Code Case Acceptability, ASME Section III Materials (Rev. 15, 5/79)
1.163	Performance-Based Containment Leak-Test Program (Rev. 0, 9/95)

The below-listed UE&C design and construction specifications applicable to the containment penetrations were prepared in accordance with applicable codes, quality control requirements and NRC Regulatory Guides.

UE&C Specifications	Title
9763.006-QAS-1	Quality Assurance Administrative and System Requirements
9763.006-QAS-3	Quality Assurance Administrative and System Requirements For Safety-Related Electrical Equipment
9763.006-MPS-1	Material and Processing Requirements for Nuclear Power Plant Components
9763.006-NSS-0185	Fuel Transfer Tube
9763.006-WS-1-NE	Requirements for Welding and Non-Destructive Examination for Nuclear Pressure Class MC Components

UE&C Specifications	Title
9763.006-SD-15-2	Seismic Requirements for Equipment Hatch and Personnel Air Lock
9763.006-SD-118-1	Seismic Requirements for Electrical Penetrations
9763.006-15-1	Containment Liner
9763.006-15-2	Containment Equipment Hatch and Personnel Locks

3.8.2.3 Loads and Load Combinations

The containment penetrations were designed to withstand all credible conditions of loading, including preoperational test loads, normal startup, operational and shutdown loads, severe environmental loads, extreme environmental loads, and abnormal loads. Design limits and load combinations are in accordance with Article NE-3000 of Division I and Regulatory Guide 1.57. The penetrations were evaluated for several combinations of loads to assure that the response of the components would remain within the limits prescribed in Subsection 3.8.2.5.

In the paragraphs that follow, these loads and load combinations are grouped according to the type of penetration.

a. Equipment Hatch and Personnel Air Lock

1. Design Loads

(a) Preoperational Test Loads

These are loads which are applied during the initial and any subsequent structural integrity or leak rate testing of the containment.

(1) Test Pressure (P_t)

The containment and components listed in Subsection 3.8.2.1 are pressurized to 115 percent of the design pressure to test their structural integrity, i.e., the test pressure is 60 psig with design pressure being 52 psig.

(2) Test Temperature (T_t)

The maximum and minimum temperatures inside the containment during the test are 100°F and 50°F, respectively. The temperature on the outside of the containment, considered to be the ambient temperature, is 0°F, the minimum temperature at the site. Any thermal loads occurring during test conditions are induced by the gradient between the containment and ambient temperatures.

(b) Normal Startup, Operational, and Shutdown Loads

Normal loads encountered during normal plant startup, operation and shutdown include the following:

(1) Dead Load (D)

This includes the weight of the component and appurtenances.

(2) Live Loads (L)

The live load on the raised floors of the personnel locks, representing personnel and equipment that may be moved through there, is taken as 100 pounds per square foot. There are no other live loads.

(3) Operational Thermal Loads (T_o)

The normal containment operating temperatures range from 50° to 120°F. Outside temperatures vary between 0°F and 90°F.

(4) Pressure Variation (P_v)

Differential pressure loads result from pressure variation either inside the containment or in the Containment Enclosure Building. This pressure variation is produced either by atmospheric fluctuations or by HVAC equipment. The internal pressure varies between -3.5 and +1.5 psig.

(c) Severe Environmental Loads

These are loads that would result from external conditions which could infrequently be encountered during the plant life. The following loads are considered in this category:

(1) Wind Load (W)

There is no wind load considered due to the presence of the containment enclosure.

(2) Operating Basis Earthquake (E_o)

These are the loads generated by the Operating Basis Earthquake, which is the earthquake that could reasonably be expected to occur at the plant site during the operating life of the plant. The applied accelerations, for static analysis, which are valid for rigid components ($f > 33$ Hz) are 0.580g for horizontal motion and 0.528g for vertical motion; damping is 2 percent for both accelerations. These accelerations represent the peak response of the containment cylinder at the elevation of the equipment hatch, the highest penetration in the containment. Only the actual dead load is considered in evaluating the seismic response forces.

(d) Extreme Environmental Loads

Extreme environmental loads are those loads which result from postulated events which are credible but highly improbable. The following loads are included in this category:

(1) Safe Shutdown Earthquake (E_s)

These are the loads generated by the Safe Shutdown Earthquake, which is the maximum potential earthquake that could occur in the vicinity of the site, based on geological and historical investigations. The applied accelerations, for static analysis, which are valid for rigid components ($f > 33$ Hz) are 0.861g for horizontal motion and 0.748g for vertical motion; damping is 3 percent for both accelerations. As with the operating basis earthquake, these accelerations also represent the peak response of the containment cylinder at the elevation of the equipment hatch.

Only the actual dead load is considered in evaluating the seismic response forces.

(2) Tornado Loads (W_t)

Due to the presence of the Containment Enclosure Building and other appurtenant structures, such as the pipe chases and the Fuel Storage Building, wind pressure and pressure variation are not considered in the design of containment penetrations.

Furthermore, due to various shield walls the equipment hatch and personnel air lock are protected from impact by tornado-generated missiles. All of the other penetrations (piping, ventilation, etc.), emerge from the containment inside the various appurtenant structures and, consequently, are also protected from impact by tornado-generated missiles.

(e) Abnormal Loads

Abnormal loads are those loads generated by postulated high energy pipe ruptures, particularly a rupture in the Reactor Cooling System resulting in a loss-of-coolant accident (LOCA).

(1) Accident Pressure (P_a)

The components were designed to withstand an internal containment accident pressure of 52.0 psig. See Subsection 6.2.1 for the development of this pressure.

(2) Accident Temperature (T_a)

The peak accident temperature of the liner plate is 268°F. However, a maximum temperature of 271°F was considered for the design of the components. See Figure 3.8-9, Figure 3.8-10 and Figure 3.8-11 for plots of the transient containment liner temperature response, as well as the transient containment pressure response.

(3) Internal Missile Loads

Internal missile loads, as described in Section 3.5, are prevented by shields, as required, which confine the missiles.

(f) Environmental Loads

These are environmental conditions which must be considered in regard to their effect on the durability of materials.

(1) Normal Relative Humidity

Under normal operating conditions, the relative humidity, both inside and outside the containment, is expected to vary between 5 percent and 100 percent.

(2) Accident Environmental Conditions

Following a loss-of-coolant accident, the following conditions will exist inside the containment.

a Relative humidity of 100 percent

b Spray solution of the following:

- Boron (calculated as boric acid), (minimum/maximum), 0.23/0.25 percent by weight
- pH (minimum/maximum), 9.0/9.6
- Sodium Hydroxide (minimum/maximum), 0.45/0.54 percent by weight

(3) Radiation

Time-integrated doses for personnel air locks and equipment hatch, are as follows:

Personnel Air Locks andEquipment Hatch

Accident Doses	Inside Containment	Outside Containment
(1 year post accident)		
Gamma TID	1.4×10^7 rads	6.5×10^4 rads
Beta TID	4.95×10^7 rads	----
Normal Doses		
(40 years period)		
Gamma TID	2×10^7 rads	5.3×10^3 rads
Beta TID	Negligible	Negligible

(g) Fatigue Requirements

The personnel air locks and equipment hatch were designed to withstand the following conditions:

- 120 cycles of plant startup and shutdown
- 400 OBE cycles
- 100 SSE cycles
- 1 accident cycle (LOCA)
- 160 pressure test cycles

2. Load Combinations

The load combinations for the equipment hatch and personnel air locks are specified in Table 3.8-6.

b. High Energy Piping Penetrations

1. Loads

Design loads applied to high energy piping penetrations were based on selected percentages of plastic load capacities of the attached piping. Seismic loads were based on the actual seismic response of the containment and piping at the elevation of the penetrations. When actual piping loads were available, having been derived from analysis of the piping systems, they were compared with the design loads; whenever a final load exceeded its design value, a re-analysis was made to validate the final piping loads. Loads were divided into two categories, Design (which includes normal, upset and emergency conditions) and Faulted, based on operating conditions. Magnitudes of individual design loads are given below.

(a) Axial Load (N)

For Design Condition, axial load varied from 5 percent to 80 percent of pipe plastic axial load. For the Faulted Condition, 65 percent to 100 percent was used.

(b) Axial Load Due to SSE Conditions (N_s)

This load was based on the mass of the flued head, and the portion of the sleeve up to the containment wall (outside) and the seismic acceleration at the elevation of the penetration.

(c) Shear Load (V_1)

This is the shear applied to the flued head. For the Design condition shear load varied from 5 percent 80 percent of the pipe plastic shear load. For the Faulted condition, 60 percent to 100 percent was used.

(d) Shear Load (V_2)

This is the shear in the process pipe at the location of the sliding support. Load V_2 was taken as 10 percent of the pipe elastic shear for the design condition and 100 percent for the faulted condition. It was determined interactively with the bending moment (M).

(e) Seismic (SSE) Shear at Flued Head (V_s)

Load V_s is based on the mass of the flued head and the portion of the sleeve up to the outside of the containment wall and the seismic acceleration at the elevation of the penetration.

(f) Pressure Inside the Cavity (P_1)

The design pressure inside the cavity was taken as 56 psi, which conservatively represents the containment design pressure (52 psi).

(g) Pipe Pressure (P_2)

This is the internal design pressure in the process pipe.

(h) Torsion (T)

This was taken as 50 percent and 100 percent of the pipe plastic torsion load for the Design and Faulted loading conditions, respectively.

(i) Moment Loads (M_1 and M_2)

These moments were considered equal and were taken as percentages of the value of moment load calculated as per the ASME B&PV Code, Section III, Division 1, Subsection NC-3652.1 and the Standard Review Plan 3.6.2, Branch Technical Position MEB 3-1, Paragraph B(1) (b)(1)(e). For both the Design condition and the Faulted condition, the percentages taken ranged from 60 percent to 100 percent.

2. Load Combinations

All loads were coidered simultaneously and were grouped into two categories. Design (which includes normal, upset and emergency conditions) and Faulted.

3. Environmental Conditions

(a) The normal temperature of the air within the containment enclosure outside the containment structure surrounding the penetration is 104°F maximum.

(b) The temperature of the air inside the containment structure is 120°F maximum (during normal and upset operating conditions).

(c) Loss of heat from the penetration to the air surrounding the penetration both outside and inside the containment structure is by natural convection.

(d) Process fluid temperatures were considered and vary from penetration to penetration.

(e) No effects of radiation were considered for high energy piping penetrations.

c. Moderate Energy Piping Penetrations

1. Loads

Loads applied to the penetration include actual piping loads, as determined from the piping analysis, and other applicable containment design loads. These loads are in compliance with the intent of Regulatory Guide 1.57 and are described below.

(a) Normal Axial Load (N)

This is the axial load imposed by the pipe onto the penetration.

(b) Shear Load (V)

This is the shear load imposed by the pipe onto the penetration.

(c) Torsional Load (T)

This is the torsional load imposed by the pipe onto the penetration.

(d) Bending Moment (M)

This is the bending moment imposed by the pipe onto the penetration.

(e) Pressure Inside the Cavity (P)

The design pressure inside the cavity was taken as 56 psig, which conservatively represents the containment design pressure (52 psig).

(f) Process Pipe Pressure Load (P_1)

This is the pressure inside the process pipe.

(g) Deadweight (D)

The deadweight of the penetrations and attachments is considered negligible.

(h) Temperature Load (T)

This is the load produced by thermal expansion of the attached piping.

(i) Operating Basis Earthquake Load (OBE)

These are the forces on the penetration from the attached piping caused by the operating basis earthquake acting on the pipe.

(j) Anchor Displacement Loads For the OBE (SAD OBE)

These are the forces produced by differential seismic displacements between the penetration and the piping anchors due to the OBE.

(k) Thrust Load (TH)

This is the load applied along the axis of the penetration.

(l) Safe Shutdown Earthquake Load (SSE)

These are the forces on the penetration from the attached piping caused by the safe shutdown earthquake acting on the pipe.

(m) Anchor Displacement Loads for the SSE (SAD SSE)

These are the forces produced from the attached piping by differential seismic displacements between the penetration and the piping anchors due to the SSE.

(n) Pressure and Thermal Anchor Displacement Load (PAD + TAD)

These are the forces on the penetration caused by differential displacement between the penetration and the piping anchors due to pressure and thermal loads.

(o) Pipe Rupture Load (PR)

This is the load on the penetration caused by the rupture of an attached pipe.

2. Load Combinations

Three design loading conditions were investigated: the Design/Normal Condition (Service Level A), the Upset Condition (Service Level B), and the Emergency/Faulted Condition (Service Level C/D). Load combinations used for each of these design loading conditions are presented in Table 3.8-7.

d. Electrical Penetrations

1. Loads

(a) Dead Load (D)

The weight of the penetration and its attachments were considered.

(b) Test Pressure (P_t)

The maximum test pressure inside the containment was 60.0 psig. This is the largest pressure to which the penetrations are subjected and, to simplify the calculations, this is the only pressure that is used for analysis.

(c) Accident Pressure (P_a)

The maximum containment pressure occurring during a design basis event is 52 psig.

(d) Test Temperature (T_t)

The temperature inside the containment during the pressure test will vary between 50°F and 100°F.

(e) Operational Temperature (T_o)

Under normal operation the temperature inside they containment will vary between 50°F and 120°F.

(f) Accident Temperature (T_a)

A peak accident temperature of 271°F was utilized for the design of components. This is a design temperature which is higher than the actual temperature to which the penetrations are subjected, and, to simplify the calculation, all material properties are based on this temperature. See Figure 3.8-27 for a plot of the actual transient temperature response of the electrical penetration seal.

(g) Operating Basis Earthquake (E)

A static equivalent load factor of 4.0 was used for both horizontal and vertical earthquakes. This value exceeded all acceleration values calculated in the response spectra at the elevation of the penetrations..

(h) Safe Shutdown Earthquake (E_{ss})

For the SSE, a static equivalent load factor of 4.0 was also used for both horizontal and vertical earthquakes. This value exceeded all acceleration values calculated in the response spectra at the elevation of the penetrations.

(i) External Pressure (P_e)

Under normal operation, the containment internal pressure varies between +1.5 psig and -3.5 psig. Thus, the maximum "external" pressure during normal operation was 3.5 psig.

2. Environmental Conditions

(a) Outside Temperature

The electrical penetrations were designed for temperatures outside the containment which vary between 40°F and 104°F. Following a DBA, the maximum temperature is 130°F.

(b) Atmospheric Pressure

Pressure on the outside of the containment varies between +0.006 psig and -0.005 psig during normal operation and between 0 psig and -0.011 psig during the containment test. Following a DBE, the minimum pressure is -0.013 psig.

(c) Relative Humidity

The maximum relative humidity inside the containment is 90 percent and outside the containment is 95 percent. Following a DBE, the relative humidity inside the containment is 100 percent.

(d) Radiation

The normal radiation rate is 50 millirads per hour. During the containment test, the radiation rate is 15 millirads per hour. The total cumulative radiation, considering 40 years of normal operation, a DBE and the post DBE period, is 1.3×10^8 Rads.

(e) Chemical Sprays

Chemical sprays include the following:

- Boric Acid	0.21 (by weight)
- pH (min/max)	8.5/10.5
- Sodium Hydroxide Solution	0.42 (by weight)

3. Loading Combinations

Loading combinations are given in Table 3.8-8.

e. Fuel Transfer Tube Assembly

1. Design Loads

(a) Preoperational Test Loads

(1) Test Pressure (P_t)

The test pressure is 60 psig.

(2) Test Temperature (T_t)

The temperature range inside the containment is 50°F to 100°F, and the minimum temperature at the site is 0°F.

(b) Normal Loads

(1) Dead Load (D)

This includes the weight of the fuel transfer tube and its support system.

(2) Live Load (L)

This includes the weight of the fuel assembly and carriage and the weight of water in the fuel transfer tube.

(3) Pressure Variation (P_v)

The containment internal pressure varies between -3.5 and +1.5 psig.

(4) Hydrostatic Pressure (H)

The maximum depth of water, above the tube, during refueling is 34'-4¼".

(c) Severe Environmental Loads

(1) Wind Loads (W)

There is no wind load due to the presence of the Fuel Storage Building.

(2) Operating Basis Earthquake (E_o)

The accelerations produced by the operating basis earthquake, which are valid for rigid components (f >33 Hertz), are 0.45g for N-S motion, 0.79g for E-W motion, and 0.42g for vertical motion.

The OBE seismic loads also included the forces produced by differential seismic displacement between the Containment Building and the Fuel Storage Building.

(d) Extreme Environmental Loads

(1) Tornado Loads (W_t)

Tornado loads are not considered due to the presence of the Fuel Storage Building.

(2) Safe Shutdown Earthquake (E_s)

The accelerations produced by the safe shutdown earthquake, which are valid for rigid components (f >33 Hertz), are 0.75g for N-S motion, 1.12g for E-W motion, and 0.61g for vertical motion.

The SSE seismic loads also included the forces produced by differential seismic displacement between the Containment Building and the Fuel Storage Building.

(e) Abnormal Loads

(1) Accident Pressure (P_a)

The containment internal accident pressure of 52.0 psig was considered in the design of the fuel transfer tube assembly.

(2) Accident Temperature (T_a)

A peak accident temperature of 271°F was considered for the design of the components.

(f) Fatigue Requirements

The fuel transfer tube was designed to withstand the following conditions:

- 400 OBE cycles
- 1 accident cycle (LOCA)
- 160 pressure test cycles
- 1000 temperature cycles

2. Load Combinations

The load combinations are given in Table 3.8-9.

f. Ventilation Penetration Assemblies

1. Design Loads

(a) Deadweight (D)

The deadweights of the sleeve, valves and attached piping, as applicable, were included.

(b) Operating Basis Earthquake (OBE)

The inertia forces produced by the operating basis earthquake were included.

(c) Safe Shutdown Earthquake (SSE)

The inertia forces produced by the safe shutdown earthquake were included.

(d) Normal Containment Pressure (P_n)

The normal containment pressure varies between -3.5 and +1.5 psig.

(e) Accident Pressure (P_f)

The accident pressure on the containment due to either a LOCA or a steam line break, whichever is higher, is used. The LOCA pressure governs and is 52.0 psig for design purposes.

(f) Test Pressure (P_t)

The containment test pressure is 60.0 psig.

2. Load Combinations

(a) Normal and Upset

$D + OBE + P_n$

(b) Faulted

$D + SSE + P_f$

(c) Test

$D + P_t$

3.8.2.4 Design and Analysis Procedures

The portions of the containment classified as steel containment were designed and analyzed using procedures described below. The components were designed to safely withstand the load combinations defined in Subsection 3.8.2.3.

a. Personnel Air Lock

The personnel air lock was designed and analyzed in accordance with Subsection NE of the ASME Boiler and Pressure Vessel Code, Section III, Division 1.

The air lock door was designed for an internal pressure of 52 psi combined with an equivalent pressure due to the OBE (0.40 psi) and the SSE (0.60 psi). The total pressure was applied to the complete cover. Design was governed by the SSE load in the Faulted Condition; hand computations were used.

That portion of the air lock barrel not backed by concrete was analyzed by hand computations for the most critical load combination in the Faulted Condition. The internal pressure and an additional pressure representing the effect of horizontal seismic loading result in a total pressure of 52.54 psi on the vertical projection of the shell. The deadweight and vertical seismic loading result in a vertical load of 20,277 lbs.

Stresses and strains within the barrel due to displacements imposed by the concrete were determined using a finite element model and the computer program ANSYS described in Subsection 3.8.2.4i. The displacements of the concrete were determined in the three-dimensional finite element analyses described in Subsection 3.8.1.4. Both elastic and elastic-plastic behaviors were considered.

Using hand computations, it was also determined that the natural frequencies associated with axial vibration of the air lock door and of the barrel as well as transverse vibration of the barrel are all much greater than 33 Hertz. Therefore, the personnel air lock was considered rigid for purposes of seismic analysis. The natural frequency for vertical vibration of the door in the open position, however, is less than 33 Hertz. This condition is not under the jurisdiction of Subsection NE but was considered for the design of the hinge plates and pins.

b. Equipment Hatch

The equipment hatch was designed and analyzed in accordance with Subsection NE of the ASME Boiler and Pressure Vessel Code, Section III, Division 1.

The equipment hatch cover was designed for the effects of accident pressure combined with the SSE, with the total applied pressure being 52.40 psi. Since live load and dead load are negligible, this combination represents the Faulted Condition with SSE, load combination 5 in Table 3.8-6.

The flange, bolts, and other components were also designed for the above indicated load combination as well as for external pressure, bolt preload, and other loads produced by gasket seating and seal testing. Hand computations were used.

Rotation of the flange was calculated to verify seal integrity, using an axisymmetric finite element model and the computer program AX2 (described in Subsection 3.8.2.4i). The critical loads for this analysis are bolt preload, with pressure on the convex side of the head, and bolt preload acting alone. In both cases, the integrity of the seal was not violated.

The design of that portion of the barrel not backed by concrete was based on analysis for accident pressure and for test pressure. Dead load and seismic load are negligible and were omitted. An axisymmetric shell model was analyzed using the computer program ANSYS. In these analyses, two sets of boundary conditions were used, one in which the flange was considered rigidly attached to the barrel and another in which the far end of the barrel was free. In this manner, the actual restraint was bounded.

A similar finite element model was used to determine the minimum thickness required for the end portion of the barrel, which is backed by concrete. Results at the concrete-to-metal junction verified the adequacy of a thickness of 1 inch.

Buckling of the head, sleeve, and barrel was checked, using the procedures of Subarticle NE-3133 of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, and found not to occur. Fatigue in the barrel was also investigated, and the design was found to be satisfactory.

To verify the design of the entire equipment hatch assembly, displacements imposed by the concrete onto the barrel were considered in an axisymmetric shell model which can accept nonaxisymmetric loading. The entire assembly, cover, flange and barrel, were modeled using ANSYS. Displacements were imposed at the intersection of the barrel with the containment liner, and accident pressure and temperature, dead load, live load, and seismic load were also included. Note that the personnel air lock was omitted from this model because simplified analyses of that region showed it to be similar in stiffness to the equipment hatch. There were four sets of imposed displacements based on four different cracking patterns in the concrete. For all load combinations, strains were added on an elastic basis and found to be below allowable values.

The air lock mounting sleeve, including its reinforcement, was analyzed by hand computation. Loads considered included accident pressure, dead load, live load, seismic load, and steady-state temperature. Stress analysis of the sleeve at the sleeve-collar boundary was performed using the AX2 axisymmetric shell program, considering only test pressure.

Transverse natural frequencies of the hatch, sleeve, and air lock were all found by hand computations to be greater than 33 Hertz. Using axisymmetric shell elements and the computer program ANSYS, the axisymmetric modes of vibration of the hatch were also found to be greater than 33 Hertz. Therefore, the equipment hatch assembly was considered rigid for purposes of seismic analysis.

c. Personnel Air Lock in Equipment Hatch

The personnel air lock in the equipment hatch was designed and analyzed in accordance with Subsection NE of the ASME Boiler and Pressure Vessel Code, Section III, Division 1.

This air lock was analyzed for accident pressure and SSE seismic loads. Overall structural response was calculated using the computer program STARDYNE (described in Subsection 3.8.1.4) and a three-dimensional model. Localized stresses at attachments, such as latches, pins and bearings, were determined using hand computations. In the analysis of the latches, an external pressure of (-)3.5 psig was applied in conjunction with the seismic loads.

The personnel air lock floor was designed for live load and dead load, and the test clamps, bolts, and plates were designed for the test pressure, all using hand computations. The open door seismic condition was also checked, considering the door in this position to be seismically flexible.

However, for all other design conditions the natural frequencies of the air lock assembly were found to be greater that 33 Hertz, and the assembly was considered rigid for purposes of seismic analysis. These calculations for natural frequency were made using the same finite element model that was used to determine overall response of the air lock.

d. High Energy Piping Penetrations

High energy piping penetrations were designed and analyzed to meet the intent of Subsection NE of the ASME Boiler and Pressure Vessel Code, Section III, Division 1. The design insured that the sleeve have a greater load carrying capacity than the process pipes.

The analysis procedure utilized three phases.

The first phase consisted of a heat transfer analysis to determine the temperature in the flued head, the sleeve and at the sleeve/containment wall interface. This was accomplished through the use of a finite element model and the computer program MARC-CDC (described in Subsection 3.8.1.4).

After establishing that concrete temperatures complied with the limit of ASME Section II, Division 2, Subsection CC-3340(b), a finite element model for stress analyses was established. This second phase consisted of a stress analysis utilizing unit load cases representing axial (N), shear (V), torsion (T), and moment (M) loadings.

Two additional load cases were run, using the containment design pressure and the actual thermal gradient, as determined in the heat transfer analysis. Total stresses for the design loads presented in Subsection 3.8.2.3 were determined by multiplying stresses from the unit load cases by the appropriate load ratio and combining the results with those from the pressure and thermal gradient runs.

The third phase consisted of using stress results from phase two to determine stress intensities at various locations in the sleeve and flued head and comparing these intensities to allowable values.

Computer programs used for stress analysis included MARC-CDC, WILSON 1 and WILSON 2 (all described in Subsection 3.8.1.4), and ANSYS (described in Subsection 3.8.2.4i). The temperature gradient which was used in the stress analyses was determined using the computer program TAPAS (described in Subsection 3.8.1.4); thermal boundary conditions were determined in the MARC-CDC heat transfer analysis of phase one.

e. Moderate Energy Piping Penetrations

Moderate energy piping penetrations were designed and analyzed to meet the intent of Subsection NE of the ASME Boiler and Pressure Vessel Code, Section III, Division 1. The design insured that the sleeves have a greater load carrying capacity than the process pipes.

The analysis procedure consisted of subjecting the penetration assembly to actual pipe loads and other applicable containment loadings and determining stresses at four critical locations, which are the junctions of the sleeve with the containment wall and the end plate and the junctions of the end plate with the sleeve and the process pipe. Weld stresses at the pipe/end plate interface weld were also determined.

All stresses were compared with appropriate allowable values to determine the adequacy of the penetrations.

All computations were performed by hand using standard formulas and techniques taken from established references and/or by the finite element program ANSYS.

f. Electrical Penetrations

Electrical penetrations were designed and analyzed in accordance with Subsection NE of the ASME Boiler and Pressure Vessel Code, Section III, Division 1.

A typical penetration was analyzed for dead load, test pressure (being the larger of maximum LOCA pressure and test pressure) and safe shutdown earthquake loads. The temperature was taken as 375°F for all loading conditions.

It was considered unnecessary to do a thermal analysis since the maximum temperature the assembly will be exposed to is only 370°F. Fatigue analysis was also unnecessary since it is expected that only a few cycles of loading will be experienced in the 40-year life of the plant.

Using appropriate hand computations, the below-listed penetration components were analyzed. Items 1 through 9 comprise the pressure boundary, and Items 1 through 8 code (Division 1) items. Items 10 and 11 are accessories which were also analyzed.

Item 1 - Installation weld which attaches the penetration assembly to the nozzle

Item 2 - 1/2-13 stud which holds the single clamp in place

Item 3 - 5/8-11 stud which holds the center clamp in place

Item 4 - Module - The module transmits the pressure applied over its projected area to the clamps.

Item 5 - Monitoring plate

Item 6 - Attaching weld (Monitor plate to flange extension)

Item 7 - Center clamp which receives loading from three modules. Each module contributed 1/3 of the load applied to it.

Item 8 - Single clamp. Each module contributed 1/3 the load applied to it.

Item 9 - Epoxy sealant material in the penetration module

Item 10 - Penetration terminal box (outboard)

Item 11 - Penetration terminal box (inboard).

To further verify the integrity of the penetration, it is pressure tested as described in Subsection 3.8.2.7.

g. Fuel Transfer Tube Assembly

The fuel transfer tube and its support system were designed and analyzed in accordance with Subsection NC of the ASME Boiler and Pressure Vessel Code, Section III, Division 1.

The fuel transfer tube assembly was analyzed for the various load combinations as listed in Table 3.8-9. Overall structural response was calculated using the computer program STARDYNE and a three-dimensional finite element model. This model was also used to determine the natural frequencies of the assembly, the lowest of which was 34.14 Hertz. Therefore, the fuel transfer tube assembly was considered rigid for purposes of seismic analysis.

Buckling of the tube was also checked, using the procedures of Subarticle NC-3133 of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC, and found not to occur.

In the load combinations which include seismic loads, the relative displacements due to differential movements between the Containment Building and the Fuel Storage Building were imposed at the sliding support.

h. Ventilation Penetration Assemblies

Ventilation penetrations were designed and analyzed to meet the intent of Subsection NE of the ASME Boiler and Pressure Vessel Code, Section III, Division 1. The design was based on system requirements, and subsequent analysis indicated that stress levels were well below an acceptable limit.

The analysis consisted of conservative static analysis procedures (hand computations) to determine stress levels at the interface with the outside of the containment wall, the critical region of the assemblies. Both primary and secondary stresses were calculated.

i. Descriptions of Computer Programs Utilized in the Design and Analysis

The computer programs used in the design and analysis of the steel components (Section III, Division 1 items) of the concrete containment which resist pressure and are not backed by concrete, are described briefly in this subsection. Programs listed below are only those programs which have not been previously described in Subsection 3.8.1.4g. A summary of the comparisons of results used to validate them is given in Appendix 3F. The program classifications are discussed in Subsection 3.8.1.4g.

1. ANSYS: Engineering Analysis System, by Swanson Analysis System, Inc. (Houston, Pa.). Documentation is available from Control Data Corporation. ANSYS provides static and dynamic analysis capabilities, including plasticity, creep and swelling; small and large deflections; steady-state and transient heat transfer; and steady fluid flow.

2. AX2: Axisymmetric Shell Program, by Pittsburgh-Des Moines (PDM) Steel Company. Documentation is available from PDM. AX2 is a computer program for the analysis of single layer, axisymmetric thin shells of revolution. Materials are isotropic and elastic, and loads are static and axisymmetric.

3.8.2.5 Structural Acceptance Criteria

The components classified as steel containment were designed to remain within the design limits specified in Subsection NE of the ASME B&PV Code, Section III, Division 1, and NRC Regulatory Guide 1.57.

Code boundaries for all penetrations except the personnel air lock are the portion of the sleeve which is backed by concrete and the attachment weld to the liner are Division 2; all remaining portions of the penetrations are Division 1. The division boundaries for the personnel air lock are described in Subsection 3.8.2.6c.

Additional acceptance criteria pertaining to specific penetrations are described below.

a. Equipment Hatch and Personnel Air Locks

Allowable stresses for the equipment hatch and personnel air locks are given in Table 3.8-10 and Table 3.8-11.

b. High Energy Piping Penetrations

The high energy piping penetrations were classified as ASME Section III, Division 1, Safety Class 2 components (NC). However, since the stress limits of Subsection NC are not directly applicable to this type of component for the specified categories of operating conditions, the penetrations were analyzed to meet the intent of the requirements of ASME Section III, Division 1, Subsection NB, and the allowable stress intensities were limited to the values given in Subsection NC. These allowables are summarized, in general terms, in Table 3.8-12.

c. Moderate Energy Piping Penetrations

The moderate energy piping penetrations were designed to meet the intent of ASME Section III, Division 1, Class MC (NE) criteria for stress limits and load combinations. Stress limits, used to determine the acceptability of the penetrations, are as follows:

	Primary Stress Intensity		Primary & Secondary Stress Intensity
	Membrane	Membrane + Bending	
Design/Normal Conditions	1.0 S_{mc}	1.5 S_{mc}	3 S_{mi}
Upset Conditions	1.0 S_{mc}	1.5 S_{mi}	3 S_{mi}
Emergency/Faulted Conditions	1.2 S_{mc} 1.0 S_y	1.8 S_{mc} 1.5 S_y	NA

d. Electrical Penetrations

All stresses are classified in accordance with Table NE-3217-1 of the ASME Code, Section III, Division 1, Subsection NE, and maximum stresses were found to be within the following limits:

1. The design limit of Subsection NE-6322 was not exceeded when the assembly was subjected to test pressure.

2. The design limits of Subsection NE-3131 (a), (b), and (d) were not exceeded when the assembly was subjected to the effects of the load combinations for test, normal, upset and faulted (abnormal/severe environmental) conditions.

3. The design limits of Subsection NE-3131 (c) (1) or (2), as applicable, were not exceeded when the assembly was subjected to the effects of the faulted (abnormal/extreme environmental) load condition.

4. The design limits of Subsection NE-3131.1 were not exceeded when the assembly was subjected to concurrently applied design loadings that produced the greatest potential for shell instability and loadings associated with the vibratory motion of the safe shutdown earthquake.

e. Fuel Transfer Tube Assembly

The fuel transfer tube assembly was designed to meet the intent of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NC, for Class 2 components. The maximum stresses and displacements at the bellows were found to be within the following limits:

1. The design limits of Subsection NC-3200, Appendix XIII-1140 were not exceeded when the assembly was subjected to the effects of the load combinations representing test, normal, refueling and upset conditions.

2. The design limits of Subsection NC-3200, Appendix XIII-1140 were not exceeded when the assembly was subjected to the effects of the faulted load condition.

3. The fuel transfer tube was analyzed in accordance with Subsection NC-3133 for the cylindrical shell subject to test pressure, accident pressure and seismic loadings produced by the SSE.

4. The deflection capabilities of the bellows were not exceeded when the assembly was subjected to the effects of the load combinations for test, normal, refueling, upset and faulted conditions.

f. Ventilation Penetration Assemblies

The ventilation penetrations were designed to meet the intent of ASME Section III, Division 1, Class MC (NE) criteria for stress limits and load combinations. For all of the indicated load conditions the stress limit was 1.0 S_m.

3.8.2.6 Materials, Quality Control and Special Construction Techniques

a. Materials

Materials used for components classified as steel containment include steel plate, bolts, pins, seamless pipe, seals, welding filler materials, coatings, and other miscellaneous items. These materials are listed in Table 3.8-11 for the equipment hatch and personnel air locks, and in Table 3.8-13 for the piping, electrical, and instrumentation penetrations, fuel transfer tube and ventilation penetration assemblies. Not listed in these tables is the protective coating system which is identical to the system described in Subsection 3.8.1.6. These coatings are applied to all exposed surfaces.

Replaceable silicone seals are used in the equipment hatch and personnel air lock.

b. Special Construction Techniques

The only special construction technique employed was the manner in which the personnel air lock was fabricated. To maintain the construction schedule, the barrel, inside bulkhead and outside bulkhead were shipped to the site and assembled and tested in place. The testing described in Subsection 3.8.2.7 verified that this assembly procedure did not affect the integrity of the lock.

The equipment hatch was also fabricated in one piece to minimize field welding. Shop and field testing for the hatch are described in Subsection 3.8.2.7.

c. Quality Control

1. Equipment Hatch and Personnel Air Locks

The contractor certified by application of the appropriate code symbol and completion of the appropriate Data Report in accordance with Article NE-8000 (Division 1) that the material used complies with the requirements of Article NE-2000 (Division 1), and that the fabrication, installation, and construction comply with the requirements of Article NE-4000 (Division 1). Specific quality control requirements are further discussed below.

Quality control procedures were in accordance with Articles NE-4000 and NE-5000 of Subsection NE of the ASME Code, Section III, Division 1. For all components, including welding and brazing materials covered under this section, the fabricators supplied Certified Materials Test Reports. These reports included chemical analyses, physical tests, mechanical tests, examinations and heat treatment.

Physical tests include a Charpy-V-Notch toughness test performed in accordance with Article NE-2300 of Division 1. The test temperature was (-)10°F in the heat-affected zone and (-)25°F in the parent metal for the equipment hatch and personnel air locks.

Longitudinal seams in welded pipe were 100 percent radiographed.

Dimensional standards for individual metallic components were in accordance with Section NE-2700 (Division 1).

All welding was performed in accordance with NE-4000 (Division 1). When the base metal temperature was less than 50°F, welding was not performed without heating the metal. When base metal temperature was within the range of 0°F to 50°F inclusive, the base metal within 3" of the place where welding was to be started was heated to a minimum temperature of 50°F.

Fabrication of the equipment hatch was performed in accordance with NE-4000 (Division 1) using the best shop practices relative to edge treatment, alignment, and general workmanship.

Due to its field fabrication, the personnel air lock was stamped as follows: the center barrel from 8" inside the containment to 1'-3" outside the containment is NPT Class CC, Division 2, 1975, with the Data Report noting that parts extending beyond the concrete were fabricated in accordance with Division 1, 1974 Edition. The other two end sections were stamped NPT Class 2 or MC Division 1, 1974 Edition.

Fabrication tolerances of linear dimensions, such as length and diameter of hatch and locks, were as listed below unless specified otherwise on drawings.

Dimensions less that 6 inches	±1/32 inch
Dimensions 6 inches to 2 feet-6 inches	±1/16 inch
Dimensions over 2 feet-6 inches to 10	±1/8 inch
Dimensions greater than 10 feet	±1/4 inch

Erection tolerances insured that the equipment hatch and personnel air locks were set plumb, square and level and at their proper elevation and plane. Angular tolerance through the containment wall, as measured from the point of attachment of each air lock or the equipment hatch to the containment liner,was not greater than ±30 minutes. Angular tolerance of the theoretical point of attachment to the containment liner, as measured from the center of the containment, was ±4 minutes in the horizontal plane. Vertical tolerance was ±1 inch. Negative radial deviation from the theoretical vertical centerline did not exceed (-)¼" at the equipment hatch.

2. Other Penetrations

Fabrication criteria for the other penetrations have been discussed in Subsection 3.8.2.1 with the descriptions of the penetrations.

3.8.2.7 Testing and In-Service Surveillance Requirements

a. Equipment Hatch and Personnel Air Locks

The structural testing and in-service inspection program consists of the following:

1. Strength test of personnel locks
2. Leak test of personnel locks
3. Leak test of leak chase system
4. Leak test of seals
5. Joint inspection
6. Other testing as described in Subsection 3.8.1.7.

Procedures are described in the paragraphs which follow.

The strength test is conducted in accordance with NE-6320 (Division 1); it consists of pressurizing the lock to 60.0 psig and then holding the pressure for 15 minutes with hold-down bars on the inner door.

Following the strength test, a leak test is conducted in accordance with NE-6215 (Division 1). During this test, the air pressure is reduced to 52.0 psig and held for 15 minutes. If a pressure drop occurs, soap bubble tests are conducted, repairs made, and the personnel lock retested.

Leak chase systems are leak tested in accordance with NE-6215 (Division 1) by pressurizing to 52.0 psig and holding for 15 minutes. If a pressure drop occurs, soap bubble tests are conducted, repairs made, and the personnel lock retested.

Seals for personnel lock doors, equipment hatch and at other locations are leak tested by pressurizing the space between the seals to 52 psig and holding for 15 minutes. If a pressure drop occurs, a halogen diode detector test was conducted, repairs made, and the seals retested.

All joints not covered by test channels are accessible for in-service inspection.

Testing of the entire containment structure, which includes the equipment hatch and personnel air lock, includes the integrated leak rate test, the in-service leak rate test, the preoperational structural integrity test, and general visual inspection of structurally critical areas. All of these tests are described in either Subsection 3.8.1.7 or in Subsection 6.2.6.

b. Other Penetrations

The portions of all the other penetrations which form part of the containment pressure boundary are subjected to the test program performed on the completed containment. This program includes the integrated leak rate test, the in-service leak rate test, the pre-operational structural integrity test and general visual inspection of structurally critical areas, all of which are described in either Subsection 3.8.1.7 or Subsection 6.2.6.

The penetrations also receive certain design verification testing as described below.

A typical electrical penetration is pressure tested in order to demonstrate that the module, epoxy and "O" ring seal can withstand the required pressure. A pressure chamber is fitted with the module and the pressure is raised to 6,000 psig without perceptible damage to the "O" rings or the penetration module. Thus a safety factor of 100 can be assumed (since the maximum design pressure is 60 psig).

3.8.3 Concrete and Structural Steel Internal Structures

The following section contains the physical description, codes, loads and load combinations, design and analysis procedures, allowable stresses, quality control and testing as they relate to the internal structures of the containment.

3.8.3.1 Description of the Internal Structures

The description of the structural configuration, materials, location and arrangement of the internal structures appears on the general arrangement drawings, Figure 1.2-2, Figure 1.2-3, Figure 1.2-4, Figure 1.2-5, and Figure 1.2-6.

The following major internal structures are located in the containment structure. They incorporate no unique or new design or construction features. Also, block or concrete masonry partitions are not utilized in the containment.

a. Reactor Support System

The reactor vessel is supported by two distinct support systems, the vertical box support and the lateral ring girder support. The vertical box support transmits vertical loads from the reactor vessel to the primary shield wall; it also transmits lateral loads from the reactor vessel to the lateral ring girder, which then transmits the loads to the primary shield wall. Design and analysis of the vertical box support are described in Subsection 5.4.14. The lateral ring girder support is described below.

The ring girder, reactor vessel lateral support, consists of a set of four curved girders resting on the primary shield wall at Elevation (-)14'-1", constructed of welded plates of ASTM A588, Grade 50 steel. As indicated above, its function is to restrain the lateral movement of the reactor vessel and to transfer the resultant lateral loads to the primary shield wall through normal contact. For details see Figure 3.8-28.

b. Steam Generator Support System

The steam generators are supported by several steel component supports, including vertical supports, upper lateral supports, and lower lateral supports. The vertical supports transmit loads to the fill mat and the upper and lower lateral supports transmit loads to both the primary and secondary shield walls. See Subsection 5.4.14 for a description of these supports.

c. Reactor Coolant Pump Supports

Reactor coolant pump supports, including vertical columns and lateral tension tie bars, are described in Subsection 5.4.14.

d. Primary Shield Wall

The primary shield wall is a circular reinforced concrete wall, varying in thickness from 4'-9" to 8'-6", enclosing and supporting the reactor pressure vessel. It is supported on the fill mat slab and extends to the refueling canal bottom elevation. In addition to providing shielding to the interior containment during normal operation or maintenance, the wall protects the reactor vessel from blowdown effects in the event of a rupture in the primary system piping outside the wall.

The reinforcing consists of several layers of hoop and meridional bars with the meridional bars extending either into the fill mat or the continuation of the fill mat down in the cavity in order to provide anchorage for the wall. See Figure 3.8-29.

e. Secondary Shield Wall

The secondary shield wall is a 4-foot thick octagon shaped reinforced concrete wall enclosing the reactor coolant piping, steam generators, reactor coolant pumps and their supports. The shield wall has openings to vent the area and to permit access from the outer annular area of the containment structure. These openings are protected by offset reinforced concrete walls which provide radiation shielding and protection from pipe whip. The secondary shield wall is supported on the fill mat slab and the above described portion extends to the underside of the operating floor.

Extending upward from the operating floor are those portions of the secondary shield wall that are referred to as the biological shield walls. They extend 7 feet above the operating floor and are shaped, in plan, like an elongated octahedron. Each one encloses two steam generators and provides radiation shielding.

f. Refueling Canal

The refueling canal consists of reinforced concrete walls and floors lined with ASTM A240 Type 304 stainless steel to provide a leak-tight membrane during refueling operations. The floor extends from the reactor cavity to the fuel transfer penetration and is supported on the primary shield wall and support walls which extend to the fill mat. The walls extend from the base of the canal to the operating floor level. During refueling the new and spent fuel elements are transported through the canal, which is flooded to Elevation 23.5 feet. Tools used during refueling are supported from the canal walls. For a more detailed description of the refueling canal and its use during refueling, see Subsection 9.1.4.

g. Pressurizer Region

The pressurizer region consists of a shield wall which provides shielding and missile protection. The wall is composed of several precast concrete sections which are placed one atop another. To maintain continuity these sections are bolted together. These sections are provided with lifting lugs and, when unbolted, may be removed to permit access to the pressurizer for maintenance and in-service inspection.

Pressurizer supports are described in Subsection 5.4.14.

h. Control Rod Drive Mechanism (CRDM) Missile Shield and Cable Bridges

The CRDM missile shield has been incorporated into the simplified head assembly. The original missile shield structure has been removed from the Containment. To support the CRDM and Drive Rod Position Indication (DRPI) System cables that run from the simplified head assembly two hinged cable bridges are provided on the operating deck, one on each side of the refueling cavity. See Figure 1.2-6A.

i. Operating Floor Slab

The operating floor slab is a reinforced concrete slab, 3'-0" to 8'-6" thick, located at Elevation +25.0 feet spanning between the crane support columns and the secondary shield walls. There are openings in the floor for access, installation, and removal of the four steam generators and the pressurizer, and access to the refueling canal and internal storage area. Removable reinforced concrete plugs are provided over the reactor coolant pumps and the in-core detector drive. The polar crane rail is located at the extremities of the operating floor concrete slab over its support structure.

j. Fill Mat

The internal structures are supported on a 4-foot thick reinforced concrete fill mat poured over the liner on the structural base mat. This fill mat is not connected to the base mat. Horizontal load transfer is provided by the keying action of the fill mat in the reactor pit, elevator pit, and sump pits. The fill mat extends under the secondary shield wall and the crane support columns to the containment cylindrical wall. An expansion joint is provided between the end of the fill mat and the containment cylindrical wall. The fill mat is thus independent of the primary containment structure walls. All supports for primary equipment and for other components and equipment located in the annulus area outside the secondary shield wall are supported on the fill mat.

k. Structural Steel

Structural steel is provided to support floors in the annulus at the operating level and in other areas of the containment structure where access to components and equipment is required, to provide a means of travel from various quadrants of the containment, and for pipe support at lower levels. Steel grating and concrete slabs are provided for a walking surface, and stairways and an elevator are required for movement between various levels of the annular steel.

l. Crane Support Structure

The crane rail is located on a continuous reinforced concrete beam at the operating floor level. The beam is supported on a series of reinforced concrete columns which extend to the fill mat outside the secondary shield wall. The beam is also connected to the biological shield wall through the floor slab at the operating floor level. This tie reduces the torsional effect on the beam, which is designed to resist the remaining torsional effects. Lateral stability of the columns is provided by reinforced concrete beams and slabs connected to the secondary shield wall at Elevations 0' and 25'.

The crane specification requires that the manufacturer's design insures that the crane will remain on the rail during either an OBE or an SSE. This is accomplished by means of kick-back plates attached to the wheel carriage, which prevent separation between the wheel and the rail. See Figure 3.8-30.

Seismic base shears are transmitted to the floor slab by means of a positive connection between the rail and the floor slab, which is part of the supporting structure. The crane rail anchor is designed to transfer all shear forces and overturning moments (resulting from the action of either an OBE or an SSE on the crane) to the concrete supporting system. Details of this connection include clamp plates, which restrain the rail and are welded to the base plate, and anchor bolts, which secure the base plate to the concrete beam. See Figure 3.8-30.

m. Neutron Shielding

The neutron shielding is a shell structure mounted on the slabs and walls above and around the reactor vessel. It consists of a plastic material encased in steel plate and provides radiation shielding.

A more complete description of the neutron shielding and its effect on structures can be found in Section 6.2.

3.8.3.2 Applicable Codes, Standards and Specifications

The design, materials, fabrication and inspection of the internal structures conform to, but are not necessarily limited to, the applicable sections of the following codes and specifications which are used to establish or implement design bases and methods, analytical techniques, material properties and quality control provisions. Dates and revisions given for the listed codes are the earliest version that was used. Subsequent issues were incorporated into the design where practicable or where the new issue directly affected the safety of the structure.

Code or Specification	Title
ACI 211.1-70	Recommended Practice for Selecting Proportions for Normal Weight Concrete
ACI 214-65	Recommended Practice for Evaluation of Compression Test Results of Field Concrete
ACI 301-72	Specification for Structural Concrete for Building (application sections)
ACI 302-69	Recommended Practice for Concrete Floor and Slab Construction
ACI 304-73	Recommended Practice for Measuring, Mixing, Transporting and Placing Concrete
ACI Committee Report 74-33	Recommended Practice for Hot Weather Concreting
ACI Report 306R-78	Recommended Practice for Cold Weather Concreting
ACI 308-71	Recommended Practice for Curing Concrete
ACI 309-72	Recommended Practice for Consolidation of Concrete
ACI Committee	Placing Concrete by Pumping Method
ACI Committee Report 72-33	Placing Concrete with Belt Conveyors
ACI 311-64	Recommended Practice for Concrete Inspections

Code or Specification	Title
ACI 315-65	Manual of Standard Practice for Detailing of Reinforced Concrete Structures
ACI 318-71*	Building Code Requirements for Reinforced Concrete (with Commentary)
ACI 347-68	Recommended Practice for Concrete Formwork
ACI 614-59	Recommended Practice for Measuring, Mixing, and Placing Concrete
ACI SP2 (1975 Edition)	ACI Manual of Concrete Inspection
CRSI	Reinforced Concrete - Manual of Standard Practice, 22nd Edition, first printing, 1976
CRSI	Recommended Practice for Placing Reinforcing Bars, 1968
ASME	ASME Boiler and Pressure Vessel Code, Section II, Material Specification, Part C - Welding Rods, Electrodes and Filler Metals (up to and including Winter 1974 Addenda)
ASME	ASME Boiler and Pressure Vessel Code, Section III, Division 2 - Code for Concrete Reactor Vessels and Containments (1975 Edition)
ASME	ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NF (up to and including Winter 1974 Addenda)
ASME	ASME Boiler and Pressure Vessel Code, Section V, Nondestructive Examination (up to and including Winter 1974 Addenda)

* Since the internal structures are reinforced and not prestressed, cannot be classified as thin shells or special ductile frames (no formation of plastic hinges allowed from seismic loads) and are not designed as special shear walls, the provisions of ACI 318-71 code Chapters 18, 19 and Appendix A are not applicable.

Code or Specification	Title
ASME	ASME Boiler and Pressure Vessel Code, Section VIII, Pressure Vessels, Part 1 (up to and including Winter 1974 Addenda)
ASME	ASME Boiler and Pressure Vessel Code, Section IX, Welding and Brazing Qualifications (up to and including Winter 1974 Addenda)
ASTM A1-68a	Standard Specifications for Carbon Steel Rails
ASTM A6-70	Specifications for General Requirements for Delivery of Rolled Steel Plates, Shapes, Sheet Piling and Bars for Structural Use
ASTM A53-73	Standard Specifications for Welded and Seamless Steel Pipe
ASTM A108-73	Standard Specifications for Cold Finished Carbon Steel Bars and Shafting
ASTM A123-73	Standard Specification for Zinc (Hot Galvanized) Coatings on Products Fabricated from Rolled, Pressed and Forged Steel Shapes, Plates, Bars and Strip
ASTM A143-74	Recommended Practice for Safeguarding Against Embrittlement of Hot-Dip Galvanized Structural Steel Products and Procedure for Detecting Embrittlement
ASTM A153-73	Standard Specification for Zinc Coating (Hot-Dip) on Iron and Steel Hardware
ASTM A185-70	Specification for Welded Steel Wire Fabric for Concrete Reinforcement
ASTM A193-75	Specification for Alloy Steel and Stainless Steel Bolting Material for High Temperature Service
ASTM A194-74	Standard Specification for Carbon and Alloy Steel Nuts and Bolts for High Pressure and High Temperature Service
ASTM A240-71	Specification for Stainless and Heat Resisting Chromium and Chromium-Nickel Stainless Steel Plate, Sheet and Strip for Fusion-Welded Unfired Pressure Vessels

Code or Specification	Title
ASTM A276-75	Standard Specification for Stainless and Heat Resisting Bars and Shapes
ASTM A307-68	Specification for Low Carbon Steel Externally and Internally Threaded Standard Fasteners
ASTM A312-74	Standard Specification for Seamless and Welded Austenitic Stainless Steel Pipes
ASTM A325-71	Specification for High Strength Bolts for Structural Steel Joints Including Suitable Nuts and Plain Temperature Service
ASTM A358-75	Standard Specification for Electric-Fusion-Welded Austenitic Chromium-Nickel Alloy Steel Pipe for High Temperature Service
ASTM A370-75a	Methods and Definitions for Mechanical Testing of Steel Products
ASTM A384-72	Safeguarding Against Warpage and Distortion During Hot-Dip Galvanizing of Steel Assemblies
ASTM A385-62	Recommended Practice for Providing High Quality Coatings (Hot Dip) on Assembled Products
ASTM A386-73	Standard Specification for Zinc Coatings (Hot Dip) on Assembled Products
ASTM A391-65	Standard Specification for Alloy Steel Chains
ASTM A394-75	Specification for Galvanized Steel Transmission Tower Bolts and Nuts
ASTM A446-72	Standard Specification for Steel Sheet Zinc Coated (Galvanized) by the Hot-Dip Process Physical (Structural) Quality
ASTM A479-75	Standard Specification for Stainless and Heat-Resisting Steel Bars and Shapes for Use in Boilers and Other Pressure Vessels
ASTM A480-74	Standard Specification for Delivery of Flat-Rolled Stainless and Heat Resisting Steel Plate, Sheet and Strip

Code or Specification	Title
ASTM A490-71	Specification for Quenched and Tempered Alloy Steel Bolts for Structural Steel Joints
ASTM A501-74	Standard Specification for Hot-Formed Welded and Seamless Carbon Steel Structural Tubing
ASTM A502-75	Standard Specification for Structural Steel Rivets
ASTM A514-70	Specification for High-Yield Strength, Quenched and Tempered Alloy Steel Plate, Suitable for Welding
ASTM A516-74a	Specification for Pressure Vessel Plates, Carbon Steel, for Moderate and Lower Temperature Service
ASTM A525-73	Standard Specification for General Requirements for Delivery of Steel Sheet, Zinc Control (Galvanized) by the Hot-Dip Process
ASTM A540-70	Alloy Steel Bolting Materials for Special Applications
ASTM A570-72	Hot-Rolled Carbon Steel Sheet and Strip
ASTM 572-75	High Strength Low Alloy, Columbium-Vanadium Steels of Structural Quality
ASTM A588-77a	Specification for High-Strength Low Alloy Structural Steel with 50,000 psi Minimum Yield Point to 4 in. Thick
ASTM A615-72	Specification for Deformed Billet-Steel Bars for Concrete Reinforcement
ASTM A759-78	Standard Specification for Carbon Steel Crane Rails
ASTM C29-71	Standard Methods of Test for Unit Weight of Aggregate
ASTM C31-69	Standard Method of Making and Curing Concrete Compressive and Flexural Strength Test Specimens in the Field
ASTM C33-71a	Specification for Concrete Aggregates

Code or Specification	Title
ASTM C39-71	Standard Method of Test for Compressive Strength of Molded Concrete Cylinders
ASTM C40-73	Standard Method of Test for Organic Impurities in Sands for Concrete
ASTM C42-68	Standard Method of Obtaining and Testing Drilled Cores and Sawed Beams of Concrete
ASTM C70-73	Standard Method of Test for Surface Moisture in Fine Aggregate
ASTM C87-69	Standard Method of Test for Effect of Organic Impurities in Fine Aggregate on Strength of Mortar
ASTM C88-73	Standard Method of Test for Soundness of Aggregates by Use of Sodium Sulfate or Magnesium Sulfate
ASTM C94-72	Specification for Ready-Mixed Concrete
ASTM C109-73	Standard Method of Test for Compressive Strength of Hydraulic Cement Mortars (Using 2-inch Cube Specimens)
ASTM C114-69	Standard Methods of Chemical Analysis of Hydraulic Cement
ASTM C117-69	Standard Method of Test for Materials Finer than No. 200 (75 M) Sieve in Mineral Aggregates by Washing
ASTM C123-69	Standard Method of Test for Light Weight Pieces of Aggregate
ASTM C125-74	Standard Definitions of Terms Relating to Concrete and Concrete Aggregates
ASTM C127-73	Standard Method of Test for Specific Gravity and Absorption of Coarse Aggregates
ASTM C128-73	Standard Method of Test for Specific Gravity and Absorption of Fine Aggregate
ASTM C131-69	Standard Method of Test for Resistance to Abrasions of Small Size Coarse Aggregate by Use of Los Angeles Machine

Code or Specification	Title
ASTM C136-71	Standard Method of Test for Sieve or Screen Analysis of Fine and Coarse Aggregate
ASTM C138-75	Standard Method of Test for Unit Weight, Yield and Air Content (Gravimetric) of Concrete
ASTM C142-71	Standard Method of Test for Clay Lumps and Friable Particles in Aggregates
ASTM C143-71	Standard Method of Test for Slump of Portland Cement Concrete
ASTM C150-71	Specification for Portland Cement
ASTM C151-74a	Standard Method of Test for Auto-clave Expansion of Portland Cement
ASTM C172-71	Standard Method of Sampling Fresh Concrete
ASTM C173-75	Standard Method of Test for Air Content of Freshly Mixed Concrete by the Volumetric Method
ASTM C186-73	Standard Method of Test for Heat of Hydration of Hydraulic Cement
ASTM C191-74	Standard Method of Test for Time of Setting of Hydraulic Cement by Vicat Needle
ASTM C192-69	Standard Method of Making and Curing Concrete Test Specimens in the Laboratory
ASTM C231-71T	Tentative Method of Test for Air Content of Freshly Mixed Concrete by the Pressure Method
ASTM C233-73	Standard Method of Testing Air-Entraining Admixtures for Concrete
ASTM C235-68	Standard Method of Test for Scratch Hardness of Coarse Aggregate Particles
ASTM C260-69	Specification for Air-Entraining Admixtures for Concrete

Code or Specification	Title
ASTM C289-71	Standard Method of Test for Potential Reactivity of Aggregates (Chemical Method)
ASTM C294-69	Description of Nomenclature of Constituents of Natural Mineral Aggregates
ASTM C295-65	Recommended Practice for Petrographic Examination of Aggregates for Concrete
ASTM C309-74	Standard Specification for Liquid Membrane-Forming Compounds for Curing Concrete
ASTM C404-70	Standard Specification for Aggregates for Masonry Grout
ASTM C494-71	Specification for Chemical Admixtures for Concrete
ASTM C496-71	Standard Method of Test for Splitting Tensile Strength of Molded Concrete Cylinders
ASTM C535-69	Standard Method of Test for Resistance to Abrasion of Large Size Coarse Aggregate by Use of the Los Angeles Machine
ASTM C566-67	Standard Method of Test for Total Moisture Content of Aggregate by Drying
ASTM C666-75	Standard Method of Test for Resistance of Concrete to Rapid Freezing and Thawing
ASTM D75-71	Standard Method of Sampling Aggregates
ASTM D512-67	Standard Methods of Test for Chloride Ion in Industrial Water and Industrial Waste Water
ASTM D1411-69	Standard Method of Test for Water-Soluble Chloride Present as Admixtures in Graded Aggregate Road Mixes
ASTM D1888-67	Standard Method of Test for Particulate and Dissolved Matter in Industrial Water
ASTM E109-63	Standard Method for Dry Powder Magnetic Particle Inspection

Code or Specification	Title
ASTM E329-72	Recommended Practice for Inspection and Testing Agencies for Concrete as Used in Construction. (Articles 7, 8 and 9 do not apply.)
AISC	Specification for the Design, Fabrication and Erection of Structural Steel for Building 1969 Edition (including Supplements 1, 2 and 3)
ANS 20.1 (ANSI N177)	Proposed Standard for the Design Basis for the Protection Against Internal and External Plant Missiles (April 1974)
ANSI A58.1-1972	American Standard Building Code Requirements for Minimum Design Loads in Buildings and Other Structures
ANSI B31.1-1973	Summer and Winter Addenda, Power Piping
ANSI N45.2-1974	Quality Assurance Program Requirements for Nuclear Power Plants
ANSI N101.2-1972	Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities
ANSI N101.4-1972	Quality Assurance for Protective Coatings Applied to Nuclear Facilities
ANSI N101.6-1972 *	American National Standard for Concrete Radiation Shield
ANSI N512-1974	Protective Coating (Paints) for Nuclear Industry
AWS B3.0-41	Standard Qualification Procedures
AWS D1.0-69	Standard Code for Arc and Gas Welding in Building Construction

* Exception is taken to those sections of ANSI N101.6 that are not applicable to nuclear power plants. The applicable sections of ANSI N101.6 are those listed in Regulatory Guide 1.69.

Code or Specification	Title
AWS D1.1-75	Structural Welding Code
AWS D1.1-75	Recommended Practice for Welding Reinforcing Steel, Metal Inserts and Connections in Reinforced Concrete Construction
American Railway Engineering Association (AREA)	Manual of Railway Engineering, Volumes I & II (1972 Revision)
American Hot Dip Galvanizing, Inc.	Hot Dip Galvanized Coatings (1973)
ASNT SNT-TC-1A (June, 1975)	American Society for Non-Destructive Testing Recommended Practice for Non-Destructive Testing-Personnel Qualifications and Certification
Bethlehem Steel	Crane Rails Catalog No. 3351 (updated to December 1979)
Bethlehem Steel	Specification for Fully Heat Treated Rails (March 1977 Revision)
Mixer Manufacturers Bureau of the Associated General Contractors of America	Concrete Plant Mixer Standards (1974 Issue)
MMB of the AGCA	Recommended Guide Specification for Batching Equipment and Control Systems in Concrete Batch Plants Publication 102 (1974 Issue)
NRMCA	National Ready-Mixed Concrete Association Certification of Ready-Mixed Concrete Production Facilities Instructions and Check List (1972)
NRMCA	Truck Mixer and Agitator Standards of Truck Mixers Manufacturer's Bureau (1971)
U.S. Army Corps of Engineers Specification CRD-C119-63	Method of Test for Flat and Elongated Particles in Coarse Aggregate

Code or Specification	Title
U.S. Dept. of Labor OSHA	Occupational Safety and Health Administration Standards, 1975 Edition
U.S. Dept. of Commerce National Bureau of Standards	PS-1-74 Construction and Industrial Plywood
U.S. Dept. of Commerce National Bureau of Standards	Handbook 44, Specifications, Tolerances and Other Technical Requirements for Commercial Weighing and Measuring Devices, 1971
SSPC	Steel Structures Painting Council Steel Structures Painting Manual, Volume 2; Systems and Specifications including supplement, 1973
SSPC	Good Painting Practice, Volume 1, 1966
Uniform Building Code	International Conference of Building Officials, Uniform Building Code, 1973 Edition
NRC TID 7024	Nuclear Reactors and Earthquakes, (August 1973)
NRC 10 CFR 50 App. B	Quality Assurance Criteria for Nuclear Power Plants
Nuclear Construction Issues Group (NCIG-01, Rev. 2, 5/7/85)	Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants

NRC Regulatory Guide No.	Title
1.10, Rev. 1, 1/73	Mechanical (Cadweld) Splices in Reinforcing Bars of Concrete Containments
1.15, Rev. 1, 12/72	Testing of Reinforcing Bars for Concrete Structures
1.29, Rev. 3, 9/78	Seismic Design Classification
1.54, Rev. 0, 8/72	Quality Assurance Requirements for Protective Coatings Applied

NRC Regulatory Guide No.	Title
	to Water-Cooled Nuclear Power Plants
1.55, Rev. 0, 6/73	Concrete Placement in Category I Structures
1.60, Rev. 1, 12/73	Design Response Spectra for Seismic Design of Nuclear Power Plants
1.61, Rev. 0, 10/73	Damping Values for Seismic Design of Nuclear Power Plants
1.69, Rev. 0, 12/73	Concrete Radiation Shields for Nuclear Power Plants
1.82, Rev. 0, 6/74	Sumps for Emergency Core Cooling and Containment Spray Systems
1.92, Rev. 1, 2/76	Combining Modal Responses and Spatial Components in Seismic Response Analysis
1.94, Rev. 1, 4/76	Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During Construction Phase of Nuclear Power Plants
1.142, Rev. 0, 4/78	Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)

The below listed UE&C design and construction specifications applicable to the containment internal structures and to the other seismic Category I structures were prepared in accordance with applicable codes, quality control requirements, and NRC Regulatory Guides:

UE&C Specification	Title
9763.006-12-1	Structural Steel Work
9763.006-12-2	Structural Steel Erection
9763.006-12-5	Fabrication of Safety-Related Structural Steel Work
9763.006-13-3	Category I Concrete Work Other Than Containment
9763.006-14-1	Furnishing, Detailing, Fabricating, and Delivering Reinforcing Bars

UE&C Specification	Title
9763.006-14-3	Installation of Reinforcing Bars in Category I Structures (Other Than Containment)
9763.006-18-1	Furnishing of Miscellaneous Embedded Steel and Weldments
9763.006-18-2	Installation of Miscellaneous Steel and Weldments
9763.006-18-3	Furnishing of Miscellaneous Steel
9763.006-18-4	Furnishing and Installing Embedded Steel and Miscellaneous Steel
9763.006-41-4	Furnishing of Protective Coating (Paint) System Materials and Related Services
9763.006-41-7	Field Painting of Containment Structure Interior
9763.006-69-1	Concrete Batch Plant
9763.006-69-3	Concrete Mixes
9763.006-69-6	Ready-Mixed Concrete from Off-Site Plant
9763.006-69-7	Standard Concrete Mixes
9763.006-69-10	Ready-Mixed Concrete for Category I, Non-Category I Structures and Systems from an Off-Site Plant
9763.006-225-2	Stainless Steel Liners
9763-MPS-1	Material and Processing Requirements (Nuclear)
9763-MPS-2	Material and Processing Requirements (Non-Nuclear)
9763-MPS-3	Material and Processing Requirements of Welded Studs, Reinforcing Bars and Anchor Bolts
9763-QAS-1	Administrative and System Requirements (Nuclear)
9763-QAS-2	Administrative and System Requirements (Non-Nuclear)

UE&C Specification	Title
9763-RM-1	Instructions for Site Records Management System
9763-WS-2	Requirements for Welding and Non-Destructive Examination for Non-Nuclear Pressure Components and Non-Nuclear Power Piping
9763-WS-3	Requirements for Welding and Non-Destructive Examination for Structural Steel
9763-WS-4C	Requirements for Cadwelding and Non-Destructive Examination of Mechanical Rebar Splice Method

3.8.3.3 Loads and Loading Combinations

The containment internal structures are designed to withstand all credible conditions of loading including normal loads, severe environmental loads, extreme environmental loads, and abnormal plant loads. These loads are determined in accordance with the applicable specifications, including ACI 318-71 and AISC-69, and are considered in Normal and Unusual Load Combinations to assure that the response of the structure remains within the limits prescribed in Subsection 3.8.3.5. Refer to Subsection 1.8 (Regulatory Guide 1.142) for a discussion concerning compliance with ACI-349.

a. Design Loads

The definitions of the loads used in the design of the internal structures include the following:

1. Normal Plant Startup, Operation and Shutdown Loads

Normal loads are those loads encountered during normal plant start-up, operation and shutdown. They include the following:

(a) Dead Loads (D)

Dead loads are all permanent gravity loads including the weight of concrete walls and slabs, structural framing, piping, cable and cable trays, permanent equipment, and static pressures of liquids.

(b) Live Loads (L)

Live loads include any movable equipment loads and other loads which vary in intensity and/or occurrence. Live loads are present only during shutdown conditions, and do not govern the design of any components.

(c) Operational Thermal Loads (T_o)

The temperature gradient through the walls under normal operating conditions is considered in the design. For a discussion of minimum and maximum operating temperatures, see Subsection 6.2.1.

(d) Operational Pipe Reactions (R_o)

These are pipe reactions due to thermal conditions existing in the piping during normal operation or shutdown. They are based on the most critical transient or steady-state condition.

2. Severe Environmental Loads

Severe environmental loads are those loads that result from events that could infrequently be encountered during the plant life. The only load included in this category is the following:

(a) Operating Basis Earthquake (E_o)

These are the loads generated by the Operating Basis Earthquake, which is the earthquake that could reasonably be expected to affect the site during the operating life of the plant. Only the actual dead load and weights of fixed equipment are considered in evaluating the seismic response forces.

The horizontal and vertical design response spectra for the OBE are derived by applying a factor of 0.5 to the response spectra given for the Safe Shutdown Earthquake (SSE) which is described below. The effects of two (2) orthogonal horizontal components and one (1) vertical component of earthquake are considered and combined by the square-root-of-the-sum-of-the-squares rule.

3. Extreme Environmental Loads

Extreme environmental loads are those loads which result from events which are credible but highly improbable. The only load included in this category is the following:

(a) Safe Shutdown Earthquake (E_{ss})

These are the loads generated by the Safe Shutdown Earthquake, which is the earthquake based upon an evaluation of the maximum earthquake potential in the vicinity of the plant. Dead and fixed equipment loads are described under the Operating Basis Earthquake, above. The horizontal and vertical forces on the internals are developed from the response spectra given in Figure 2.5-38 and Figure 2.5-39 the development of which is described in Subsection 2.5.2.6.

The effects of two (2) orthogonal horizontal earthquakes and one (1) vertical earthquake are considered and combined by the square-root-of-the-sum-of-the-squares rule.

4. Abnormal Loads

Abnormal loads are those loads generated by a postulated high energy pipe break resulting in a loss-of-coolant accident (LOCA). The design basis accident results in the highest postulated pressures and temperatures, and is determined by considering a LOCA.

(a) Accident Pressure (P_a)

Immediately following a postulated primary pipe break and prior to pressure equalization, a differential pressure occurs on the interior structures. See Subsection 6.2.1.2 for a description of these pressure loads, including the effects of the neutron shielding on pressures. Note that the differential pressure load on the primary shield wall is greater than the differential pressure load for the secondary shield wall; however, in this subsection one symbol is used to represent this loading with the understanding that the value is as specified in Subsection 6.2.1.2. Accident pressures resulting from a main steam line break are also considered.

(b) Accident Temperature (T_a)

Also following a postulated primary pipe break is an increase in temperature on the interior structures. These loads are also further described in Subsection 6.2.1.3.

(c) Accident Pipe Reactions (R_a)

Pipe reaction loads due to thermal conditions generated by the postulated pipe break, including R_o, are considered in the design. The magnitude of these loads is determined by the piping design.

(d) Pipe Break Loads (R_r)

Pipes, other than the Primary Coolant System, are anchored in sleeves through the interior concrete structures and transmit thrust loads to the structure during normal operating conditions and for the postulated pipe break. In addition, loads transmitted by pipe whip restraints are considered. The three components of pipe break loads are as follows:

(1) R_{rr} = load on the structure generated by the reaction of a ruptured high energy pipe during the postulated pipe break. The time-dependent nature of the load and the ability of the containment to deform beyond yield are considered in establishing the structural capacity necessary to resist the effects of R_{rr}.

(2) R_{rj} = load on the structure generated by jet impingement from a ruptured high energy pipe during the postulated pipe break. In general, direct impingement of steam on the structure does not produce significant design loadings due to the distance between the wall and the break location. Where a break is postulated to occur close enough to a wall to produce a critical loading, a shield, or deflector, is provided, and the loading is transferred to the embedment for the pipe whip restraint to which the shield is attached.

(3) R_{rm} = load on the containment resulting from the impact of a ruptured high energy pipe during the postulated pipe break. Since all high energy lines are constrained by pipe rupture restraints, missile loadings of this nature are prevented.

5. Other Loads and Load Considerations

(a) Internal Missile Loads (M)

Internal missile loads, other than those defined as R_{rr}, R_{rj}, and R_{rm}, are considered including an appropriate dynamic factor to account for the dynamic nature of the load. For a discussion of specific missiles generated inside the containment and the missile design procedures, see Section 3.5. Note that the sides and floor of the refueling canal are designed for the impact of the neutron shielding.

(b) Primary Equipment Supports

The primary equipment supports transmit loads to the fill mat, to the primary and secondary shield walls and to the operating floor slab. For a discussion of these loads and of the design criteria for the supports see Subsection 5.4.14.

(c) Load Considerations for Internal Structures

(1) Time dependent loads such as thermal effects, creep and shrinkage do not have any significant effects on the internal structures since the accident loads are generally resisted in tension by the reinforcing steel. In addition, the accident loads are short-term, once-occurring loads, which have negligible creep effects.

(2) The containment structure functions as an independent structure with complete physical separation from the internal structures, and therefore there are no loading interactions between the two.

(3) Compartmentalization is considered in the design of the internal structures by using the peak subcompartment differential pressures, plus a safety margin. This is further discussed in Subsection 6.2.1.2.

b. Load Combinations

Various load combinations are considered in design to determine the greatest strength requirements of the structure. Where varying loads occur, the combinations producing the most critical loading are used. Basic combinations in the design of the containment internal structures are given in Table 3.8-14. These load combinations are in agreement with Subsections II.3 and II.5 of the Standard Review Plan for Subsection 3.8.3 of the Updated FSAR. The factors which are to be applied to allowable stresses have been transposed and applied as load factors instead, resulting in acceptance criteria as indicated in the table. Two categories of loading conditions and criteria are used in the design of the containment internal structures as described below.

1. Normal Load Conditions

Normal load conditions e those encountered during testing and normal operation. They include dead load, live load and anticipated transients or test conditions during normal and emergency startup and shutdown of the Nuclear Steam Supply, Safety and Auxiliary Systems. Normal loading also includes the effect of an Operating Basis Earthquake. Normal load conditions are referred to in Division 2 as service load conditions.

Under each of these loading conditions, the structure is designed so that stresses will be within the elastic limits. Design assumptions are presented in Subsection 3.8.3.4 and stress limitations are presented in Subsection 3.8.3.5.

2. Unusual Load Conditions

Unusual load conditions are those conditions resulting from combinations of the LOCA, SSE and OBE, high-energy pipe failures, and live and dead loads. They are referred to in Division 2 as factored load conditions.

For each of the unusual loading combinations, the internal structures are designed to remain below their ultimate capacity so that the behavior of structural components is in the small deformation elastic range. Design assumptions are presented in Subsection 3.8.3.4 and stress limitations are presented in Subsection 3.8.3.5.

3.8.3.4 Design and Analysis Procedures

a. Design and Analysis

After preliminary design, the internal structures are analyzed to determine the maximum stress and displacements in reinforcing steel, concrete and structural steel for the loading criteria described in Subsection 3.8.3.3. Each structural system, either an individual component or a group of connected components, is analyzed separately using appropriate design assumptions and boundary conditions.

The analyses include fixed base conditions, uncracked concrete, load superposition, modeling and selective interpretation of results. Material properties and boundary conditions are selected to be representative of the particular structure or component being analyzed. Either conventional methods and formulae of structural mechanics or the finite element direct stiffness method are used. All analyses consider elastic behavior only. The design is based on codes, specifications and documents listed in Subsection 3.8.3.2.

The methods of design and analysis, including assumptions, boundary conditions, loads resisted and analysis techniques are given below. Seismic analysis is discussed in Subsection 3.7.2.

Reinforcing in concrete components is proportioned in accordance with the responses obtained from the analyses using the strength design method of the ACI 318-71 code, as applicable, and the requirements of Subsections 3.8.3.3 and 3.8.3.5. Bond and anchorage requirements are in accordance with ACI 318-71. As a result, reinforcing patterns in walls and floors, in general, consist of orthogonal layers on both the inside and outside faces of the concrete.

The computer program CONCOL (described in Subsection 3.8.3.4) was used throughout the internals for ultimate strength design of concrete. Other computer programs used in the design and analysis of specific internal structures are described below.

Structural steel is designed in accordance with the AISC Specification, using elastic methods, except as noted below for the reactor support system.

Refer to Subsection 1.8 (Regulatory Guide 1.142) for a discussion concerning compliance with ACI-349.

b. Internal Structures

1. Reactor Support System

Design and analysis of the vertical supports are described in Subsection 5.4.14.

The reactor vessel lateral support, the ring girder, is analyzed using a three-dimensional finite element model. Since the support possesses bilateral symmetry and is subjected only to unilateral horizontal loading, only half of the support was required for the model. Loadings used in the analyses were supplied by the NSSS vendor, Westinghouse.

Design and analysis procedures for the reactor vessel lateral support are in accordance with the ASME Boiler & Pressure Vessel Code, Division 1, Subsection NF and Appendix XVII.

2. Steam Generator Support System

Design and analysis of the steel supports are described in Subsection 5.4.14.

3. Reactor Coolant Pump Supports

Design and analysis of the reactor coolant pump supports are described in Subsection 5.4.14.

4. Primary Shield Wall

The reactor pressure vessel is enclosed and supported by a reinforced concrete primary shield wall. The primary shield wall is designed under the accident condition considering the effects of temperature, jet forces and forces transmitted by the reactor vessel, seismic loads, etc. The design pressure loads are conservatively considered as peak transient pressure differential on the wall. Other LOCA forces on the structure are determined using dynamic time-dependent analysis. The peak values of these loads are conservatively used as equivalent static loads.

The wall is analyzed as an axisymmetric structure resting on the fill mat using the WILSON 2 computer code (described in Subsection 3.8.1.4). The analysis is performed assuming base fixity and top restraint provided by the floor of the refueling canal. Differential pressure loads and thermal loads are analyzed as axisymmetric load cases. Differential pressures are considered as nonaxisymmetric loads and are represented by a Fourier series.

An additional consideration is the effect of radiation in generating heat in the primary shield wall. This wall is the only concrete subjected to relatively high irradiation. The attenuation of the integrated neutron and gamma rays within the wall will produce a temperature rise which will be limited to a maximum of 150°F at the concrete surface adjacent to the reactor vessel by the convection of cooling air circulated between the wall and the reactor vessel.

Studies on the behavior of concrete under irradiation (Reference 5) indicate that irradiated specimens behave in a manner similar to control specimens subjected to temperature without radiation. Thus, the effects on the shield wall concrete from heat generated by irradiation were evaluated in the same manner as temperature rise occurring in the concrete from a nonradiating source.

5. Secondary Shield Wall

The secondary shield wall is analyzed as part of a larger three-dimensional model which includes the secondary shield wall, the refueling canal, the operating floor slab, the crane support structure and the biological shield wall. The computer program used was STARDYNE (described in Subsection 3.8.1.4).

The model is composed of plate elements (which include bending and transverse shear) for the walls and slabs and beams for the crane support structure and other uses as required. Restraint is provided at the level of the fill mat and at the interfaces between the operating floor and the primary shield wall. Differential pressures, thermal loads, pipe break loads, and normal operating pipe loads are applied to the finite element model. Force and displacement boundary conditions imposed by the primary shield wall onto the operating floor slab are considered.

6. Refueling Canal

The refueling canal is analyzed as part of the model described above under the heading, "Secondary Shield Wall."

7. Pressurizer Region

The pressurizer region is analyzed for the established load criteria using an independent three dimensional finite element model and the general procedures described above under the heading, "Secondary Shield Wall."

8. Control Rod Drive Mechanism (CRDM) Missile Shield

The CRDM missile shield has been integrated into the simplified head assembly. The missile shield has been evaluated to the criteria presented in the Standard Review Plan (SRP) Section 3.5.3. The missile shield and supporting structure have been modeled using the ANSYS computer code.

9. Operating Floor Slab

The operating floor slab is analyzed as part of the model described above under the heading, "Secondary Shield Wall."

10. Fill Mat

The fill mat is analyzed as an independent circular reinforced concrete slab; the model consists of plate elements (which include bending and transverse shear). It is designed to resist and transfer to the containment base mat all reaction loads from the internal structures including the primary equipment supports. The internal structures are generally rigidly connected to the fill mat.

11. Structural Steel

Structural steel framing and platforms, including connections, are conventionally designed using applicable codes and specifications, as listed in Subsection 3.8.3.2. Those members which support piping, electrical cable trays and ducts were further evaluated for rigidity considerations as described below.

All members of the containment annulus steel framing that support piping or piping and electrical cable trays and/or ducts were initially designed to have natural frequencies of 20 hz or greater, in order to minimize the seismic loads in the supported components. For members supporting only ducts and/or electrical cable trays the minimum frequency was 15 hz.

Where meeting these criteria is not practicable, members supporting piping were initially designed for a minimum frequency of 15 hz, if practicable.

Members that were not sized using the frequency approach were initially designed for static loads and an additional 2.0 g factor (horizontal and vertical).

Final member sizes were all checked for conformance with applicable codes against final loadings obtained from dynamic analyses of steel framing, piping, ducts and equipment.

12. Crane Support Structure

The crane support structure is analyzed as part of the model described above under the heading, "Secondary Shield Wall."

13. Control Rod Drive Mechanism (CRDM) and Drive Rod Position Indication (DRPI) Cable Bridges

To support the CRDM and DRPI cables that run to the simplified head assembly, two hinged cable bridges have been mounted on the operating deck, one on each side of the refueling cavity. The bridges have been modeled using the STAAD III computer code.

c. Description of Computer Programs Utilized in the Design and Analysis

The computer programs used in the design and analysis of the containment internal structures are described briefly in this subsection. Programs listed below are only those programs which have not been described previously in Subsections 3.8.1.4g and 3.8.2.4i. A summary of the comparisons of results used to validate them is given in Appendix 3F. The program classifications are discussed in Subsection 3.8.1.4g.

1. CONCOL (SYSTEM PROFESSIONAL): Documentation is available from Control Data Corporation. CONCOL is a computer program for ultimate strength design of concrete columns.

2. STRAP: "Static Analysis of Linear Elastic Structures," by United Engineers & Constructors, Inc. STRAP is used to perform static analyses of structures which can be represented as an assemblage of linear elastic members. It uses the stiffeners matrix methods of analysis.

3. GENSAP (STRU-PAK): "Static Analysis of Elastic Structures." Documentation is available from Control Data Corporation. GENSAP is used for general static analysis of elastic structures composed of beams and columns.

4. STAAD III Structural Analysis and Design Software: A proprietary computer program of Research Engineers, Inc. (REI), California, for the analysis and design of structures. The code has been placed under configuration control by Westinghouse and specific features of this software utilized in the simplified head assembly calculations have been independently verified by Westinghouse. (Westinghouse Letter Number EDRE-CSE-134(97), Software Release of STAAD III (22.0W) on Windows NT System, 9/25/97, and Westinghouse Calculation #CSE-06-98-0001, Rev. 0, titled: "STAAD Verification Problem-Response Spectra Analysis.")

5. GOTHIC Generation of Thermal-Hydraulic Information for Containments: GOTHIC is a general purpose thermal-hydraulic computer program for design, licensing, safety and operating analysis of nuclear power plant containments and other confinement buildings. GOTHIC has been developed for the Electric Power Research Institute (EPRI) by Numerical Applications Inc, Richland, Washington. The code has been placed under configuration control by Westinghouse. (GOTHIC Qualification Report for Version 5.0e(NAI 8907-09, Rev. 3, Dec. 1995.)

3.8.3.5 Structural Acceptance Criteria

The bases for the development of the following stress-strain criteria are the ACI 318-71 and AISC codes.

a. Normal Load Conditions

Internal structures are proportioned to remain within the elastic limits under all normal loading conditions described in Subsection 3.8.3.3.

Reinforced Concrete - designed in accordance with ACI 318-71 Strength Method, which insures flexural ductility by control of reinforcing steel percentages and stresses.

Structural and Miscellaneous Steel - designed in accordance with AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, Part I.

b. Unusual Load Conditions

Internal structures are designed to maintain elastic behavior under all unusual load conditions shown in Subsection 3.8.3.3. The upper bound of elastic behavior is taken as the yield strength capacity of the load carrying components. The yield strength of structural and reinforcing steel is taken as the minimum guaranteed yield stress as given in the appropriate ASTM Specification.

Reinforced Concrete - designed in accordance with ACI 318-71 Building Code. Member yield strength is considered to be the strength capacity calculated by the ACI Code.

Structural and Miscellaneous Steel - designed in accordance with AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, Part 1.

Overall stability of steel structures designed for unusual loading is verified using the AISC Specification, Part 2, and the load factors in Table 3.8-14, with the exception of the reactor vessel lateral support, which is designed in accordance with ASME B&PV Code, Division 1, Subsection NF.

c. Deformations

Each of the structures is designed to remain in the small deformation elastic range so that no gross deformations will occur and cause contact with other structures or pieces of equipment.

d. Seismic Analysis

The seismic analysis techniques used for seismic Category I structures and systems are described in Subsection 3.7.2. The seismic analysis techniques used for seismic Category I subsystems are described in Subsection 3.7.3.

The shear produced by the seismic response of the internal structures is accounted for in design. If the available shear strength of the concrete section is not adequate, shear reinforcement is used to increase the shear resistance of the section. In general, seismic loads are not the governing factor in the design of the internal structures. Consequently, shear produced by seismic response is resisted by orthogonal layers of reinforcing provided for the applicable design loading combination; diagonal seismic reinforcing is not required. The shear response of the internal structures is transferred to the fill mat which is keyed into the containment base mat at the reactor cavity pit.

3.8.3.6 Materials, Quality Control and Special Construction Techniques

The primary materials of construction for the internal structures are concrete, reinforcing steel, structural steel (rolled shapes and plates), stainless steel (liner plate), and field coatings. Descriptions of these materials and respective quality control are discussed below. There are no special construction techniques.

a. Concrete

Concrete work is in accordance with ACI 318-71 and ACI 301 codes, except as noted in Subsection 1.8 (Regulatory Guide 1.55). The concrete is a dense, durable mixture of sound, coarse aggregate, cement and water. Admixtures were added, where required, to improve the quality and workability during placement and to retard the set of the concrete. Engineering approval was required prior to the use of admixtures.

Aggregate conforms to ASTM C33. It consists of inert materials that are clean, hard and durable, free from organic material and uncoated with clay or dirt. Fine aggregate consists of natural sand and the coarse aggregate consists of crushed stone.

Portland cement conforms to ASTM C150, Type II (moderate heat of hydration requirements).

Water is clean and free from any deleterious amounts of acid, alkali, salts, oil, sediment, organic matter or other substances which may be harmful to the concrete or steel.

The reinforced concrete has a nominal density of 150 lb/ft^3, which is used for determination of dead load. Shielding calculations for the primary shield wall are based on a dry concrete density of 139 lb/ft^3; other shielding calculations are based on a dry concrete density of 147 lb/ft^3. The 28-day standard compressive strength of the concrete is 4000 psi.

To assure that adequate means of control were used in the manufacture and that the properties described above were realized, the following were required:

1. Suppliers, fabricators and contractors were required to have written quality surance procedures, which were reviewed and approved by United Engineers. Material certifications were required in accordance with the applicable portions of the quality assurance plan described in Chapter 17 of the Updated FSAR and in the material specifications.

2. The following tests and certifications were required:

(a) Aggregates were tested to comply with ASTM C33.

(b) Cement was tested in accordance with ASTM C114 to conform to ASTM C150.

(c) Concrete samples were taken from the mix in accordance with ASTM C172.

(d) Cylinders were made in accordance with ASTM C31.

(e) Compressive tests were made in accordance with ASTM C39.

(f) Slump tests were in accordance with ASTM C143 and air content tests were in accordance with ASTM C231.

(g) Evaluation of the tests was in accordance with the material specifications, the codes discussed above, and the ACI 318-71 code, as applicable.

3. All making and testing of concrete samples was accomplished by an independent testing laboratory.

All concrete operations during cold weather conditions followed the practice defined in ACI 301 and 306R-78.

During cold weather curing of the concrete, concrete surfaces whose temperatures are below 50°F by accident for short periods of time, but remain 40°F or above, have had the 7-day curing period extended by the amount of time the concrete was below 50°F (rounded out to the nearest whole day).

The strength maturity method as defined in ACI 306R-78 was allowed as an alternate to specified curing requirements. The required maturity factor was selected to insure that concrete attained 70 percent of the design strength. This method was also used to determine the attainment of in-place concrete strength required for removal of forms in lieu of field cured cylinder as required in ACI 301.

b. Reinforcing Steel

Reinforcing steel consists of high-strength deformed billet bars conforming to ASTM A615, Grade 60. Certified Material Test Reports and user tests, as required by Regulatory Guide 1.15, were provided by the material manufacturer.

All reinforcing bars were spliced in accordance with UE&C Specification 9763-WS-4C, and No. 14 and 18 bars were joined by mechanical butt splices (Cadweld splices).

Additional information on requirements for reinforcing steel can be found in Subsection 3.8.1.6.

c. Structural Steel

All structural and miscellaneous steel, including stainless, conforms to the following specifications unless otherwise noted.

Structural steel conforming to ASTM A36 was used except where high strength is required. ASTM A588 or A572, Grade 50 was used for these applications, such as the reactor vessel lateral support, the cross-over leg restraint and the hot leg restraint.

To assure that steel plates, such as embedded plates for pipe restraints, are able to transmit orthogonal loads, a material with good through-plate tensile properties, ASTM A36 steel, was used.

Bolts were made of ASTM A325 steel except in special applications. Where high strength was required ASTM A193, A490, or A540 bolts were used. ASTM A307 bolts were used for stair stringers, stair treads, grating, handrail, toe plates, girts and purlins.

Stainless steel liner plate was fabricated from material conforming to ASTM A240, TP-304 steel. A liquid penetrant test is performed on the liner plate on the refueling canal to assure leak tightness.

The crane rail is a 171 lb/ft rail fabricated from A-759 steel; it was fully heat treated.

All welding conforms to either the American Welding Society Code (AWS) or to the ASME Boiler and Pressure Vessel Code (Section III).

For structural steel governed by AWS D1.1 and not subject to fatigue loads, welds shall be visually inspected in accordance with NCIG-01, "Visual Weld Acceptance Criteria (VWAC)," as specified in UE&C Specification WS-3. The only exception is for structural steel governed by AWS D1.1 which are subject to fatigue; no undercut is permitted.

Welders of carbon steel are qualified in accordance with the "Standard Qualification Procedure" of the AWS; welders of stainless steel liner plate are qualified in accordance with Section IX of the ASME Boiler and Pressure Vessel Code.

Certified Material Test Reports giving chemical composition and physical properties were supplied by the manufacturers for all structural steel.

Fabrication and erection was in accordance with AISC standards and the applicable material specifications.

d. Steel and Concrete Coating System

Materials used for coating the internal structures are the same as those described in Subsection 3.8.1.6. These materials meet the requirements of ANSI Standard N101.2.

3.8.3.7 Testing and In-Service Surveillance Requirements

Quality control testing as discussed in Subsection 3.8.3.6 will be employed. No additional testing or in-service surveillance is required.

3.8.4 Other Seismic Category I Structures

This section contains physical descriptions, codes, loads and load combinations, design and analysis procedures, allowable stresses, quality control, and testing requirements as they relate to seismic Category I structures exclusive of the containment structure and its internals. Nonseismic Category I structures which received special design considerations to prevent collapse onto adjacent Category I structures are also included in this section.

3.8.4.1 Description of the Structures

Locations of all of the major plant structures are shown on the plant arrangement plan, Figure 1.2-1; overall dimensions of the seismic Category I structures are given in Table 3.8-15.

All seismic Category I structures are separated from adjacent structures above the point of fixity by means of isolation joints, except for a few structures described below in which two portions of the same structure are used for two different functions. The resulting multi-function structure is isolated from adjacent structures. See Subsection 3.8.5.1 for a detailed discussion of seismic isolation.

The seismic Category I structures are described in the following paragraphs. They incorporate no unique or new design or construction features. Also, block or concrete masonry partitions are not utilized in any Category I structure.

a. Containment Enclosure Building

The Enclosure Building is a reinforced concrete right cylindrical structure with a hemispherical dome. The inside diameter of the cylinder is 158 feet. The vertical wall varies in thickness from 36 inches to 15 inches; the dome is 15 inches thick. The inside of the dome is 5'-6" above the top of the containment dome.

Located on the outside of the Enclosure Building is the plant vent stack, consisting of a light steel frame with steel plates and varying in cross section. The stack carries exhaust air from various buildings.

The space between the containment and the Enclosure Building is maintained at a slight negative pressure during accident conditions. All joints and penetrations are caulked or gasketed to ensure air tightness. See Figure 1.2-2, Figure 1.2-3, Figure 1.2-4, Figure 1.2-5 and Figure 1.2-6 for sections and elevations of the Enclosure Building.

b. Containment Equipment Hatch Missile Shield

This shield is a removable, precast, reinforced concrete wall located outside the equipment hatch. It serves to protect the hatch from tornado-generated missiles. See Figure 1.2-4 and Figure 1.2-5 for a plan and a section.

c. Containment Enclosure Ventilation Area

The containment enclosure ventilation area is an irregularly shaped reinforced concrete building that houses ventilation equipment (fans, filters, etc.) for the Enclosure Building and is located on the southwest side of the containment. Its overall dimensions are 116.0 feet long, 55.75 feet wide at its widest point, and 31.5 feet high. See Figure 1.2-2, Figure 1.2-3, and Figure 1.2-4.

d. Control and Diesel Generator Building

The Control and Diesel Generator Building is a reinforced concrete structure approximately 233 feet long by 90 feet wide. All load carrying walls and columns are founded on fill concrete and rock below grade. This is a multi-function structure in which the two portions, the control room area and the diesel generator area, are separated by a common wall in the north-south direction and are not seismically isolated. The building was analyzed and designed as a unit.

The east portion of the structure (Control Building), which is 138 feet long, has three floors and extends from grade to approximately 79 feet above grade. The two intermediate floors and roof are supported on steel columns in the center and on concrete walls all around. The ground floor contains electrical switchgear, motor generator sets and battery rooms; the second floor is for cable spreading; and the third floor is the main control room.

The west portion of the structure (Diesel Generator Building), which is 95 feet long, has two floors and extends from 36 feet below grade to approximately 59 feet above grade. The portion below grade houses storage tanks for diesel fuel. The area between elevations 20'-0" and 50'-0" is divided, north and south, by a 2 feet thick reinforced concrete wall which supports the second floor and provides protection for each diesel generator against missiles generated by the other. The second floor contains air intakes for the diesel generators and building ventilation equipment. The roof is supported by concrete walls all around and by steel columns in the center extending from the second floor and located directly over the dividing missile wall below.

See Figure 1.2-31, Figure 1.2-32, and Figure 1.2-33, Figure 1.2-34, Figure 1.2-35 and Figure 1.2-36 for plans and sections of the Control Building and Diesel Generator Building, respectively.

e. Control Room Makeup Air Intake Structure

The control room makeup air intake structures are reinforced concrete structures which serve as terminals for buried ductwork that provides air for the control rooms during accident conditions. The intake is located near the demineralized water tank, north of the Circulating Water Pumphouse.

The top of the sleeve for the air intake is located at Elevation 22'-7", higher than the maximum flood elevation.

See Figure 3.8-31 for a plan and section.

f. Emergency Feedwater Pump Building Including Electrical Cable Tunnels and Penetration Areas (Control Building to Containment)

The Emergency Feedwater Pump Building is a reinforced concrete building on the north end of the Enclosure Building. It extends to approximately 28 feet north of the Enclosure Building and is 84 feet wide. In elevation, the building extends from Elevation (-)26'-0" (top of floor slab) to elevation 47'-0" (top of roof slab). The building consists of the emergency feedwater pump room located above a two-story high electrical cable tray tunnel. The roof is supported on three exterior walls and on columns, on the south side. Portions of the emergency feedwater pump rooms which extend beyond the electrical cable tray tunnels are supported on walls founded on rock or on concrete fill.

The emergency feedwater pump room contains emergency feedwater pumps, demineralized water makeup pumps, valve stations and an auxiliary control panel. A monorail is provided for servicing the pumps. The electrical penetration areas are approximately 84 feet wide and are situated one on top of the other. They penetrate the Enclosure Building and join with the containment structure.

See Figure 1.2-2, Figure 1.2-3, Figure 1.2-4, and Figure 1.2-6, and Figure 1.2-51.

g. Enclosure for Condensate Storage Tank

The enclosure is a cylindrical reinforced concrete wall, 2 feet thick and 43 feet high, which surrounds each tank and extends up to the springline of the tank, providing protection from horizontal tornado-generated missiles. The enclosure is capable of retaining the contents of the tank should the tank rupture due to a missile penetrating it from above. See Figure 3.8-32.

h. Fuel Storage Building

The Fuel Storage Building is a reinforced concrete structure approximately 98 feet square that extends approximately 44 feet below grade to 66 feet above grade. The building contains the new fuel storage area and the spent fuel pool. The spent fuel pool is constructed of concrete walls of a minimum thickness of 6 feet and a 4'-10" thick concrete floor; for leaktightness the inside surface is lined with stainless steel plates 3/16" thick on the walls and 1/4" thick on the floor. The building superstructure is constructed of concrete walls and roof.

New and spent fuel storage and handling are described in Section 9.1. Refer to Figure 1.2-15, Figure 1.2-16, Figure 1.2-17, Figure 1.2-18, Figure 1.2-19, Figure 1.2-20 and Figure 1.2-21 for plan and sectional elevations of the building.

i. Main Steam and Feedwater Pipe Chase (East) Including East Penetration Area

The main steam and feedwater pipe chase (east) is a reinforced concrete structure which houses and protects the main steam and feedwater piping. The overall dimensions are 127.58 feet long, 22.25 feet wide for most of the length, and 61.5 feet high. It is 41.25 feet wide at the widest point.

The east penetration area is a room located at the southern end of the pipe chase which houses the control panels for the hydrogen recombiner. See Figure 1.2-3 and Figure 1.2-4.

j. Main Steam and Feedwater Pipe Chase (West) Including Mechanical Penetration Area and Personnel Hatch Area

The main steam and feedwater pipe chase (west) is a reinforced concrete structure which houses and protects the main steam and feedwater piping. The pipe chase is 113.75 feet long, 20.0 feet wide for most of the length, and 61.5 feet high (overall); the pipe chase area is 59 feet high. It is 23.75 feet wide at the widest point.

Located below the chase area is the mechanical penetration area which houses piping running between the Containment and the Primary Auxiliary Building. This region is partitioned into several smaller areas which include the radiation and nonradiation shield tunnels. The mechanical penetration area is 78.0 feet long, 34.71 feet wide at the widest point, and 37.5 feet high.

The personnel hatch area is an irregularly shaped reinforced concrete structure located outside the personnel hatch of the containment, for which it provides protection from missiles and illegal entry. It is connected to the west pipe chase and has approximate overall dimensions of 46.75 feet length, 37.25 feet width, and 29.25 feet height. See Figure 1.2-2, Figure 1.2-3, Figure 1.2-4, and Figure 1.2-5.

k. Piping Tunnels

The piping tunnels are underground reinforced concrete structures, rectangular in section, which are located throughout the plant. They house the piping systems found in the plant. See Figure 3.8-33.

l. Pre-Action Valve Building

The Pre-Action Valve Building is an irregularly shaped reinforced concrete building which contains the deluge valve for the Fire Protection System. It is approximately 34.83 feet long, 34.75 feet wide and 27.33 feet high in overall dimensions and is located on the east side of the Emergency Feedwater Building. See Figure 1.2-51.

m. Primary Auxiliary Building Including Residual Heat Removal (RHR) Equipment Vault

The Primary Auxiliary Building is a reinforced concrete structure. The major portion is approximately 79 feet wide by 145 feet long and extends from 13 feet and 46 feet below grade to 88 feet above grade. The equipment vault is attached to the structure and is 57 feet by 43 feet. It extends from 81 feet below grade to 5.5 feet above grade. The entire building is founded on rock or concrete fill.

The RHR equipment vault is subdivided into six compartments by continuous concrete walls as follows: two for containment spray pumps and heat exchangers, two for residual heat removal pumps and heat exchangers and two for access stairs. Plugs are provided in the reinforced concrete roof for removal of the heat exchangers.

Located on the east and north sides of the RHR equipment vault are a personnel walkway and an electrical cable tray chase located above it. This structure is supported on fill concrete over rock and connected to the RHR equipment vault at its base, Elevation 20'-8"; it extends to Elevation 53'-0". The east portion is approximately 13.0 feet wide and 41.0 feet long; the north portion is approximately 10.0 feet wide and 38.0 feet long.

That portion of the building which is 79 feet by 145 feet, the Primary Auxiliary Building, has two intermediate reinforced concrete floors which support miscellaneous auxiliary nuclear equipment, such as heat exchangers, pumps, demineralizers, filters, tanks and ventilation equipment. Reinforced concrete walls and steel columns support the intermediate floors and reinforced concrete roof slab.

Below-grade reinforced concrete pipe tunnels connect the building to the Containment and Waste Processing Building. Monorail hoists are provided to handle materials and servicing of equipment.

See Figure 1.2-9, Figure 1.2-10, Figure 1.2-11, and Figure 1.2-12 for plans and sectional elevations of the Primary Auxiliary Building and Figure 1.2-13 and Figure 1.2-14 for plans and sectional elevations of the RHR equipment vault, including the personnel walkway and electrical cable tray chase.

n. Safety-Related Electrical Duct Banks and Manholes

Safety-related electrical duct banks are reinforced concrete structures, rectangular in cross section, which enclose cables running between various seismic Category I buildings other than the containment. Cross-sectional dimensions vary from duct to duct and lengths are as required by the separation of the buildings being connected. The safety-related electrical duct banks are completely below grade and principally supported on engineered fill. They are isolated from the buildings and manholes by flexible connections at the connection points.

Direction changes of the duct banks between buildings are accomplished by intermediate reinforced concrete manholes, which are generally rectangular in cross section. Overall dimensions vary depending on the size and number of duct banks entering the manhole.

See Figure 3.8-34 and Figure 3.8-35 for the duct banks and manholes respectively.

o. Service Water Cooling Tower Including Switchgear Rooms

The Service Water Cooling Tower is a rectangular building approximately 300 feet x 54 feet in plan, extending 28 feet below grade, and projecting 57.5 feet above ground. It is located on the south side of the plant and is founded on rock. The cooling tower serves as an ultimate heat sink in the unlikely event that the cooling water tunnels are rendered inoperative. The cooling tower houses pumps, fans, water distribution system and nozzles.

The switchgear rooms are approximately 54 feet x 26 feet in plan and extend from Elevation 22.0 feet to Elevation 46.0 feet. They are located on the east and west ends of the cooling tower and are an integral part of the structure. The switchgear rooms house the switchgear, substation, and motor control center for the cooling tower. See Figure 1.2-56.

Operational characteristics of the cooling tower are described in Subsection 9.2.5.

p. Service Water Pumphouse

The Service Water Pumphouse (SWPH) is located on the plant site east of the Containment Building and is founded on rock. It is attached to and shares a foundation with the Circulating Water Pumphouse (CWPH), a non-Category I structure which is designed so that its loss or failure will not impair the Service Water Pumphouse or System.

The structure under the SWPH is a reinforced concrete basin and is approximately 91 feet wide by 74 feet long; it extends from the operating floor, 1 foot above grade, to 63 feet below grade. The Service Water Pumphouse itself is approximately 118 feet wide by 78 feet long and extends 28 feet above the operating floor to Elevation 49'-0". The walls, roof and interior columns are reinforced concrete.

A service water electrical control room on the west end of the SWPH is a reinforced concrete structure founded on rock and is integrally connected to the SWPH. It is approximately 48 feet wide by 38 feet long and extends to 19.5 feet above grade.

The Circulating Water Pumphouse is approximately 119 feet wide by 123 feet long and extends 28 feet above the operating floor to Elevation 49'-0". The building is non-Category I and consists of a steel frame covered with metal siding, from the grade to the roof level, and roofing. A non-Category I trash removal room on the north end is a steel framed structure with metal siding and is approximately 39 feet wide by 17 feet long extending to 14 feet above grade to Elevation 34'-0".

The structure under the CWPH is a reinforced concrete basin and is approximately 110 feet wide by 123 feet long; it extends from the operating floor to 63 feet below grade.

The basin under the SWPH is integrally connected to the basin under the CWPH by a common east-west wall. This common basin, the SWPH and the electrical room are designed as a unit.

See Figure 1.2-46, Figure 1.2-47,and Figure 1.2-48 for plans and sectional elevations of the Service Water Pumphouse as well as the Circulating Water Pumphouse.

Operational characteristics of the Service Water System are described in Subsection 9.2.1, and operational characteristics of the Circulating Water System are described in Subsection 10.4.5.

q. Tank Farm (Tunnels) Including Dikes and Foundation for Refueling Water Storage Tank (RWST) and Reactor Makeup Water Storage Tank (RMUWST)

The tank farm area consists of a reinforced concrete portion and structural steel framing portion. The reinforced concrete portion including the foundation, dike walls, pipe tunnels and pipe chases are structures associated with safety-related systems and are designed as Seismic Category I.

Structural steel framing portion which includes steel framing, concrete roofing and metal siding, is used to enclose the area above the tanks and to form the motor control center and switchgear room. The steel framing portion is designated nonseismic Category I and designed and constructed so that the safe shutdown earthquake (SSE) would not cause the steel framing portion to collapse upon any safety-related structure, system or component within or surrounding the tank farm area.

The tunnels provide a passageway for piping which runs between the Primary Auxiliary Building and either the Service Water Pumphouse or the Service Water Cooling Tower.

The dikes are reinforced concrete walls surrounding the tanks and extending to Elevations 42'-0" and 30'-0" for the RWST dike and the RMUWST dike, respectively. Structural steel framing is used between the tops of the dikes and the roof framing, which is also structural steel.

Overall dimensions are 152 feet in length, approximately 66 feet in width and 92 feet in height. See Figure 1.2-23, Figure 1.2-24, Figure 1.2-25 and Figure 1.2-27.

r. Waste Processing Building

The Waste Processing Building is approximately 108 feet wide by 189 feet long and extends 51 feet below grade to 91 feet above grade.

The building consists of a reinforced concrete portion and a structural steel portion with a 4" nominal concrete slab.

The reinforced concrete portion between column lines 1 to 2 and A to D, housing radioactive gaseous waste equipment and carbon delay bed, is designed as seismic Category I. To maintain integrity of this portion, the entire structure is designed as seismic Category I, except for siding, girts and roofing on steel at El. 111'-0. However, local effects of seismic loads from system supports is not accounted for, since they do not affect overall safety of the structure.

The safety-related portion of the concrete between column lines 1 to 2 and A to D is designed to withstand tornado loads. The structural main frame above El. 53'-0" is designed to withstand tornado wind loads.

The building contains liquid and radioactive gaseous waste processing and solid waste systems. A trucking facility is provided along the south wall for shipping drums and containers.

See Figure 1.2-22, Figure 1.2-23, Figure 1.2-24, Figure 1.2-25, Figure 1.2-26, Figure 1.2-27, Figure 1.2-28, Figure 1.2-29 and Figure 1.2-30 for plans and sectional elevations of the Waste Processing Building.

s. Nonseismic Category I Structures

Several nonseismic Category I structures were designed against collapse onto seismic Category I structures due to tornado wind and SSE loadings. These structures can be found on the site arrangement plan, Figure 1.2-1, and include the following:

- Turbine Building
- Nonessential Switchgear Building
- Tank Farm Area (Steel Framing Portion)
- Steam Generator Blowdown Recovery Building

The Turbine Building is described in Subsection 1.2.2, and the tank farm area (steel framing portion) is described earlier in this subsection.

The Nonessential Switchgear Building is located on the north side of the Control Building and houses and protects the electrical equipment used to provide lighting for the plant.

The Steam Generator Blowdown Recovery Building is located on the south side of the Waste Processing Building and tank farm area, and houses the Steam Generator Recovery System.

Specific design measures taken to protect the effected seismic Category I structures are given in Subsections 3.3.2 and 3.7.2 for tornado wind and SSE loadings, respectively.

t. Service Water Access Vault

The Service Water Access Vault (SWAV) is located underground on the plant site north of the Cooling Towers. It is a reinforced-concrete structure approximately 14 feet wide by 38 feet long; it extends from 15 feet to 29 feet below grade. A precast concrete manway extends from the top of the vault to approximately 2 feet below grade. The vault is founded on fill concrete and rock. The SWAV provides access to the underground safety-related A and B Train 24" service water piping to permit entry into the piping for field weld joint refurbishment and joint inspection. See Figure 3.8-37.

3.8.4.2 Applicable Codes, Standards and Specifications

Codes, standards and specifications listed in Subsection 3.8.3.2, as well as the following US NRC regulatory guides, are also applicable to other seismic Category I structures:

Regulatory Guide No.	Title
1.76 (4/74)	Design Basis Tornado for Nuclear Power Plants
1.117 (4/78)	Tornado Design Classification

3.8.4.3 Loads and Loading Combinations

The seismic Category I structures discussed herein were designed, as applicable, for all credible conditions of loadings, including normal loads, loads due to severe and extreme environmental conditions, abnormal loads and missile loads. These loads were determined in accordance with the applicable codes, including ACI 318-71 and AISC-69, and were considered in normal and unusual load combinations to assure that the structures remain within the limits prescribed in Subsection 3.8.4.5.

a. Design Loads

The definitions of the loads used in the design of the other seismic Category I structure include the following:

1. Normal Startup, Operational, and Shutdown Loads

Normal loads are those loads encountered during normal plant operation, startup and shutdown. They include the following:

(a) Dead Loads (D)

Dead loads are all permanent gravity loads including, but not limited to, concrete walls and slabs, structural framing, piping, cable and cable trays, permanent equipment and miscellaneous building items. Hydrostatic pressures of liquids are also included in this category.

(b) Live Loads (L)

Live loads are all temporary gravity loads including but not limited to normal snow loads, conventionally distributed and concentrated floor loads, and movable equipment loads, such as cranes and hoists.

Equipment operating loads and impact factors are the greater of those recommended by the manufacturer or the applicable building codes.

Unusual snow load (L_s), which is greater in magnitude than normal snow load, was also used where applicable. Lateral earth pressures due to soil backfill (H) were used where applicable. Three types of lateral earth pressure loading, active, at rest and passive, were considered, with pressures determined by acceptable theories of soil mechanics.

(c) Operational Thermal Loads (T_o)

These are the thermal effects and loads occurring during normal operating or shutdown conditions, based on the most critical transient or steady-state condition.

(d) Operational Pipe Reactions (R_o)

These are the pipe reactions occurring during normal operating or shutdown conditions, based on the most critical transient or steady-state condition.

2. Severe Environmental Loads

Severe environmental loads are those loads which could infrequently be encountered during the plant life. The following loads are included in this category:

(a) Operating Basis Earthquakes (E_o)

These are the loads generated by the operating basis earthquake, which is the earthquake of an intensity of the same level as the highest intensity that has been experienced historically at the site. Only the actual dead load and weights of fixed equipment were considered in evaluating the seismic response forces. The horizontal and vertical design response spectra for the OBE were derived by applying a factor of 0.5 to the response spectra given for the safe shutdown earthquake (SSE), which is described below. The effects of two (2) orthogonal horizontal components and one (1) vertical component of earthquake were considered and combined by the square-root-of-the-sum-of-the-squares method. Lateral dynamic soil loads, including hydrodynamic loadings, due to the OBE (H_e) were also used where applicable.

(b) Wind Load (W)

Wind loads generated by the design wind specified for the plant site were considered. See Subsection 3.3.1 for a discussion of wind velocity and applied forces.

3. Extreme Environmental Loads

Extreme environmental loads are those loads which result from postulated events which are credible, but highly improbable. The following loads are included in this category:

(a) Safe Shutdown Earthquake Loads (E_s)

These are the loads generated by the safe shutdown earthquake, which is the maximum potential earthquake that could occur in the vicinity of the plant, based on geological and historical investigations. Only the actual dead load and weights of fixed equipment were considered in evaluating the seismic response forces. The horizontal and vertical responses on the structures were developed from the response spectra given in Figure 2.5-38 and Figure 2.5-39, the development of which is described in Subsection 2.5.2.6. The effects of two (2) orthogonal earthquakes and one (1) vertical earthquake were considered and combined by the square-root-of-the-sum-of-the-squares method. Lateral dynamic soil loads, including hydrodynamic loadings, due to the SSE (H_s) were also used where applicable.

(b) Tornado Loads (W_t)

Wind loads generated by the design tornado specified for the plant site were considered along with the differential pressure loads due to rapid atmospheric pressure drop and the tornado-generated missile effects.

See Subsection 3.3.2 for a discussion of the design basis tornado and its associated wind and pressure loadings. See Section 3.5 for a discussion of tornado-generated missiles.

4. Abnormal Loads

Abnormal loads are those loads generated by postulated high energy pipe ruptures, particularly a rupture in the Reactor Coolant System resulting in a loss-of-coolant accident (LOCA).

(a) Accident Pressure (P_a)

This is the design pressure load occurring within a structure due to the DBA.

(b) Accident Temperature (T_a)

This includes the thermal effects and loads generated by the DBA including T_o.

(c) Accident Pipe Reactions (R_a)

Pipe reaction loads due to thermal conditions generated by the postulated pipe break, including R_o, were considered in the design.

(d) Pipe Break Loads (R_r)

These are local effects on the structures due to the postulated pipe break, as follows:

(1) R_{rr} = load on the structure generated by the reaction of a ruptured high energy pipe during the postulated pipe break. The time-dependent nature of the load and the ability of the structure to deform beyond yield were considered in establishing the structural capacity necessary to resist the effects of R_{rr}.

(2) R_{rj} = load on the structure generated by jet impingement of a ruptured high energy pipe during the postulated pipe break. In general, direct impingement of steam on a wall of a structure does not produce significant design loadings due to the distance between the wall and the break location. Where a break is postulated to occur close enough to a wall to produce a critical loading, a shield, or deflector, is provided, and the loading is transferred to the pipe whip restraint to which the shield is attached.

(3) R_{rm} = the load on the structure resulting from the impact of a ruptured high energy pipe during the postulated pipe break. Since all high energy lines are constrained by pipe restraints, loading of this nature is prevented.

5. Site-Related Loads

(a) Flood Loads (F)

These are the loads resulting from the design basis flood including uplift and wave runup. See Section 3.4. for further discussion.

(b) Missile Loads (M)

Internal missile loads, other than those defined as R_r, are considered including an appropriate dynamic load factor. See Section 3.5 for a detailed discussion of these missiles.

b. Load Combinations

Various load combinations were considered in design to determine the strength requirements of the structure. Where varying loads occur, the combinations producing the most critical loading were used. The basic combinations considered in the design of each seismic Category I structure are given in Table 3.8-16.

Two categories of loading conditions and criteria were used in the design of the seismic Category I structures other than the containment, as described below:

1. Normal Load Conditions

Normal load conditions are those encountered during testing and normal operation and are referred to in the standard review plan as service load conditions. They include dead load, live load and anticipated transients, loads occurring during normal startup and shutdown, and loads occurring during emergency shutdown of the nuclear steam supply, safety and auxiliary systems. Normal loading also includes the effect of an operating basis earthquake and normal wind load. Under each of these loading combinations the structures were designed so that stresses are within the elastic limits. Design and analysis procedures are presented in Subsection 3.8.4.4 and stress limitations are presented in Subsection 3.8.4.5.

2. Unusual Load Conditions

Unusual load conditions are those resulting from combinations of accident, wind, tornado, earthquake, live and dead loads and are referred to in the standard review plan as factored load conditions.

For these loading combinations, the structures were designed to remain below their ultimate yield capacity such that deformations will be small and structural components will respond elastically. Design and analysis procedures are presented in Subsection 3.8.4.4 and stress limitations are presented in Subsection 3.8.4.5.

c. Other Load Considerations

1. Creep

Effects of concrete creep are negligible due to the low sustained concrete stresses associated with conventionally reinforced concrete structures and, therefore, were not a governing factor in design.

2. Stability

The other seismic Category I structures were checked for overturning, sliding, and flotation using the load combinations of Table 3.8-16 with the exception that the coefficient for live load is zero. Buoyant forces were considered to decrease the dead load in computing both overturning and sliding.

3.8.4.4 Design and Analysis Procedures

a. Design and Analysis

Category I structures other than the containment are constructed of reinforced concrete with structural steel framing used to support vertical loading on floor slabs. Structural steel is also used to provide enclosure for some areas, as described in Subsection 3.8.4.1, and for other miscellaneous purposes. Reinforced concrete structures consist of a system of walls and slabs generally to provide a continuous, integral framing system. Vertical forces are transferred to the foundation mats through the walls and structural steel and reinforced concrete columns. Lateral forces are transferred to the foundation mats primarily by the action of shear walls; some load is also transferred by means of flexural action of the walls, all of which are rigidly attached at the mat.

The Containment Enclosure Building, due to its cylindrical and hemispherical shape and relative dimensions, was analyzed as a three-dimensional, thin-shell structure. Boundary conditions were consistent with the support on rock and the lateral restraint provided by backfill concrete placed against the structure. Internal resultant forces and moments were determined by integration of the appropriate shell stresses through the thickness. Critical transverse shear force was derived by considering the variations in bending moments across the surface, in conjunction with the applied hydrostatic load (which produces additional local shear not reflected in the finite element analysis). Reinforcing was subsequently designed for these internal forces.

Columns are designed to resist other lateral loads, such as pipe loads or building displacements, in addition to those forces transmitted to the columns at floor levels. Steel columns are generally pin-connected at the foundation mat, and reinforced concrete columns are rigidly attached. Structural steel framing for floor systems primarily consists of pin-connected framing with some members being continuous. Rigidity is provided by the box-like concrete walls and slabs.

Several computer programs were used for static analysis and are described in Appendix 3F.

Table 3.8-17 contains a list of these programs and the respective structures for which they were used. The load combinations are given in Subsection 3.8.4.3.

The idealization of each structure depended on its geometric configuration and applied loading. Both two and three-dimensional analyses were performed. When the loads are considered to act principally in a given plane the floor diaphragms are rigid in their own planes; and when the rotations between adjacent frames are uncoupled, two-dimensional analyses were used. Three-dimensional analyses were used when other analyses were not suitable.

Detailed analyses of local areas of these structures, which either are subjected to local loads such as thermal hydrostatic and hydrodynamic loads, or have unusual geometry, such as numerous openings, were performed where required. Such areas included the north wall of the main steam and feedwater pipe chase (east) and the dividing wall in the Control and Diesel Generator Building. In these analyses, a representative area of a structure was isolated, and force and/or displacement boundary conditions, consistent with the overall behavior of the structure, were imposed on the substructure in addition to the appropriate loading.

Structural systems of the buildings were designed for dead load, live load, and lateral loads, such as those loads produced by wind, tornado, and earthquake. All structural elements were designed to resist the effects of internally generated missiles, where applicable. Tornado loads consist of applied pressure and missile impact, for which all seismic Category I structures are designed except as indicated below.

The safety-related electrical duct banks and manholes, the mechanical penetration area and the pipe tunnel were not designed for any tornado loads. The duct banks and manholes are covered by a nominal 5 feet of backfill, minimum, which protects them from tornado loads. The mechanical penetration area is located beneath the west pipe chase, and the pipe tunnel is covered by approximately 15 feet of backfill. The manhole covers, however, were designed for tornado-generated missiles.

The Service Water Cooling Tower and the Containment Enclosure Building were designed for tornado pressure but not for tornado-generated missiles. Since the circulating water tunnels will be in operation during a tornado instead of the cooling tower and since the containment was designed for tornado-generated missiles, these structures need not be designed for missiles.

The Pre-Action Valve Building is designed for tornado-generated missiles and tornado depressurization loading (Elevation 27'-0" slab). The elevation 27'-0" slab is a tornado depressurization barrier for safety-related equipment and components located in the electrical tunnels.

In the Waste Processing Building only those areas housing radioactive gaseous waste equipment were designed for tornado loads; the other areas do not contain safety-related equipment. Procedures by which the structures were checked for missile loads, tornado-generated as well as internally generated, are described in Section 3.5. Determination of pressures on structures due to tornado is described in Subsection 3.3.2. Pressure loadings from wind are described in Subsection 3.3.1.

The seismic analysis of seismic Category I structures is described in Subsection 3.7.2. All cranes in these structures are furnished with hold-down devices to ensure that they are not dislodged by earthquake forces. Monorails, by nature of their support mechanisms, cannot be dislodged by earthquake forces.

Using methods outlined in TID-7024, "Nuclear Reactors and Earthquakes," the effects of hydrodynamic forces were included in the seismic analyses of the Service Water Cooling Tower. Also using methods outlined in TID-7024, the weight of constrained water and sloshing effects of water in motion were included as equivalent static loads in the final design of the Service Water Cooling Tower, Service Water Pumphouse and Fuel Storage Building.

Reinforced concrete design of Category I structures was in accordance with the strength design procedures of the ACI 318-71 code, except as indicated in Subsection 3.8.4.5. Structural steel design was in accordance with the provisions of the AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings (1969 Edition). Refer to Subsection 1.8 (Regulatory Guide 1.142) for a statement concerning compliance with ACI-349.

b. Material Properties

Material properties were selected from the normal range of values to produce a conservative design. See Subsection 3.8.1.4 for a detailed discussion of the influence of material properties on design and analysis.

c. Computer Programs

The computer programs used in analysis and design of other Category I structures are listed in Table 3.8-17. These programs are further described in Appendix 3F.

3.8.4.5 Structural Acceptance Criteria

The basis for the acceptance criteria is the ACI 318 and AISC Codes. However, under the action of seismic or wind loadings, in accordance with the standard review plan (Section II.5), the 33 percent increase in allowable stresses was not permitted.

a. Normal Load Conditions

Structures were proportioned to remain within the elastic limits under all normal loading conditions described in Subsection 3.8.4.3. Reinforced concrete structures were designed in accordance with ACI-318 Strength Method, which insures flexural ductility by limiting reinforcing steel percentages and stresses.

Structural and miscellaneous steels were designed in accordance with AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings, Part 1.

b. Unusual Load Conditions

Structures were proportioned to maintain elastic behavior under all unusual load conditions shown in Subsection 3.8.4.3. The upper bound of elastic behavior was taken as the yield strength capacity of the load carrying components. The yield strength of structural and reinforcing steel was taken as the minimum guaranteed yield stress as given in the appropriate ASTM Specifications. Reinforced concrete structures were designed in accordance with ACI-318 Building Code. Member yield strength was considered to be the strength capacity calculated by the ACI Code.

Structural and miscellaneous steels were designed in accordance with Part 1 of AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings.

c. Deformations

Since each of the structures was designed to be in the small deformation, elastic range, no gross deformations will occur that will cause significant contact with other structures or pieces of equipment. All deformations, however, were evaluated considering the relationship of the subject component to both adjacent and supporting structures and equipment.

d. Additional Stress-Strain Considerations

Stress-strain limits and design parameters were applied to the design of each applicable element for the specific values identified with each loading combination and design condition identified in Subsection 3.8.4.3.

No special allowance has been made for variation of material properties over the life of the structure, beyond that which is taken into account in establishing allowable stresses, strains, capacity reduction factors, concrete protection of reinforcing, and crack control as outlined in the referenced ACI and AISC codes. Additional corrosion protection is provided to concrete structures by means of waterproofing for parts of the structure below grade and by painting, coating or installing of liners for structural concrete tanks (such as the spent fuel pool in the Fuel Storage Building).

Variations in stress and strain, due to scheduled plant shutdowns and startups, have negligible effect on the overall structural behavior because of the small variation in the average structure temperature and loading. Since the designs of the structures were governed by extreme, infrequently occurring loadings, such as tornadoes and earthquakes, and normal cyclical changes in stress levels are comparatively small, no reduction in the margin of safety will occur over the life of the plant.

All connections and joints were designed to transfer all design forces (shear, tension and compression) and moments with a safety margin and degree of conservatism that is required by the applicable code.

e. Stability

Acceptance criteria for stability are given in Subsection 3.8.5.5.

3.8.4.6 Materials, Quality Control and Special Construction Techniques

The primary materials of construction are concrete, reinforcing steel and structural steel (rolled shapes and plates).

Descriptions of the materials and basic quality control procedures are discussed in Subsection 3.8.3.6.

3.8.4.7 Testing and In-Service Surveillance Requirements

Normal quality control testing is discussed in Subsection 3.8.3.6. A general visual inspection of the exposed accessible interior and exterior surfaces of the Containment Enclosure Building will be periodically conducted as discussed in Subsection 6.2.6.1.

3.8.5 Foundations

The following sections discuss the physical descriptions of the foundations, applicable codes, standards and specifications, loads and load combinations, design and analysis procedures, structural acceptance criteria, materials, quality control and special construction techniques, and testing and in-service inspection requirements for the foundations of seismic Category I structures.

3.8.5.1 Descriptions of the Foundations

The locations and relationships of the various seismic Category I foundations are shown on the plot plan, Figure 1.2-1. Details of individual foundations, including type and dimensions, are given in Table 3.8-15. This table also contains a list of figures showing plans and profiles of the foundations.

Foundations for seismic Category I buildings are conventionally reinforced concrete mats of varying thicknesses supported on sound bedrock or on fill concrete extending to sound bedrock. The walls of the Containment Enclosure Building extend to a spread footing, 10'-3" wide by 10' deep, which carries the load from the walls to sound rock. This footing is not continuous, having openings for the pipe chases and electrical tunnels below the Emergency Feedwater Pumphouse.

The bottoms of this footing and most mats are embedded in the rock in order to transfer horizontal shear forces. If the bottom of a foundation does not extend to sound rock, fill concrete was placed from the sound bedrock to the elevation of the underside of the structure.

The only exceptions to the above criteria are five safety-related electrical manholes, certain sections of service water pipes, and most of the 8350 feet of safety-related electrical duct banks which are supported on engineered backfill consisting of offsite borrow or tunnel cuttings (see Subsections 2.5.4.5c and 3.7(B).1.4). Compaction requirements for this engineered backfill ensure firm support which, along with flexible couplings between the manholes and ducts, prevent any seismic response in one of these components from inducing a response in an adjacent component or structure.

Since all other seismic Category I structures are founded on sound rock, or on fill concrete over sound rock, they are inherently isolated from each other. There is no connecting soil medium which could foster seismic coupling.

Similarly, the foundations of nonseismic Category I structures, which are also founded on rock, do not have seismic coupling with the foundations of seismic Category I structures, except for the Circulating Water Pumphouse. This is a non-Category I structure which is attached to the Service Water Pumphouse, shares a common foundation it, and it is designed as a unit. Seismic isolation of the buildings above the point of fixity is ensured by means of isolation joints; these joints are provided between adjacent buildings above the point of fixity.

The point of fixity is the elevation on the building to which rock extends or fill concrete was placed. There is no correlation between this point and the natural rock elevation; it may be lower than natural rock or higher if fill concrete was used to increase the embedment. Within the region of a building that is fixed, the walls and foundation are rigid and accelerate along with the rock.

There are no unique features used for foundations.

The load transfer for the foundations is as follows:

a. Load is applied to the wall of a seismic Category I structure.

b. The wall transfers the load to the foundation (mat or footing).

c. The foundation transfers the load to the rock, fill concrete, or engineered fill.

A more detailed description of the mechanism of load transfer is given in Subsection 3.8.5.4.

For a detail of the typical reinforcing pattern at the junction of a reinforced concrete wall and its foundation, see Figure 3.8-36.

Piles are not used under seismic Category I structures. The structures transmit load directly to rock by contact or through fill concrete or engineered fill.

3.8.5.2 Applicable Codes, Standards and Specifications

For applicable codes, standards and specifications, see Subsections 3.8.1.2 and 3.8.4.2 for the containment structure foundation and for other seismic Category I structure foundations, respectively.

3.8.5.3 Loads and Load Combinations

For the containment structure foundation, see Subsection 3.8.1.3 for loads, load combinations, load factors and the design approach used with the load combinations and load factors.

For other seismic Category I structure foundations, see Subsection 3.8.4.3 for loads, load combinations, load factors and the design approach used with the load combinations and load factors.

Foundations and structures are checked for sliding and overturning due to earthquakes, winds, and tornadoes and for flotation due to floods and high water table using load combinations described in Subsections 3.8.1.3 and 3.8.4.3 for the containment structure and the other seismic Category I structures, respectively.

Lateral earth pressures are considered as applicable. Nonrigid walls are designed for active earth pressure under static conditions and for an equivalent static earth pressure based on a coefficient of dynamic earth pressure that dampens with depth under seismic conditions. Rigid walls are designed for earth pressure at rest under static conditions and for an equivalent static earth pressure based on a constant coefficient of dynamic earth pressure under seismic conditions. Lateral pressure diagrams for these conditions are shown in Figure 2.5-52 and Figure 2.5-53 and discussed in Subsection 2.5.4.11.

3.8.5.4 Design d Analysis Procedures

The foundations of seismic Category I structures are analyzed and designed in accordance with Subsections 5.2 and 3.8.5.5 to determine maximum stresses in reinforcing and concrete, using the load combinations discussed in Subsection 3.8.5.3.

a. Boundary Conditions and Expected Behavior

Most foundations of seismic Category I structures are founded directly on sound rock or on fill concrete; five safety-related electrical manholes and most of the safety-related electrical duct banks are supported on engineered fill. The entire length of the duct banks, however, is designed with the assumption of support on engineered fill.

Design and analysis, including idealization and boundary conditions, for the circular base mat of the containment structure are described in Subsection 3.8.1.4. The base mat is designed to sustain all credible loads resulting from the containment and internal structures.

Design procedures for all structures insured that foundation mats and footings were sized to limit bearing pressure on the rock to 60 tons/square foot on horizontal surfaces and 10 tons/square foot in weathered zone and 60 tons/square foot below weathered zone on vertical surfaces. These bearing pressures were established on the basis of results of tests of unconfined compressive strength with a factor of safety on the order of 10. Foundations for those structures supported on engineered fill are designed to limit bearing pressures to the allowable limits given in Subsection 2.5.4.5.

Each building has an individual foundation; no common foundations are employed except for the Service and Circulating Water Pumphouse. In a few additional cases, as noted in Subsection 3.8.4.1, a structure is partitioned into a dual-function structure which is designed as a unit. Also, adjacent foundations such as those for the containment structure and the Enclosure Building, even though they are not continuous, may transfer horizontal forces through bearing.

All loads are transferred to the foundation through the walls, or in the cases of some equipment, through the supporting pedestals. Reinforcing continues from the wall into the foundation mat and is embedded in the mat and wall by hooks, bends or straight extensions of sufficient length to develop the design strength of the bars. Thus, forces are adequately transferred from the wall to the foundation.

Soil-structure interaction is not applicable for rock-supported seismic Category I structures. Thus, for these structures, the foundation mats are designed as flat plates supported on rigid non-yielding foundation media.

In the case of electrical duct banks in engineered fill, the ability of the elastic fill to impose a seismically induced strain on the duct bank was considered. This is further described in Subsection 3.7(B).2.

For safety-related electrical manholes which are supported on engineered fill, amplification of the ground motion is considered. The slab, however, is essentially rigid, and bearing pressures are very low. (See Subsection 3.7(B).2.)

Behavior of the foundation mats when subjected to the load combinations given in Subsection 3.8.5.3 is within the acceptance criteria of Subsection 3.8.5.5.

b. Vertical Loads, Lateral Loads and Overturning Moments

Vertical loads are carried by direct bearing on the rock. Overturning moments and bearing pressures on the rock (including maximum toe pressure) were investigated in accordance with the safety factors discussed in Subsections 3.8.1.3 and 3.8.4.3 for the containment foundations and other seismic Category I structure foundations, respectively.

Base shears and torsional moments are transferred to rock by one or both of the following:

1. Embedment in rock which provides resistance to shearing loads through direct bearing

2. Friction between the bottom of the foundation and the top of the rock (the coefficient of friction takes into consideration the reduced shear resistance due to the presence of the waterproofing membrane).

In the containment structure, the walls of the reactor cavity bear against the rock through fill concrete to provide additional resistance for transfer of base shears. The walls of the cavity are designed to carry these loads.

Conventional hand calculations were employed for the design of the foundations and were augmented, for some of the larger foundations, by finite element techniques to determine uplift patterns.

This uplift was determined by static methods using forces which were determined in the dynamic seismic analysis. The mat was modeled by a series of flat plate bending elements which include stiffening effects of the walls or shell. An uplift pattern was assumed, loads applied, and final displacements checked. When the calculated displacements were in agreement with the assumed uplift pattern, the analysis for that load combination was then completed by calculating shears and bending moments.

Methods of determining the overturning moments which were used in the uplift analysis include all three components of the earthquake and are discussed in Subsection 3.7(B).2.

Design factors of safety against sliding, overturning and buoyancy are defined in Subsections 3.8.1.3 and 3.8.4.3 for the containment structure and other seismic Category I structures, respectively.

c. Computer Programs

See Appendix 3F for a description of the programs. See Table 3.8-17 for a list of structures and the computer programs which were used for them.

3.8.5.5 Structural Acceptance Criteria

The acceptance criteria relating to stress, strain, gross deformation and shear loads are described in Subsections 3.8.1.5 and 3.8.4.5 for the containment and other seismic Category I structure foundations, respectively.

Safety factors for buoyancy, sliding, and overturning are as follows:

	Factor of Safety		
Load	Overturning	Sliding	Flotation
Service/Normal load combinations	1.5	1.5	--
Factored/Unusual load combinations	1.1	1.1	--
Dead load and design basis flood load	--	--	1.5

3.8.5.6 Materials, Quality Control and Special Construction Techniques

The primary materials of construction are concrete and reinforcing steel. Their descriptions and basic quality control procedures are discussed in Subsections 3.8.1.6 and 3.8.4.6 for the containment foundation and other seismic Category I structure foundations, respectively. Engineered fill, fill concrete and backfill concrete are described in Subsection 2.5.4.5.

There are no special construction techniques.

3.8.5.7 Testing and In-Service Surveillance Requirements

The ability of the containment foundation to resist 1.15 times the design pressure is demonstrated during the structural integrity test as described in Subsection 3.8.1.7.

For other seismic Category I structure foundations, no preoperational or in-service surveillance is required.

Structures which are founded on sound rock or on fill concrete over sound rock do not have any potential areas of settlement or displacement which should be monitored. Similarly, gradation requirements, compaction criteria and compaction tests for engineered fill ensure a foundation material which will support the design loads with negligible settlement. Piles were not used. For these reasons there are no potential settlements or displacements which should be monitored for any foundation.

3.8.6 References

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3. Duchon, N.B., "Analysis of Reinforced Concrete Membrane Subject to Tension and Shear," ACI Journal, September, 1972, pp. 578-583.

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5. Alexandria, S. C., Effects of Irradiation of Concrete, Final Results, Research Reactor Division UKAEA, Harwell, AERE-R-4490, December 1963.

7.5 SAFETY-RELATED DISPLAY INSTRUMENTATION

7.5.1 Introduction

Display instrumentation is provided in the main control room to enable the operator to monitor plant status under all operating conditions and to take any necessary manual actions. This display instrumentation consists of analog and digital indicators, recorders, status lamps, indicating lights, Video Alarm System (VAS) alarms, video displays and annunciators. Display instrumentation is also provided in the Technical Support Center (TSC) and the Emergency Operations Facility (EOF) to support the functions to be performed by the personnel in the TSC/EOF.

7.5.2 Definitions

a. Design Basis Accident Events

Those events postulated in the plant safety analyses, any one of which may occur during the lifetime of the plant, and those events not expected to occur, but postulated in the plant safety analyses because their consequences would include the potential for release of significant amounts of radioactive material to the environs. These events are listed in Updated FSAR Chapter 15 as Conditions III and IV occurrences. Excluded are those events (defined as "normal" and "anticipated operational occurrences" in 10 CFR 50) expected to occur more frequently than once during the lifetime of the plant.

b. Task Analysis

Process of identifying and examining tasks that must be performed by the control room operating crew when interacting with the plant systems.

7.5.3 Discussion

An Accident Monitoring Instrumentation (AMI) list, Table 7.5-1, has been developed to define the instrumentation required by the operator for design basis accident events. The AMI enables the operator to monitor safety functions, take any manual actions required to support the accomplishment of safety functions and determine the effect of manual actions during and following a design basis accident event. The AMI also enables the operator to maintain the plant in a hot shutdown condition, or to proceed to cold shutdown. Details are provided in Subsection 7.5.4.

Table 7.5-2 lists additional information available to the operator for monitoring conditions in the reactor, the Reactor Coolant System, the containment and key process systems throughout all normal operating conditions of the plant, including anticipated operational occurrences.

Status lamp arrays are used to indicate both a demand for a protective function/ESF actuation and the appropriate valve position and equipment status for ESF actuations. These arrays are functionally arranged on the control board to enable the operator to quickly and accurately monitor system status. Status lamp arrays are provided to monitor bistable trips for the following safety functions:

- Reactor Trip
- Safety Injection
- Containment Isolation
- Steam Line Isolation
- Feedwater Line Isolation

To monitor valve position, actuated equipment status and emergency power availability, status lamp arrays are provided for the following:

- Cold Leg Injection
- Cold Leg Recirculation
- Hot Leg Recirculation
- Containment Isolation, Phase A
- Containment Isolation, Phase B
- Main Steam and Feedwater Isolation
- Cooling Tower Actuation
- Diesel Generator Status
- Emergency Power Sequencer

A computer-based Video Alarm System (VAS) is provided to alert the operator when various process limits are exceeded. The incoming alarms are prioritized to allow the operators to focus on high priority alarms during major plant upsets. Three levels of priority have been established. Incoming alarms are also broken down into primary and secondary sides; primary side alarms are displayed on the alarm displays in main control board Sections A and D, while secondary side alarms are displayed on the alarm displays in Sections F and I.

An alarm suppression scheme is normally active on the VAS computer software to minimize the number of alarms that the operators must respond to during routine and transient plant operation. Two types of alarm suppression are provided: one based on mode of plant operation and the other for selected plant transient events. Only alarms that are expected and do not represent an abnormal plant condition for a mode or event are suppressed. If required, the capability for operators to deactivate mode and/or event alarm suppression is available through the main control board VAS workstations.

An alarm horn silence switch has been provided at the Unit Supervisor's desk in the Control Room. This switch allows the Unit Supervisor to disable the main control board alarm audibles during periods of high alarm activity. When activated, this switch will reduce Control Room noise levels and the repetitive operator actions required to silence alarms. An amber lamp mounted next to the switch indicates when the alarm horn audibles have been silenced.

Various dynamic displays are provided to serve the needs of the operating crew. These displays supplement those described above. Video displays are provided in the Technical Support Center (TSC) and Emergency Operations Facility (EOF) to support the functions to be performed by the personnel in the TSC/EOF.

The computer system consists of two host computers, each of which is fed from a separate uninterruptible power supply. An automatic failover scheme is provided. The remainder of the system is configured so that system peripherals can be manually aligned to the available UPS.

Annunciators back up the VAS should a complete Computer System failure occur. The annunciators also have a limited "First Out" capability to assist the operator in determining the cause of a reactor trip or safety injection. A limited set of essential parameters is monitored for the following:

- Reactor Trip Signals
- ESF Actuation Signals
- Certain Technical Specification Deviations
- Important Systems

The annunciators are powered from instrumentation power sources that are independent of the power sources for the VAS.

Bypassed/inoperable condition of safety systems is displayed on the VAS and on status lamp arrays on the MCB - one per train. Refer to Subsection 7.1.2.6 for a complete discussion of compliance with Regulatory Guide 1.47.

7.5.4 Accident Monitoring Instrumentation

7.5.4.1 Compliance with Regulatory Guide 1.97

Regulatory Guide 1.97, Revision 3 endorses, subject to certain clarifications, ANSI/ANS 4.5-1980, "Criteria for Accident Monitoring Functions in Light-Water-Cooled Nuclear Reactors." The guidance provided in Regulatory Guide 1.97 and ANS 4.5, with certain exceptions, and NUREG-0737 has been used in selecting the Seabrook Accident Monitoring Instrumentation (AMI).

The exceptions to the guidance provided in Regulatory Guide 1.97 and ANSI/ANS 4.5 are:

a. Not all the variables recommended by Regulatory Guide 1.97, Table 3 have been included in the AMI List. Specific deviations and the associated justifications are provided in Appendix 7A.

b. Not all the AMI characteristics recommended by Regulatory Guide 1.97, Table 3 have been met. Specific deviations and the associated justifications are provided in Appendix 7A.

c. The determination of performance requirements for AMI did not follow the guidance of Regulatory Guide 1.97, Section C.2.4 in that:

1. Required accuracy of measurement was not determined in procuring the instrumentation. Instead, the accuracy of the as-procured instrumentation was determined and then reviewed for acceptability. Further details are provided in Subsection 7.5.4.4e.5.

2. Except for meteorological monitoring instrumentation, response characteristics (time) have not been determined for instrumentation channels that provide monitoring functions only. The response time for these channels is similar to the response time determined for ESF actuation channels since similar hardware is used. Therefore, determination of the response time for each channel is not necessary. See Subsection 2.3.3.3a for a description of the meteorological monitoring system.

7.5.4.2 Description of Variable Types

The accident monitoring variables are classified into five types (A, B, C, D or E) according to the monitoring function they perform. A definition of each type is provided in the following subsections.

a. Type A Variables

Type A variables for Seabrook Station are those variables to be monitored that provide the primary information for the control room operators to take specific preplanned manual actions for which no automatic control is provided. These actions are required for safety systems to accomplish their safety function for design basis accident events. Actions taken as a result of equipment failures, e.g., the "Response Not Obtained" column in the Emergency Response Procedures (ERPs), are excluded.

b. Type B Variables

Type B variables provide the most direct indication to monitor the accomplishment of the critical safety functions (CSFs). CSFs are those safety functions that are essential to prevent a direct and immediate threat to the health and safety of the public. The accomplishment of these functions ensures the integrity of the physical barriers against radiation releases.

The six CSFs for Seabrook are:

1. Subcriticality
2. Containment Integrity (including radioactive effluent control)
3. Heat Sink
4. Core Cooling
5. RCS Integrity
6. RCS Inventory

c. Type C Variables

Type C variables provide the most direct indication of the potential for or the actual breach of the barriers to fission product releases. These barriers are: fuel cladding, primary coolant pressure boundary, and Containment.

d. Type D Variables

Type D variables are those variables that provide information to indicate the operation of individual safety systems and nonsafety systems used in the mitigation of design basis accidents.

e. Type E Variables

Type E variables are those variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and continually assessing such releases.

7.5.4.3 Development of Accident Monitoring Instrument List

As part of the Detailed Control Room Design Review (DCRDR), a task analysis was performed on the ERPs to identify the needed instrumentation and controls to support the execution of these procedures. For each instrument needed, a determination of the variable type was made based on its use in the ERPs and the definitions for each variable type. The task analysis included the ERP contingency guidelines (ECAs); instrumentation used only to support the execution of the ECAs is not considered AMI.

For each variable, a determination is made whether it is a key variable or backup variable in accordance with the following criteria.

Key variables are those variables that provide the primary information required to permit the control room operating crew to:

a. Perform the diagnosis specified in the ERPs for design basis accidents

b. Take any manual action required to mitigate the consequences of an accident

c. Monitor the operation of safety systems.

Primary information is information that is essential for the direct accomplishment of the specified safety functions.

Backup variables are those variables that also provide information in addition to the key variables to assist the control room operating staff in:

a. Performing the diagnosis specified in the emergency operating procedures for design basis accidents

b. Taking any manual actions required to mitigate the consequences of an accident

c. Monitoring the status of individual components and ESF demand signals

d. Resolving instrument ambiguity.

Variables are then assigned a design category using the following matrix:

	Design Category	
Variable Type	Key Variables	Backup Variables
A	1	3
B	1	3
C	1	3
D	2 for safety systems	3
	3 for nonsafety systems	3

Variable Type	Design Category Key Variables	Backup Variables
E	3	3

The AMI list contains the instrumentation classified as Design Category 1 and 2, the instrumentation identified to monitor the performance of safety systems (Type D, Design Category 2) and Design Category 3 instrumentation included in Regulatory Guide 1.97, Table 3, Revision 3.

7.5.4.4 Design and Qualification Criteria

a. Discussion

The AMI are assigned design categories as discussed in Subsection 7.5.4.3. The design and qualifications criteria for each design category are provided in the following subsections.

b. Design Category 1 - Design and Qualification Criteria

1. Equipment Qualification

Design Category 1 instrumentation is environmentally qualified in accordance with IEEE 323-1974 and associated daughter standards. This instrumentation is seismically qualified in accordance with IEEE 344-1975. Further details on the methods used and compliance with associated regulations and Regulatory Guides are provided in Sections 3.10 and 3.11 of the Updated FSAR.

2. Redundancy

No single failure within the AMI, its auxiliary supporting features, or its power sources concurrent with the failures that are a condition or result of a specific accident, will prevent the operators from being presented the information necessary to determine the safety status of the plant and to bring the plant to and maintain it in a safe condition following the accident. The electrical independence and physical separation of redundant channels is discussed in Sections 8.3 and 7.1.

Where failure of one accident monitoring channel results in information ambiguity (i.e., the redundant displays disagree) that could lead operators to defeat or fail to accomplish a required safety function, backup information is provided to allow the operators to deduce the actual conditions in the plant. This is accomplished by providing additional independent channels of information of the same variable (an identical channel) or by providing an independent channel to monitor a different variable that bears a known relationship to the multiple channels (a diverse channel). Information on redundant/diverse channel availability is included in the operator training program.

For systems having redundant components, single channel monitoring of the redundant parts of the system is provided. Verifying the proper functioning of one the redundant parts of the system is sufficient to monitor the accomplishment of the safety function.

3. Power Source

Design Category 1 instrumentation is powered from safety-related power sources. Where momentary power interruption is not tolerable, uninterruptible power sources are used.

4. Availability

The Design Category 1 instrumentation channels will be available prior to an accident except for testing and maintenance as provided in Paragraph 4.11 of IEEE Standard 279-1971 or as specified in the Technical Specifications.

5. Quality Assurance

Quality Assurance for Design Category 1 instrumentation is provided in accordance with the QA Program described in Chapter 17 of the Updated FSAR. Conformance to appropriate regulatory guides is discussed both in Chapter 17 and Section 1.8 of the Updated FSAR.

6. Display and Recording

Indication: For design Category 1 variables, continuous, redundant indication is provided. This indication meets the applicable requirements for design Category I instrumentation.

Recording: Recording of instrumentation readout information is provided for at least one of the redundant channels.

Trend Indication: Where direct and immediate trend or transient information is essential for operator information or action, this information is available from multiple displays such as:

- Dedicated recorders, or
- Dedicated ratemeters, or
- Video display (via the plant computer) available on demand, or
- Plasma displays - available on demand by use of dedicated function push buttons.

For trend display channels, at least one of the display devices meets the applicable requirements for design Category 1 instrumentation.

7. Identification

Type A, B, C, & D instrumentation displays provided for operator use during accident conditions are identified by an orange nameplate containing black lettering.

8. Interfaces

The transmission of signals to the accident monitoring equipment from protection equipment is through isolation devices, which are classified as part of the protection system.

No credible failure at the output of an isolation device will prevent the associated protection channel from meeting the minimum performance requirements considered in the design bases. Examples of credible failures include short circuits, open circuits, grounds, and the application of the maximum credible AC or DC potential (140V DC or 129V AC). Refer to Updated FSAR Subsection 7.2.2.2c.7 for further discussion.

c. Design Category 2 - Design and Qualification Criteria

1. Equipment Qualification

Design Category 2 instrumentation is environmentally qualified in accordance with IEEE 323-1974 and associated daughter standards. Further details on the methods used and compliance with associated regulations and Regulatory Guides are provided in Section 3.10.

2. Power Source

Design Category 2 instrumentation is powered from highly reliable power sources, very often Class 1E. Where momentary power interruption is not tolerable, uninterruptible power sources are used.

3. Quality Assurance

Quality Assurance for Design Category 2 instrumentation is provided by United Engineers and Constructors for the design, procurement and installation phases. Their QA Program contains the measures necessary to insure that the instrumentation has been properly specified, procured and installed. This program contains the applicable elements of 10 CFR 50, Appendix B.

Quality Assurance for the testing phase is provided by the standard testing procedures of the NHY Startup and Test Department. Auditable records are available for each Design Category 2 instrument.

Quality Assurance during the operational phase is provided under the FPLE Seabrook Operational Quality Assurance Program (OQAP). Further details are provided in Section 17.2.

4. Display and Recording

Indication

For Design Category 2 instruments, either display on demand or continuous indication is provided.

Recording

Effluent radioactivity and area radiation variables are recorded.

Trend Indication

Where direct and immediate trend or transient information is essential for operator information or action, trend indication is provided. This indication consists of either dedicated recorders or video displays.

5. Identification

Types A, B and C instrumentation displays provided for operator use during accident conditions are identified by an orange nameplate containing black lettering.

6. Channel Availability

Design Category 2 instrumentation channels will be available prior to an accident as provided in the plant administrative procedures.

7. Interfaces

Same as Design Category 1.

d. Design Category 3 - Design and Qualification Criteria

1. Quality Assurance

This instrumentation is of high-quality commercial grade and is selected to withstand the expected plant service environment.

2. Display and Recording

Indication

The information display can be either continuous or available on demand.

Recording

Effluent radioactivity variables and meteorological variables are recorded.

Trend Indication

Where direct and immediate trend or transient information is essential for operation information or action, trend information is provided. Trend information may be from a dedicated recorder or available on demand from the plant computer system.

e. Design and Qualification Criteria Applicable to Design Categories 1, 2, and 3

1. Range

The range of the read-outs extends over the maximum expected range of the variable being measured. Where two or more instruments are needed to cover a particular range, overlapping of the instrument spans is provided.

2. Servicing, Testing, and Calibration

Means are provided for checking, with a high degree of confidence, the operational availability of each sensor during reactor operation.

This may be accomplished in various ways, for example:

- By perturbing the monitoring variable; or
- By introducing and varying, as appropriate, a substitute input to the sensor of the same nature as the measured variable; or
- By cross-checking between channels that bear a known relationship to each other and that have read-outs available.

The AMI is designed to permit any channel to be maintained when required during power operation.

3. Human Factors

The AMI is designed to facilitate the recognition, location, replacement, repair, or adjustment of malfunctioning components or modules. The AMI is designed to minimize the development of conditions that cause meters, annunciators, recorders, etc., to give anomalous indications potentially confusing to the operator.

The displays are functionally arranged on the control board to provide the operator with ready understanding and interpretation of plant conditions. Comparisons between duplicate information channels or between functionally related channels will enable the operator to readily identify a malfunction in a particular channel.

In accordance with the guidance provided in NUREG-0737, an integrated effort for both the Detailed Control Room Design Review (DCRDR) and the AMI review was undertaken. The results of this effort identified the instrumentation needed by the operating crew during the course of an accident or in the recovery phase. The DCRDR reviewed the adequacy of these instrumentation displays for use by the operating crews against human factors criteria. The AMI review determined the adequacy of instrumentation channels against the design criteria stated in this subsection. Changes made after the completion of the DCRDR will be subjected to human factors review.

4. Direct Measurement

To the extent practicable, monitoring instrumentation inputs are from sensors that directly measure the desired variables. Indirect measurements are generally used to provide backup information only.

5. Instrument Accuracy

The plant-specific background documents prepared for the ERPs verify and document that the installed AMI has sufficient accuracy to support the ERPs.

The accuracy of the AMI is addressed as part of the Operator Training Program.

8.3 ONSITE POWER SYSTEMS

8.3.1 AC Power Systems

8.3.1.1 Description

The onsite AC power systems include the 13,800V Distribution System, including the connections from the unit auxiliary transformers (UAT) and reserve auxiliary transformers (RAT); the 4160V Distribution System, including the standby diesel generators, the Supplemental Emergency Power System and connections from the UATs and RATs; the 480V Distribution System; and the 120V Vital Instrumentation and Control Power System. The onsite AC power system one-line diagram is shown in Figure 8.3-1. The 120V Vital Instrumentation and Control Power System is shown in Figure 8.3-2 and Figure 8.3-4.

Qualification of the Class 1E electrical equipment used in the onsite AC power system is discussed in Section 3.11.

Post accident monitoring of the onsite AC power system is discussed in Section 7.5.

a. 13,800V Distribution System

The 13,800V Distribution System is nonsafety-related, and is shown in Figure 8.3-1, Figure 8.3-5 and Figure 8.3-6. Each of the two 13.8-kV buses is normally fed from a UAT. A reserve source is available to each bus through a RAT.

1. Automatic and Manual Transfers

The automatic and manual transfers of power from one source to the other are as follows:

(a) An automatic transfer from the main generator supply to the offsite source through the generator step-up transformers (GSUs) is accomplished by the opening of the main generator breaker; no bus breaker action is required.

(b) A manually initiated live bus transfer is provided for transferring the 13.8-kV buses from the offsite source through the GSU transformers to the main generator source and vice versa, by the operation of the main generator breaker. The transfer from offsite to main generator source is contingent upon the two sources being in synchronism.

(c) The opening of the UAT incoming line breaker, either manually or automatically, initiates an automatic transfer from UAT to RAT source, provided that the RAT is energized, no fault exists on the bus, and the bus voltage is in synchronism with the RAT voltage at the time of transfer initiation, or has decayed to an acceptable value.

A high-speed static synchronism check relay will allow the RAT incoming line breaker to close, if both the bus voltage and phase angle between bus and RAT source are within limits. If the transfer is not completed 20 cycles after initiation, the reactor coolant pump motors are tripped to reduce inrush on the RAT when the bus is re-energized. If the transfer continues to be blocked due to lack of synchronism, the RAT breaker will be automatically closed when the residual bus voltage decays to an acceptable value.

(d) A manually initiated, synchronism check relay-supervised live bus transfer is provided for transferring a 13.8-kV bus from the UAT source to the RAT source, and vice versa, provided that the source to which the bus is being transferred is energized, and the two sources are in synchronism.

2. Switchgear

The 13,800V Distribution System is comprised of two metal-clad indoor switchgear assemblies containing 15 kV, 500 MVA nominal interrupting rating air circuit breakers. The incoming line breakers and the switchgear buses are rated 2000-amperes continuous current. Motor and unit substation feeder breakers are rated 1200-amperes continuous current. The 13.8-kV switchgear is non-Class 1E, and is located in the noncategory I, nonessential switchgear room adjacent to the north wall of the Control Building. Connections to the auxiliary transformers are made by means of 15-kV nonsegregated phase bus duct. This duct is routed through the Turbine Building.

3. Grounding and Ground Detection

The 13,800V System is a low-resistance grounded system. Grounding resistors for limiting the ground fault current to 600 amperes are located at the UAT and RAT transformers. Occurrence of a line to ground fault will cause the faulted portion of the system to be isolated by tripping appropriate circuit breakers. Inspection of the ground fault detector relays at the switchgear indicates the location of the fault.

4. Protection of 13.8 kV Containment Electrical Penetrations

Fuses mounted in Class 1E metal-clad fuse enclosures are provided in the feeders to the reactor coolant pump motors. These fuses are located in a seismic Category I building and are part of the protection for the containment electrical penetrations as required by Regulatory Guide 1.63. In addition, a measure of backup protection is provided by the reactor coolant pump circuit breaker and the 13.8-kV bus incoming line circuit breaker. DC control power from separate battery sources is provided for these breakers to preclude the loss of a single DC source from preventing the tripping of both the RCP and the incoming line breaker. Although these breakers are not Class 1E, the construction of the 13.8-kV switchgear is similar to the construction of the Class 1E 4-kV switchgear. Periodic testing of these breakers according to the Technical Specifications further verifies their reliability.

A fault on one of the RCP motors assuming failure of its switchgear will be isolated by the fuses provided. Backup protection is provided by the incoming feeder to the switchgear; credit is taken for the 13.8-kV breakers mentioned above to provide backup protection because if it is assumed that the seismic event damages the nonseismic qualified 13.8-kV switchgear, then it will have to be assumed that the nonseismic qualified power sources will also be damaged by the same event; thus, the circuit will be de-energized. For the penetration protection coordination curve, see Figure 8.3-47.

5. Contingency Power Source

During an extended loss of off-site power (LOP) event, emergency diesel generator EDG-1A may be aligned to provide power to 13.8 kV Buses 1 & 2 (see Figure 8.3-1). This contingency alignment is described in Subsection 8.3.1.1.b.9.

b. 4160V Distribution System

1. Arrangement

The 4160V Distribution System is shown in Figure 8.3-1, Figure 8.3-7, Figure 8.3-8 and Figure 8.3-9. The system consists of four buses, two of which are the redundant Class 1E emergency buses supplying the redundant engineered safety features loads. These safety loads are divided into two separate and independent Trains A and B, as shown on Figure 8.3-8 and Figure 8.3-9. The preferred power supply to each 4160-volt bus is from a UAT. An alternate source is available to each bus through a RAT. A standby power supply, consisting of a diesel generator, is available to each emergency bus. A non-safety related supplemental emergency power system (SEPS) is available as a backup power source to either Bus E5 or E6 when the emergency diesel generators fail to start and load. This alignment is described in Subsection 8.3.1.1.b.9. Buses E5 and E6 are the equipment designations of the redundant Class 1E buses.

Redundant Class 1E Buses E5 and E6 are located in completely separate, but adjacent rooms in the seismic Category I Control Building, as shown on Figure 8.3-27. Buses E5 and E6 are connected to the auxiliary transformers via non-Class 1E nonsegregated phase bus duct.

The bus duct is supported by seismically qualified supports in the Control Building. Taps in the bus duct provide the power to nonsafety-related Buses 3 and 4 from the bus duct runs to Buses E5 and E6, respectively. The tie between the nonsafety-related bus ducts and the Class 1E switchgear is through Class 1E air circuit breakers.

2. Switchgear

All Class 1E switchgear has identical electrical ratings:

(a) Buses - 2000-ampere continuous rating, braced for 80,000 amperes momentary.

(b) Incoming line breakers - 2000-ampere continuous rating, 350-MVA nominal interrupting capacity. SEPS incoming breaker – 1200-amperes continuous rating, 350 MVA nominal interrupting capacity.

(c) Feeder breakers - 1200-ampere continuous rating, 350-MVA nominal interrupting capacity.

The switchgear is selected in accordance with the criteria of ANSI C37.010 and C37.4.

3. Automatic and Manual Transfers

The automatic and manual transfers of power to each bus from one source to the other are as follows:

(a) An automatic transfer from the main generator supply to the offsite source through the GSUs is accomplished by the opening of the main generator breaker; no bus breaker action is required.

(b) A manually initiated live bus transfer is provided for transferring the 4.16-kV buses from the offsite source through the GSUs to the main generator source, and vice versa, by the operation of the main generator breaker. The transfer from offsite to main generator source is contingent upon the two sources being in synchronism.

(c) The opening of the UAT incoming line breaker, either manually or automatically, initiates an automatic transfer from UAT to RAT source, provided that the RAT transformer is energized, no fault exists on the bus, and the bus voltage is in synchronism with the RAT voltage at the time of transfer initiation or has decayed to an acceptable value.

A high-speed static synchronism check relay will allow the RAT incoming line breaker to close, thus completing the transfer, if both the voltage and phase angle between bus and RAT source are within limits. If the transfer is blocked due to lack of synchronism, the RAT breaker will be automatically closed when the residual bus voltage decays to an acceptable value. On the emergency buses, if the RAT breaker has not closed 1.2 seconds after transfer initiation, then the motors on the bus are tripped and the automatic transfer circuitry is disabled to prevent inadvertent RAT breaker closure.

(d) A manually initiated, synchronism check relay-supervised, live bus transfer is provided for transferring a 4.16-kV bus from the UAT source to the RAT source, and vice versa, provided that the source to which the bus is being transferred is energized, and the two sources are in synchronism. Synchronism is checked at the main control board (MCB), prior to manual initiation, by energizing the appropriate synchronism check relays.

After synchronism is verified by the synch scope, the transfer is initiated by manually closing (by means of the MCB-mounted control switch) the desired source breaker, which automatically trips the other source breaker.

4. Undervoltage and Load Shedding

(a) First Level Undervoltage Protection

Upon loss of voltage on a 4.16-kV emergency bus, 1.2 seconds are allowed for the automatic transfer described in Subsection 8.3.1.1b.3(c) above to be completed. If undervoltage persists after this time, the first level of undervoltage protection will be activated. Channel response time includes consideration of the bus voltage decay time due to generated Electro-Motive Force (EMF) from motors connected to the bus as the motors coast down. The following actions occur simultaneously:

(1) Bus loads are tripped as required

(2) UAT and RAT breakers are tripped to isolate the bus

(3) Automatic transfer schemes are disabled

(4) Standby power supply (diesel generator) is started and subsequently connected to the emergency bus as described in Subsection 8.3.1.1e.

On an emergency bus, if the UAT incoming line breaker trips open and the RAT source is unavailable, the transfer schemes are not initiated. The standby power supply (diesel generator) is immediately started and connected to the emergency bus as described in Subsection 8.3.1.1e.

(b) Second Level Undervoltage Protection

If the voltage on a 4.16-kV emergency bus is below that required to ensure the continued operation of safety- related equipment, the second level undervoltage protection scheme is activated. If the activation occurs coincidentally with an accident signal, then the UAT and RAT incoming line breakers are automatically tripped after a time delay to prevent spurious operation due to transients such as starting of large motors. This will result in total loss of voltage to the bus with ensuing actions described in Subsection 8.3.1.1b.4(a) above.

If the second level undervoltage protection scheme is activated without the coincident presence of an accident signal, then only an alarm is received. Established plant procedures require the operator to take specific steps to assess the magnitude and expected duration of the disturbance causing the undervoltage. If he is not assured that the disturbance is transitory, and that recovery is imminent, he may choose to manually trip the offsite power circuit breakers after ensuring that further deterioration of safety will not result from his proposed action.

Systems and equipment used for safe shutdown as listed in Section 7.4 will be available in the event of a degraded grid voltage because of one or more of the following reasons:

(1) Not powered by the degraded power source

(2) Does not rely on electric power

(3) In standby and, therefore, not connected to the degraded source

(4) Equipment will continue to run unimpeded under degraded voltage conditions because of margin between equipment rating and duty

(5) Not sensitive to degraded voltage (resistive load)

(6) Time is available for corrective action.

This equipment will not be exposed to or rendered inoperative by degraded voltage and, therefore, would be available to place the plant in a safe shutdown status under nonaccident conditions.

5. 480-Volt Substation Feeders

Feeders from the 4.16-kV emergency buses are taken to 480-volt unit substations to supply the engineered safety feature loads requiring 480 volt supply. The 4.16-kV breakers feeding load centers are normally closed during the plant operation, and control is provided at the 4.16-kV switchgear. Breaker position indication is provided on the main control board.

6. Bus Ties

A non-safety related 4160 volt transfer switch (SEPS-CP-1) for the Supplemental Emergency Power System is located on elevation 21'-6" of the Train B switchgear room for the purpose of selecting a connection to 4.16 kV bus E5 or E6. The transfer switch consists of three cubicles; an incoming termination section and two switching sections. The termination cubicle is the center section which contains the connections for the incoming cables and bus bar which connects to the line side of the two disconnect switches located in the outer cubicles. A mechanical interlock is used that allows operation of only one switch closure at a time. These switches are manual assisted and do not have any automatic controls for operation. A 4.16 kV safety related circuit breaker on bus E5 and E6 is used to connect the SEPS output to the selected bus. A Kirk key interlock is provided to prevent both breakers from being racked in at the same time. One of these breakers is normally racked in. These design features prevent connection between the two safety-related buses.

7. Grounding, Ground Detection and Protective Relaying

The 4.16-kV system is a high resistance grounded system. In addition, the diesel generator neutral is grounded through a single-phase grounding transformer. The SEPS diesel generators neutrals are grounded through a single phase grounding resistor. In order to detect grounds, ground sensors have been provided on each motor circuit and load center feeder and each incoming line, including the diesel generator. Inputs from ground sensors and ground detection circuits are furnished to the station computer. The computer is programmed to recognize various combinations of inputs and to provide an alarm to alert the operator of the ground fault and its location. The grounding scheme used allows single ground faults to be alarmed only, and the equipment to continue operation.

The 4.16-kV motor and load center feeders are protected by overcurrent relays with long-time characteristics and instantaneous elements. The buses and incoming feeders are protected by inverse-time overcurrent relays.

8. Control Power Supplies

DC power supplies, as shown on Figures 8.3-3 and 8.3-37, are used to provide power for the operation of breakers and control circuitry associated with each of the redundant 4.16-kV emergency buses.

9. Contingency Connections

(a) Startup Feedpump

The startup feedwater pump (SUFP) P-113 can be supplied from either Bus 4 or Bus E5. Through Bus E5, the SUFP can be supplied power from emergency diesel generator EDG-1A under contingency conditions (see Subsections 6.8 and 10.4.12). Connection of the SUFP to Bus E5 is through a Class 1E circuit breaker. These connections are shown in Figures 8.3-7 and 8.3-8.

Paralleling of the Class 1E and non-Class 1E buses will be positively prevented by an interlocking system. A two-position (Bus E5 - Bus 4) key-locked switch must be operated to be able to close the breaker on Bus E5 or Bus 4. In addition to this interlock, when the switch is placed in position "Bus E5" it will send a trip signal to the circuitry of the SUFP on Bus 4 and vice versa. Kirk Key interlocks assure that only one breaker can be racked in at anytime. During transfer from one bus to the other, the "from" bus circuit breaker will have to be racked out before the "to" circuit breaker can be racked in preventing paralleling the buses. Transferring the SUFP power supply between Buses 4 and E5 will be manual actions controlled by procedure.

(b) Extended Loss of Offsite Power

An extended loss of offsite power (LOP) in the winter months such as could occur during a severe winter storm or from a Y2K grid disturbance results in concern with equipment freezing in the non-safety related balance-of-plant (BOP) areas. Heating in these areas is not supplied by an emergency power supply. An LOP at 100% power would cause a plant trip with both safety related EDGs automatically starting to supply safe shutdown loads.

As a contingency action, emergency diesel generator EDG-1A can be aligned to supply the BOP loads (e.g., heating and lighting) on the non-safety related electrical buses (see Figure 8.3-1). This alignment is administered under procedural control and can be used only after EDG-1B is verified stable and supplying the loads required to maintain the plant in a safe shutdown condition. EDG-1A is still available to supply safety-related loads if problems are encountered with operation of EDG-1B.

EDG-1A can be connected to the non-safety related electrical buses using both reserve auxiliary transformers (RAT) or only using RAT-3A and the unit substation (US) cross tie circuit breakers to supply the Bus 2 US from the Bus 1 US. A study and analysis has been performed to demonstrate the capability of EDG-1A and RAT(s) to supply the BOP loads. EDG-1A loading will be maintained within its continuous rating while supplying the BOP loads. Because this contingency connection is not a normal system lineup, various control circuit features, some of which are discussed in Subsection 8.3.1.1.b.3(c) & (d); 8.3.1.1.b.4(a) & 8.3.1.1.e.6, must be bypassed (jumper, lifted lead, etc.) to permit EDG-1A to supply the RAT(s). Installation of the circuit bypasses will be a controlled evolution according to an approved procedure. These circuit bypasses do not degrade the electrical protection (e.g., overcurrent, differential, etc.) for EDG-1A, the electrical buses and RAT(s). The same degree of electrical protection is provided for EDG-1A, the electrical buses and RAT(s) for this contingency lineup as is available when EDG-1A is paralleled with offsite for routine surveillance testing.

Control room bypass/inop status indication per Regulatory Guide 1.47 is not required for these circuit bypasses because the LOP condition is not expected to occur more frequently than once per year. Also, EDG-1A is not required to be operable since EDG-1B is operable/operating to supply the safety-related equipment required to cope with the LOP.

After offsite power is restored and verified stable and reliable, normal offsite power connections to the plant buses can be restored. EDG-1A would be synchronized with offsite power, its circuit breaker opened and EDG-1A shut down.

Subsection 8.3.1.4.a indicates that protective devices for non-Class 1E loads connected to Class 1E buses are coordinated such that failure of all of the non-Class 1E loads will not result in tripping the incoming breaker to the Class 1E bus. The RAT, Buses 1 & 2 and their connected loads are non-safety related so the Bus E5 RAT circuit breaker potentially falls under this requirement. However, since EDG-1A and Bus E5 are considered inoperable for this contingency alignment, there is no need for coordination or testing of the protective device since trip of this circuit breaker to Bus E5 would not affect loads performing a safety related function.

(c) Supplemental Emergency Power System (SEPS)

The non-safety related supplemental emergency power system (SEPS) is designed as a backup power source to either Bus E5 or E6 when the EDGs fail to start and load. SEPS is capable of providing the required safety related loads in the event of a loss of offsite power (LOP) coincident with the loss of both emergency diesel generators. During these events it is assumed that there is no seismic event or an event that requires safeguards actuation (SI, CBS, CVI, CI, etc.). This is considered a non design basis event. Although SEPS may be used during a station blackout, it will not be credited as a station blackout power source. Seabrook Station will continue to be an AC independent plant (See UFSAR Section 8.4)

Subsection 8.3.1.4a indicates that protective devices for non-Class 1E loads connected to Class 1E buses are coordinated such that failure of all of the non-Class 1E loads will not result in tripping the incoming breaker to the Class 1E bus. Non-safety related SEPS is connected to bus E5 or E6 when the offsite power sources and both EDGs fail to start and load so the SEPS circuit breakers potentially fall under this requirement. However, since EDG-1A, EDG-1B, and Buses E5 and E6 are not operable during this contingency alignment, there is no need for coordination or testing of the protective devices since trip of these breakers would not effect loads performing a safety related function.

The SEPS supply connects to a transfer switch for the purpose of selecting a connection to 4.16 kV Bus E5 or E6. Mechanical interlocks prevent closing both switches at any one time. A 4.16 kV safety-related circuit breaker on buses E5 and E6 is used to connect the SEPS system output to their respective buses. Connection to both E5 and E6 will be positively prevented by mechanical interlock. The mechanical interlock assures that only one breaker can be racked in at anytime.

c. 480V Distribution System

1. Arrangement

The 480V Distribution System comprises unit substations, 460-volt motor control centers, and distribution panels. The unit substations consist of a transformer and adjacent 480-volt switchgear.

The 480V Engineered Safety Features Distribution System consists of two separate and independent Trains A and B, to supply redundant engineered safety features load groups, consistent with the separation of the 4160V Engineered Safety Features Distribution System. The system configuration, shown on Figure 8.3-1, Figure 8.3-11, Figure 8.3-12, Figure 8.3-13, Figure 8.3-14, and Figure 8.3-15, shows the motor loads and motor control centers fed from the unit substations. The loads supplied from the motor control centers are shown on Figure 8.3-16, Figure 8.3-17, Figure 8.3-18, Figure 8.3-19, Figure 8.3-20, Figure 8.3-21, Figure 8.3-22, Figure 8.3-23, Figure 8.3-24, Figure 8.3-25, Figure 8.3-26, Figure 8.3-52, Figure 8.3-53, and Figure 8.3-54.

2. Unit Substation Transformers

Transformers for 480-volt emergency buses are supplied from the 4160 volt emergency buses. The transformers are rated 1000/1333 kVA (AA/FA) and are air insulated, dry type. The transformer impedance is selected to limit the maximum short-circuit current at the 480-volt load center bus to the motor control center (MCC) breaker rating of 25,000 amperes rms symmetrical (except as noted in Section 8.3.1.1.c.3), and to maintain voltage at the motors of 414 volts (90 percent of 460) under normal operating conditions and 368 volts (80 percent of 460) during starting of motors except for unit substation E64. A lower transformer impedance is selected for unit substation E64 to maintain voltage at the motors of 414 volts (90 percent of 460) under normal operating conditions and 368 volts (80 percent of 460) during starting of motors. The maximum short-circuit current at the 480-volt unit substation bus is limited to the motor control center breaker rating of 25,000 amperes rms symmetrical by limiting the load on unit substation E64 to 600 kVA.

3. Unit Substation Switchgear

The 480-volt safety-related switchgear is of metal-clad indoor design and has three-pole, metal frame, low-voltage power circuit breakers. Electrically operated breakers receive control power from one of the redundant DC power supplies, as shown in Figure 8.3-3. The switchgear is selected in accordance with ANSI C37.13, where applicable, and has ratings compatible with the normal current and expected fault duty as follows:

	Frame Size Amperes	RMS Sym. Amps. with Inst. Trip	RMS Sym. Amps. without Inst. Trip
Transf. secondary bkr.	2000	65,000	55,000
Feeder breaker	600	30,000	22,000*
Feeder breaker	1600	50,000	50,000

* Design calculations confirm that the actual available fault current is less than 22,000 A where 600 A frame breakers without instantaneous trip devices are used since their interrupting rating is less than the 25,000 A available short circuit current given in Section 8.3.1.1.c.2.

The 460-volt motors connected to the load centers are protected by tripping devices having a combination of long time delay and instantaneous elements. Feeder breakers to MCCs are equipped with the long-time and short-time tripping devices.

4. 460 Volt Motor Control Centers

460 volt motor control centers are of metal-enclosed indoor design, and are provided with molded-case circuit breakers with magnetic or thermal-magnetic tripping devices. Combination motor starters, consisting of a magnetic-only breaker and a starter having one thermal overload element per phase, are provided for motors located outside the containment. (For motors located inside the containment, see Subsection 8.3.1.1c.7 on special 480-volt circuits.) Thermal magnetic breakers, with a contactor as required, are provided for all other loads.

The motor control center buses are braced to withstand 42,000 amperes rms symmetrical. The combination starter units and feeder breaker units are rated to withstand a three-phase fault at the load terminals, with a fault current contribution of 25,000 amperes rms symmetrical at the line terminal. Contactors and starters are rated to withstand voltage dips to 70 percent of 460 volts without dropping out.

Distribution panels are standard metal-enclosed panelboards with molded case breakers protecting the branch circuits. Breakers are rated to withstand a three-phase fault at the load terminals, with the maximum available system fault current contribution at the line terminals.

5. Classification

The 480-volt load centers, 460-volt motor control centers and distribution panels which supply power to Engineered Safety Features or reactor protection systems are all Class 1E and are located in seismic Category I structures.

Certain other motor control centers (MCC 111 and MCC 231) and distribution panels (RC-PP-6A, 6B, 6C, 6D and 6E) are qualified to meet Class 1E requirements and are located in Category I structures specifically because they are used to feed loads inside the containment as described in paragraph 7(a) below.

6. Bus Ties

No bus ties exist between redundant buses. Manual bus tie breakers provide the capability to interconnect load center buses within a single train.

Bus tie breakers provide manual interconnection capability between unit substations E51, E52 and E53, all of Train A. Similarly, interconnection capability exists between unit substations E61, E62 and E63, all of Train B.

Bus ties may be used when any unit substation transformer is out of service for maintenance or repair. Bus ties are provided only for operational flexibility. The unit substations are not designed to supply the total load of both buses when bus ties are used. When a bus tie breaker is used, loading on each unit substation will be administratively controlled to be within the FA rating of the unit substation transformer.

7. Special 480-Volt Criteria

(a) Protection of Containment Electrical Penetrations

The Class 1E and non-Class 1E 480-volt unit substations, 460-volt motor control centers and the distribution panels which feed loads inside the containment are all qualified to meet Class 1E requirements and are located in seismic Category I structures. 460-volt loads inside the containment are fed from distribution equipment with special provisions to satisfy the requirements of Regulatory Guide 1.63 for containment electrical penetration protection. These provisions are outlined below.

460-volt loads inside containment which are fed from unit substations, motor control centers or distribution panels are provided with one of the following special arrangements to insure that the penetration integrity is maintained:

(1) Circuits of motors 5 hp and less are provided with two identical combination starters. Both units are located in the same compartment of the MCC. See Figure 8.3-50 for typical coordination curves. The thermal overload relays identified in the figure by numbers (2) and (4) are utilized to protect the penetration from a fault whose magnitude is insufficient to trip the magnetic part of the protective device.

(2) Circuits of motors greater than 5 hp are provided with a thermal magnetic breaker in series with a combination starter. Both the breaker and the combination starter are located in the same compartment of the MCC. See Figure 8.3-49 for typical coordination curves.

(3) Feeder circuits, except for the pressurizer heater circuits, are provided with two identical thermal magnetic breakers. Both breakers are located in the same compartment of the MCC or panel.

(4) Feeder circuits for the pressurizer heater Groups A and B circuits are provided with fuses in series with an air circuit breaker. Feeder circuits for the pressurizer heater Groups C, D, and E are provided with dual fuses. The fuses are located in panels and the air circuit breakers are located in unit substations.

The motor control centers and unit substations containing these special protective devices are located in the Control Building switchgear area, with one exception. The panels for the pressurizer heater circuits are located in the electrical penetration area outside the containment. Both the primary and the backup protective devices are qualified 1E devices.

There are no high or moderate energy lines in the above areas; therefore, only faults within the electrical devices could conceivably damage these protective devices. If a protective device fails catastrophically while clearing a short circuit, the second protective device may possibly be affected because of its proximity. However, in this instance, no penetration damage can occur because all short-circuit current flow will be diverted to the new fault located at the protective device. Therefore, there is no conceivable electrical failure that could prevent both the protective devices from operating and at the same time allow the fault current to flow through the penetration.

460-volt loads inside the containment, which are fed directly from the 480-volt unit substation, satisfy the requirements of Regulatory Guide 1.63 by utilizing the load breaker as primary protection and the unit substation incoming feeder breaker as backup protection. See Figure 8.3-48 for typical coordination curves.

Feeders for 460-volt distribution panels (lighting panels), located inside containment are provided with two thermal magnetic molded-case breakers in series to satisfy the requirements of Regulatory Guide 1.63.

460-volt loads inside the containment which are normally used only during shutdown (e.g., cranes, refueling machines, welding receptacles, etc.) are not provided with redundant protection because their circuits are de-energized and padlocked at the unit substation or motor control center during normal plant operation. Verification of the circuits being de-energized is part of the Technical Specifications. Though some of these circuits may be required for brief durations during plant operation, such as prior to or after refueling outages, lack of redundant protection is justified because of the very limited usage in this mode and the fact that such usage will be under Technical Specification requirements.

Control circuits powered from 120V AC or 125V DC distribution panels have dual protective devices (circuit breakers and/or fuses) to provide penetration protection in accordance with Regulatory Guide 1.63. The protective devices are located in seismically qualified buildings.

The CRDM circuits (lift coil, stationary and movable gripper) which are powered from the CRDM power distribution system have dual non-Class 1E protective devices (fuses) to provide penetration protection in accordance with IEEE Standard 741-1986, Subsection 5.4.3. These fuses are part of the CRDM Power Distribution System.

The manufacturer of the penetrations has furnished damage curves which establish the duration and magnitude of overcurrents that the penetrations can sustain. Typical coordination curves are shown on Figure 8.3-47, Figure 8.3-48, Figure 8.3-49 and Figure 8.3-50. Except for Figure 8.3-47 which shows the coordination curves for the RCP electrical penetrations, the sizes, setpoints, and response curves of the protective devices in the figures do not correspond to any specific load applications. Those figures are intended to be representative of how various types of protective devices are used to coordinate with the electrical penetration damage curves. The specific sizes, setpoints, and response cures are controlled by design calculations and drawings.

It can also be seen in these figures, that the curves of the protective devices are, in all cases, to the left of the penetration damage curves. Thus, the protection provided will assure that long or short duration overcurrents that are capable of damaging the penetration will be interrupted before they cause damage.

Low energy circuits, i.e., control circuits powered from limited-capacity power sources such as control power transformers (maximum capacity 300 VA) and instrumentation circuits, do not require dual protection because the short circuit versus time capacity of their power sources is within the penetration capabilities.

(b) Protection Against Single Failure of Manually Controlled Electrically Operated Valves (BTP ICSB 18)

Separate, normally de-energized motor control centers (E522-Train A, E622-Train B) are provided for motor- operated valves in the Safety Injection and Residual Heat Removal Systems whose spurious movement due to random single failures is deemed unacceptable. These valves are normally aligned to their safe positions. Following an accident, the operator can energize these MCCs from the control room to accomplish any necessary valve repositioning. Redundant valve position indication is provided on the main control board for each valve. One set of indicating lights is powered from a 120V AC uninterruptible power supply; the redundant set of lights is powered from a 120V AC power panel in a safety-related MCC other than E522 or E622.

In addition to the above circuits, another special circuit is used to support BTP ICSB-18. Valve SI-V93, in the common miniflow return line from the SI pumps SI-P6A and P6B to the RWST is provided with a contactor controlled by a key-operated switch in addition to the normal combination motor starter. For details see Figure 8.3-45. With this arrangement, a single failure will not cause a valve movement and, furthermore, the operator is required to perform two distinct operations to change the valve position. The circuit for SI-V93 is powered from MCC E621.

(c) Special Provisions in Response to NUREG-0737

The design complies with the guidelines of NUREG-0737 and the "clarifications" to NUREG-0737.

(1) Item II.E.3.1, Pressurizer Heaters

A description of the pressurizer heaters is provided in Section 5.4.

One pressurizer heater bank can be supplied from the Train A diesel generator and one bank can be supplied from the Train B diesel generator during loss of offsite power. Each bank can establish and maintain natural circulation at hot standby conditions. Each bank can be supplied from either offsite power or from one diesel generator.

As demonstrated in Table 8.3-1, the standby power supply has the capacity to supply the pressurizer heaters without load shedding.

Changeover of the pressurizer heaters from normal offsite power to emergency onsite power can be accomplished manually in the control room.

Motive and control power connections to the Class 1E buses from the pressurizer heaters are through Class 1E devices.

The pressurizer heaters are not automatically shed from the emergency buses upon the occurrence of a safety injection actuation signal. However, they are load shed on loss of offsite power.

(2) Item II.G.1, PORV and PORV Block Valve Power and Control

Motive and control components of the PORVs and the PORV block valves can be supplied from the offsite power source or the onsite power source.

Motive and control power connections to the Class 1E buses for the PORVs and block valves are through Class 1E devices.

The pressurizer level indication instrument channels are powered from the vital instrument buses. The vital buses can be powered from the offsite power source or onsite power sources.

The design of the PORV block valves provides the capability to close the valves and retains to the extent practical the capability to open the valves. These capabilities are maintained by providing two redundant motor-operated PORV block valves located in parallel flow paths. One PORV valve is supplied power by the Train A emergency bus and the other PORV block valve is supplied power by the Train B emergency bus.

The motive and control power for the block valves is supplied from a different emergency bus than the source supplying the PORVs. See Figure 8.3-57.

Changeover of power to the PORV and block valves from offsite power to onsite power can be accomplished from the control room.

8. Undervoltage and Load Shedding

Selected loads on 480-volt safety-related load centers are tripped upon detection of sustained undervoltage on the 4160-volt emergency bus. Automatic reclosing of the unit substation motor feeder breakers is described in Subsection 8.3.1.1.e.6.

Upon loss of voltage on a 460-volt MCC bus, all starters will drop out automatically. Combination starters having maintained start circuits with an over-riding trip will reclose upon restoration of voltage. However, other combination starter units will require start signals for reclosing. Refer to Subsection 8.3.1.1.e.6 for automatic reclosing on the standby power supply.

9. Grounding and Ground Detection

The 480V system is a high-resistance grounded system. The grounding equipment is located at the 480-volt unit substations. This grounding configuration allows the system to operate effectively as an ungrounded system, i.e., service is not interrupted in case of a single ground fault, but an alarm is initiated.

d. 120V Vital Instrumentation and Control Power System

The 120V Vital Instrumentation and Control Power System comprises the uninterruptible power supply (UPS) units and the 120-volt distribution panels, as shown in Figure 8.3-4. The vital instrument panels and the vital UPS units are Class 1E and are located in the seismic Category I Control Building at elevation 21'-6". The physical arrangement of these UPS units and panels is shown in Figure 8.3-27. The six vital UPS units are normally powered from the 480V system and can also convert 125-volt DC power from station batteries to 120-volt AC power. These UPS units feed six electrically independent 120-volt AC vital instrument panels which serve as instrument and control power supplies.

Four of the vital UPS units provide separate and independent power supplies to the four NSSS instrumentation channels (designated as channels I, II, III and IV). The two additional vital UPS units provide redundant power supplies to the balance of plant Train A and Train B vital instrument panels. Each vital UPS unit has adequate capacity to carry the associated load continuously. Loads supplied from each UPS unit are as shown in Figure 8.3-28, Figure 8.3-29, Figure 8.3-30, Figure 8.3-31, Figure 8.3-32, Figure 8.3-33, Figure 8.3-34 and Figure 8.3-35.

One of the NSSS channel-associated UPS units and one of the balance-of-plant UPS units also feed separate panels for nonvital instrumentation and controls. The nonsafety-related panels are supplied from Class 1E panels through Class 1E circuit breakers. Manually operated maintenance feeds are provided to each of the four vital instrument panels associated with the NSSS instrumentation channels from 480/120-volt AC transformers connected to nonsafety-related MCCs. Each of the two balance-of-plant vital instrument panels are provided with a static transfer switch for automatic and fast transfer of these buses to a maintenance supply from a 480/120-volt AC transformer connected to a nonsafety-related MCC in the event of unavailability of the associated UPS. In addition to the automatic transfer switch, the manual transfer capability to maintenance supply is also provided to bypass and isolate the static transfer switch for maintenance. On each UPS, instrumentation is provided to monitor AC and DC input currents, as well as output current and voltage. Alarms are provided on the station computer for loss of AC voltage on the vital instrument panels. The 120V AC Vital System is a two-wire ungrounded system with a ground detection scheme. Each branch circuit at the distribution panel is protected by a thermal magnetic breaker.

In addition to the six vital UPS units, there are three nonsafety-related UPS units feeding the station computer and miscellaneous auxiliary loads which require a reliable AC source. These units are also normally powered from the 480V AC system and can also convert 125-volt DC power from station batteries to 120-volt AC power.

e. Standby Power Supply (Diesel Generator Units)

1. Diesel Generator Ratings and Capabilities

The standby power supply is provided by two redundant diesel engine generator systems of identical design and characteristics which supply onsite power of sufficient capacity and capability to reliably shut down the reactor. The load rating of each diesel generator is as follows:

8760 hours per year (continuous)	6083 kW
Short time	6697 kW

The generator itself is rated 8375 kVA at 0.8 pf lagging continuous, with a 2-hour short time rating of 9213 kVA at 0.8 pf.

The basis for sizing the diesel generator is consistent with the regulatory position of Regulatory Guide 1.9, the application criteria set forth in IEEE 387, and the "single generator driven by a single prime mover" philosophy that conforms to the regulatory position of Regulatory Guide 1.6.

The capacity of each diesel generator is adequate to support operation of engineered safety feature loads within the short time rating, and is determined on the basis that the sum of the predicted loads needed to be powered at any one time does not exceed the short time rating.

Each diesel generator is connected to a 4160-volt emergency bus as shown in Figure 8.3-1 and Figure 8.3-10. The capacity of each diesel generator is sufficient to meet the safety features demand caused by a loss of offsite power with or without a coincident loss-of-coolant accident. The diesel generator safety features loading sequence is shown in Table 8.3-1 and Table 8.3-2.

The diesel generator control circuits provide the capability for both fast and slow starts. A fast start is where rated voltage and frequency is attained within a maximum of 10 seconds after receipt of a start signal. A

slow start involves starting to idle speed, operation at idle speed for a predetermined time period, and then automatic speed increase to rated speed at a predetermined rate. Slow start capability is provided to reduce engine wear and tear during periodic surveillance testing. Emergency start signals (manual, safety injection or loss of bus power) are always fast starts. Emergency starts always override a slow start signal.

2. Diesel Generator Auxiliaries

Each diesel generator system comprises the auxiliaries necessary for fast start operation, connection to the 4160-volt emergency bus, and connections to the required services. No auxiliaries are shared between the diesel generator systems. External power sources, other than DC control power from the unit's station batteries, are not required for starting or subsequent operation.

To ensure availability of starting air, each diesel engine is supplied with air starting systems, as described in Subsection 9.5.6. Elapsed time from receipt of start signal to a condition of rated voltage and frequency is a maximum of 10 seconds.

The diesel engines are cooled by a closed-jacket water system which in turn is cooled by water from the Service Water System. Each diesel is provided with a thermostatically controlled electric heater for the jacket water. Lubricating oil is continuously circulated through the engine and receives heat from an electric heater. This procedure provides maximum assurance of starting with minimum engine wear. The diesel is capable of operating without cooling water flow through the heat exchanger during loss of offsite power for a period of time in excess of that required to start the service water pumps (minimum of 52 seconds) from the emergency bus. A further description of the Diesel Cooling Water System is presented in Subsection 9.5.5.

Each diesel generator system is provided with a seven-day fuel oil storage tank and a day tank of capacity sufficient for a minimum of three hours of engine operation at continuous full load. The day tank is equipped with level switches to provide automatic actuation (starting and stopping) of the associated fuel oil transfer pump, and low level and high level alarms. Overflow recirculation piping is provided from the day tank to the fuel oil storage tank. The fuel oil storage tanks have a capacity suitable for operating one emergency generator at post-accident load for at least seven days. Fuel oil is transferred from each oil storage tank to its respective day tank by a pump powered from the associated emergency bus. Valving is provided to permit either transfer pump to supply one or both day tanks from either storage tank.

Procedural controls and scheduled periodic testing are established to maintain the integrity of the system, and to confirm that the tanks have an adequate fuel supply when the diesel generators are in the standby condition. A further description of the Diesel Engine Fuel Oil System is presented in Subsection 9.5.4.

Each diesel generator unit and its auxiliaries are located in a separate and independent enclosure within a seismic Category I building. The reinforced concrete enclosure wall between diesel generators has a three-hour fire rating and is designed to withstand explosions and stop postulated missiles from the adjoining diesel generator and its auxiliaries, such as a crank-case door created by a crank-case explosion, or rupture of one of the air receiver vessels. The diesel generator and its auxiliaries which are essential for the operation are designed in accordance with Category I seismic requirements.

Separate and independent heating and ventilating equipment is provided for each diesel generator system to supply adequate air for control of the ambient temperature. The heating and ventilation system for the diesel generator system is described in Subsection 9.4.8.

Electrical components associated with the starting of the diesel generators (e.g. auxiliary relay contacts, control switches, etc.) are located within NEMA 12 control cabinets which minimize the accumulation of dust and dirt on electrical contacts. The cabinet doors are gasketed and, where louvers are provided, the openings are covered by filters which will prevent passage of particulate matter including products of combustion which could degrade engine starting or operation.

3. Arrangement

The power connection between the diesel generator and the 4160-volt emergency bus is made with Class 1E nonsegregated phase bus duct. The bus duct and power and control cables for each diesel unit are routed to maintain physical separation of the diesel generator systems, and to prevent a single event from disabling both of the redundant systems. The arrangement of the diesel generator systems, demonstrating physical separation and isolation, is shown in Figure 8.3-36.

4. Protective Devices

Diesel generator protective trips are in compliance with the requirements of Branch Technical Position ICSB 17. They consist of low lube oil pressure, high lube oil temperature, high jacket coolant temperature, generator overcurrent, reverse power, generator differential current, loss of field and mechanical overspeed.

The generator overcurrent, reverse power, loss of field, high lube oil temperature, and high jacket coolant temperature protective devices are bypassed when the diesel generator is required to mitigate the effects of an accident. The overspeed, differential and low lube-oil pressure protective devices are not bypassed in any mode of operation. To preclude the possibility of a spurious trip from lube-oil pressure, three independent low-pressure signals are conditioned by a two-out-of-three coincidence logic.

The operability of the bypass circuitry can be tested during diesel generator routine testing. With the diesel generator running and connected to the emergency bus, first an accident signal is simulated to activate the bypass circuitry and then protective signals are simulated. Failure of the engine to shut down and the diesel generator breaker to trip is evidence of the proper functioning of the bypass circuitry.

During an accident the bypassed protective devices provide annunciation. Other monitoring devices provide annunciation in any mode of operation. Mechanical and electrical surveillance devices are located at the local panel and in the control room so operators can observe conditions and take appropriate action.

Because the diesel generator cannot supply power to a faulted bus, it should be disconnected in the event of a bus fault to avoid possible damage or fire to the 4.16-kV switchgear. For this reason, an additional protective relay has been provided to trip the diesel generator circuit breaker on a 4.16-kV bus fault. This bus fault relay is not bypassed in any mode of operation. The relay does not trip the diesel generator but only trips the generator circuit breaker. The diesel engine continues to run and can be reconnected manually to the bus if no fault exists. The bus fault relay is set to actuate at a fault current sufficiently high to avoid spurious trips of the diesel generator breaker.

An additional trip (operational) when the diesel generator is connected to the bus during testing with UAT or RAT supply connected, assures isolation of the diesel generator on receipt of an accident signal.

Status monitoring lights are provided on the main control board to indicate diesel generator operating mode, availability, and bypassed conditions during all modes of plant operation.

Alarms are provided in the control room to indicate the availability of diesel generator auxiliaries.

Conditions that can render the diesel generator unable to respond to an emergency start signal have been evaluated. These conditions and the resulting alarms and monitoring lights presented to the control room operator have been summarized in Table 8.3-8.

Other conditions that can make emergency power unavailable, but do not necessarily render the diesel generator unable to respond to an automatic start signal, and the resulting alarms are presented in Table 8.3-9.

Table 8.3-8 and Table 8.3-9 list specific alarms actuated by disabling conditions as well as common alarms that use logic to provide one alarm from multiple inputs. The alarms listed in the tables are those which are actuated by disabling conditions. Each common alarm clearly indicates the status of the emergency diesel generator and Emergency Power System. All disabling conditions are clearly distinguishable from conditions that are abnormal but not disabling. All conditions that render the diesel generator incapable of responding to an automatic emergency start signal are alarmed in the control room.

5. Reliability and Testing

Prior to shipment of the diesel generator sets, a type qualification testing program meeting the requirements of IEEE 387-1977 was performed on one diesel generator unit. The type qualification testing program consisted of load capability qualification, start and load acceptance qualification, and margin qualification. The test results are summarized below.

Load capability qualification was demonstrated by performing the following tests:

(a) No Load Test

Acceptance Criteria: The diesel generator unit must be operated for 6 hours at "ready to accept load" status, then operated for a period of 1 hour at 100% load with no abnormalities encountered.

Conclusion: Performance of the diesel generator unit met the acceptance criteria.

(b) 100% Load Test

Acceptance Criteria: The unit must operate at 100% load for 22 hours (8414 bhp - 6083 kW) at equilibrium temperatures without exceeding 180°F jacket water temperature out of the engine and 145°F lube oil to the engine.

Conclusion: Seven hourly load readings were noted that were below the 100% load rating. The worst being 98.9 percent or 6066 kW which was 17 kW below rated load. Despite this discrepancy, it is felt that the performance of the diesel generator has met the intent of the acceptance criteria.

(c) 110% Load Test

Acceptance Criteria: Immediately following the 100% load test, the unit must operate at 110% load (9264 bhp - 6697 kW) for a period of two hours without exceeding 180°F jacket water temperature out of the engine and 145°F lube oil to the engine.

Conclusion: Performance of the diesel generator met the acceptance criteria.

(d) 100% Load Rejection Test

Acceptance Criteria: The unit, when operating at 100% load (8414 bhp - 6083 kW) must remove the load instantaneously in one step without exceeding 560 rpm.

Conclusion: The diesel generator met the acceptance criteria.

Start and load acceptance qualification was demonstrated by performing the following test:

(a) 300 Start and Load Acceptance Qualification Test

Acceptance Criteria: A total of 300 valid start and loading tests shall be performed with no more than 3 failures allowed. A valid start and load test is defined as a start with loading to at least 50 percent of the continuous rating within the required time interval, and continued operation until temperature equilibrium is attained. The starts were made from both keep-warm temperature conditions as well as when it was at its normal operating temperatures.

Conclusion: The diesel generator unit successfully passed the 300 start and load acceptance qualification test.

Margin qualification was demonstrated in conjunction with the load acceptance tests.

Acceptance Criteria: Start the diesel generator and apply an initial load 10 percent greater than the initial design load and then, when the diesel is at approximately 75 percent of its rated load, apply an additional load 10 percent greater than the worst design step load:

(1) The voltage shall not drop to less than 80 percent of normal and frequency not less than 95 percent of normal

(2) The time to recover from 80 percent rated volts to 90 percent rated volts and from 95 percent frequency to 98 percent frequency after each load step shall not be more than 3 seconds.

Conclusion: The diesel generator successfully met the stated criteria during the load test.

In addition to the above and the manufacturer's standard tests, the following tests were performed at the factory on each unit prior to acceptance.

Load acceptance test - to demonstrate the capability to accept the loads that make up the design load, in the sequence and time indicated in Table 8.3-1, and to maintain voltage, speed, and frequency within limits specified in Subsection 8.3.1.2b.2. For this factory test, motor loads and resistive loads were connected to the diesel generator in a combination which exceeded the requirements of the actual loading sequence.

Rated load test - to demonstrate the capability to carry the continuous rated load for a time required to reach temperature equilibrium plus one hour, and to carry the 2000-hour rating, 6503 kW, for two hours.

Load rejection test - to demonstrate the capability to reject the largest single load without exceeding the speed limits specified in Subsection 8.3.1.2b.2.

Overload test - to demonstrate the capability to carry the short time rating for two hours.

Each diesel generator has been tested at the site as part of the preoperational tests to demonstrate that it is capable of performing its intended function. Site tests have been performed in accordance with IEEE Standard No. 387 and Regulatory Guide 1.108, with clarification as explained in Subsection 8.1.5.3.

Since the diesel generator has been placed in service, periodic testing is being performed as described in Subsection 8.3.1.1j. This periodic testing also provides input to the Maintenance Rule EDG performance criteria which support the target reliability levels required for the Station Blackout analysis (see Section 8.4.2).

6. Loading Description

Loads important to safety are connected to emergency buses and are arranged so that they are capable of being powered by the diesel generators in the event of failure of the offsite power sources. The operation of either diesel generator provides the minimum Engineered Safety Features required for abnormal and accident conditions.

If undervoltage is experienced on a 4160-volt emergency bus, a 1.2 second time delay is allowed for the transfer scheme to function. If the transfer is unsuccessful, the diesel generator is started. If undervoltage is experienced and offsite power is not available, the diesel generators are started immediately.

Independent manual starting and control devices are provided in the control room and at the local control panel near each unit to allow the generator to be synchronized to the bus for load testing without interrupting normal power to the bus. The capability exists in the control room and at the local control panel to synchronize the diesel generator to the offsite power supply and to close the offsite source breaker.

In the event of loss of offsite power, emergency buses are automatically cleared of all incoming power feeders and selected loads prior to the closing of the diesel generator circuit breaker. The feeder breakers from the 4160-volt emergency bus to the 480-volt load centers are not tripped, and thus these loads become energized when the diesel generator is connected to the 4160-volt bus. Following the automatic closing of the diesel generator circuit breaker, loads are connected to the emergency buses in a predetermined sequence dictated by plant conditions. Protective relays block the connection of a diesel generator to a faulted bus.

The automatic connection of the predetermined loads to the emergency bus is accomplished by emergency power sequencers (EPS). The emergency power sequencers function only upon loss of offsite power to sequence loads on the diesel generator. See Figure 8.3-46 for EPS logic. An EPS is furnished for each safety load group and is located in the Control Building Train A or B switchgear room. The EPS monitors critical emergency bus parameters such as voltage and incoming line circuit breaker positions (open or closed). Depending upon whether an accident condition (SI) is also present, the EPS provides appropriate contact outputs to the various safety-related loads to start them in a programmed time sequence. Momentary signals are provided to circuit breakers and starters of loads which are required to start at a specific time ("definite start loads"; see Table 8.3-1 and Table 8.3-2). Maintained permissive contacts are provided for loads whose starting is also dependent upon the presence of a process signal ("indefinite start loads," see Table 8.3-1 and Table 8.3-2).

The indefinite start loads assigned to the first load step at the 12-second time sequence point are not interlocked with a sequence timer contact and may, therefore, be loaded on the diesel generator at any time. By assigning these loads to the first load step in Table 8.3-1 and Table 8.3-2, they are assumed to start at that time or anytime thereafter throughout the loading sequence. This first step is the most heavily loaded step and, therefore, is the limiting step. If these loads start randomly at any other time, other than at the first step, they will have less impact on the diesel generator loading capability. Indefinite start loads assigned to steps other than the first load step are interlocked with sequencer timing contacts to prevent these loads from starting prior to their assigned sequence point. As in the first step, these loads may start at any time after their assigned step.

Indicating lights for the sequencing steps are provided on the main control board to assist operation. Loading is started when the diesel generator reaches rated speed and voltage and the generator circuit breaker closes (approximately 10 seconds after the diesel start signal).

Table 8.3-1 shows the order and time at which the loads are automatically and sequentially applied to the diesel generator during a combined loss of offsite power and accident condition.

During the diesel generator load sequence testing, which is performed at least every 18 months, the design accident load will be tested by simulating a loss of offsite power with a safety injection signal. In this way, the test will simulate the load of Table 8.3-1 as close as practical.

Table 8.3-2 shows the order and time at which loads are automatically and sequentially applied to the diesel generator during a loss of offsite power.

Whenever a tower actuation (TA) signal is received, the cooling tower pumps receive an automatic start signal and the service water pumps are automatically tripped and blocked from starting until the TA signal is reset.

As noted on Table 8.3-1 and Table 8.3-2, either the cooling tower pump or the service water pump, but not both, will be loaded on the diesel generator. Upon loss of offsite power, all service water pumps and cooling tower pumps receive a trip signal. At sequence interval 52 seconds (step 8), both the cooling tower pump and the service water pump receive a start permissive from the EPS. If a TA signal is also present, the cooling tower pump will start; otherwise the service water pump will start.

The diesel generator has been tested and/or analyzed to demonstrate its ability to successfully start a load larger than the 800-hp cooling tower pump at the 52-second loading sequence interval.

During the diesel generator loading process, 4160 and 480-volt emergency bus undervoltage tripping circuits are disabled to prevent inadvertent tripping due to momentary voltage dips caused by application of large motor loads. During the SEPS DGs loading process the under voltage tripping circuits are also disabled. If for any reason the diesel generator breaker trips open during or subsequent to the loading process, undervoltage tripping is restored and the bus is cleared, as in the original loss of offsite power. Upon reclosing of the diesel generator breaker, the loading process is re-initiated and proceeds as before.

The diesel generator is also capable of starting and powering the startup feed pump (SUFP) P-113 when carrying the maximum Train A load listed in Table 8.3-1. In addition, procedures for operating the SUFP on the diesel generator (refer to Subsection 8.3.1.1b.9) will require that the operator verify diesel generator loading to ensure that adequate margin is available for running this pump. There is also an emergency power sequencer interlock to permit SUFP operation on the diesel generator only after load sequencing has been completed. Subsection 8.3.1.1.b.9 also describes a contingency alignment where emergency diesel generator EDG-1A may be aligned to provide power to the non-safety related electrical buses.

In the event of a safety injection signal, the diesel generators are automatically started and operated at idle. Should the offsite power supply subsequently fail, the diesel generators are automatically connected to the emergency buses and the loading sequence as described in Table 8.3-1 is initiated.

Upon receipt of an automatic diesel start signal (LOP or SI) during load testing, the diesel generator breaker is automatically tripped, and the diesel generator continues to run. The diesel generator controls, including voltage regulator control, are automatically returned to the automatic control mode. Should this be accompanied by a loss of offsite power, relays sense the loss of voltage on the emergency bus and respond by initiating the loading sequence described in Table 8.3-1.

If a safety injection signal is received during or after a loss of offsite power load sequencing, the sequencer will reset and resequence all the required engineered safety loads.

The diesel generator is equipped with an auto-tracking manual voltage regulator. As the automatic voltage regulator responds to load variations, the manual voltage level is automatically adjusted to the same level so that in the event of loss of the automatic voltage regulator, voltage transients will be minimized upon switching to manual.

f. Separation of Control Power Sources

The control systems for the Train A Engineered Safety Features are separate and independent from control systems for Train B Engineered Safety Features, as shown in Figure 8.3-3. Control power for the Train A control systems is provided by the Train A 125-volt DC Bus 11A and 120-volt AC vital instrument panel 1A. For Train B, control power is supplied from the 125-volt DC Bus 11B and 120-volt AC vital instrument panel 1B. A detailed description of instrumentation and controls for Engineered Safety Features Systems is included in the description of the respective system.

g. Circuit Protection System

All protective devices and relays are selected on the basis of compatibility with the type of equipment to be protected (motor, transformer, bus) and the equipment characteristics (motor locked rotor current, starting time, etc.). These protective devices are coordinated with the protective devices in adjacent zones to provide selective tripping of breakers.

Relaying schemes for each zone of protection are designed to provide overlap between the zones. This assures that no electrical feeder or equipment is outside a protective relay zone.

Protective relays are specified in accordance with ANSI C37.90 - 1971, "IEEE Standard for Relays and Relay Systems Associated with Electrical Power Apparatus." The following standards are used in developing relaying schemes:

1. ANSI - C37.91 - Guide for Protective Relay Applications to Power Transformers (IEEE 273)

2. IEEE 288 - Guide for Induction Motor Protection

The Following paragraphs contain general design criteria for the sizing of circuit protective devices. Any exceptions to the specified criteria are evaluated within the applicable design calculations.

Safety-related 4000-volt motor long-time overcurrent protection is set at a minimum of 125 percent of the full load current to the extent practical. Motor instantaneous overcurrent protection is set at a minimum of 1.75 times the locked rotor current. When locked rotor current is not available, instantaneous protection is set at a minimum of 12 times full load current.

The design criteria for the selection of thermal overloads is that all motors will be equipped with thermal overload protection, which will be selected so as to protect the motor against failure in the event of an overload condition. Ambient compensated thermal overload relays are used to protect both continuous and short-time rated motors that are connected to motor control centers. These ambient compensated thermal overload relays are responsive to current and their time-current characteristics are, for practical purposes, unaffected by temperature variations.

The unit substation breakers which feed motor control centers have overloads set at the bus rating of the MCCs (600 amps) except for MCC E511 that has its overload set at 480A and MCCs E522 and E622 that have their overloads set at 240A for coordination purposes. For safety-related motors fed from unit substations, the feeder breaker overload setting is a minimum of 125 percent of the full load current times the equipment service factor. For non-safety related motors fed from unit substation motors, the overload setting is a minimum of 125 percent of full load current.

The trip point setting criteria for engineered safety features motors connected to motor control centers is as follows:

1. For motors rated for continuous duty, the trip setpoint of the thermal overload relay is selected so that it is unresponsive to currents below 125 percent of the nameplate full load current times the service factor.

2. For short-time rated motors such as motor-operated valves (MOVs), the trip setpoint is determined by establishing the values for motor nameplate full load and locked rotor current, thermal time limit for carrying locked rotor current, and the actual stroking time of the MOV. The motor trip set point is then determined so that:

   (a) When carrying a current equal to nameplate full-load current times the service factor, the motor will not trip in a time period less than three times the MOV stroking time.

   (b) When carrying locked rotor current, the thermal overload relay should actuate in a time within the motor's limiting time for carrying locked rotor current.

If it is not possible to achieve both (a) and (b) due to relay characteristics, then condition (b) will be relaxed. Condition (a) will not be compromised in any circumstances.

During preoperational testing, the tripping times and currents were measured to verify the accuracy and repeatability of a representative sample of thermal overload relays as follows:

1. The motor was operated to verify that the actual current does not exceed the nameplate full-load current which was used in determining the trip setpoint. Should the actual current exceed the nameplate current, the higher current was evaluated and, if acceptable, the trip setpoint was recalculated based on the actual current measured during dynamic testing.

2. Preselected values of currents were applied to the thermal overload relay. If the relay operating time was in accordance with the criteria established and the relay curve, the relay was considered satisfactory.

Since the plant began operation and at intervals specified in the Technical Specifications, a representative sample of at least 25 percent of the thermal overload relays used on the motor-operated valves listed in the Technical Requirements Manual (TR-14) will be selected for testing and calibration in accordance with the requirements of technical specification subsection 3/4.8.4.3. The selected relays will undergo a testing program at the maintenance shop or other designated location prior to the 18-month test interval. A preselected value of current will be applied and it will be observed that the relay operates (it opens the circuit) in accordance with the criteria established and the relay curve. These pretested relays will then be installed in the selected circuits (see Technical Specification Subsection 3/4.8.4.3) whose thermal overload relays have been removed. The functional operation of the circuit will then be tested (i.e., valve closed or open) with the new relay in place. These test procedures comply with Position C2 of Regulatory Guide 1.106 (Rev. 1).

Because of the nature of application of the continuous duty motors such as pumps, fans, etc., misoperation or misapplication of thermal overload relays will be detected in time by either alarm or other process signals and corrective action will be taken. Therefore, there will be no specific surveillance procedures for periodically testing the thermal overload relays used with the continuous duty motors located outside the containment. These relays will be inspected as part of the plant's regular maintenance program under ANSI N18.7-1976. In addition, redundant motors that are in a standby status will be periodically rotated so that any abnormal condition will be detected.

The thermal overload protection for continuous duty motors located inside containment is part of the design provided to satisfy the requirements of Regulatory Guide 1.63 for containment electrical penetrations. These thermal relays will be periodically tested as defined by Technical Specification 4.8.4.2.

h. Equipment Grounding

Copper, copperweld cable and copper bus provide low resistance ground paths wherever electrical equipment is located. All electric equipment and nonelectrical conductive material such as structures, enclosures, tanks, and raceways are grounded in conformance with IEEE Standard 142-1972 and IEEE Standard 80-1971. The building grounding system is provided with adequately sized ground cables for peripheral connections to the station ground grid.

The method of system grounding utilized at the various voltage levels is discussed in the applicable sections.

i. Safety-Related System Motor Selection

All motors are sized for continuous operation of the running load and operate successfully at 90 percent of rated motor voltage. Motors are capable of starting their rated loads with 80 percent voltage and 95 percent frequency at the motor terminals. The system design and diesel generator specification assures that this voltage will be present at the motor terminals when needed. The calculated continuous brake horsepower is not greater than 95 percent of the horsepower rating of the motor. The starting torque for the motor is based on the inertia and speed-torque characteristics of the driven equipment. Exceptions to the above criteria are evaluated on a case-by-case basis to show that there are adequate voltage and horsepower to start and operate the load.

The motor-torque curve, at its closest approach to the load-torque curve, and at the required starting voltage, is greater than the torque required by the load at that speed. This permits the motor to develop a margin of torque over that required by the load to ensure successful starting and acceleration. The insulation system for motors is NEMA Class B, as a minimum, with the actual insulation class selected on the basis of environment and service conditions in which the motor is required to operate. The factors taken into consideration in selection of the insulation system are resistance to radiation, resistance to moisture, resistance to chemicals, ambient temperature and pressure. The motor enclosure is selected to protect against adverse environmental conditions. Winding temperature detectors and bearing thermocouples are provided on large motors to alarm high temperature conditions.

The motor suppliers are required to verify that actual test data confirm that the torque margin is equal to or greater than that of the calculated data. If this requirement is not met, as an alternate, the vendor is required to furnish an analysis showing that margin between motor torque and load torque is adequate to accelerate the load within an acceptable time period. A further check of motor capability was the preoperational testing conducted at the site under plant light-load conditions to simulate the maximum voltage practically obtainable, and under plant heavy-load conditions to simulate the minimum voltage practically obtainable (reference Subsection 14.2.7, exceptions to Regulatory Guide 1.68).

j. Provisions for Periodic Testing and Maintenance

The onsite AC distribution system for engineered safety features loads is designed and installed to permit periodic inspection and testing in accordance with General Design Criterion 18, IEEE Standard 308, Regulatory Guide 1.118, and IEEE 338 (except as noted in Subsections 8.1.5.2 and 8.1.5.3) to ensure:

1. The operability and functional performance of the components of the system, and

2. The operability of the system as a whole under design conditions.

Switchgear and accessories for the Auxiliary Power System are easily accessible for inspection and testing.

The 13.8-kV, 4160-volt and 480-volt switchgear circuit breakers may be tested when the individual equipment is de-energized. The breakers can be placed in the test position and tested functionally.

The first and second level undervoltage schemes (see Subsection 8.3.1.1b.4) are designed to permit periodic testing during normal plant operation.

Breakers for engineered safety features auxiliaries are exercised on a schedule similar to that for the auxiliaries controlled by the breakers. Transfer schemes can be exercised during normal operation, or by simulation of the necessary conditions. Timing checks can be performed on transfer schemes. Protective relays are provided with test plugs or test switches to permit testing and calibrating the devices.

Containment penetration conductors overcurrent-protective devices are periodically tested according to the requirements of the Technical Specifications.

The control circuits of the emergency diesel generators are designed to permit testing during operation of the plant as well as while the plant is shut down. Periodic tests are performed to demonstrate the availability and capability of the unit to perform its intended function. These tests are performed in accordance with Regulatory Guide 1.108, with clarification as explained in Subsection 8.1.5.3.

This periodic testing also provides input to the Maintenance Rule EDG performance criteria which support the target reliability levels required for the Station Blackout analysis (see Section 8.4.2).

Station procedures require fully loaded operation during routine diesel surveillance testing. This guards against the accumulation of incomplete combustion product buildup in the engine and exhaust systems.

During extended no-load or light-load operation (less than 20% load), the diesel will be loaded to a minimum of 50% load for one hour following each six hours of continuous no-load or light-load operation.

During troubleshooting, no-load operation will be minimized. If the troubleshooting operation takes place over an extended period of time (i.e., greater than four hours), the engine will be loaded to a minimum of 25% load for at least 30 minutes.

The Emergency Power Sequencer (EPS) is designed to permit periodic testing of the sequencer logic during operation of the plant. During the EPS test, combinations of EPS inputs are simulated and the corresponding EPS outputs are verified. During this testing, the continuity of the actual EPS output relay coils is verified and the accuracy of the interval between each sequence step is determined. If a bona fide EPS input is received during EPS testing, testing ceases and the EPS automatically performs its design function.

Every 18 months the actual input and output relays of the EPS are tested as part of the diesel generator load testing program.

k. Lightning Protection

Adequate lightning protection for all structures is provided by a system of air terminals, coursing and down conductors which are connected to the Plant Grounding System.

Lightning protection for the onsite AC distribution system is provided by a combination of metal-enclosed bus design and careful placement of lightning arrestors in the critical areas.

From the point where the 345-kV transmission lines drop down to make the transition to gas-insulated metal-enclosed bus, all outdoor power buses at 345 kV, 25 kV, 13.8 kV and 4.16 kV are of metal enclosed design. These metal enclosures are solidly connected to the plant ground system.

Lightning protection for the 345 kV overhead transmission lines is provided by static wires on top of the transmission line poles. Lightning protection for equipment in the air termination yard, including the air bushings for the gas-insulated metal-enclosed bus, is provided by poles with air terminals mounted in the yard.

In addition, lightning arrestors are provided in the following locations:

1. 25-kV isolated phase bus duct

2. At the transition structures where the overhead 345-kV lines end and the gas insulated bus begins, and at the end of the gas-insulated bus before its connection point to the switching station (line side of the motor-operated disconnect switch)

3. On the primaries of all 480-volt unit substation transformers

These lightning arrestors, together with the protection of medium and high voltage conductors afforded by the grounded metal enclosures, minimize the likelihood of a lightning strike which could jeopardize the onsite power system.

8.3.1.2 Safety Analysis of the Onsite AC Power System

a. Compliance with Applicable General Design Criteria

1. Criterion 17 - Electric Power Systems

The onsite AC power system is designed to permit the functioning of structures, systems, and components important to safety under all normal and accident conditions. The system provides sufficient capacity and capability to assure that specified fuel design limits and design conditions of the reactor pressure boundary are not exceeded as a result of anticipated operational occurrences, and that the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite AC system has sufficient independence, redundancy, and testability to perform its safety functions assuming a single failure. Independence is provided by physical separation of components and cables to minimize vulnerability of redundant engineered safety features systems to single credible accidents. Systems and components which comprise the onsite AC distribution system have been designed to afford maximum in-service testability. Where in-service testability cannot be provided due to adverse impact on plant operation, systems and components are tested during plant shutdown.

The onsite AC source of electrical power consists of two diesel generators, one connected to each of the redundant emergency buses. One diesel generator is capable of supplying sufficient power for the operation of the minimum safety features required during a postulated loss-of-coolant accident and/or loss of offsite power.

During a postulated loss-of-coolant accident, each diesel generator starts automatically on a safety injection signal and, if offsite power is not available, it is connected to its associated emergency bus. The safety feature equipment is then sequentially started.

The Supplemental Emergency Power Supply is a defense-in-depth power source that is beyond the requirements of GDC-17.

There are no ties between redundant engineered safety features load groups. See Section 8.3.1.1.b.6 for a discussion of the SEPS system connections to the emergency buses.

A failure mode and effect analysis of the onsite AC power system is presented in Table 8.3-3.

2. Criterion 18 - Inspection and Testing of Electrical Power Systems

Class 1E electric equipment is designed and located to permit appropriate periodic inspection and testing to assure availability of systems and condition of components, in line with the provisions for testing and maintenance listed in Subsection 8.3.1.1j.

These tests will assure the operability and functional performance of the components, and the operation of the system as a whole.

During plant shutdown, and under conditions as close to normal operation as practical, the full operational sequence that brings the system into operation including portions of the protection system and transfer of power among various offsite and onsite power supplies will be tested.

During reactor operation, the capability to transfer power from the unit auxiliary transformer source to the reserve auxiliary transformer source is continuously monitored. Alarms are provided in the main control room to alert the operator if synchronism is lost between the switchgear and the reserve source or if control power is lost to the reserve source circuit breaker.

Transfer of power from the unit auxiliary transformer source to the reserve auxiliary transformer source is not periodically tested at power because such transfers may introduce unwarranted challenges to the electric power system that may result in a plant trip.

For a discussion of transfer of power initiated by operation of the generator breaker, see Subsection 8.2.1.3.e.1.

3. Criterion 5 - Sharing of Systems or Components Between Units

Seabrook Station is a single unit plant; therefore, electrical structures, systems and components important to safety in the onsite power system are not shared.

4. Criteria 33, 34, 35, 38, 40, 41, 44

Onsite electric power interconnection and transfers are designed so that the safety functions of the Reactor Coolant Makeup System, the Residual Heat Removal System, the Emergency Core Cooling System, the Containment Heat Removal System, the Containment Atmosphere Cleanup System and the Cooling Water System can be accomplished, assuming a single failure. Two independent power trains at each distribution voltage level supply redundant load groups with power during normal, abnormal and post-accident conditions. These load groups comprise engineered safety features and protection systems in such a way that loss of one group does not prevent the minimum safety function from being performed.

5. Criterion 2 - Design Basis for Protection Against Natural Phenomena

   (a) The components of the onsite AC power system are located in seismic Category I structures which provide protection from the effects of tornadoes and external floods, and other natural phenomena.

   (b) These components are Class 1E.

   (c) These components have been designed to be fully qualified for the seismic and natural environmental conditions appropriate to their location (see Section 3.11).

6. Criterion 4 - Environmental and Missile Design Bases

   (a) The components of the onsite AC power system are located in seismic Category I structures which provide protection from the effects of tornado missiles, turbine missiles and other events and conditions which may occur outside the nuclear power unit.

   (b) These components are Class 1E.

   (c) These components are designed to accommodate the effects of, and to be compatible with or are protected against, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents including loss-of-coolant accidents. Criteria are presented in Chapter 3. Environmental conditions are presented in Chapters 3 and 6.

   (d) These components are protected, as appropriate, against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures (postulated accidents).

b. Compliance with Applicable Regulatory Guides

1. Regulatory Guide 1.6 - Independence Between Redundant Standby Power Sources

Two diesel generators constitute the AC standby power sources. Each diesel generator serves as the standby power supply for a redundant load group. In addition, each redundant load group is connected to the preferred (offsite) power supply through different UATs or RATs. The design is based on the concept of independent, redundant groups of engineered safety feature loads and, as such, one redundant load group or power source is never automatically connected to the other redundant load group or power source. (See Figure 8.3-1 and Figure 8.3-3.)

The Supplemental Emergency Power System (SEPS) uses a transfer switch that has the capability of connecting the non-safety related SEPS DGs to a safety related circuit breaker on either redundant load group. The switch consists of three cubicles, an incoming termination section and two switching sections. The termination cubicle is the center section which contains the connections for the incoming cables and bus bar which connects to the line side of the two disconnect switches located in the outer cubicles. A mechanical interlock is used that allows operation of only one switch closure at a time. Unless in service to supply an emergency bus, the redundant load group circuit breakers which connect to the SEPS system are open. The Train B breaker is normally racked in and the Train A breaker is normally racked out. A Kirk key interlock is provided to prevent both breakers from being racked in at the same time. The racked out breaker and Kirk key interlock provides the Regulatory Guide 1.6 required interlock to prevent connecting redundant load groups together.

2. Regulatory Guide 1.9 - Selection of Diesel Generator Set Capacity for Standby Power Supplies

Each diesel generator set has been selected on the basis that the total running load at any time will not exceed the short time rating of the diesel generator.

During preoperational testing, the maximum continuous load demand has been verified by tests.

Each diesel generator is capable of starting and accelerating all Class 1E loads to rated speed, in the required sequence.

The excitation and governor controls furnished for the diesel generators are designed so that, during and after sequential loading of Class 1E loads, the voltage and frequency decrease to no less than 80 percent and 95 percent of nominal values, respectively.

The voltage may dip to less than 80 percent of nominal value when the diesel generator breaker closes and energizes the 1000/1333 kVA, 4160/480V unit substation transformers. The diesel generators are designed to recover from this dip (due to transformer magnetizing inrush current) to at least 80 percent of nominal value in about 6 cycles, causing a negligible delay to the acceleration of the first load group.

Voltage and frequency are restored to within 10 percent and 2 percent of nominal values in less than 60 percent of each load sequence time interval.

During recovery from transients caused by step-load increases, or resulting from the disconnection of the largest single load, the increase in speed of the diesel generator set will not exceed 75 percent of the difference between nominal speed and the overspeed trip set point, which is set at 110.7 percent of nominal speed.

Diesel generator protective trips, other than engine overspeed, generator differential current, 4.16-kV bus fault and low lube oil pressure, are bypassed during accident conditions. For more information on design and testing of the bypass circuitry, refer to Subsection 8.3.1.1e. Generator overcurrent and reverse power have a common alarm in the main control room. Loss of field, high lube oil temperature, and high jacket coolant temperature each have a separate alarm in the control room. The station computer provides information as to which protective trip is activated first.

Periodic testing of the diesel generator is in accordance with Regulatory Guide 1.108, with clarification as explained in Subsection 8.1.5.3.

3. Regulatory Guide 1.32 - Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants

Two immediate access circuits are available from the transmission network to the Class 1E Emergency Distribution System.

4. Regulatory Guide 1.63 - Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants

The electric penetration assemblies are designed to withstand, without loss of mechanical integrity, the maximum fault current vs. time conditions that could occur as a result of single random failures of circuit overload devices. The 600-volt system X/R ratio used in specifying the electrical penetrations is 4. Calculations show that this value is conservatively applied because the actual ratio is considerably less than 4.

To preclude damage to electric penetrations due to single failures of circuit overload protection devices, each penetration circuit, with the exception of CRDM, 15-kV RCP, instrumentation and low energy circuits, is provided with dual Class 1E overload protective devices. Seismically qualified Class 1E fuses protect 15-kV RCP penetrations. Additional protection is provided by two non-Class 1E breakers in series. These breakers are coordinated and derive their control power from different batteries. For more details refer to Subsections 8.3.1.1a and 8.3.1.1c.

5. Regulatory Guide 1.75 - Physical Independence of Electric Systems

The design is consistent with the criteria for physical independence of electric systems established in Attachment C of AEC (NRC) letter dated December 14, 1973. Attachment C which is incorporated as Appendix 8A, is in general conformance with Regulatory Guide 1.75.

For clarification of position C4 as it relates to associated circuits, refer to Updated FSAR Subsection 8.1.5.3b.

The Seabrook cable and raceway separation criteria (see Updated FSAR Subsection 8.3.1.4) is a combination of the standard criteria given in Attachment C of AEC Letter dated December 14, 1973 (see Updated FSAR Appendix 8A) and IEEE 384-1974 and criteria established by analysis and testing as permitted by Attachment C and IEEE 384-1974.

Physical separation and identification of circuits are described in detail in Subsections 8.3.1.3 and 8.3.1.4, respectively.

c. Compliance to Branch Technical Position PSB-1 - Adequacy of Station Electric Distribution System Voltages

1. Position B1

An acceptable alternative to the second level undervoltage protection system described in Position 1 is provided. This alternative system is described in Subsection 8.3.1.1b.4(b).

2. Position B2

The Seabrook Station design meets Position 2 of Branch Technical Position PSB-1 except as noted below. The bypass of the load shedding feature during sequencing, and its restoration in the event of a subsequent diesel generator breaker trip, is discussed in Updated FSAR Subsection 8.3.1.1e.6. In addition, the load shed feature is reinstated after load sequencer action when the operator resets the sequencer override pushbutton. This action permits the operator to reassume control of diesel generator loading.

Position 2 specifies that the Technical Specifications must include a test requirement to demonstrate the operability of the automatic bypass and reinstatement features at least once per 18 months during shutdown. During development of Seabrook Station's Technical Specifications, the NRC deleted this test requirement.

3. Position B3

The voltage regulation study was performed to meet the requirements of Position 3. The voltage levels at the safety-and nonsafety-related buses are optimized for full load and minimum load conditions that are expected throughout the anticipated range of voltage variations of the offsite power source by appropriate adjustment of the voltage tap settings of the station transformers.

4. Position B4

The analytical techniques and assumptions used in the voltage analysis cited in item 3 (Position B3) above have been verified by actual measurement as part of the pre-operational test program. The guidelines of Position 4 of Branch Technical Position PSB-1 have been followed and good correlation between the analytical results and the test results have been demonstrated.

Seabrook Station's commitment to perform this testing is also described in Subsection 14.2.7, which describes Seabrook Station's interpretation of Regulatory Guide 1.68, Appendix A, Section 1.g.

d. Environmental Effects on Electric Equipment

All equipment that must operate in a hostile environment during and/or subsequent to a design-basis event are identified with their ambient environmental conditions, and their qualifications are discussed in Section 3.11.

e. Effects of Submergence on Electrical Equipment

Analysis has been performed to determine the effects of submergence as a result of a LOCA on electrical equipment. The results of this study indicate no detrimental effect upon the Class 1E electrical power sources as a result of submergence of electrical equipment following a LOCA.

8.3.1.3 Physical Identification of Safety-Related Equipment

All cables, raceways and safety-related equipment are assigned to a particular channel or train. There are two redundant trains of power and controls, and four redundant channels of instrumentation. Each channel or train is assigned a particular color, as shown below:

Separation Group		Equipment Nameplate	Raceway Tag	Cable Color
A.	Channel I and Train A	Red	Red	Red
	Train A Associated	Black		Black w/Red Tracer
B.	Channel II and Train B	White	White	White
	Train B Associated	Black		Black w/White Tracer

Separation Group	Equipment Nameplate	Raceway Tag	Cable Color
C. Channel III	Blue	Blue	Blue
D. Channel IV	Yellow	Yellow	Yellow

The equipment nameplate colors described above represent the color assigned to identify each separation group. In the original nameplate design, the nameplate background color was used to identify the separation group. As a result of labeling improvements, including the addition of bar codes, a redesign of the background color was required. Newer nameplates may use different methods, such as black letters on a white background with a border color that identifies the separation group. In this way, the same basic separation group color is maintained for different nameplate styles.

Each piece of electrical equipment is marked with the node number indicated on the design drawings, in the particular color corresponding to the channel or train to which that equipment is assigned. Similarly, trays and exposed conduits are marked with color-coded markers. The cable jacket color code serves as its identification. The operator or maintenance craftsman needs only to observe the color of the nameplate of any piece of equipment or the cable jacket color to determine which channel or train it serves. For exceptions to the above cable and raceway identification criteria, see Subsection 8.3.1.4k. For additional information on physical identification of safety-related equipment, see Subsection 7.1.2.3.

8.3.1.4 Independence of Redundant Systems

a. General

Seabrook Station complies with the requirements of Updated FSAR Appendix 8A, IEEE 384-1974 and Regulatory Guide 1.75, Rev. 2. These documents describe acceptable methods of complying with IEEE 279-1971 and Criteria 3, 17 and 21 of Appendix A to 10 CFR Part 50 with respect to the physical independence of the circuits and electrical equipment comprising or associated with the Class 1E power system, the protection system, systems actuated or controlled by the protection system, and auxiliary or supporting systems that must be operable for the protection system and the systems it actuates to perform their safety-related functions. Preservation of independence of redundant systems within the control boards and all other field-mounted racks is discussed in Subsection 7.1.2.2.

In accordance with the provisions of Section 4.5a and Subsection 4.6.2 of Updated FSAR Appendix 8A, Section 4.5(1) and Subsection 4.6.1 of IEEE 384-1974, and Position C4 of Regulatory Guide 1.75, Revision 2, we have elected to associate all of the non-Class 1E circuits with Class 1E circuits. This application of associated circuits allows the plant to be designed with one less separation group; that is, instead of having five separation groups consisting of four safety-related separation groups and one nonsafety-related separation group, Seabrook Station has only four separation groups. The major advantages of this approach are the ability to provide greater separation distances between the groups, as well as to reduce the raceway system's exposure to fire.

As a result of this design, all plant circuits are specifically assigned to one of the following four separation groups as noted in Figure 8.3-55:

Group A - Train A, Channel I and Train A Associated Circuits

Group B - Train B, Channel II and Train B Associated Circuits

Group C - Channel III

Group D - Channel IV

The great majority of associated circuits are with Group A, a very limited number are with Group B, and none are with Groups C and D.

The circuits that are associated with Train A consist of:

1. Non-Class 1E power, control, instrument circuits contained within the Nuclear Island

2. Non-Class 1E power, control, and instrumentation circuits that traverse the Nuclear Island boundary

3. Non-Class 1E power, control, and instrument circuits outside the Nuclear Island.

The circuits that are associated with Train B consist of:

1. Non-Class 1E power, control, and instrument circuits contained within the Nuclear Island

2. Non-Class 1E power, control, and instrumentation circuits that traverse the Nuclear Island boundary.

3. Non-Class 1E power, control and instrumentation circuits outside the Nuclear Island.

The Nuclear Island boundary is shown in Figure 8.3-56. This figure denotes the buildings, structures, duct banks, etc., which are part of the Nuclear Island. All other buildings, structures, etc., are considered to be outside the Nuclear Island.

The four separation groups are routed through four separate raceway systems per the separation criteria given in Table 8.3-10. These separation criteria are based on a combination of the following:

1. Standard separation criteria given in Subsections 5.1.3, 5.1.4, and Section 5.6 of Updated FSAR Appendix 8A and IEEE 384-1974

2. Separation criteria established by analysis and testing as permitted by Subsection 5.1.1.2 and Section 5.6 of Updated FSAR Appendix 8A and IEEE 384-1974. This analysis and testing are documented in References 1 through 5 (see Updated FSAR Subsection 8.3.4).

Cable, raceway and internal wiring installations which do not meet the separation criteria given in Table 8.3-10 are analyzed based on the analyses and testing documented in References 1 through 6 to show that the lesser separation distances do not compromise the ability to achieve a safe plant shutdown under design basis event (DBE) conditions. These analyses are documented in the appropriate design basis documents.

Separation criteria for temporary cables have also been developed based on the analysis and testing documented in References 1 through 5. These temporary cable separation criteria are contained in the appropriate design and operating documents.

The following analysis examines the design features and modes of failure of associated circuits of each separation group to determine any interaction and challenges with other separation groups. The overall objective is to assure that the ability to achieve a safe plant shutdown under design basis event (DBE) conditions is not compromised.

There are two (2) classifications of associated circuits, those that directly interface with a Class 1E circuit and those that do not directly interface with a Class 1E circuit. An associated circuit has a direct interface with a Class 1E circuit if it shares power supplies, enclosures or raceways, or if it does not meet the minimum separation criteria. Only associated circuits that directly interface with Class 1E circuits have the potential to degrade a Class 1E circuit. This type of associated circuit is provided with at least one protective device to prevent degradation of the Class 1E circuit unless it can be shown by analysis that failure of the associated circuit will not degrade the Class 1E circuit. These protective devices are required to perform their current interrupting function to prevent failure of the associated circuit, which has a direct interface with a Class 1E circuit, from degrading that Class 1E circuit. The special design, procurement and testing requirements imposed on these protective devices are described below.

Devices such as control power transformers and instrument power supplies which limit the current available to a circuit to less than a cable's ampacity are considered low energy sources. Because of this current limitation, protective devices are not required for circuits supplied by low energy sources since there is no potential to degrade Class 1E circuits.

Cables for both Class 1E circuits and associated circuits are procured to the same requirements including applicable environmental qualification. Exceptions to the cable procurement practices are described in Subsection 8.3.1.4f.

When Class 1E power supplies are used for an associated circuit, failure of a non-Class 1E motor, load, or device connected to this power supply will be promptly isolated by operation of Class 1E protective devices. The Class 1E protective devices protecting these non-Class 1E loads are coordinated so that failure of all nonqualified, non-Class 1E loads, with proper operation of their own breakers, will not result in tripping of the incoming breaker to the Class 1E bus.

When non-Class 1E power supplies are used for an associated circuit that directly interfaces with a Class 1E circuit, non-Class 1E protective devices perform a current interrupting function. These non-Class 1E protective devices are of a similar design as their Class 1E counterparts. While these protective devices are not procured as environmentally or seismically qualified, sufficient controls are imposed in the procurement process to ensure that these protective devices will be capable of performing their current interrupting function. These non-Class 1E protective devices will be tested to ensure operability of their current interrupting function when such testing is nondestructive in nature. Periodic testing of the current interrupting function for non-Class 1E protective devices with associated circuit requirements will be performed on a ten-year frequency in accordance with Station procedures. Protective devices, such as fuses, which cannot be nondestructively tested will have their current interrupting function verified on a sampling basis during the procurement process prior to installation in the plant.

Since Class 1E and non-Class 1E protective devices are similar, any generic degradation such as setpoint drift, manufacturing deficiencies, and material defects will be detected and corrected as a result of the Station's procurement, maintenance and trending programs for the Class 1E protective devices and the non-Class 1E protective devices with associated circuit requirements.

Based on the above analysis, the protective devices for directly interfacing associated circuits can be assumed to function properly under design basis conditions. The following provides additional specific analysis for associated circuits contained within the Nuclear Island, traversing the Nuclear Island and those completely outside the Nuclear Island.

The SEPS equipment is associated with Train B because SEPS is normally aligned to Train B and plant design criteria requires that all equipment non-safety related be associated to either train. However, SEPS output can be connected to either Train A or Train B depending on the distribution alignment. When SEPS output is aligned to either Train A or B, the SEPS design is such that the output circuit is completely isolated from the other train and a single failure will not adversely impact the opposite train. A single train alignment is accomplished through strict administrative controls and use of key interlocks during SEPS output feeder alignment. Based on the system design, the SEPS output feeders can be aligned to Train A even though the SEPS is associated with Train B. Also, since the event for which SEPS is used to provide power to an emergency bus is considered a beyond design event, the loads supplied by the emergency bus would not be performing a safety related function and the emergency bus would not be considered operable when powered from SEPS.

b. Train A Associated Circuit Analysis

1. Associated Circuits Contained within the Nuclear Island that Have Direct Interface with Class 1E Circuits

Non-Class 1E circuits that remain within the Nuclear Island are permitted to share the same raceway as Train A Class 1E circuits. These circuits are classified as Train A Associated Circuits and are designed and installed to meet all the requirements placed on associated circuits as required by the compliance documents listed earlier.

Challenges to Class 1E circuits, because of failure in an associated circuit, have been examined and determined to have no detrimental effect because of the following:

(a) Mounting of non-Class 1E power supplies (such as switchgear, motor control centers and distribution panels) within the Nuclear Island is similar to the mounting of their Class 1E counterparts; therefore, credit can be taken for this equipment to perform its current interrupting function under DBE conditions. Mounting of fuses as part of vendor packages may not be similar to the mounting of similar Class 1E fuses. However, there is no credible failure mechanism which could prevent these fuses from performing their current interrupting function under DBE conditions.

(b) The probability of an ensuing fire is minimized because all cables used for associated circuits that have direct interface with Class 1E circuits are specified, designed, manufactured, and installed to the same criteria as Class 1E cables. Factors that have been taken into consideration include flame retardancy, nonpropagating and self-extinguishing properties, splicing restrictions, appropriate limitations on raceway fill, cable pulling and termination requirements, appropriate cable derating, and environmental qualifications. The above provisions and considerations used for the associated circuits during the construction phase of the plant are also used during the operations phase.

(c) Degradation of an associated circuit because of a raceway failure during a DBE, has been eliminated because all electrical raceway systems within the Nuclear Island are seismically analyzed.

(d) Other design considerations that contribute to the integrity of these associated circuits are as follows:

(1) Cables associated with one train are never routed in raceways containing Class 1E or associated cable of another train or channel.

(2) All cables for instrumentation circuits use shielded construction which minimizes any unacceptable interaction between Class 1E and associated circuits.

(3) All circuits entering the reactor containment are provided with protective devices complying with Regulatory Guide 1.63. For exceptions see Subsection 8.3.1.1c.7(a).

Based on the above design features and analysis, we do not consider these associated circuits to pose any challenges to any Class 1E circuits. Therefore, the ability for safe plant shutdown under DBE conditions has not been jeopardized.

2. Train A Associated Circuits that Traverse the Nuclear Island Boundary and Have Direct Interface with Class 1E Circuits

For analysis purposes, the associated circuits that traverse the Nuclear Island boundary can be further subdivided into two basic types: (a) those that have their protective device located in the Nuclear Island, and (b) those that have their protective device outside the Nuclear Island. It should be noted that there are a limited number of power cables in these categories.

(a) Associated Circuits that Have Protective Devices Located in the Nuclear Island and that Have Direct Interface with Class 1E Circuits

These circuits are also designed and installed to meet all the requirements as outlined above. Though the raceway system outside the Nuclear Island is not seismically analyzed, this is of no concern because the circuit protective devices inside the Nuclear Island are assumed to perform their protective function. Concerns that design-basis events such as a seismic event may cause high voltage cables that are not in seismically analyzed raceways and not located in Category I buildings to interact with lower voltage cables are analyzed below.

Seismic tests, performed on raceways representing typical installations on SEP plants, proved that the raceways can withstand seismic events with no significant failures. Since the typical nonseismic installation at Seabrook is superior to the tested SEP installations, it can be assumed that they will survive a seismic event. Failures of raceways resulting from collapse of the nonseismically designed buildings can be dismissed because the conservative criteria and UBC seismic loading used in the construction of the building will ensure little likelihood of collapse.

Not withstanding the preceding, any event involving the raceway system that can cause a higher voltage cable to come in contact with another lower voltage cable will first cause the higher voltage cable to be grounded. Contributing factors to this are: (1) the cables are in grounded metallic trays or enclosures, (2) the 13.8-kV and 4.16-kV power cables are of armored construction, and (3) as indicated in Figure 8.3-55, separate raceways are designated for the different voltage levels.

A ground fault in the low resistance grounded 13.8-kV system will cause protective circuit breakers to open and isolate the fault. In the high resistance grounded 4-kV and 480V system, although a single ground fault will not cause circuit breaker operation, it is highly probable that under such a failure, the faults will be such to cause breaker operation.

In view of the above design considerations and analysis, any possible interaction between cables of different voltage levels is deemed nonexistent.

It is, therefore, concluded that the ability for safe plant shutdown under DBE conditions will not be jeopardized by these associated circuits.

(b) Associated Circuits that Have Protective Devices Outside the Nuclear Island and that Have Direct Interface with Class 1E Circuits

Protective devices outside the Nuclear Island which are not located in a LOCA environment are similar in design to Class 1E devices except for seismic requirements. Other design basis events such as pipe break, fire, flood, etc., will not cause failure of the protective device located outside the Nuclear Island simultaneously with the failure of load which is located in the Nuclear Island. Hence, credit can be taken for their proper operation.

Although the protective devices might not be in Category I buildings, they are similar in design to the Class 1E devices and, based on operating experience of protective devices that have been subject to actual and simulated seismic conditions, it is highly probable that the protective devices will maintain their structural integrity and perform their function.

However, if one postulates their misoperation under a seismic event, such an event is likely to disable the power source itself which is also not seismically qualified.

Analysis of concerns on interaction between cables of different voltage levels is shown in 8.3.1.4b.2(a) above.

We conclude, therefore, that these circuits will not degrade Class 1E circuits, since the non-Class 1E power supply will be lost and all non-Class 1E equipment will become de-energized.

For the above reasons, the ability for the safe plant shutdown under DBE conditions will not be jeopardized by these circuits.

3. Train A Associated Circuits Completely Outside the Nuclear Island that Have Direct Interface with Class 1E Circuits

There are no Train A associated circuits completely outside the Nuclear Island that have a direct interface with a Class 1E circuit. Therefore, there are no special requirements related to separation for the protective devices for these associated circuits. As described in UFSAR Section 7.2.1.1, there are some Class 1E circuits with a portion of their route in the Turbine Building, which is outside the Nuclear Island. The associated circuits completely within the Turbine Building are not directly interfacing circuits because they meet the minimum separation criteria to these Class 1E circuits. Inherent in application of the separation criteria is the use of flame retardant cable. Therefore, the cables for the associated circuits within the Turbine Building must be flame retardant unless it can be shown by analysis that the lack of such flame retardancy will not degrade Class 1E circuits.

c. Train B Associated Circuit Analysis

1. Associated Circuits Contained Within the Nuclear Island that Have Direct Interface with Class 1E Circuits

Non-Class 1E circuits that remain within the Nuclear Island are permitted to share the same raceways as Train B Class 1E circuits. These circuits are classified as Train B associated circuits and are designed and installed to meet all the requirements placed on associated circuits as required by the compliance documents listed earlier in Subsection 8.3.1.4a. Therefore, using the analysis performed for Train A associated circuits, we conclude that the ability for the safe plant shutdown under DBE conditions will not be jeopardized by these circuits.

2. Associated Circuits that Traverse the Nuclear Island Boundary and Have Direct Interface with Class 1E Circuits

For analysis purposes, the associated circuits that traverse the Nuclear Island boundary can be further subdivided into two basic types: (a) those that have the protective devices located in the Nuclear Island and (b) those that have their protective devices outside the Nuclear Island.

(a) Associated Circuits that have Protective Devices Located in the Nuclear Island and Have Direct Interface with Class 1E Circuits

There are very few Train B associated circuits that traverse the Nuclear Island boundary. These circuits are unavoidable either because of plant design constraints, such as the need for interlocks and permissives for the preferred power supply circuits to Train B emergency buses, or because of features provided to improve plant reliability, such as power supply and control for the station service air compressors fed from Train B buses. The portion of these circuits which are outside the Nuclear Island are routed in dedicated embedded or exposed conduits; therefore, the potential or harmful interactions with other associated circuits or other voltage level cables is minimized. For applicable details on the interaction between cables of different voltage levels, see analysis under Subsection 8.3.1.4b.2(a).

The design features described in Subsection 8.3.1.4c.1 for associated circuits contained within the Nuclear Island are also applicable to these circuits. Though the conduit system outside the Nuclear Island is not seismically analyzed, this is of no concern because the circuit protective devices located in the Nuclear Island are assumed to perform their protective function.

Based on the above, we conclude that the ability for the safe plant shutdown under DBE conditions will not be jeopardized by these few circuits.

(b) Associated Circuits that have Protective Devices Outside the Nuclear Island and Have Direct Interface with Class 1E Circuits

The only circuits under this category are the 15-kV cables to the reactor coolant pumps for motor feeders and potential transformers. These interlocked armor cables are routed in embedded conduit outside the Nuclear Island and are in dedicated seismically analyzed raceway systems in the Nuclear Island. Furthermore, the portion of the circuit entering the containment is protected by qualified fuses located in the electrical penetration area which would open the circuit in the event of a catastrophic failure of a reactor coolant pump.

The 15-kV cables used on these circuits meet all the construction and material requirements placed on the 5-kV Class 1E cables, i.e., flame retardancy, etc., but do not have documented LOCA/MSLB qualifications.

Based on the above design features and additional engineering analyses, we conclude that these circuits do not pose any challenges to Class 1E circuits and, therefore, the reactor coolant pump 13.8kV circuit breakers are excluded from having special associated circuits requirements.

3. Train B Associated Circuits Completely Outside the Nuclear Island that have Direct Interface with Class 1E Circuits

There are no Train B associated circuits completely outside the Nuclear Island that have a direct interface with a Class 1E circuit. Therefore, there are no special requirements related to separation for the protective devices for these associated circuits.

d. Groups C and D Circuits

Separation Groups C and D, which are comprised of circuits for Channels III and IV, do not have any associated circuits. Since these channels meet all requirements as defined in the compliance documents listed in Subsection 8.3.1.4a, these channels are not susceptible to any challenges from any associated circuits; therefore, the ability for the safe plant shutdown under a DBE cannot be jeopardized.

e. Cables

Medium voltage (5 kV and 15 kV) power cables are installed in raceways separate from those used for low voltage power and control cables, low level signal cables, and nuclear instrumentation cables. Medium voltage power cables for different voltage levels are installed in separate raceways. In vertically stacked trays, the highest voltage cables are in the highest position in the tray stack.

Low voltage (480 volts AC and some 120 volts AC, and 125 volts DC) power cables in vertically stacked trays are located below the medium voltage power cables, and are separated from control, low level signal and nuclear instrumentation cables.

Low voltage power cables associated with the control rod drive mechanism (CRDM) are routed in raceways separated from control and instrument raceways.

Control cables in vertically stacked trays are located below the low voltage power cables. Control cables are separated from low level signal and nuclear instrumentation cables, as well as from medium and low voltage power cables.

Low level signal cables are run in raceways separate from all other cables. In vertically stacked trays, the low level signal tray is generally at the lowest level in the stack.

In general, the above order of cable trays in a vertical stack is maintained with very few exceptions except where physical interferences within the plant necessitate alteration of the above order.

Nuclear instrumentation cables are routed in steel conduits for their entire distance.

The two redundant trains (Train A and B) and the four redundant channels (Channels I, II, III and IV) are routed through four physically separated raceway systems, called separation groups, as shown in Table 8.3-4. Physical separation of the four groups is maintained by means of one or more of the following:

1. Separate exposed rigid metal conduits, flex conduits, and wireways

2. Separate concrete-encased plastic or metal ducts in the same duct bank

3. Cable trays separated by a wall, a floor, or an equivalent barrier with a three-hour fire rating

4. Separate cable trays in the same room where a minimum of three feet horizontal or five feet vertical separation exists between trays or redundant systems

5. Separate cable trays in the cable spreading room (as defined in Appendix 8A, Subsection 5.1.3) where a minimum of two inches horizontal or six inches vertical separation exists between trays of redundant systems.

   Exceptions to the above separation distances are discussed in Section 8.3.1.4.a.

Subsection 5.1.1.3(c) of Appendix 8A specifies that cables shall not fill a cable tray above the side rails. Note 1 under Subsections 5.1.3 and 5.1.4 indicates that tray separation should be measured from the top of the tray side rail, i.e., the side rail is zero reference point for measuring separation. Various conditions in the raceway system, for example, fire stops and cable crossing at tray fittings, result in cables being above the tray side rail. For these cases, the zero reference point for separation will be the top of the cables, not the tray side rails. Since separation will be maintained, this exception to Appendix 8A is considered acceptable.

f. Selection of Cable Insulation

Insulation systems for cables comprise materials or combinations of materials for primary insulation, jackets, shielding, tapes, fillers and armoring. The factors considered in selecting a cable insulation system include stability and length of life, dielectric properties, resistance to ionization and corona, resistance to high temperatures, resistance to moisture, resistance to chemicals, resistance to radiation, mechanical strength, flexibility, self-extinguishing and nonpropagating fire characteristics, and general environmental considerations.

Cables for both Class 1E circuits and associated circuits are purchased to exactly the same requirements, including applicable environmental qualification, except as follows: the 15-kV and 5-kV SEPS power cables meet all the construction and material requirements placed on the 5-kV Class 1E cables, i.e., flame retardancy, etc., but do not have documented LOCA/MSLB qualifications. The 15-kV and 5-kV SEPS power cables are routed in dedicated raceways. Other exceptions such as certain cables in the Administration Building have been analyzed to show that they cannot challenge Class 1E circuits. In addition, other cables, for example General Electric cables for the turbine EHC System and certain Westinghouse supplied cables which may not have exactly the same qualifications as the Class 1E cables, are routed in dedicated raceways.

The basis for selecting insulation is to obtain insulation systems that are appropriate for each application and are compatible with their individual operating environments.

Cables installed inside the reactor containment which are required to be functional prior to, during, and/or after an accident are suitably selected and qualified to operate in the expected environment.

Cables are typically constructed with the following insulation and jackets:

1. Power cables: Ethylene propylene rubber (EPR) insulation and chlorosulforated polyethylene (CSPE) (Hypalon) on conductors with an overall Hypalon jacket.

2. Control cable: Ethylene propylene rubber insulation with an overall chlorinated polyethylene jacket.

3. Instrumentation cables: Cross-linked polyethylene (XLP) insulation with an overall Hypalon jacket.

4. Control rod drive power cables: Ethylene propylene rubber insulation with Hypalon jacket overall up to the reactor head area where they change to Tefzel insulation with a stainless steel braid jacket overall.

5. Pressurizer heater power cables in the pressurizer area: Ozone and radiation-resistant silicone rubber insulation and a glass braid jacket.

6. Coaxial and triaxial transmission cables: Cross-linked polyethylene insulation and a flame retardant cross-linked polyolefin jacket.

g. Sizing of Conductors

All cables are sized to operate within their normal rating and temperature rise with respect to current carrying capacities and insulation properties.

Ampacity reduction factors using ICEA recommendations are utilized based on the cable arrangement and spacing in tray. Where a cable is routed through several types of raceways and in various arrangement of conductors, the cable size is based on the most restrictive situation. The effect of fire stops and fire barriers is also considered when determining the cable ampacity.

Considering the above and the effects of short circuit current heating, voltage regulation, and voltage drop, the power cables to motors are sized to carry continuously a minimum of 125 percent of the nameplate load current. Feeder cables to load groups are sized for a minimum of 125 percent of the calculated circuit load current.

The basis for correct sizing of cable conductors is to ensure that the connected equipment and the cable operate within their permissible ratings.

h. Cable Routing

A computer-generated conduit and cable schedule is the basic cable drawing. All cables are included in this schedule with various exceptions, e.g., vendor-supplied specialty cable for the Radiation Monitoring System and portions of various other systems (for example, telephone system, lighting [see Subsection 9.5.3], temporary cables and fire protection/detection). This program is designed to maintain records of the power, control and instrumentation circuits through both cables and raceways and to accumulate, sort, collate and print out the information. The computer performs the following functions, all in accord with the limits established in the input data:

1. Calculates the shortest route and length for each cable, given a network of raceways and the origin and destination of the cable

2. Maintains the predetermined maximum fill percentage for each raceway type and size

3. Properly routes the circuits assigned to the four separation groups (see Table 8.3-4), to maintain the required physical independence

4. Properly routes various circuit types (heavy power, medium/light power, control, instrumentation, and signal levels) to the designated raceways

5. Reduces the need for installers to refer to drawings in order to pull cables.

6. Gives the plant maintenance personnel useful information for startup, operation and identification.

As part of the conduit and cable input, the network of raceways is developed connecting the various items of electrical equipment. The bulk of this raceway system is cable trays as listed above. This network is governed by the following:

1. Requirements for the power, control and instrumentation systems
2. Requirements for physical separation of redundant circuits
3. Physical loading requirements
4. Avoidance of hazardous areas where possible
5. Protection from missiles, fire and irradiation
6. Simplicity of layout for ease in installation and access.

Cable routing is governed by the same criteria listed above for raceways.

i. Raceways

Raceways consist of exposed rigid or flexible steel conduit, electrical metallic tubing, concrete encased plastic or metal duct, steel cable trays and wireways. Cable trays are ladder type, except for those containing low-level instrumentation, which are solid bottom trays with covers, except in limited areas of the plant where it has been determined that the noise interaction should not be a problem and that the elimination of the covers will not degrade the installation or violate any separation criteria if applicable. Solid bottom steel cable trays with solid covers are used for the Supplemental Emergency Power System (SEPS) power cables within the plant. Raceways are laid out throughout the plant between electrical equipment to support and protect the cables that traverse these locations.

The factors that are considered when selecting a raceway for an application include mechanical strength to support and protect the cables, resistance to chemicals, resistance to moisture, resistance to high temperature, length of life, flexibility, internally generated heating, vulnerability to fire, and general environmental considerations.

The criteria for thermal and physical loading of raceways are based on IEEE/ICEA recommendations or test results for cables installed in different raceways. In cable trays, the percentage fill requirements are as follows:

1. For all 15-kV and 5-kV power cables and 480 volt heavy power cables (4/0 and larger in size) - one layer with a nominal one-quarter diameter air spacing [Note 2].

2. For 480-volt medium and small size power cables (2/0 AWG and smaller in size) - 40 percent [Note 1] of usable tray volume fill

3. For control and instrumentation cables and control rod drive power cables - 40 percent (see [Note 1]) of usable tray volume fill.

Note 1 Forty percent fill is based on nominal cable diameters. Actual fill percentage might vary because of manufacturing tolerances in cable construction. Cable ampacity reduction factors remain unaffected by decreased or increased fill percentages. Percent fill greater than 40% is allowable in limited cases where engineering analysis verifies the criteria for thermal and physical loading are maintained.

Note 2 It is understood that there will be occasional crossing and touching (point contact) of cables installed in ladder type trays. This has no appreciable effect on cable ampacity and therefore is acceptable.

In the Nuclear Island, raceways that carry nuclear safety-related circuit cables are embedded in Category I walls, floors, and duct banks or supported by steel members which are qualified by means of seismic analysis.

The bases for selecting, laying out, and loading raceways are to minimize the loss of function of cables in the raceway due to adverse conditions external or internal to the raceway.

j. Electrical Penetrations

The electrical penetration assemblies provide the means to allow passage of power, control and instrument circuits through the containment pressure barrier while maintaining the integrity of the pressure barrier. The criteria for physical separation of electrical penetrations is the same as for raceways as described above.

Penetrations for 600-volt service and below are modular type with a header plate welded to the outside of a 12-inch containment sleeve. Because of the concern regarding leakage currents of terminal blocks during accident conditions, low level instrumentation circuit conductors inside containment are connected to the penetration conductors with qualified splices. Safety-related 480-volt power, 120-volt AC and 125-volt DC control circuit conductors inside containment required to function for LOCA and main steam line break conditions are also connected to the penetration conductors with qualified splices. The balance of medium power 480-volt conductors, and control and instrumentation conductors are terminated on terminal blocks inside terminal boxes both inside and outside containment. 480-volt heavy power conductors are terminated with lugs on special termination plates inside terminal boxes both inside and outside containment. Nuclear instrumentation detector circuits are terminated with connectors inside terminal boxes both inside and outside containment. Penetrations for medium voltage have header plates welded to the outside of an 18-inch containment sleeve. Each penetration consists of three 1000-MCM conductors terminated with premolded stress cones inside terminal boxes both inside and outside containment.

The capability of the electrical penetrations to withstand the total range of time versus fault current without loss of containment integrity under worst-case environmental conditions was demonstrated by test. These test results substantiate the capability of the electrical penetration to withstand the total range of time versus fault current without seal failure.

The penetrations are arranged in two levels, with one power train and two channels entering above the intermediate floor of the Containment Building, and the redundant train and two channels entering below the intermediate floor. Once inside the containment, this floor provides the necessary physical separation and protection between the redundant trains; outside the containment, this separation is continued by separate tunnels connecting the penetration area to the switchgear and cable spreading areas of the Control Building.

Penetration conductors are sized using ICEA guidelines with an additional restriction of a 65°C nozzle-concrete interface temperature.

The design, construction, and installation of the penetration assemblies are in accordance with IEEE 317 and Regulatory Guide 1.63. (See Subsections 8.1.5.3, 8.3.1.1, and 8.3.1.2 for further details on compliance to Regulatory Guide 1.63.)

k. Cable and Raceway Identification

The computerized conduit and cable schedule provides a permanent record of the routing and termination of cables. Circuit level coding identifies the individual channel or train assigned to each raceway and cable. These data are entered into the conduit and cable program, which in turn produces reports designating the unique number with origin, destination, channel or train, and specific path for every cable. Every cable is identified by a tag affixed at each end, bearing the unique cable number.

Each channel or train is assigned a particular color, as described in Subsection 8.3.1.3.

All safety-related cables have jackets of the color assigned to the particular channel and train so there is no difficulty in distinguishing between cables of redundant channels. Nonsafety-related cables are associated with either Train A or B and have black jackets with a red trace for cables associated with Train A and a white trace for cables associated with Train B. It is immediately evident to the operator or maintenance man, by observing the color of the cable jacket, that a given cable is safety-related and that it is a particular channel or train. This system also prevents placing a cable of one channel or train with cables of another, by the obvious dissimilarity of jacket color. For SF-P10C, which can be powered from either the A or B train emergency bus, an exception has been taken to the general color code convention. The color code of the common cable is selected as Train A.

Per UFSAR Appendix 8A Section 5.1.2, cables were color coded to facilitate initial verification that the installation was in conformance with the separation criteria. Per cable specifications, cable color was only guaranteed for 10 years. In the event that certain cable colors change with age, example: white cables may yellow with age, the separation group can still be identified by the cable code that is printed on the jacket, or by the permanent identification tag at both ends of every cable.

Alternate methods of color coding cable have been made necessary by government regulations restricting the use of paints due to environmental concerns. New installations of safety-related cables may have black jackets stamped with their assigned color at intervals along their length. These cables will be further identified by the application of colored tape at both ends.

Occasionally installations may justify use of a different train cable code. At a minimum, the outer jacket of these cables will be taped with the appropriate color tape at each end along with a tag indicating the intended cable code. Such installations will be evaluated on a case by case basis and will include specific identification details for the application.

Each cable is further identified by a footage and cable code on the jacket of the cable at intervals of approximately five feet. Reference to pulling records reveals the cable number, routing, separation, circuit type, and use of any cable at any accessible point in the raceway system where the footage marker and cable code can be identified.

Exceptions to the above cable identification criteria exist for vendor-supplied specialty cables for the Radiation Monitoring System and portions of various other systems (for example telephone system, lighting, temporary cables and fire protection/detection). For these exceptions, the necessary information to ensure adequate control of separation, installation, inspection, etc., is provided in the construction documents.

Raceways which are part of the computerized cable and conduit schedule are marked to identify their number and circuit level. Conduit raceways are identified at each end where conduit terminates and at both sides of walls, floors and in-line boxes. Tray raceway markers are spaced at 15-foot or less intervals. These markings are in the same colors assigned to the channels and trains. For example, a raceway with a red section marking is utilized only by cables with red (or black with red tracer) jackets. Hence, it is readily apparent that a given cable is routed with its respective channel.

Tray raceway markers are color coded with the same colors assigned to the channels and trains. The markers have a white background with a border color that corresponds to the separation group color code. For example, a raceway section with a marker with a red border is used only by cables with red (or black with red tracer) jackets. It is noted that for Train B/Channel II raceway markers there is no demarcation between the white border and the white background.

Raceways which are not part of the computerized conduit and cable schedule may not be marked with a unique identification number, but their function is obvious by tracing the raceway to its end device. These raceways may be used to carry vendor-supplied specialty cables for the Radiation Monitoring System and portions of various other systems such as telephone system, lighting and fire protection/detection. For these raceways, the necessary information to ensure adequate controls of separation, installation, inspection, etc., is provided in the construction documents.

Since Seabrook Station is a single unit plant there is no portion of the wiring or its components that is shared; therefore it is in compliance with GDC-5.

l. Administrative Responsibility and Control

Administrative responsibility for assuring compliance with applicable design criteria and bases relative to independence of redundant systems during the initial construction phase rested with the A/E's Project Electrical Engineer. He was responsible for coordination with the A/E's field electrical supervisor to verify that the independence, separation and availability of Class 1E equipment was preserved during installation of the electric power system.

The following control procedures were established by the A/E's Project Electrical Engineer to ensure compliance of the electric power system with the design criteria and bases:

1. Periodic design reviews with the cognizant engineer, the design supervisor, and the reviewing engineer to ensure the criteria are being interpreted and followed

2. Issuance of periodic administrative and design directives covering procedures

3. Periodic field reviews at the job site by the Project Electrical Engineer and/or the cognizant engineer to check field installation procedures, to provide interpretation of design drawings and guidance for solution of field installation problems, and to verify compliance with criteria.

The design of the conduit and raceway system is guided by the recommendations of applicable IEEE, ICEA and NEC standards. For instance, the limiting percentages of fill of internal area of the various size conduits or cable trays are fixed in one of the input forms of the computer conduit and cable schedule and these limits are automatically applied to all conduits and cable trays by the computer. If the conduit or cable tray is one which the computer is free to size, it designates the size which accommodates the cables to be enclosed. If the conduit or cable tray size is designer-designated and the fill exceeds the limiting percentage, the computer indicates an error message so that either the conduit can be made a larger size, or the cables routed by another path. By these methods, all raceways are assured of being of adequate capacity.

Correct installation practice assured that the design criteria by which the equipment was selected were not violated during construction. Installation bases were prescribed, where necessary, by the A/E's Project Electrical Engineer or the field electrical supervisor, and guidelines were established to ensure compliance with the above. For example, maximum pulling tensions for cables are limited, dependent on the method of pulling. In designing the duct runs, anticipated tensions were calculated using industry standard formulae to verify that they are below permissible maximum tensions for the cable being installed. Minimum bending radii were established for the cables being used. These indicate the minimum bend to which a given cable may be bent for safe electrical operation without physically damaging the insulation or coverings. Larger radii were normally implemented whenever installation conditions permit. Precautionary measures to prevent nicks, cuts, abrasion and damage to cables during installation were established.

Inspections were made by the A/E's Project Electrical Engineer and the A/E's field electrical supervisor to verify that cables were being located and routed in accordance with the design criteria.

The design control program and the work control program provide the necessary controls to ensure that the original design bases of the plant are maintained during the operational phases.

8.3.2 DC Power System

8.3.2.1 Description

The station DC power system is comprised of the battery chargers, station batteries and 125V Distribution System. It provides the sources of power for direct current load groups, vital control and instrumentation systems, and control and operation of Class 1E and non-Class 1E electrical equipment. It is a two-wire ungrounded system.

The battery chargers (rectifiers) provide the normal steady-state DC power; the station batteries provide for normal transient loads and also act as the reserve source upon failure of the rectifier or the AC supply to it. Figure 8.3-2, Figure 8.3-37 and Figure 8.3-38 present the one-line diagrams for the station DC electric power system, and show the connections to the AC Vital Instrumentation and Control Power System.

The safety-related portion of the station DC power system shown on Figure 8.3-37 consists of four 125-volt batteries, chargers and DC buses. The loads supplied from the buses include inverters for redundant vital instrument buses, distribution panels for power to the Class 1E direct current loads, power for control and operation of the Class 1E systems for Engineered Safety Features, and power for selected non-Class 1E loads.

Each DC bus consists of metal-enclosed 125-volt DC switchgear consisting of vertical sections housing buses, circuit breakers, instruments and accessory equipment. The breakers are low voltage manual power circuit breakers. Figure 8.3-37 shows that the safety-related DC system incorporates mechanically interlocked manual circuit breakers which will permit the connection of two DC supply buses within the same train to a single battery, but prevents paralleling the two batteries in the train.

The nonsafety-related portion of the station DC systems shown on Figure 8.3-38 consists of two 125-volt batteries, chargers and DC buses. The loads supplied from the buses include inverters for the computer and auxiliary power panels feeding nonvital equipment requiring constant supply and control power feeders to nonvital equipment (13.8-kV switchgear, turbine generator emergency oil pump, etc.).

a. System Separation, Ventilation and Redundancy

Four safety-related 125-volt batteries are supplied. Each battery is housed in an individual room in the seismic Category I Control Building. Separate ventilating systems are provided for the battery rooms of each train (see Subsection 9.4.10). The batteries are seismically qualified and are mounted on seismic Category I racks. The safety-related battery chargers and DC buses are also seismically qualified.

Each battery has its own charger and DC bus. The battery chargers and DC buses of each train are located in an area adjacent to their associated battery rooms and are physically separated from the chargers and buses associated with the redundant train (see Figure 8.3-27 and Figure 8.3-36).

Four DC supplies are provided for the four NSSS inverters for vital instrument buses and the power and control requirements of the two engineered safety features trains (see Subsection 8.3.1.1d). Equipment is located and cables are routed in a manner to assure continued independence and separation so that the loss of DC supply to either train does not prevent the minimum safety function of the other train from being performed.

One nonsafety-related inverter for the station computer is powered from the Train A DC system through a Class 1E breaker on Bus 11C. One nonsafety-related DC power panel is powered from the Train B DC system through a subfeed from safety-related DC power panel PP-111B. All remaining nonsafety-related loads (DC motors, other nonsafety-related inverters, nonvital control panels) are connected to the nonsafety-related batteries (Figure 8.3-38).

b. Station Battery Capacity

The safety-related station batteries are lead-calcium, power station type. Each battery consists of 59 cells, and has a nominal 8-hour rating of 2280-ampere hours.

Each safety-related battery is sized to supply its safety-related and nonsafety-related loads for the durations indicated in Table 8.3-5. Battery B-1C is capable of providing power to the nonvital computer inverter, I-2A, for 15 minutes while supplying its safety-related loads; the inverter load is automatically disconnected from the DC system after the 15-minute period. This disconnection is accomplished by a safety-related trip circuit on the Class 1E breaker feeding inverter I-2A. This circuit, which monitors the time the inverter draws power from the battery, is testable.

In addition, each safety-related battery is sized to have sufficient capacity to serve as the source, for the duration indicated in Table 8.3-5, for two load groups of the same train during the period when one battery is out of service (see Figure 8.3-37). Figure 8.3-51 shows the separate and combined load profiles for the safety-related batteries.

The safety-related station batteries also have sufficient capacity for the four-hour Station Blackout coping duration. The Station Blackout battery sizing evaluation includes the one battery/two bus configuration (see Section 8.4.4.2).

There are two nonsafety-related batteries (B-2A and B-2B) provided in the Turbine Building. The nonsafety-related station batteries are lead calcium, power station type, consisting of 59 cells. Battery B-2A supplies various DC motors for the turbine auxiliaries, various control panels and the Turbine Building DC lighting. Battery B-2B supplies the computer inverter I-2B, the nonvital instrument inverter I-4, Control Building DC lighting and various control panels.

Each Class 1E battery was sized in accordance with the recommended practices in IEEE Standard 485-1978. These practices were applied as follows:

1. The system maximum voltage (140 volts) and maximum equalizing cell voltage (2.33V per cell) were selected. This resulted in a selection of 59 cells which include margin between the equalizing voltage (137.5V), and the system's maximum voltage.

2. A duty cycle diagram was developed, based upon the combined known and anticipated loads for both DC buses of the same train (see Figure 8.3-51).

3. The battery capacity data were selected from the manufacturer's data, based upon the minimum cell voltage (1.78V per cell permitted by the system minimum voltage of 105V).

4. The calculated minimum required cell size was increased by 25 percent for end-of-life compensation.

5. Temperature correction factors were applied to the calculated minimum required cell size, to allow for operation at the minimum design temperature (65°F for batteries B-1A and B-1C, and 60°F for batteries B-1B and B-1D). These temperatures also apply to Station Blackout.

6. Sizing calculations were performed using methods similar to Figure 3 of IEEE 485-1978 to determine the minimum required cell size.

7. A minimum design margin of 15 percent was included in the original battery purchase specification calculated cell size to allow capacity for future loads.

c. Battery Charging

Power for each DC distribution bus is normally supplied through a battery charger dedicated to that bus. Each safety-related battery charger is rated for 150 amps and has been sized to charge its associated battery from the design discharged state (i.e. the state of a battery following a service discharge test) back to the charged state while carrying the largest combined demand of the steady-state loads under all plant operating conditions. If the battery has reached the design minimum charge state (i.e., the state of a battery following a performance discharge test), the charger will restore the battery back to the fully charged state while carrying the largest combined demand of the steady-state loads under all plant operating conditions. Transient emergency peak loads are adequately carried with assistance from the battery if these loads exceed the charger full load output capability.

Each cell is maintained on a float-charge of 2.23 volts. The battery manufacturer's data indicate that when this type cell is float-charged above 2.20 volts per cell it requires little or no equalizing charge. If equalizing is required, the cell voltage will be raised to 2.33 volts per cell (137.5 volts total). Equalized charging of each battery can be provided by the dedicated battery charger with the battery connected to the bus. All DC equipment has been specified and purchased with a maximum operating voltage of 140 volts DC.

Each charger is fed from a separate 460-volt emergency motor control center. On loss of the normal and preferred power supplies, the chargers are energized by standby power. See standby power supply loading sequence charts, Table 8.3-1 and Table 8.3-2.

During the period when a charger requires maintenance, a portable spare charger can be used to replace the normal charger. The spare charger can be positioned adjacent to the charger it is replacing. 460V AC power is supplied to the portable battery charger by means of a plug and receptable connection. The portable charger is connected to the same power train as the fixed charger it is replacing. Alarms from this unit replace the alarms from the fixed charger.

In addition to being used as a one-for-one replacement for a normal charger that is taken out of service, the portable chargers can also be used during the performance of safety-related charger surveillance tests or to recharge the station safety-related batteries following on-line battery maintenance and testing. During this mode of operation, the portable chargers are operated coincident with the normal chargers. Maintenance and test activities on the nonsafety-related chargers and batteries do not require that the portable chargers be operated coincident with the normal chargers. When a portable charger is used for an application other than the replacement of a normal charger, only the local alarms on the portable charger are functional.

d. DC Power System Loading

The safety-related portions of the DC loads are divided into redundant load groups, as listed in Table 8.3-5 and detailed on Figure 8.3-37, Figure 8.3-39, Figure 8.3-40, Figure 8.3-41, Figure 8.3-42, Figure 8.3-58, Figure 8.3-59, Figure 8.3-60 and Figure 8.3-61. The batteries are sized to accommodate both the safety and nonsafety loads, as noted in Subsection 8.3.2.1b. The operator may manually load shed selected loads to extend the discharge time, if required. Load shedding is not required to meet the four-hour Station Blackout duration (see Section 8.4.4.2). Two low voltage alarms, a battery ammeter and a DC bus voltmeter are available in the control room for each DC bus to aid the operator in this decision.

e. DC Power System Testing

The batteries and other equipment associated with the DC system are easily accessible for periodic testing and inspection. Surveillance and testing are performed in accordance with the plant Technical Specifications in compliance with the guidelines of IEEE Standard 338, 450, Regulatory Guides 1.118 and 1.129. except as described in Subsections 8.1.5.2 and 8.1.5.3.

The preoperational testing of the safety-related portion of the DC system has been performed in accordance with Regulatory Guides 1.68 and 1.41.

f. Surveillance and Monitoring

The DC power system (batteries, distribution systems and chargers) is monitored continuously to show that it will be ready to perform its intended function when called upon. Table 8.3-6 summarizes the surveillance and monitoring provisions for the DC power system. The following indications and/or alarms are present locally or in the control room:

1. Battery current (ammeter-charge/discharge) - an ammeter is located in the control room and at the DC switchgear to indicate battery charge and discharge. A local digital ammeter is also provided with sufficient resolution to read normal float charging current.

2. Battery charger output current (ammeter) - an ammeter is located at the charger. The loss-of-charger input AC voltage computer alarm and the battery discharge computer alarm, provide sufficient indication to show when the battery charger is capable of performing its intended function.

3. DC bus voltage (voltmeter) - a voltmeter is located in the control room and at the DC switchgear to indicate the bus voltage. In addition, there is a voltmeter at the charger showing the charger DC output voltage.

4. Battery discharge alarm - a control room alarm is provided to indicate battery discharge. This alarm is derived from the local digital ammeter (see Item 1). This alarm is part of the loss of charging current alarm described in Item 7.

5. DC bus undervoltage and overvoltage alarms - two undervoltage computer alarms are provided for the DC buses; one alarms low voltage as a result of battery discharge. A second alarms on low voltage as a result of a bus fault; this also trips the charger feeder breaker to prevent the charger from feeding a fault. An overvoltage computer alarm is provided by an overvoltage relay at the charger (Device No. 59/62); the charger being the most probable cause of an overvoltage condition. There is also an undervoltage alarm for the battery charger output.

6. DC bus ground alarm (for ungrounded system) - for ground detection, a computer alarm and meters are provided in the control room with additional meters mounted locally in the DC switchgear.

7. Battery breaker(s) or fuse(s) open alarm - the battery supply breaker to the bus has a computer alarm for the breaker open position. A control room alarm is provided to indicate loss of battery charging current. This loss could be caused by an opening in the battery circuit (blown fuses, cut cable, etc.) or by loss-of-charger output. This alarm is part of the battery discharge alarm discussed in Item 4.

8. Battery charger output breaker(s) or fuse(s) open alarm - the battery charger supply breaker to the bus has a computer alarm for the open position. The charger output breaker (integral to the charger) does not have an open alarm; however, it is nonautomatic and would therefore not open on a fault, and if inadvertently left open, it would be alarmed by the battery discharge alarm since the charger could not feed the bus (see 4. above).

9. Battery charger trouble alarm (one alarm for a number of abnormal conditions which are usually indicated locally) - the battery charger trouble alarms (high DC output voltage, low DC output voltage, and loss of AC input voltage), all have separate computer and local alarms.

8.3.2.2 Analysis

The DC System Failure Mode and Effect Analysis is found in Table 8.3-7.

a. Compliance with General Design Criteria

1. Criterion 2 - Design Basis for Protection Against Natural Phenomena

(a) The components of the onsite DC power system are located in seismic Category I structures which provide protection from the effects of tornadoes and external floods, and other natural phenomena.

(b) These components are Class 1E.

(c) These components have been designed to be fully qualified for the seismic and natural environmental conditions appropriate to their location (see Section 3.11).

2. Criterion 4 - Environmental and Missile Design Bases

(a) The components of the onsite DC power system are located in seismic Category I structures which provide protection from the effects of tornado missiles, turbine missiles and other events and conditions which may occur outside the nuclear power unit.

(b) These components are Class 1E.

(c) These components are designed to accommodate the effects of, and to be compatible with or are protected against, the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents including loss-of-coolant accidents. Criteria are presented in Chapter 3. Environmental conditions are presented in Chapters 3 and 6.

(d) These components are protected, as appropriate, against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids that may result from equipment failures (postulated accidents).

3. Criterion 5 - Sharing of Systems and Components

Seabrook Station is a single unit plant; therefore, no portion of the station DC system or its components important to safety is shared.

4. Criterion 17 - Electric Power Systems

Compliance with the requirements of the DC portion of the electric power supplies criterion and the associated independence, redundancy and testability are covered by Subsections 8.3.2.1a through 8.3.2.1e.

The station safety-related DC power system provides separate and independent DC power supplies and channels for redundant load groups during abnormal and accident conditions. These redundant load groups comprise engineered safety features and plant protection systems, grouped in such a way that loss of one group does not prevent the minimum safety functions of redundant groups from being performed. In the event of loss of DC power from the chargers, the batteries pick up the load on the DC buses.

5. Criterion 18 - Inspection and Testing of Electric Power Systems

The Class 1E DC electric equipment is designed and located to permit appropriate periodic inspection and testing in line with the provisions for testing listed in Subsection 8.3.2.1e.

These tests assure the operability and functional performance of the DC components, and the operation of the DC system as a whole. During unit shutdown, and under conditions as close to normal operation as practical, the full operational sequence that brings the system into operation, including portions of the protection system, is tested.

b. Compliance with Regulatory Guides

1. Regulatory Guide 1.6 - Independence Between Redundant Standby Power Sources and Between their Distribution Systems

The safety-related portion of the station DC system includes four batteries. The redundant safety-related load groups are each fed by a separate battery and battery charger. There is no provision for automatically connecting one battery-charger combination to any other redundant load group, nor is there any provision for interconnecting batteries either manually or automatically. To further enhance safety and reliability, two DC supply buses of the same train may be connected together manually, but circuit breaker interlocks prevent an operator error which would parallel two batteries. (See Figure 8.3-37).

2. Regulatory Guide 1.32 - Criteria for Safety Related Electric Power Systems for Nuclear Power Plants

The design is consistent with the requirements of this regulatory guide. For details, refer to Subsections 8.3.2.1c and 8.3.2.1e except as noted in Subsection 8.1.5.3.b.

3. Regulatory Guide 1.75 - Physical Independence of Electric Systems

The design is consistent with the criteria for physical independence of electric systems established in Attachment C of AEC letter dated December 14, 1973. Attachment C is incorporated as Updated FSAR Appendix 8A and is considered similar to Regulatory Guide 1.75.

For clarification of position C4 as it relates to associated circuits, refer to Updated FSAR Subsection 8.1.5.3b.

The Seabrook cable and raceway separation criteria (see Updated FSAR Subsection 8.3.1.4) is a combination of the standard criteria given in Attachment C of AEC Letter dated December 14, 1973 (see Updated FSAR Appendix 8A) and IEEE 384-1974 and criteria established by analysis and testing as permitted by Attachment C and IEEE 384-1974.

4. Regulatory Guide 1.129 - Maintenance, Testing and Replacement of Large Lead Acid Storage Batteries for Nuclear Power Plants

For compliance to this regulatory guide, refer to Subsection 8.3.2.1e.

5. Regulatory Guide 1.155 - Station Blackout

The design is consistent with the requirements of this Regulatory Guide. The safety-related station batteries have sufficient capacity for the four-hour Station Blackout coping duration.

c. Compliance with IEEE 308, Class 1E Electric Systems

The station DC system conforms to the requirements of IEEE 308 except as noted in Subsection 8.1.5.2. The power supplies, distribution system, and load groups (see Subsection 8.3.2.1) are arranged to provide direct current electric power to the Class 1E direct current electric loads, and for the control and operation of the Class 1E systems. Sufficient physical separation, electrical isolation, and redundancy are provided to prevent the occurrence of common failure modes in the Class 1E systems.

d. Conformance with Appropriate Quality Assurance Standards

The equipment of the DC system conforms to the controls for electrical equipment listed in Chapter 17.

e. Independence of Redundant Systems

The criteria and bases of minimum requirements to preserve the independence of redundant Class 1E electric systems are those outlined in the General Design Criteria and IEEE 308. Safety loads are divided into redundant groups and equipment is physically separated from its redundant counterpart to prevent the occurrence of a common failure mode.

Batteries are in individual rooms, and chargers and distribution equipment are separated by physical barriers, as indicated on Figure 8.3-27.

The criteria and bases for the installation of raceways and electrical cable for this system are the same as those listed for the AC power system in Subsection 8.3.1.4. Train separation throughout the safety-related portions of the plant is indicated on Figure 8.3-36, Figure 8.3-43 and Figure 8.3-44, which shows electrical arrangements at the three critical elevations.

f. Physical Identification of Safety-Related Equipment

The methods used to physically identify the DC safety-related equipment to assure its appropriate treatment are the same as that for the AC safety-related equipment listed in Subsection 8.3.1.3.

Identification systems distinguish between redundant separation groups; it is clearly evident to the operator or maintenance craftsman which equipment is safety-related and, if safety-related, which separation group is involved.

8.3.3 Fire Protection for Cable Systems

The fire prevention and protection system for cables is part of the integrated fire detection and protection system for the entire plant, and is described in Subsection 9.5.1.

Design aspects used in the prevention of fires in cable systems include separation between redundant trains and voltage levels, cable material selection and cable sizing. This is described in Subsections 8.3.1 and 8.3.2.

8.3.4 References

1. "Analysis of Separation Criteria for Seabrook Station" dated March 24, 1986.

2. Wyle Test Report No. 47966-02 dated January 24, 1986.

3. Wyle Test Report No. 48361-02, dated November 11, 1986.

4. Wyle Test Report No. 48361-03, dated November 11, 1986.

5. Letter No. SBN-1107, Electrical Separation Criteria; Additional Information, J. DeVincentis to V. S. Noonan - NRC, dated June 13, 1986.

6. Foreign Print 34957, Electrical Isolation Test Report

8.4 COMPLIANCE WITH 10 CFR 50.63, LOSS OF ALL ALTERNATING CURRENT POWER (STATION BLACKOUT)

8.4.1 Basic Requirements

This section describes Seabrook's compliance with 10 CFR 50.63 which requires that each light-water-cooled nuclear power plant be able to withstand and recover from a loss of all alternating current power or station blackout (loss of both offsite power and onsite emergency power). Regulatory Guide 1.155 (RG 1.155), "Station Blackout," provided a method for complying with 10 CFR 50.63. RG 1.155 stated that NUMARC 87-00, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," also provided acceptable guidance for meeting the requirements of 10 CFR 50.63. Seabrook followed NUMARC 87-00 except where the Regulatory Guide took precedence.

Station Blackout is considered a non-design basis accident. No other active or passive failures or design basis events are required to be considered during Station Blackout. Safe shutdown for Station Blackout (see 10 CFR 50.2 and NUMARC 87-00) means bringing the plant to a hot shutdown or hot standby condition. The Seabrook analysis and procedures proceed with plant cooldown until secondary side pressure is reduced to about 250 psig and the plant is in a hot standby condition.

Seabrook responds to Station Blackout as an AC Independent plant relying only on the station batteries as a source of electrical power for the coping duration specified in Section 8.4.2. When the Station Blackout analysis was initially performed there were no alternate AC power sources to support response as an Alternate AC (AAC) plant. The Supplemental Emergency Power System (SEPS) was subsequently installed and may be used as a source of power during a station blackout event but will not be credited as an alternate AC power source.

8.4.2 Station Blackout Duration

Seabrook Station's Blackout coping duration is four hours. This is based on evaluation of the offsite power design characteristics, emergency AC power system configuration and emergency diesel generator (EDG) reliability. The offsite power design characteristics included the expected frequency of grid-related loss of offsite power, the estimated frequency of loss of offsite power from severe and extremely severe weather, the number of switchyards and the type of bus transfers. Site-specific weather data were used to evaluate the reliability of offsite power relative to weather-caused outages. One out of two emergency diesel generators is required to operate safe shutdown equipment following a loss of offsite power. A target EDG reliability of 0.975 will be maintained by implementation of the Maintenance Rule EDG performance criteria.

8.4.3 Procedures

Station procedures address the action necessary to cope with a Station Blackout (loss of all AC power) including actions such as opening cabinet doors. Also, as required by RG 1.155, procedures address AC power restoration and severe weather conditions.

8.4.4 Coping Assessment

8.4.4.1 Condensate Inventory for Decay Heat Removal

Decay heat removal for the four-hour Station Blackout coping duration requires 137,000 gallons of water which is less than the Condensate Storage Tank (CST) 196,000 gallon dedicated EFW supply. The 196,000 gallons is the usable volume from the 212,000 gallon Technical Specification limit. The 137,000 gallon value includes decay heat removal, removal of sensible heat and steam generator level shrinkage.

8.4.4.2 Battery Capacity

The safety-related station batteries have sufficient capacity for the four-hour Station Blackout coping duration. The methodology for battery sizing is described in more detail in Section 8.3.2.1.b. The Station Blackout battery sizing includes the one battery/two bus configuration described in Section 8.3.2.1.b.

8.4.4.3 Compressed Air

Air-operated valves required during the Station Blackout four-hour coping duration can either be operated manually or have sufficient backup sources of air independent of AC power. The only valves requiring air are the ASDVs whose backup supply is described in Section 9.3.1.1. Operation of the required valves is addressed in plant procedures.

8.4.4.4 Effects of Loss of Ventilation

The areas containing equipment required to cope with a Station Blackout were evaluated for the effects of loss of ventilation. These areas include the emergency feedwater pumphouse, vital switchgear rooms, battery rooms, containment structure, main control room, electrical tunnels including electrical penetration area, mechanical penetration area and main steam/feedwater pipe chases including east electrical room and west stairwell. For all of these areas, the final calculated temperature at the end of the four-hour Station Blackout coping duration was less than the minimum environmental qualification temperature for the equipment located in the area. Procedures require opening of control cabinet doors in the main control room and the Train B essential switchgear room to enhance cabinet cooling. Minimum battery room temperatures were also determined for input into the battery sizing calculations.

8.4.4.5 Containment Isolation

Although it is not specifically required to isolate containment in response to Station Blackout, the capability to establish containment integrity must be provided. Capability means the ability to close and have position indication, independent of offsite power and onsite emergency EDG power, for valves that may be open at the onset of the Station Blackout. RG 1.155 and NUMARC 87-00 permit exclusion from further review the following containment isolation valves:

a. valves normally locked closed during operation,

b. valves that fail closed on a loss of power (Seabrook did not exclude valves powered from DC power because they would not lose power during a Station Blackout),

c. check valves,

d. valves in nonradioactive closed-loop systems not expected to be breached in a Station Blackout (this does not include lines that communicate directly with containment atmosphere), and

e. valves of less than 3-inch diameter.

UFSAR Table 6.2-83 was reviewed against these exclusion criteria. None of the valves required to establish containment integrity are required to be opened during a Station Blackout. Once these valves are closed or verified closed, they would remain in that position for the duration of the Station Blackout. An action to close or verify closed the position of these valves is included in the Station Blackout procedure.

8.4.4.6 Reactor Coolant Inventory

The expected rates of reactor coolant inventory loss under Station Blackout conditions do not result in the core's becoming uncovered in the four-hour Station Blackout duration. The analysis includes loss or leakage of reactor coolant inventory through the reactor coolant pump seals, the sources described in the Technical Specifications and the letdown line. Therefore, makeup systems are not required during Station Blackout to maintain core cooling under natural circulation (including reflux boiling).

8.4.5 Quality Assurance

All equipment required to cope with Station Blackout is safety related. All safety-related equipment is within the scope of the Operational Quality Assurance Program which complies with the requirements of 10 CFR 50, Appendix B, which exceeds the quality assurance requirements described in RG 1.155.

11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

11.5.1 Design Bases

The process and effluent radiological monitoring and sampling systems are designed to provide radiation measurements, records, alarms, and/or automatic line isolation required to handle, process, and/or release station liquid and gaseous radioactive effluents, in compliance with the requirements of 10 CFR Parts 20 and 50, General Design Criteria 60, 63 and 64, and Regulatory Guide 1.21. The systems are designed to be in general compliance with the guidelines of Regulatory Guides 4.15 and 1.97.

The systems are designed to continuously monitor and/or sample process and effluent streams wherever a potential for a significant release of radio-activity exists during normal operations, including anticipated operational occurrences, and during postulated accidents. For certain effluent streams for which the potential release of radioactivity is determined to be insignificant relative to the design objectives of 10 CFR 50 Appendix I, process monitoring/sampling and airborne sampling are utilized to conservatively assess the releases.

11.5.1.1 Performance Requirements

The functional performance requirements for the Radiation Monitoring Systems are to:

a. Warn of leakage from process systems containing radioactivity

b. Monitor the amount of radioactivity released in effluents

c. Isolate lines containing liquid and gaseous activity when activity levels reach a preset limit

d. Record the radioactivity present in various station systems and effluent streams

e. Provide a means for leakage detection

f. Provide information on failed fuel.

11.5.1.2 Design Provisions

The components of the process and effluent radiation monitoring and sampling systems are designed for the following environmental conditions:

a. Temperature: An ambient temperature range of 40°F to 120°F

b. Humidity: 0 to 95 percent relative humidity

c. Pressure: Components designed for normal atmospheric pressure.

Radiation monitors, utilizing G.M. tubes are of a nonsaturating design so that they register full scale if exposed to radiation levels up to 100 times full-scale indication.

Radiation monitoring equipment is designed and located so that radiation damage to electrical insulation and other materials will not affect their usefulness over the life of the plant. Where possible, electronic components beyond the detector are mounted near the detector in a background radiation level of less than 2.5 mR/hr.

Each radiation monitoring channel is designed so that it can be checked on a daily basis, tested monthly, and recalibrated at refueling intervals.

Access to each of the radiation monitoring channel alarm setpoints is under administrative control.

Process and effluent radiation monitors provide annunciation and indication in the main control room.

Process and effluent monitors continuously monitor radiation levels in the various process streams and effluent release points.

Process and effluent monitors provide instrument failure annunciation in the main control room.

All online process and effluent radiation monitors are implemented with the capability to replace or decontaminate these monitors without opening the process stream or losing the capability to isolate the effluent stream.

This system is non-Class 1E and nonsafety-related, with the exception of the monitors identified in Subsection 11.5.2.1n (Containment Online Purge). The containment online purge monitors are Class 1E, safety-related and supplied from Class 1E uninterruptible power supplies.

The resin sluicing monitors (RM-6560, -6561 and -6564) and the Storm Drainage System monitor (RM-6454) are the only monitors that do not interface with the RDMS host computer system (Subsection 11.5.2.1i)

11.5.2 System Description

Process and effluent radiological monitoring systems consist of multiple channels which monitor radiation levels in various plant operating systems. The digital computer-based Radiation Data Management System (RDMS) consists of local microprocessors for each channel interconnected by redundant communication loops to a redundant (two computers) host computer system. Either of the two computers can provide, by itself, the total computing capacity required for satisfactory operation of the RDMS. The host computer system, in turn, is connected to an operator display/control console in the control room, the health physics checkpoint, the RDMS computer room, the Main Plant Computer System (MPCS) computer room and the hot chemistry lab. The process and effluent radiation monitoring system instrument engineering diagram (Figure 11.5-1) shows an overview of the system, its components and location.

Table 11.5-1 lists the various processes and effluent radiation monitoring channels provided and their pertinent design information, such as detector type, ranges, reference isotopes which the detectors are keyed to, and sensitivities.

Ranges and sensitivities have been selected using the following bases:

a. Maximum calculated concentrations during normal operations, anticipated operational occurrences and postulated accidents

b. State-of-the-art limitations of the commercially available detectors

c. Minimum concentrations that must be detected to permit timely automatic or operator manual responses tabulated in Table 11.5-2 and to avoid exceeding Technical Specification limits.

Shielding is provided on all the monitors to reduce the effect of background radiation, so that the minimum sensitivities specified are met. Table 11.5-2 shows the automatic system response and the operator response to annunciated radioactivity level limits.

A modular assembly with a microprocessor is provided in a locally mounted cabinet for each channel. The assembly converts pulse rate from the detectors to engineering units suitable for indication and recording. The following functions and components are included:

a. Indication

Radiation level is indicated by digital readout. Units are microcuries per cubic centimeter, milliroentgens per hour, counts per minute or roentgens per hour.

b. Alarms

Alarms on (1) rising signal (the setpoint is adjustable to any point on the scale), and (2) loss of signal implying circuit failure are provided.

c. Functional Test and Calibration Requirements

Each radiation monitoring channel has the capability to expose the detector to a radiation check source by energizing a solenoid actuated device. This will cause an up-scale indication verifying the operability of the channel. The check source has a long half-life and an energy emission with the spectra of the radiation being monitored.

For both safety and nonsafety-related monitors, check sources can be activated locally from the RM-23s or the RM-80s. The RDMS Host Computer can initiate check source actuation for nonsafety-related monitors and is prevented from initiating check source actuation of safety-related monitors by a disabling switch at the RM-80s. On high background radiation the RDMS Host Computer will disable check source actuation from the workstations for non-safety related monitors.

Calibration test is accomplished by inputting a pre-calibrated pulse signal to the channel. Reading at the local meter will verify the calibration of the channel.

d. Indicating Lights

Indicating lights at the local radiation monitoring cabinets monitor individual channel high radiation alarms and circuit failures.

e. Power Supplies

Power supplies are mounted at the local radiation monitoring cabinets, and provide the voltages for the modular component circuitry, relays and alarm lights. The power supplies also supply high voltage for the detector. Internal battery backup is provided to prevent loss of stored information in the event of loss of AC power.

f. Fail-Safe

Fail-safe circuits in each monitoring channel indicate channel failure caused by signal or power failure.

11.5.2.1 Channel Descriptions

a. Waste Gas Processing Monitor - Channels 6502, 6503 and 6504

Radiogas monitors are located online at three points within the Radioactive Gaseous Waste System. Monitor 6502 is located up-stream of the carbon delay beds, 6503 is located downstream of the carbon delay beds, and 6504 is located downstream of the waste gas compressors. These monitors serve as indicators of carbon bed performance, with control room annunciation to alert station operators of abnormal operation or conditions. Remote indication and annunciation are provided on the control panel for the Radioactive Gaseous Waste System and in the control room.

A high radiation signal on 6504 terminates waste gas system discharges to the ventilation stack by automatic closure of the waste gas discharge valve.

b. Condenser Air Evacuator System Gas Monitor – Channel 6505

This channel monitors the discharge from the shell-side vacuum pump exhaust header of the condenser for gaseous radioactivity, which is indicative of a primary-to-secondary system leak. During normal plant operation, the gas discharge is routed to the Primary Auxiliary Building exhaust filter system. During startup operation (hogging), the gases removed by the evacuation system are discharged to the atmosphere via the Turbine Building vent. A beta scintillator is used to monitor the gaseous radioactivity level. Remote indication and annunciation are provided locally and in the control room.

c. Boron Recovery System Monitors – Channels 6500 and 6501

Radiation monitors are located at two points within the Boron Recovery System (BRS). Monitor 6500 is located downstream of the boron recovery filters and upstream of the boron waste storage tanks. Monitor 6501 is located between the distillate cooler and the recovery test tanks. These monitors serve as indicators of BRS processing performance, with control room annunciation and indication to alert station operators of abnormal operation or conditions. Remote indication and annunciation are provided on the control board for the BRS.

d. Primary Component Cooling Liquid Monitors – Channels 6515 and 6516

These two channels continuously monitor trains A and B of the Primary Component Cooling System for radioactivity indicative of a leak from the Reactor Coolant System or one of the other radioactive systems which exchange with the Primary Component Cooling System. Indication and annunciation are provided locally and in the control room.

The gamma scintillation detectors are located in offline liquid samplers.

e. Waste Processing System Liquid Effluent Monitor-Channel 6509

All discharges from the station via the WL Test Tank discharge header are monitored by an online gamma scintillation detector. This includes all discharges from the Test Tank itself, as well as steam generator blowdown demineralizer regenerant solution from the waste holdup sump or the bottom of the demineralizer beds. (See Subsection 10.4.8.2.)

Automatic valve closure action is initiated by this monitor to prevent further release after a high-radiation level is indicated and alarmed. Control room and remote indication and annunciation are provided.

f. Steam Generator Blowdown Liquid Sample Monitor – Channels 6510, 6511, 6512, 6513 and 6519

These channels monitor the liquid phase of the secondary side of the steam generator for radioactivity concentrations, which would indicate a primary-to-secondary system leak, providing backup information to that of the condenser air evacuation system gas monitor. A sample from the bottom of each steam generator is continuously monitored by a scintillation counter mounted in line with an offline type sample assembly.

Monitor 6519 is an offline detector with pumping system which monitors the flash tank discharge.

High activity alarm indications are displayed at the detector location and in the control room.

In the event of a high activity alarm from any monitor, the isolation valve in the blowdown flash tank discharge closes.

g. Reactor Coolant Letdown Gross Activity Monitor – Channels 6520-1, 6520-2

The reactor coolant letdown monitoring system is in service whenever normal letdown is in service and has no automatic functions. This monitor provides indication of primary coolant radioactivity concentration over a wide range of operating conditions and assists in the detection of failed fuel.

The design utilizes an adjacent-to-line detector (Geiger Mueller type) positioned in a shield that provides a collimated view of the letdown line. The detector is placed far enough from the RCS to allow for sufficient N-16 decay. The monitor readout indication is in mR/hr and has a range of 10^{-1} to 10^{4} mR/hr. This detector has a minimum sensitivity of about $1x10^{0}$ μCi/cc in a 15 mR/hr background depending upon the isotope of interest. The detector exhibits good energy linearity for gamma rays between 210 keV and 1333 keV and provides acceptable response to the isotopes listed in Table 11.5-1. The upper detection limit of the monitor is about $1x10^{3}$ μCi/cc depending upon the isotope of interest thereby providing a range of 1 to 1000 μCi/cc.

h. Liquid Waste From Evaporators to Waste Test Tanks – Channel 6514

A scintillation detector in an online sampler continuously monitors the waste liquid transferred to the waste test tanks. Increasing radioactivity concentrations indicate a potential problem upstream or a need to filter the tank contents or add water to the tanks to reduce the radioactivity concentration. Control room and remote indication and alarms are provided. A high alarm level closes the waste test tank inlet valves.

i. Resin Sluicing Operation Monitors - Channels 6560, 6561 and 6564

Geiger-Mueller tubes clamped to the process pipe monitor the resin sluicing operation. They provide indication and alarm locally and at the waste management panel. They do not interface with the RDMS host computer system. The function of these detectors is to monitor filter failure and to indicate completion of sluicing operation.

j. Main Steam Line Radiation Monitors - Channels 6481-1, 6481-2, 6482-1, and 6482-2

Online gamma-sensitive detectors are located on each main steam line upstream of the safety relief valves. As required by NUREG-0737, these monitors provide a method of quantifying high-level releases of radioactive noble gasses after an accident. Control room indications and alarms are provided.

The monitors display steam line dose rates in "mr/hr." The noble gas release rate is calculated from the dose rate by using a procedure.

k. Plant Vent Monitor - Channels 6528-1, 6528-2, 6528-3, and 6495

The monitoring capability associated with the main plant vent is described in Subsection 12.3.4.

l. Fuel Storage Building Exhaust Monitor – Channel 6562

The monitoring capability associated with air exhaust is described in Subsection 12.3.4.

m. Turbine Building Sump Liquid Radiation Monitor - Channel 6521

This online gamma scintillation detector is located on the Turbine Building sump effluent line. At a pre-determined radioactive concentration, this monitor will alarm and automatically terminate the discharge and isolate the sump.

n. Containment Online Purge Monitor - Channels 6527A, 6527B

These detectors monitor the air exhausted via the containment purge. They utilize GM tubes sensitive to Xe-133. These detectors provide measurement of the activity of the containment purge and provide isolation on a high signal. The detectors and their associated microprocessors are Class 1E. Each monitor utilizes a two-out-of-two detector logic such that two detectors must be in alarm before the monitor initiates an isolation signal.

o. Auxiliary Condensate Monitor - Channel 6490

An offline, skid-mounted monitor draws a sample from the auxiliary condensate return line. In the event that the auxiliary steam should become contaminated, this monitor will automatically terminate the condensate return to the auxiliary steam boiler and isolate the return piping.

p. Storm Drains Monitor - Channel 6454

This is an offline, skid-mounted monitor which continuously samples the storm drainage. The monitor design is identical to the steam generator blowdown liquid sample monitor. An auxiliary pump provides the necessary sample flow. Indication and alarm are provided only locally. This monitor does not interface with the RDMS host computer system. On a high alarm a composite grab sample is automatically obtained. An automatic continuous operating composite sampler is also provided.

q. Water Treatment Liquid Effluent Radiation Monitor – Channel 6473

This online gamma scintillation detector is located on the Water Treatment effluent line that receives water from the Water Treatment Neutralization Tank, Condensate Polishing Low Conductivity Tank or Condensate Polishing Resin Regeneration Megarinse Waste. High activity alarm indications are displayed at the detector location and in the Main Control Room. At a pre-determined radioactive concentration, this monitor will alarm and provide a signal to CPS PLC (1-CPS-CP-563) to automatically terminate the discharge. **(UFCR 03-039)**

11.5.2.2 Alarm Setpoints

The alarm setpoints for the process and effluent radiation monitoring system are provided in Table 11.5-1.

In establishing the site boundary concentration, it is assumed for continuous releases that average annual meteorology exists for gaseous discharges and average annual circulating water flow exists for diluting liquid discharges. For intermittent or off-normal releases, short-term meteorology and actual dilution water flow is assumed for gaseous and liquid discharges, respectively.

11.5.2.3 Design Evaluation

a. The reactor coolant letdown gross activity monitor (RM6520) serves as a failed fuel advisory and provides a function independent of the discharge monitoring system. This monitor is not required for detection of fuel cladding breach (see Appendix 7A, Deviation No. 8). The liquid and gaseous waste discharge monitoring system is employed to maintain surveillance over the release of radioactivity, and is provided with the following features:

1. The check source is operated by command from the display/control console or by command at the remote cabinet.

2. If the reading falls off scale at any time, an indicator visible to the operator in the control room will alarm.

3. Power failure is indicated by its own indicator, and does not alarm as high radiation failure.

b. An evaluation of instrumentation function, relative to monitoring and for controlling release of radioactivity from various plant systems, is discussed below.

1. Liquid and Gas Wastes

For ruptures or leaks in the Waste Processing System, station area monitors and the vent stack monitor (see Subsection 12.3.4) will alarm on an increase in radiation level over a preset level. For cases where leaks are involved, the operator may control activity release by system isolation. For more severe postulated accident cases, such as rupture of a carbon delay bed, activity release is not controlled. The environmental consequences of the postulated accidents are based on no instrument action. For inadvertent releases relative to violation of administrative procedures, monitors provide means for limiting radioactivity release as well as alarming functions. The waste gas vent monitor will trip the flow control valve in the discharge line when the radiation level exceeds a preset level. Where liquid waste releases are involved, the waste processing system liquid discharge monitor trips shut a valve in the liquid waste discharge line when the radiation level in the discharge line exceeds a preset level.

2. Liquid Waste Release Procedure

The release of liquid waste is under administrative control. The normal procedure for discharging liquid waste is:

(a) A batch of waste is collected in one waste test tank.

(b) The tank is isolated.

(c) The tank contents are recirculated to mix the liquid.

(d) Samples are taken for analysis before release.

(e) If analysis indicates that release can be made within the terms of the operating license, the quantity of activity to be released is recorded on the basis of the liquid volume in the tank and its activity concentration. Release is made when it is determined that the release will be within the operating license.

(f) To release the liquid, an operator must unlock and open the last stop valve in the discharge line (which is normally locked shut); open a second valve, which trips shut automatically on high radiation signal from the monitor (6509); start a test tank pump and establish the normal flow rate using the flow indicator provided; and finally, close the recirculation valve. Liquid is now being discharged.

11.5.2.4 Sampling

Sampling provisions are installed at the locations shown in Table 11.5-3. The samples are drawn from lines or tanks, and transported to the radiochemistry laboratory where the samples are analyzed for radioactivity content.

The sampling program is defined by a series of procedures for obtaining and analyzing representative samples. The administrative and procedural controls are in accordance with Regulatory Guide 4.15 (Position C) and Regulatory Guide 1.21 (Position C). Prior to sampling, large tanks of liquid waste are well mixed to assure uniform distribution of particulate solids. Sample lines are flushed for a sufficient period of time prior to sample extraction in order to remove sediment deposits and air and gas pockets. Sample collection techniques which preclude losses of radionuclides are employed.

Effluent ventilation release points are monitored continuously. Particulate sampling is done isokinetically for major release points.

11.5.2.5 Laboratory Analytical Instrumentation and Capabilities

Samples of process and effluent gases and liquids are analyzed in the laboratory. Laboratory instrumentation includes the appropriate means of detection for alpha and beta analysis, and gamma isotopic analysis.

Sample volume and counting time are chosen to yield the required sensitivities. Corrections are made for sample-detector geometry, sample self-absorption, and other parameters as necessary to assure accuracy. Gross alpha analysis of all liquid effluent samples is performed by liquid scintillation or by direct counting of evaporated deposits.

Gross alpha analysis of air particulate filters is performed by direct counting of the filters.

Alpha isotopic analysis is performed using the silicon surface barrier detector with the multichannel analyzer system, or by liquid scintillation.

Gamma spectrometry is used for isotopic analysis of liquid, gaseous and airborne particulate and iodine samples. A high efficiency, high-resolution HpGe detector is available, in conjunction with a multichannel analyzer, for resolving complex gamma spectra.

Effluent tritium samples are collected by various methods and analyzed by liquid scintillate.

11.5.2.6 Calibration and Maintenance of Effluent Radiation Monitors

A primary calibration is performed on a one-time basis, using typical isotopes of interest to determine proper detector response. Further primary calibrations are not required since the geometry cannot be significantly altered within the sampler. Calibration of samplers is then performed based on a known correlation between the detector responses and multiple secondary standards.

Secondary standard calibrations are performed with radiation sources of known activity. This calibration confirms the channel sensitivity. The secondary standard calibration is performed by placing the secondary standards on the sensitive area of the detector and comparing detector response to the detector response at the time of primary calibration.

The radiation monitoring system channels will be status checked at least daily and calibrated periodically. If a monitor functionally tested quarterly provides a control function on release, it will be functionally tested prior to that release.

Calibration of the indicating channels is performed following any equipment maintenance which could result in reducing the accuracy of the instrument indication. It is also done any time use of the ion chamber precalibrated pulse test signal or the radioactive check source indicates instrument drift.

A burn-in test, operational test and isotopic calibration of the Complete Radiation Monitoring System are performed at the factory. Field calibration after system installation will be performed using calibration sources and their decay curves provided with the system. The sample chambers will be decontaminated in situ periodically and, if required, are easily replaceable.

11.5.3 Effluent Monitoring and Sampling

General Design Criterion 64 requires monitoring of effluent discharge paths. Compliance with requirements is discussed in Subsection 11.5.2.

Airborne radioactivity monitoring is discussed in Subsection 12.3.4.

11.5.4 Process Monitoring and Sampling

Means are provided to control and monitor the release of radioactivity to the environment in accordance with the requirements of General Design Criteria 60 and 63 as discussed in Subsection 11.5.2.

TABLE 11.5-1 PROCESS AND EFFLUENT RADIATION MONITORS

Instrument Tag No. Re-	Description	Detector Type	Det Back Grd mr/hr	Range Low-High (µci/cc)	(Note 5) Alarm Set Point (µci/cc)	Reference Isotope	Detector Qty	Safety Class	Energy* Level	Loop Diag. 1-NHY	P&Id 1-NHY
6454	Storm Drains	Gamma Scint	0.5	10^{-6} 10^{-2}		Co^{58},I^{131},CS^{137}	1	Non 1E	Note 2	506765	SD-20404
6502	Waste Gas Inlet to Carbon Delay Beds	Gamma Scint	15.0	10^{-2} 10^{+2}		Xe^{133}	1	Non 1E	Note 1	506897	20772
6503	Waste Gas Compressor Inlet	Gamma Scint	15.0	10^{-3} 10^{+1}		Kr^{85}	1	"	Note 1	506898	20770
6504	H_2 Gas Compressor Disch.	Gamma Scint	15.0	10^{-3} 10^{+1}		Kr^{85}	1	"	Note 1	506899	20773
6505	Condenser Air Evac	Beta Scint	0.5	Note 3		Xe^{133}	1	"	Note 1	506055	20774
6500	Boron Recovery Stor. Tank Inlet	Gamma Scint	1.0	$10^{-5}10^{-1}$		Co^{58},I^{131},CS^{137}	1	"	Note 2	506105	20856
6501	Boron Recovery Test Tank Inlet	Gamma Scint	2.5	10^{-6} 10^{-3}		Co^{58},I^{131},CS^{137}	1	"	Note 2	506113	20861
6515,6516	Primary Component Cooling Water	Gamma Scint	2.5	10^{-7} 10^{-3}		Co^{58},I^{131},CS^{137}	2	"	Note 2	506190, 506194	20211, 20205
6509	Liquid Waste Test Tk Disch to CWS	Gamma Scint	2.5	10^{-6} 10^{-2}		Co^{58},I^{131},CS^{137}	1	"	Note 2	506927	20831
6514	Waste Liquid From Evaporators	Gamma Scint	2.5	10^{-6} 10^{-2}		Co^{58},I^{131},CB^{137}	1	"	Note 2	506931	20831

* See Table 11.5-1 (Sheet 3) for notes.

Instrument Tag No. Re-	Description	Detector Type	Det Back Grd mr/hr	Range Low-High (µci/cc)	(Note 5) Alarm Set Point (µci/cc)	Reference Isotope	Detector Qty	Safety Class	Energy* Level	Loop Diag. 1-NHY	P&Id 1-NHY
6510, 6511, 6512, 6513	Steam Gen Blowdown Sample Loops 1,2,3,4	Gamma Scint	2.5	10^{-6} 10^{-2}		Co^{58},I^{131},CS^{137}	4	"	Note 2	506815	20521
6519	Steam Gen Blowdown Flash Tank Drain	Gamma Scint	2.5	10^{-7} 10^{-3}		Co^{58},I^{131},CS^{137}	1	"	Note 2	506734	20626
6520	Reactor Coolant Gross Activity Monitor	GM	15	10^{-1} 10^{+4} mr/hr		Co^{58},I^{131},CS^{137}		"	Note 2	506269	20722
6481-1, 6482-1 6481-2, 6482-2	Main Steam Line Monitor	Gamma Scint	2.5	10^{-0} 10^{+5}mr/hr		Xe^{133}		"	-	506551 -2, -3, -4	20580 20581
6490	Aux Steam Cond	Gamma Scint	0.5	$10^{-7}10^{-3}$		Co^{58}, I^{131}, CS^{137}	1	Non 1E	Note 2	507165	20908
6521	Turb. Bldg. Sump Liq. Monitor	Gamma Scint	2.5	10^{-6} 10^{-2}		Co^{58}, I^{131}, CS^{137}	1	Non 1E	Note 2	506713, 506716	20195
6527A1, A2 B1, B2	COP Monitors	GM	2.5	10^{1} -10^{6}cpm		Xe^{133}	4	1E	Note 1	506211	20504
6560	Resin Sluice Line	GM	100	Note 4		Xe^{133}	1	Non 1E		506694	20252
6561	Resin Transfer Line	GM	100	Note 4		Xe^{133}	1	Non 1E		506694	20735
6564	Sluice Pump Line	GM	100	Note 4		Xe^{133}	1	Non 1E		586692	20252
6473	Water Treatment Liquid Effluent Radiation Monitor	Gamma Scint	2.5	10^{-6} 10^{-2}		Co^{58}, I^{131}, CS^{137}	1	Non 1E	Note 2	506976	20040

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Note	Isotopes	Max. Beta Energy (Mev)	Predominant Gamma Energy (Mev)
1	Xe^{133}	0.346	0.081
	Xe^{135}	0.92	0.249
	Kr^{85}	0.67	0.514
	Kr^{85m}	0.82	0.150
2	I^{131}	0.606	0.364
	I^{133}	1.27	0.53
	Cs^{134}	0.662	0.604
	Cs^{137}	0.514	0.662
	Co^{58}	0.474	0.81
	Co^{60}	0.314	1.17, 1.33

3 Condenser Air Evacuation Monitor to have output in counts per min (cpm) (10^1 to 10^6).

4 Monitors 6560, 6561, 6564 have output in mr/hr (10^0 to 10^5).

5 Radiation monitoring setpoints are varied during operation to follow station operating conditions. Setpoints are maintained within the bounds established in the Technical Specifications. The methodology for establishing the setpoints is found in the Station Offsite Dose Calculation Manual (ODCM) and/or station operating procedures.

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 Facility Design Features

The radiation protection philosophy for the design of Seabrook Station is to restrict radiation exposure to plant personnel and the general public to within the limits of 10 CFR 20 and 10 CFR 50, while ensuring high flexibility and availability within the power generation and safety objectives of plant operation.

This philosophy can be summarized in several basic design goals intended to minimize exposures:

a. To minimize, to the extent possible, the production of radioactive isotopes, e.g., the steps taken to reduce crud production described in Subsection 12.2.1.3.

b. To minimize the extent of areas housing radioactive equipment and piping through efficient arrangement of equipment and systems.

c. To shield the normally occupied areas from radiation.

d. To minimize exposures within high radiation areas by, first, controlling access to those areas, and secondly, through design of systems and equipment for reliability and ease of maintenance.

The plant is designed to permit periodic online equipment inspection and maintenance, radioactive material handling, decontamination and cleanup, and access to vital plant areas during normal plant operation, including anticipated operational occurrences. Postulated accidents are also considered in the determination of radiation exposure of Engineered Safety Features and other materials. In addition, these accidents are evaluated for access to, and habitability of, the control room, including ingress and egress, for the duration of the accident.

Offsite radiation exposures following postulated accidents are discussed in Chapter 15; those from processed radioactive material releases during normal operation are discussed in Chapter 11.

The primary objective of plant shielding is to provide for the protection and safety of all plant personnel and the general public under all normal and anticipated abnormal plant operating conditions. Reactor shielding, along with the radiation monitoring system and access control procedures, supplemented by periodic radiation surveys and radiochemical analysis, ensure that radiation exposures of the general public and plant personnel do not exceed the limits set by the federal regulatory agencies. The maximum allowable design dose rates for all plant areas, in conjunction with anticipated occupancy, limit the integrated whole body dose to less than 5 rem per calendar year. All areas that house radioactive materials are appropriately marked in accordance with Part 20 of Title 10 of the Code of Federal Regulations (10 CFR 20).

Plant operating personnel and the general public are protected by radiation shielding wherever a potential radiation source may exist. The shielding design philosophy embodies the following objectives:

a. To restrict the potential radiation dose to operating personnel during normal operation to within the limits of 10 CFR 20.

b. To adequately protect the operating personnel in the unlikely event of an accident, in order to allow termination of accident conditions and mitigation of the consequences without undue risk to the general public.

c. To protect equipment from excessive radiation exposure to prevent malfunctions due to radiation-induced failures.

d. To maintain the radiation exposure of the general public from normal operation to within the limits of 10 CFR 20.

The guidance provided by Regulatory Guide 8.8 has been utilized extensively in the plant radiation protection design philosophy, as described in the following pages of this section.

The radiation sources and associated input parameters, assumptions, and methodology described in Section 12.2 represent those used to establish original shielding design. The radiation sources utilized for the design of the primary shield were developed by the reactor vendor and are based on a standard four loop Pressurized Water Reactor. The radiation sources in the spent fuel and the N-16 source were also developed by the reactor vendor, and are based on a reactor power of 3565 MWt. The radiation sources in the primary coolant system and the auxiliary systems are based on a reactor power of 3654 MWt, a one-year fuel cycle, and 1 percent fuel element defects.

The impact on plant shielding requirements was evaluated for an analyzed reactor power of 3659 MWt and operation with an 18-month fuel cycle. This represents a minor change from the original design basis. The conservative analytical techniques used to establish the original shielding requirements, and the Station Technical Specification which will restrict the reactor coolant activity to levels significantly less than the 1 percent fuel defects, ensure that operation at the licensed core power level will have no significant impact on shielding requirements and safe plant operation.

12.3.1.1 Radiation Zones and Access Control

The shielding design bases for work areas are a combination of the design radiation level and anticipated occupancy times. The plant is divided into zones dependent upon the intensity of radiation within the given area. Areas within these zones are posted in accordance with the regulations of 10 CFR 20. Occupied areas within a zone are limited to the same radiation ranges as prescribed for that zone. Zone classifications are presented in Table 12.3-1 and Table 12.3-2.

Zone boundaries, decontamination facilities and location of radiation monitors are shown in Figure 12.3-1, Figure 12.3-2, Figure 12.3-3, Figure 12.3-4, Figure 12.3-5, Figure 12.3-6, Figure 12.3-7, and Figure 12.3-8, Figure 12.3-9, Figure 12.3-10, Figure 12.3-11, Figure 12.3-12, Figure 12.3-13, Figure 12.3-14, Figure 12.3-15, Figure 12.3-16, and Figure 12.3-17. The arrangement of the chemistry lab, health physics facilities and counting room is shown in Figure 12.3-17. Shield wall thicknesses for all major sources of radiation are given in Table 12.3-3, Table 12.3-4, Table 12.3-5, Table 12.3-6, Table 12.3-7, Table 12.3-8, Table 12.3-9, Table 12.3-10, Table 12.3-11, Table 12.3-12, and Table 12.3-13.

Access control points are also shown in Figure 12.3-1, Figure 12.3-2, Figure 12.3-3, Figure 12.3-4, Figure 12.3-5, Figure 12.3-6, Figure 12.3-7, Figure 12.3-8, Figure 12.3-9, Figure 12.3-10, Figure 12.3-11, Figure 12.3-12, Figure 12.3-13, Figure 12.3-14, Figure 12.3-15, Figure 12.3-16, and Figure 12.3-17, while discussion of access control and its implementation can be found in Section 12.5. The entire Containment Building is a controlled access area.

The turbine generator structure areas, administrative offices, turbine plant service areas and the control room are designated Zone I. Areas such as the local control space in the Primary Auxiliary Building, the waste disposal area, and the operating deck of the spent fuel storage area, are generally designated Zone II. Intermittently occupied work areas, such as valve galleries, are designated Zone III. Typical Zone IV areas include steam generator compartment areas (after reactor shutdown) and areas outside of containment pipe penetrations. A typical Zone V is the volume control tank area. Certain areas of the Containment are accessible for a limited time during normal plant operation.

The radiation counting room is designated Zone I (less than 0.5 mrem/hr); however, sufficient shielding is provided to assure that the background dose rate is low enough (less than 0.1 mrem/hr) to permit accurate operation of counting equipment.

Shielding has been designed by identifying source strengths within an area and then providing sufficient shielding to achieve the specified dose rate in adjacent zones. The source strengths are based on 1 percent failed fuel and maximum expected activation product levels, so that the actual radiation levels experienced within the station are expected to be less than the design values. Concrete shield thickness was in most cases determined by rounding from the calculated required thickness to the next higher 6-inch increment. In a few cases, where space was limited, the next 3-inch increment was used.

12.3.1.2 Handling of Nuclear Materials

Most systems included in the Primary Auxiliary and Waste Processing Buildings are used to process the radioactive byproducts produced in, and which leak from, the Reactor Coolant System during normal power operation. The design of the systems in the letdown purification and general waste processing systems reflects the concept of minimizing the exposure of plant personnel.

New fuel handling is discussed in Subsection 9.1.4.

Storage and handling of radioactive sources is discussed in Subsection 12.5.3.7.

12.3.1.3 Shielding and Layout Features

a. Plant Layout

Plant layout includes optimal location of radioactive components. The most radioactive systems are located toward the interior and on the lower plant levels, with less radioactive systems located toward the outside.

The plant layout provides for personnel access which maintains occupational doses as low as reasonably achievable (ALARA). Passage through a higher radiation area to obtain access to a lower radiation area is avoided, thus minimizing unnecessary accumulation of occupational exposure.

Wherever practical, components are shielded individually to keep exposure ALARA during maintenance periods. Shielded pipe chases are utilized extensively to segregate radioactive piping from normal occupied areas. Reach rods are used where required to place the operator in a low radiation area during valve operation. Auxiliary control boards are located with ALARA occupational exposure in mind.

When sources of sufficient strength are present, labyrinth entrances are utilized to minimize the contribution in the walkways. One or two scatter labyrinths are used as required by scattering calculations.

Periodic review of plant design and equipment arrangement aimed at maintaining occupational radiation exposures (ORE) ALARA resulted in a number of design changes, as illustrated in the examples below:

1. Access to the stairways in the RHR Vaults may be restricted below elevation 3'-2" due to equipment and pipe shine.

2. Shielding above the demineralizers in the Primary Auxiliary and Waste Processing Buildings was increased.

3. Primary sample heat exchanger and sink room was rearranged to reduce shine.

4. Evaporator equipment was rearranged to minimize radiation levels in adjacent walkways.

5. Areas of potentially excessive radiation levels were identified and space for possible future shielding was reserved.

b. Equipment Layout

The criteria for the arrangement of equipment containing radioactive sources were developed specifically for maintaining occupational doses ALARA and for ease of maintenance.

1. Filters (Liquid)

   (a) Each potentially radioactive filter is located inside an individual shielded compartment. This minimizes the contribution to radiation levels from adjacent filters during maintenance periods. One exception to this are the vendor supplied waste liquid processing system filters. These filters may be in a common shielded compartment with demineralizer vessels, or outside a shielded area if the expected dose is Zone III or lower.

   (b) For PAB and WPB filters, where changeouts are required, adequate space is provided for use of the remote filter handling cask in removing the filter (lateral room to swing the vessel head clear, and head room for lifting the cartridge), loading the filter into the cask, and transportation to the solid waste area. Attention is given to ensure that there are no interferences or obstructions in the path. The shield wall for waste liquid processing equipment is vendor supplied for the vendor system. This ensures interferences or obstructions in filter/resin manipulations are accounted for.

   (c) All valves and instrumentation associated with filters are located outside the compartment. Normally, filter process valves are operated by remote manual mechanical linkage which extends to a low radiation zone, to minimize operator exposure during normal operation.

   Where practical, filters are located near the solid waste area to minimize the chance of spillage in transit.

2. Demineralizers

   The primary means of processing radioactive water to be discharged is through a vendor-supplied system. This system meets the intent of the design requirements in Regulatory Guide 1.143 and NUREG 0800. (see Section 1.8)

   Other aspects of radioactivity processing systems are:

(a) Capability is provided to remotely remove all primary- side resins by flushing. All initially installed resin transport lines are designed to avoid resin traps. Butt welds instead of socket welds and five diameter bends instead of fittings are used in downward-sloping pipe without any sections horizontal or upward-sloping. Procedural controls are in place to prevent normal access during times of elevated dose rates.

(b) The resin fill line is shaped to prevent direct radiation streaming.

(c) Each potentially radioactive primary-side demineralizer in the Primary Auxiliary Building and Waste Processing Building is located inside shielded compartments or cubicles in order to reduce shielding problems during maintenance periods.

(d) All valves and instrumentation associated with these demineralizers are located outside the compartment. All demineralizer process valves, except for the vendor-supplied system, are operated by remote manual mechanical linkage which extends to a low radiation zone to minimize operator exposure during normal operation. The vendor-supplied system is designed to process water from floor drain tanks. The vendor-supplied components which concentrate radionuclides are all shielded to reduce any potentially elevated dose rates.

(e) Only the required process lines enter into, or pass through, the demineralizer cubicles, as access to the cubicles during normal operation will be strictly controlled.

(f) The demineralizers associated with the Steam Generator Blowdown System are not located in individual shielded cubicles, and are not provided with the capability for remote sluicing. These units normally treat secondary coolant with only minor or no contamination present. Processing blowdown through these demineralizers is based on maintaining the general area near the vessels as a radiation Zone II (<2 mr/hr). If significant primary-to-secondary leakage occurs, the primary method to process radioactive secondary liquid from the steam generators is to direct steam blowdown flash tank bottoms cooler discharge to the floor drain tanks. If no secondary pressure is available, the steam blowdown and wet lay-up pumps can be used. From the floor drain tanks, processing through the installed vendor system (WL-SKD-135) to the waste test tanks is the preferred method (reference Subsection 11.2.2.1). In addition blowdown processing may be through the blowdown evaporators in place of the Blowdown Demineralizer System.

3. Adsorber Beds

(a) The first and second waste gas adsorber beds are located in individual shielded compartments.

(b) The third-through-fifth beds are located in a common compartment with the most active bed farthest from the entrance.

(c) All valves are located outside the bed cubicles, and can be operated from a low radiation area.

4. Recombiners

(a) Post-accident recombiners are located inside the Containment Building, and are designed for operation in the accident environment.

(b) Shielding is provided by the 4½-foot-thick containment walls.

5. Tanks

   (a) Tank overflow lines are connected to prevent spillage on the floor and to prevent dissolved gases from escaping the tank. In limited cases the overflow lines are directed toward floor drains.

   (b) Controlled ventilation is provided for tanks containing aerated or hydrogenated fluids.

   (c) Tanks located outside of heated buildings are protected from freezing by steam heating panels.

   (d) Manual valves are located in, or have handwheel extensions to, low radiation zones.

6. Evaporators

   (a) Shielding is provided for individual evaporator units.

   (b) Instruments, valves in service lines, and evaporator sample points are located in low radiation zones.

   (c) Connections are provided for flushing and draining of pipe and equipment prior to maintenance.

7. Pumps

   (a) In order to perform major maintenance in as low a radiation zone as reasonably achievable, the pump or motor can be decontaminated, if necessary, and moved to a low radiation area. Temporary local shielding may also be used.

   (b) Pumps are designed with double mechanical seals to give a minimum leakage of radioactive fluid.

   (c) Remote instrumentation and switching is provided, as required.

8. Steam Generators

   The portions of the steam generators containing reactor coolant are shielded by the 4-foot-thick secondary shield walls.

9. Sampling Station

   (a) The sample sink room is separated from the sample heat exchangers by a shield wall.

   (b) Sample rates are limited to 1.5 gallons per minute by design.

   (c) The sample hood is ventilated to prevent the accumulation of gases.

   (d) The sample system is designed for a closed system line purge prior to sampling.

   (e) Shielding is provided at local sampling points, as required.

10. Penetrations

   (a) Where possible, penetrations through shield walls are offset from line-of-sight of the source.

   (b) Where necessary, the annulus between a pipe and its sleeve is packed with lead wool or lead-silicone foam with a nominal density of 150 lb/ft^3.

11. Instrumentation

   (a) Drains from instrument blowdowns are routed to radioactive drains.

   (b) Diaphragm seals or clean water seal legs are used, wherever practical, to minimize the volume or radioactive fluids entering low radiation areas via instrument impulse lines.

   (c) Radioactive gas and liquid samples are returned to process lines wherever practical.

   (d) Wherever practical, instruments are located in low radiation zones to permit extended access for calibration and testing.

12. Piping and Valves

In order to minimize concentrated pockets of crud in permanently installed radioactive systems:

(a) Piping 2½" and larger is butt-welded.

(b) Spent resin sluicing lines utilize five diameter bends to minimize the number of fittings.

(c) Valves are selected to avoid crud pockets.

(d) Piping layout avoids pockets wherever possible.

12.3.2 Shielding

The material most commonly employed for shielding is concrete. Where space is limited, steel or lead is substituted for ordinary concrete in equivalent thicknesses. Whenever cast-in-place concrete is replaced by concrete blocks (removable or fixed), the design assures protection on an equivalent shielding basis.

Analytical models were selected according to the source geometry under consideration. Tanks, vessels, and large pipe-sections containing radioactive materials were shielded by considering uniform cylindrical volumetric sources. Appropriate line source approximations were used for small pipes and tall vessels with small diameters. For a conservative estimate of dose rate, sources of irregular geometry were modeled by a point source of strength equivalent to the volumetric source.

The following techniques, codes, models, and assumptions were used:

- Point kernel integration methods were used.
- Buildup factors were accounted for inside the integrals.
- Self-shielding was taken into consideration.
- Concrete density was assumed to be 2.35 gm/cc.
- The maximum calculated shield thickness was specified for each component or radiation area.

- Source term data corresponds to 1 percent failed fuel with a power level of 3654 MWt.

Special protective design features to ensure that occupational radiation exposures will be ALARA are described in Subsection 12.3.1.3. The guidance given in Regulatory Guide 8.8 can be seen in these features.

Compliance with Regulatory Guide 1.69 is addressed in Section 1.8.

12.3.2.1 Shield Configurations

a. Reactor Shielding

1. Primary Shield

The primary shield is a large mass of reinforced concrete, 7½ feet thick at core midplane, that surrounds the reactor vessel and extends upward from the containment floor to form the walls of the refueling cavity. The primary shield is designed to:

(a) Reduce, in conjunction with the secondary shield, the radiation level from sources within the reactor vessel and Reactor Coolant System, and allow limited access to the Containment during normal operation.

(b) Limit the radiation level after shutdown from sources within the vessel, and permit limited access to the areas containing reactor coolant system equipment.

(c) Limit neutron flux activation of component and structural materials over the life of the plant.

2. Secondary Shield

The secondary shield is a reinforced-concrete structure that surrounds the reactor coolant equipment, pipes, pumps and steam generators. This shield protects personnel from gamma radiation emanating from reactor coolant activation products and fission products that are transported from the core by the reactor coolant. The neutrons emitted by the decay of carried-over N-17 are effectively eliminated by the concrete shielding employed for N-16. The secondary shield also supplements the primary shield function of attenuating direct core radiation. In addition, it permits limited access to the Containment during normal operation so that inspection of essential equipment may be accomplished without requiring plant shutdown.

3. Neutron Shield

In order to reduce dose rates and equipment activation on the containment operating floor during power operation, a supplementary shield has been designed to minimize the streaming of neutrons from the reactor cavity. A neutron shield consisting of Reactor Experiments Type 277 borated concrete, and which is integral to the permanent reactor cavity seal ring, is installed around the reactor vessel refueling flange. The neutron shield is suspended from the permanent seal ring and fills the annular area between reactor vessel refueling flange and the cavity wall. The neutron shield is approximately fourteen inches thick and supported from the bottom by a one-inch thick steel plate. Sectional and plan views of the permanent reactor cavity seal ring/neutron shield are shown in Figure 6.2-26 and Figure 6.2-28, respectively.

b. Containment Shielding

The containment shielding is a steel-lined, reinforced-concrete containment structure that completely surrounds the reactor building equipment. At full power operation, this shield attenuates the radiation level outside the primary-secondary shield complex, including radiation sources which become airborne during normal operation due to primary system leakage, to ensure that radiation levels outside the Containment are less that 0.5 mrem/hr. The containment structure also shields against radiation sources inside the Containment due to fission products released following postulated accidents. The integrated direct dose is less than 260 mrem immediately outside the Containment over a period of two hours after the design basis accident (DBA), which will permit access to such vital areas as the control room. The containment wall and dome are 4½-feet and 3½-feet thick, respectively.

c. Spent Fuel Shielding

This shielding provides protection during all phases of spent fuel removal and storage. Operations that require shielding of personnel are spent fuel removal from the reactor, spent fuel transfer through the refueling canal and transfer tube, spent fuel storage, and spent fuel shipping cask loading prior to transportation. All spent fuel removal and transfer operations are performed under borated water to provide radiation protection.

All accessible areas around the tube and canal are shielded. All shields were designed for a contact radiation dose rate of less than 100 mr/hr. Four inches of lead plate were added between the liner and concrete at the bottom of the canal. In the Enclosure Building a shield box was designed around the tube. This box consists of approximately 300 bricks weighing 50 pounds each. These bricks will be explicitly marked with a sign stating that potentially lethal radiation fields are possible if the bricks are removed during fuel transfer. The access point noted in Figure 1.2-3 is an inspection hatch (manway). This hatch is shielded with a three-foot concrete plug. This plug shall also be marked as noted above.

Minimum allowable water depth above a fuel assembly during fuel handling is 10 feet in the reactor cavity. This limits the dose at the water surface to less than 10.0 mrem/hr for an assembly in a vertical position. The minimum water depth in the spent fuel pool is 13 feet above the top of the fuel assemblies in the storage racks. For this depth, the dose rate at the water surface is less than 2.5 mrem/hr. Normal water depth above the stored assemblies is about 25 feet.

The 5-foot thick concrete walls of the fuel transfer canal and the 6-foot thick spent fuel pool walls supplement the water shielding, and limit the radiation dose levels in most working areas to less than 2.5 mrem/hr, and a maximum dose in some areas less than 100 mrem/hr.

The refueling water and concrete walls also shield personnel from activated control rod clusters and reactor internals that are removed at refueling times. Dose rates are generally less than 2.5 mrem/hr in working areas. However, certain manipulations of fuel assemblies, control rod clusters, or reactor internals may produce short-term dose rate levels in excess of 2.5 mrem/hr. Radiation levels in the working areas will be closely monitored during refueling operations to ensure that exposures for plant personnel do not exceed the integrated doses specified in 10 CFR 20.

d. Control Room Shielding

The control room shielding is designed in accordance with applicable regulations, to permit continuous occupancy by control room personnel following a DBA. This enables control room operators to maintain full control and to shut down the plant without personal hazard. The control room shielding is 2-feet thick, based upon an integrated dose during the 30 days following the DBA which does not exceed 5 rems whole body, or its equivalent to any part of the body, as required by General Design Criterion 19 of 10 CFR 50, Appendix A. A layout drawing of the control room is shown in Figure 1.2-32.

e. Plant Auxiliary Systems Shielding

Auxiliary shielding includes all concrete walls, covers and removable blocks that shield the numerous radiation sources in the radioactive waste disposal, makeup and purification, and chemical addition and sampling systems. Typical components that require shielding include the volume control tank, thermal regeneration demineralizers, waste drumming area and reactor coolant system drain tank. Shield wall thicknesses for components in auxiliary systems are given in Table 12.3-3, Table 12.3-4, Table 12.3-5, Table 12.3-6, Table 12.3-7, Table 12.3-8, Table 12.3-9, Table 12.3-10, Table 12.3-11, Table 12.3-12, and Table 12.3-13.

f. Turbine Shielding

The radioactive material inventory in the Turbine Building is very small since only secondary steam enters the area with small amounts of primary coolant leakage. Shielding is not, therefore, a major concern here, with wall and floor thicknesses determined from structural considerations.

g. General Plant Yard Areas

All shielding is designed so that the dose rates in plant yard areas which are frequently occupied by plant personnel remain below 0.5 mr/hr. These areas are surrounded by a security fence, and are closed off from areas accessible to the public for general safety.

12.3.2.2 Plant Shielding to Provide Access to Vital Locations for Post-Accident Operations

Following an accident, significant radioactivity may be released from the reactor core, presenting unusual hazards to operating personnel. A review was conducted to assess the projected amount of activity released, systems involved in transport of this activity, effect of the transported activity on plant dose rates and acceptability of dose rates in locations requiring access for necessary operations (vital locations).

This assessment employed core fission product release source terms consistent with NUREG-0737, Section II.B.2 (100% noble gas, 50% halogen, 1% other fission product). The assessment addressed both pressurized and depressurized accidents, and projected consequences of the release at post-accident times ranging from the onset of cold leg recirculation to 1 year. The assessment was based on an analyzed power level of 3565 MWt and a 1-year fuel cycle length.

The systems considered in this assessment included containment spray, chemical and volume control, safety injection, residual heat removal and combustible gas control.

Using the source term and system transport information described above, dose rates in various plant areas where projected. These projections considered shine, scatter and radiation streaming, including effectiveness of facility shielding. Levels in areas which must be accessed for operational tasks (vital locations) were tabulated, along with occupancy times, to verify projected exposures are within applicable limits. Such locations included the control room, technical support center, post-accident sample station, chemistry laboratory, switch gear room, radwaste control station, radiation controlled area tunnels and hydrogen analyzer area. High dose-rate areas are graphically depicted on area zone maps of the plant. These maps will aid in projecting exposures for potential post-accident operations not explicitly identified in the vital location table. Results of these projections demonstrate projected exposures in vital locations are within the GDC-19 and NUREG-0737 (Item II.B.2) criteria.

The assessments described above were incorporated into the Post-Accident Dose Engineering Manual, which is used in planning for post-accident operations. Rationale for not including several areas noted in NUREG-0737 (Item II-B.2) is delineated in this manual. A copy of the manual was provided to the NRC. The information in this document will be factored into the overall post-accident response actions.

The impact of an analyzed core power level of 3659 MWt and operation with an 18-month fuel cycle was evaluated, and it was determined that the post-accident operator exposure will continue to remain within regulatory limits. The information from the Post-Accident Dose Engineering Manual, modified to reflect the licensed core power level, is factored into the overall post-accident response actions.

12.3.3 Ventilation

The station ventilation system has been designed to provide a maximum of safety and convenience for operating personnel, construction workers and site visitors working both within the station radiologically controlled area and in station buildings outside the radiologically controlled area during normal operating and anticipated operational occurrences. The potential exposure to onsite personnel and to members of the general public resulting from airborne radionuclides from station operation complies with 10 CFR Part 20 and 10 CFR Part 50, respectively.

12.3.3.1 Ventilation Design Bases

Descriptions of the ventilation systems for each building which can be expected to contain radioactive materials, including design bases, are contained in Section 9.4. Diagrams associated with the descriptions show equipment, air flow patterns, and expected flow rates for normal and emergency conditions.

A description of the ventilation systems for the control room complex is contained in Subsection 9.4.1, Figure 9.4-1, Figure 9.4-2, and Figure 9.4-3, and shows equipment, air flow patterns, and expected flow rates. Section 6.4 discusses the habitability and life support systems of the control room complex with respect to NRC General Design Criterion 19.

In each case, air flow has been directed from areas of low potential airborne radioactivity to areas of higher airborne radioactivity by exhausting from the areas of higher radioactivity. The ventilation rate for the areas of higher radioactivity was determined both from the ventilation rate required to remove equipment heat, piping and electrical losses and from the ventilation rate required to control the concentration of airborne radioactivity. The ventilation was designed to meet the exposure limits for airborne concentrations listed in 10 CFR 20. Maintenance of a negative pressure by the exhaust systems in the areas of higher radioactivity induces an air flow from corridors and operating areas preventing the exfiltration of airborne radioactivity to clean areas normally occupied by operating or maintenance personnel.

Testing and maintenance will be performed in accordance with the criteria presented in Regulatory Guide 1.52 for the safety-related filter systems.

Failure to meet these in-place testing criteria will necessitate the change-out of filters or adsorbers.

12.3.3.2 Provisions for Localized Ventilation

Provisions for localized ventilation during maintenance and refueling operations are provided to the extent practicable to reduce concentrations of airborne radioactivity in areas not normally occupied where maintenance of in-service inspection has to be performed.

12.3.3.3 Exhaust Filtration

The exhaust from each area which can be expected to contain significant airborne radioactivity is processed through HEPA or HEPA and carbon air cleaning systems before being discharged to the unit plant vent.

Air cleaning units are provided for the containment enclosure emergency exhaust, the fuel storage building emergency exhaust, the containment purge exhaust, the primary auxiliary building exhaust for areas with a potential for significant airborne radioactivity, and the waste processing building exhaust for areas with a potential for higher airborne radioactivity.

In addition, recirculation air cleaning units are provided for the main control rooms and the containment structures.

Descriptions of the air cleaning systems, including design bases for the containment enclosure emergency exhaust and the fuel storage building emergency systems, including the design bases for the remainder of the buildings, are contained in Section 9.4.

Compliance with Regulatory Guide 1.52, Revision 2, of the containment enclosure emergency air cleaning unit, the fuel storage building air cleaning unit, and the control room emergency filtration subsystem is detailed in Table 6.5-1, Table 6.5-2 and Table 6.5-3 respectively.

The containment purge exhaust air cleaning unit, the primary auxiliary building normal exhaust air cleaning unit and the waste processing building air cleaning unit are equipped with quick-release filter clamps to minimize exposure of maintenance personnel when changing prefilters, medium efficiency filters and HEPA filters. The remainder of the air cleaning units employ standard threaded filter clamping devices because they are not expected to be exposed to more than minimal quantities of radioactive particulates.

The containment purge exhaust air cleaning unit, the primary auxiliary building normal exhaust air cleaning unit, the fuel storage building emergency exhaust air cleaning units and the containment enclosure emergency exhaust air cleaning unit are provided with bulk fill adsorber beds and guard beds, where applicable. The carbon for the adsorber and guard beds is pneumatically removed and filled, which minimizes exposure of the maintenance personnel to contaminated carbon. The control room emergency recirculation air cleaning unit and the containment recirculation air cleaning unit employ tray-type carbon adsorbers. The waste processing building air cleaning unit has no adsorber bed.

Access from both sides of the air cleaning units for maintenance and changing of filters is provided for the containment purge exhaust air cleaning unit, the fuel storage building emergency exhaust air cleaning unit and the primary auxiliary building normal exhaust air cleaning unit. Aisle space and clear means of ingress and egress are provided for the handling of filters and carbon bed carbon removal/fill equipment.

A layout of the primary auxiliary building normal exhaust unit provided in Figure 12.3-18 is an example of the filter bank spacing and access for maintenance.

The radiation control area (RCA) of the Administration and Service Building is provided with a once-through ventilation system. The exhaust system maintains a negative pressure on the entire RCA portion of the building, preventing the exfiltration of airborne radioactivity to the clean areas. The exhaust air is processed through 55 percent medium efficiency filters and then through HEPA filters. There is no adsorber bed provided for this system. The air is directed within the RCA from areas of low potential airborne radioactivity to areas of higher potential airborne radioactivity.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

12.3.4.1 Area Radiation Monitoring Instrumentation

a. Objectives and Design Basis

1. Detectors are located in areas that may be normally occupied without restricted access and which may have a potential for radiation fields in excess of the radiation zones described in Subsection 12.3.1.

2. The detectors provide on-scale readings of dose rate that include the design maximum dose rate of the radiation zone in which they are located, as well as the maximum dose rate for anticipated operational occurrences.

3. Each monitor has local visual and audible alarms, with variable setpoint.

4. Indication and annunciation are available in the main control room.

5. The design objectives and location criteria are in conformance with 10 CFR Part 20, Part 70 and Part 50, Appendix A, General Design Criteria 63 and 64, and Regulatory Guides 1.21, 8.2 and 8.8.

6. Post-accident monitoring instrumentation is provided as discussed in Section 7.5.

b. System Description

The digital computer-based Radiation Data Management System (RDMS) consists of local microprocessors for each channel, interconnected by a redundant communication loop to a redundant host computer system. Either of the two computers can by itself provide the total computing capacity required for satisfactory operation of the RDMS. The host computer system, in turn, is connected to an operator display/control console in the control room, the health physics control point, the RDMS computer room, the Main Plant Computer System (MPCS) computer room and the hot chemistry lab. The area radiation monitoring system instrument engineering diagram, Figure 12.3-19, shows an overview of the system, its components and location.

Table 12.3-14 lists the various area radiation monitoring channels provided and their pertinent design information, such as detector type, range, background radiation, safety class, alarm setpoints, referenced drawings for location of area radiation detectors, etc.

Class 1E area radiation equipment is supplied from Class 1E uninterruptible power supplies (UPS).

Except for the post-LOCA containment monitors and other high-range monitors, each channel is equipped with a radioactive check source which can be actuated from the main control room during test. The post-LOCA monitors and other high-range monitors use an electronic signal to test the circuit.

A typical channel is shown in Figure 12.3-19.

High radiation levels during refueling at the manipulator crane area in the containment structure initiates isolation of the containment purge and vent system.

Those detectors which are designated as non-Class 1E, and are located inside the containment structure are not designed to operate following a major LOCA, and are assumed to be not available to monitor post-LOCA conditions inside Containment.

Refer to Subsection 11.5.2 for a discussion of the local microprocessor provisions and operating details.

1. Area Monitor Detectors

The area monitors employ Geiger-Mueller and ion chamber gamma detectors, as indicated on Table 12.3-14.

2. Class 1E Requirements

Separate redundant cabinets are provided in the control room for control, recording and remote indication for those monitors in Table 12.3-14 designated as Class 1E. These cabinets and Class 1E area monitors are powered from their respective Class 1E inverters. Class 1E monitors supply their data to the RDMS host computer through an IEEE 279 acceptable isolation device. No information or alarm setting is permitted between the RDMS host computer and the Class 1E equipment. All setpoint changes and check-source insertions are performed locally or from hard-wired modules in the control room.

3. In-Containment High Range Monitoring

Redundant Class 1E monitors are provided to monitor containment conditions under accident situations. The detector range is 10^{0}-10^{8} R/hr. The electronics cabinet is located outside Containment in the electrical tunnels. Indication is provided on the RDMS video displays and the RDMS racks in the main control room. These monitors will be designed, located, calibrated and qualified in accordance with Table II.F.1-3 of NUREG-0737.

The detectors are located on the steam generator biological shield wall (the "A" detector is near steam generator "D" and the "B" detector is near steam generator "B") at an approximate elevation of +31'. These locations were selected to provide the detectors as large a view of Containment as possible, consistent with affording ease of access for maintenance and calibration.

4. Area Monitor Channel Description

(a) Containment Manipulator Crane Area Monitor-Channels 6535 A and B

Redundant Class 1E detectors are located on the manipulator crane. In the event of a fuel handling accident, these monitors in conjunction with safeguards actuation signals isolate the containment online and offline purge isolation valves, trip containment air pre-entry, refueling supply and containment online purge fans. Indication and alarm are provided locally and in the main control room.

(b) Personnel Hatch (Post-LOCA) - Channels 6536-1,2

This area monitor is located external to the Containment and is aligned with the personnel hatch. This radiation monitor is intended to monitor ambient radiation conditions following a LOCA.

(c) Containment Post-LOCA - Channels 6576A, B

These detectors are intended to monitor conditions inside Containment for Post-LOCA and are Class 1E.

(d) Volume Control Tank - Channel 6540

The detector is located inside the volume control tank area.

(e) Cavity Beneath Reactor Vessel - Channel 6529

This radiation monitor is located in the cavity beneath the reactor vessel. The associated components for this monitor are located behind the shield walls which provide protection from the very high ambient radiation in this cavity during reactor operation.

(f) High Range Area Monitors - Channels 6508-1,2, 6563-1,2, 6517-1,2 and 6518.

There are seven high range ion chamber detectors. In compliance with Regulatory Guide 1.97, these detectors have an upper range of 10^4 R/hr. These monitors are located in areas which may require entry after an accident or which contain recirculating post-accident fluids. These monitors are:

1) PAB - High Range Area Monitor - Channels 6508-1,2 and 6563-1,2

2) RHR - High Range Area Monitor - Channel 6517-1,2

3) FSB - High Range Area Monitor - Channel 6518

(g) Other Area Radiation Monitors - Channels 6534, 6537, 6538, 6539, 6541, 6543, 6544, 6545, 6546, 6547, 6549, 6550, 6551, 6552, 6553, 6554, 6555, 6556, 6557, 6558, 6559, 6570, and 6571

These channels use Geiger-Mueller detectors and monitor the ambient radiation at various points throughout the facility as listed in Table 12.3-14.

5. Calibration and Maintenance

Refer to Subsection 11.5.2.6 for calibration and maintenance details.

12.3.4.2 Airborne Radioactivity Monitoring Instrumentation

a. Objectives and Design Basis

The ventilation airborne Radioactivity Monitoring System provides radiation measurements, indications, records, alarms and controls at selected locations to detect and control radiation levels within Containment, Service Building, Radwaste Building and the plant vent, and to verify compliance with applicable limits of 10 CFR 20 and 10 CFR 50, General Design Criteria 19, 63 and 64.

On May 21, 1991, a complete revision to 10 CFR 20 was issued. Several design bases reference 10 CFR 20 and specific terms or parts of 10 CFR 20. Design bases information provides a historical perspective of the information used to formulate a particular design. References to 10 CFR 20 when used in a historical or design bases context have not been changed to reflect the revised 10 CFR 20.

Monitored points within the station ventilation system are in areas where potential personnel exposure to radiation is most likely and in several ventilation exhaust ducts. A tabulation of the airborne Radiation Monitoring System is found in Table 12.3-15, Table 12.3-16 and Figure 12.3-20.

Those monitors which are Class 1E are listed in the above tables and further discussed in Subsection 12.3.4.1b.2.

The sensitivity of the airborne radioactivity monitors is such that they should be capable of detecting ten MPC-hours of particulate and gaseous radioactivity in those plant areas that have contained sources of airborne radioactivity and which may be occupied by personnel. Typical airborne concentrations for various plant areas are given in Table 12.2-31, Table 12.2-32, Table 12.2-33, Table 12.2-34, Table 12.2-35, Table 12.2-36, and Table 12.2-37.

As discussed in Subsection 12.5.3.1, the Health Physics Program includes requirements to perform sampling and analysis for airborne radioactivity, routinely and during specific evolutions such as opening of the primary system. Sampling equipment includes portable continuous air monitors and portable samplers. The monitoring and sampling capabilities, when combined, provide sufficient information to permit adequate protection of personnel from exposure to airborne radioactivity.

b. System Description

Subsection 12.3.4.1b describes the digital computer based RDMS. Subsection 11.5.2 describes the local microprocessor provisions. The airborne Radioactivity Monitoring System consists of two basic types of monitoring systems:

- Particulate and gaseous monitors (with iodine sampling) which are skid-mounted and utilize pumping systems.
- Gross activity monitors which consist of detectors mounted directly in duct air stream.

A typical channel is shown in Figure 12.3-20.

1. Particulate and Gaseous Monitors

Each airborne particulate and gaseous monitor has common equipment as follows:

(a) Isokinetic Sampler

Sampler and lines adhere to requirements of ANSI N13.1. Sample line sizes are one-half inch with flow rate designed for 2-3 scfm. All sample lines slope from high point (isokinetic sampler) to the low point (sample pump).

(b) Pumping System

(1) The flow control assembly includes a pump unit and selector valves that provide a representative sample (or a "fresh" sample) to the detector.

(2) The pump unit consists of:

a A pump to obtain the air sample

b A flowmeter to indicate the flow rate

c A flow control valve to provide flow adjustment

d A flow alarm assembly to provide low and high flow alarm signals.

(3) Selector valves are used to direct the desired sample to the detector for monitoring and to block flow when the channel is in the maintenance or "purging" condition.

(4) Purging is accomplished with a valve control arrangement whereby the normal sample flow is blocked and the detector purged with a "fresh" sample.

(5) A sample flow rate indicator is calibrated linearly from 0 to 4 scfm.

(6) Indicator lights are actuated by the following:

a Flow alarm assembly (low or high flow),

b The filter paper sensor (paper drive malfunction), or

c The pump power control switch (pump motor on).

(c) Detectors

The particulate channel air sample is drawn in a closed system monitored by a scintillation counter-filter paper detector assembly. The filter paper collects 99 percent of all particulate matter greater than 0.3 micron in size on its continuously moving surface, and is viewed by a photomultiplier-scintillation crystal combination.

The air sample is returned after it passes through the series-connected iodine filter and gas monitors.

The detector is a hermetically sealed scintillator crystal combination. The pulse signal is transmitted to the radiation monitoring system local cabinet.

Lead shielding reduces the background radiation level to prevent interference with the sensitivity of the detector. The filter paper mechanism, and electro-mechanical assembly which controls the filter paper movement, is provided as an integral part of the detector unit. The unit contains an approximate 25-day filter paper supply at normal speed.

The particulate filter is followed by a cartridge-type charcoal iodine filter. The iodine filter is a charcoal filter which will meet certain specifications for iodine removal. The particulate and iodine filters are suitable for laboratory analysis. The removal of all particles greater than 0.3 microns eliminates the need for maintaining isokinetic conditions downstream of the particulate and iodine filters.

The iodine filter is not monitored continuously. The iodine filter is periodically removed and analyzed as appropriate.

Next, the gaseous channel views the air sample. The sample is constantly mixed in the fixed, shielded volume, where it is viewed by a beta scintillator. The sample is then returned to its environment.

The detector assembly is in a completely enclosed housing containing a beta scintillator mounted in a constant gas volume container. Lead shielding reduces the background radiation level to prevent interference with the detector's sensitivity.

2. Particulate and Gaseous Monitor Channel Descriptions

(a) Containment Monitor - Channel 6526 and 6548

Monitor 6526 draws a containment air sample through redundant pumps from the containment atmosphere. The sample is then returned to the containment atmosphere.

Monitor 6548 is located inside Containment at zero foot elevation and acts as a backup to monitor 6526. Monitor 6548 draws air sample from the Containment via the sample pump and then discharges back to the Containment.

These monitors are classified seismic Category I.

Indication and alarm is available locally and in the main control room.

See Subsections 5.2.5.3b.2, 5.2.5.5b and 5.2.5.5c for a further discussion of monitoring requirements.

(b) Waste Process Building Monitor - Channel 6531

The major potential release of airborne radioactivity in the Waste Processing Building is that associated with the Gaseous Waste Processing System. The gas dryers, carbon delay beds and the two gas compressors are situated in their individual compartments, and these compartments are ventilated in such a way that they are at a negative pressure with respect to surrounding areas. The ducted ventilation exhaust is continuously sampled and monitored. The sample is returned to the ducted ventilation exhaust line which is directed to the plant vent. Both the sampling point and the return are downstream of the filters in the ventilation exhaust line outside the building. Information from this channel is displayed and alarmed on the radiation monitoring system panel in the main control room and locally.

(c) Primary Auxiliary Building Monitor - Channel 6532

Three minimum ventilation areas have been defined for the Primary Auxiliary Building:

(1) Heat exchanger, thermal regeneration demineralizer, and mixed bed demineralizer area

(2) Volume control tank area

(3) Charging pump area.

These areas, which are potential sources of airborne activity, are maintained at a negative pressure with respect to surrounding areas. The PAB ventilation system collects potentially contaminated air through a duct system and discharges it to the plant vent via filter train F-16. The sample withdrawal point for this monitor (RM-6532) is downstream of filter train F-16. The location of this sample withdrawal point provides an early warning to the operating personnel in the event that radioactive material becomes airborne in the PAB.

Indication and alarm is available locally and in the main control room. An alarm indication on these monitors would trigger a radiological evaluation within the areas served by these monitored ventilation lines. The evaluation would be performed by station HP personnel using portable survey and/or air sampling equipment, as necessary, to locate the source of the elevated ventilation line indication.

(d) Administration and Service Building Monitor-Channel 6522

The Service Building, which houses the RCA shop, counting room, hot chemistry lab, decontamination facility, RCA showers and health physics control point, is a potential minor source of airborne contamination. This building has its own ventilation system which exhausts to the atmosphere through a single run of ventilation ducting. A continuous air sample is removed from the ducting downstream of the filters and monitored for particulate and gaseous activity. The sample is returned to the ventilation exhaust lines, again, downstream of the filters. The channel is displayed on the radiation monitoring system panel in the main control room.

(e) Plant Vent Monitors - Channels 6528-1, 6528-2, 6528-3, and 6495

These detectors monitor the air exhausted by the Primary Auxiliary Building, Waste Process Building, Fuel Storage Building, containment structure and containment enclosure via the plant vent. An isokinetic probe, supplemented with an integral pumping system is used to withdraw an air sample from the plant vent. The air quantities exhausted via the plant vent are indicated on Figure 9.4-5, Figure 9.4-6, Figure 9.4-7, Figure 9.4-8, Figure 9.4-9, and Figure 9.4-10.

Multisensor flow transmitters and microprocessor provide a signal to the radiation monitor (RM-6528) to permit this monitor to calculate the microcuries per cubic centimeter flowing in the duct, microcuries per second and the integrated microcuries released through the plant vent.

The air collected by the isokinetic probe passes through the wide-range gas monitor, WRGM, (RE-6528-1, RE-6528-2, and RE-6528-3). Sampling provisions are located downstream of the isokinetic nozzle. The air flow enters the sample conditioning skid of monitor RM-6528 at a flow rate of ≈0.06 cfm during postulated accident conditions. This skid is intended to provide representative particulate and radioiodine samples for laboratory analysis (for normal operation as well as accident conditions) and to prevent contamination of the gas monitors. A multiple filter arrangement is provided to allow sampling capabilities for the duration of the measurement period. Each filter is equipped with a 4π solid lead shielding and quick disconnect fittings to minimize personnel exposures. In addition, all functional control is performed remotely.

The wide-range gas monitor has the capability of detecting a wide range of radiogas concentrations over 12 decades. The monitor meets the requirements of NUREG-0737, item II.F.1 by providing an upper range of 10^5 μCi/cc for noble gases and the capability to collect post-accident plant vent grab samples. A seven-day composite grab sampling system for normal operation is also provided for particulates and iodine determination. Indication and alarm from the WRGM are available locally and in the main control room.

A portable pump with detector (RE-6495) provides backup to the WRGM mid and high range. Indication and recording are provided in the Technical Support Center (TSC).

3. Gross Activity Monitors

These units do not utilize a pumping system. The detectors are located directly in the duct air stream. These monitors employ Geiger-Mueller type detectors. The local microprocessor cabinet provisions are described in Subsection 12.3.4.1b.

4. Gross Activity Monitor Channel Descriptions

(a) Administration and Service Building Fume Hoods Monitor -Channels 6523, 6524 and 6525

These detectors are Geiger-Mueller counters and are located in each of the chemistry fume hoods. They measure the gross activity of air exhausted from the shop fume hoods to atmosphere. Local alarm and indication are available near the fume hoods. No control function is provided.

These monitors provide data to the RDMS host computer for alarm, display and documentation. Remote indication and alarm are available in the main control room.

(b) Fuel Storage Building Exhaust Monitor - Channel 6562

This detector is a Geiger-Mueller counter and is located in the fuel storage building ventilation exhaust duct downstream of the fans. This detector measures the gross activity vented from the Fuel Storage Building to the plant vent. Indication and alarm is available locally in the Fuel Storage Building near the spent fuel storage pool, and remotely in the main control room.

(c) Containment Enclosure Emergency Exhaust Monitor – Channel 6566

This detector is a Geiger-Mueller counter and is located downstream of the containment enclosure emergency exhaust filter fans and measures the gross activity exhausted to the plant vent stacks. Indication and alarm are available locally near the filter fans and remotely in the main control room.

(d) Primary Auxiliary Building, Miscellaneous Ventilation Exhaust - Channel 6567

This detector is a Geiger-Mueller counter and is located at the inlet to the primary auxiliary building cleanup filter. The following areas are monitored by this detector: valve aisle, volume control tank area, sample heat exchanger room, sample room fume hood, degasifier area, PAB lower level elevation (-)6', and PAB filter and heat exchanger area.

Indication and alarm are available locally near the cleanup filter, and remotely in the main control room.

(e) Containment Enclosure Monitor - Channel 6568

This detector is a Geiger-Mueller counter and is located in the exhaust duct from the containment enclosure at the inlet to the cleanup filter. The detector monitors the gross activity exhausted from the containment enclosure. Indication and alarm are available locally near the cleanup filter, and remotely in the control room.

(f) Control Room Air Intake Monitors - Channels 6506A and B, 6507A and B

Four detectors are located in the east air intake piping and four detectors are located in the west air intake piping. These detectors are located in the Control and Diesel Building. These GM detectors, which are Class 1E, monitor the control room air intake and automatically shut down, on a high radiation signal, the control room ventilation fans and isolation dampers. Each monitor utilizes a two-out-of-two detector logic such that two detectors must be in alarm before the monitor initiates an isolation signal. These detectors are directly mounted in the air intake stream and do not require shielding.

Indication and alarm are provided locally. Indication, recording and alarm are provided in the main control room.

(g) Containment Online Purge Monitor - Channels 6527A, 6527B

For a description of this monitor see Subsection 11.5.2.1.

5. Portable Continuous Air Monitors (CAM)

Four portable continuous air monitors are available. The CAMs are equipped to monitor particulate and noble gas.

The normal locations for the CAMs are as follows:

- Waste Process Building
- Primary Auxiliary Building
- Fuel Storage Building
- Containment (on the operating floor during refueling outages)
- Control Building (during normal operations)

CAMs may be moved to other station locations as radiological conditions dictate.

6. Calibration and Maintenance

Refer to Subsection 11.5.2.6 for calibration and maintenance details.

12.3.4.3 Post-Accident In-Plant Iodine Assessment

The capability exists for the determination of airborne radioiodine levels in-plant under accident conditions. This capability includes the use of air samplers with radioiodine-specific sample cartridges and the use of gamma spectroscopy instrumentation for sample analysis. Information on portable air sampling and counting room equipment is discussed in Subsection 12.5.2.

This sampling and analysis is described in station procedures in which station personnel are trained. Training includes the proper handling and preparation of high-level radioactive samples and the operation and calibration of gamma ray spectroscopy equipment for post-accident sampling in addition to normal sampling techniques.

TABLE 12.3-14 AREA RADIATION MONITORS

INSTRUMENT TAG NO. RE-	DESCRIPTION	DETECTOR TYPE	DETECTOR BACK-GRD. mr/hr	RANGE LOW-HIGH mr/hr	(Note 5) ALARM SET POINT mr/hr	DETECTOR QTY.	IEEE CLASS	UPDATED FSAR FIGURE REFERENCE
	Containment Structure							
6534	In-Core Instrument Seal Table	GM (Note 4)	15	10^{-1}-10^{4}		1	Non 1E	12.3-2
6535A, B (Note 1)	Manipulator Crane	GM (Note 4)	15	10^{-1}-10^{4}		2	1E	12.3-3
6536-1, 2	Personnel Hatch (Post-LOCA)	Ion Chamber	2.5	10^{+1}-10^{9}		2	Non 1E	12.3-3
6529	Cavity Beneath Reactor Vessel	Ion Chamber	$3x10^{6}$ (Note 3)	10^{+1}-10^{7}		1	Non 1E	12.3-1
6576A, B	Containment (Post-LOCA)	Ion Chamber	25	10^{0}-10^{8} r/hr		2	1E	12.3-3
	Primary Auxiliary Building							
6537	Sampling Room	GM	2.5	10^{-1}-10^{4}		1	Non 1E	12.3-6
6538, 6539	RHR Pump Area	GM	>100	10^{-1}-10^{4}		2	Non 1E	12.3-4
6540	Volume Control Tank Area	Ion Chamber	$8x10^{4}$	10^{1}-10^{7}		1	Non 1E	12.3-7
6541	Lower Level	GM	2.5	10^{-1}-10^{4}		1	Non 1E	12.3-5
6543	Entrance	GM	>100	10^{-1}-10^{4}		1	Non 1E	12.3-5
6544	Entrance	GM	2.5	10^{-1}-10^{4}		1	Non 1E	12.3-6

INSTRUMENT TAG NO. RE-	DESCRIPTION	DETECTOR TYPE	DETECTOR BACK-GRD. mr/hr	RANGE LOW-HIGH mr/hr	(Note 5) ALARM SET POINT mr/hr	DETECTOR QTY.	IEEE CLASS	UPDATED FSAR FIGURE REFERENCE
6545, 6546 6547	Charging Pump Room	GM	110	10^{-1}-10^{4}		3	Non 1E	12.3-5
6508-1,2	PAB-HRAM	Ion Chamber	>100	10^{-2}-10^{4} r/hr		2	Non 1E	12.3-5
6563-1,2	PAB-HRAM	Ion Chamber	>100	10^{-2}-10^{4} r/hr		2	Non 1E	12.3-5
6517-1,2	RHR - Pump Vault HRAM	Ion Chamber	>100	10^{-2}-10^{4} r/hr		2	Non 1E	
	Fuel Storage Building							
6549	Spent Fuel Pool Area	GM	2.5	10-1-10^{4}		1	Non 1E	12.3-15
6518	Spent Fuel - HRAM	Ion Chamber	2.5	10^{-2}-10^{4} r/hr		1	Non 1E	12.3-15
	Control Room							
6550	Main Control Board Area	GM	0.5	10^{-1}-10^{4}		1	Non 1E	1.2-32
	Waste Processing Building							
6551	Waste Gas Processing Area	GM	2.5	10^{-1}-10^{4}		1	Non 1E	12.3-11
6552	Truck Loading Area	GM	1.5	10^{-1}-10^{4}		1	Non 1E	12.3-10
6553	Radwaste Control Room	GM	0.5	10^{-1}-10^{4}		1	Non 1E	12.3-10
6554	Waste Management Control Panel Area	GM	5	10^{-1}-10^{4}		1	Non 1E	12.3-10

INSTRUMENT TAG NO. RE-	DESCRIPTION	DETECTOR TYPE	DETECTOR BACK-GRD. mr/hr	RANGE LOW-HIGH mr/hr	(Note 5) ALARM SET POINT mr/hr	DETECTOR QTY.	IEEE CLASS	UPDATED FSAR FIGURE REFERENCE
6570	Extruder/Evaporator	GM	(Note 6)	10^0-10^5		1	Non 1E	12.3-10
	Manifold Area							
6571	Compacted Rad Waste Storage Area	GM	2.5	10^{-1}-10^4		1	Non 1E	12.3-10
	Administration & Service Building							
6555	Hot Chemistry Laboratory	GM	0.5	10^{-1}-10^4		1	Non 1E	12.3-16
6556	Decontamination Room	GM	0.5	10^{-1}-10^4		1	Non 1E	12.3-16
6557	RCA Shop (Note 2)	GM	0.5	10^{-1}-10^4		1	Non 1E	12.3-16
6558	RCA Personnel Decontamination Area	GM	0.5	10^{-1}-10^4		1	Non 1E	12.3-16
6559	RCA Women's Locker Room	GM	0.5	10^{-1}-10^4		1	Non 1E	12.3-16

(Note 1) 6535-A and 6535-B will automatically terminate containment purge in the event of high radiation during fuel handling operations.

(Note 2) RCA – Radiologically Controlled Area.

(Note 3) Background 16 hours after shutdown 200 mr/hr.

(Note 4) GM – Geiger-Mueller.

(Note 5) Radiation monitoring setpoints are varied during operation to follow station conditions. Setpoints are maintained within the bounds established in the Technical Specifications. The methodology for establishing the setpoints is found in the station operating procedures.

(Note 6) >100