NT SECTION 2

SITE AND ENVIRONS

Table of Contents

2. SITE AND ENVIRONRONS . . . 1
2.1 INTRODUCTION . . . 1
2.2 GENERAL DESCRIPTION OF SITE AND ENVIRONS . . . 1
2.3 TOPOGRAPHY . . . 1
2.4 SEISMOLOGY . . . 1
2.4.1 Seismic Geology and History . . . 1
2.4.2 Seismicity . . . 1
2.4.3 Seismic Design Criteria . . . 3
2.4.4 Conclusions . . . 3
2.5 METEOROLOGY . . . 1
2.5.1 General Climate . . . 1
2.5.2 Local Meteorology . . . 2
2.5.2.1 Wind Direction and Speeds . . . 2
2.5.2.2 Precipitation . . . 3
2.5.2.3 Temperatures . . . 3
2.5.2.4 Relative Humidity . . . 4
2.5.2.5 Thunderstorms . . . 4
2.5.2.6 Atmospheric Stability . . . 4
2.5.2.7 Topographical Description and Its Influence on Site Meteorology . . . 5
2.5.3 Meteorological Monitoring Program . . . 6
2.5.3.1 Preoperational and Initial Monitoring Program . . . 6
2.5.3.2 Permanent Meteorological Monitoring Program . . . 8
2.5.4 Short Term (Accident) Diffusion Estimates . . . 8
2.5.4.1 The Diffusion Model (8 Hours or Less) . . . 9
2.5.4.2 The Diffusion Model (Longer Than 8 Hours) . . . 10
2.5.4.3 Hourly Dispersion Factors Based On On-site Data . . . 11
2.5.4.4 Dispersion Factors for Periods Up to 30 Days . . . 11
2.5.5 Long Term (Routing) Diffusion Model . . . 12
2.6 GEOLOGY . . . 1
2.6.1 Physiography . . . 1
2.6.2 Regional Geology . . . 1
2.6.3 Regional Structure . . . 1
2.6.4 Areal Structure . . . 2
2.6.5 Geologic History . . . 2
2.6.6 Glacial History of the Missouri Valley . . . 3
2.6.7 Site Geology . . . 3
2.6.8 Conclusions . . . 4

2.7 HYDROLOGY 1
2.7.1 Surface Drainage 1
2.7.1.1 General 1
2.7.1.2 River Stage and Flow 2
2.7.1.3 River Temperature 5
2.7.1.4 River Water Analyses 5
2.7.2 Ground Water 7
2.7.2.1 General 7
2.7.2.2 Site Water Table and Transmissibility 8
2.7.2.3 Well Water Analyses 9
2.8 DEMOGRAPHY 1
2.9 LAND USE 1
2.9.1 General 1
2.9.2 Agronomy 1
2.9.3 Industry 1
2.9.4 Wildlife 6
2.9.4.1 Fish 6
2.9.4.2 Animals and Birds 6
2.10 ENVIRONMENTAL RADIATION MONITORING 1
2.10.1 General 1
2.10.2 Preoperational Survey Program 1
2.10.3 Preoperation Survey Results 2
2.10.3.1 Trial Monitoring Period 2
2.10.3.2 Preoperational Monitoring Period 6
2.10.3.3 Operational Survey Program 6
2.11 SECTION 2 REFERENCES 1

List of Tables

Table 2.5-1 - "Climatological Normals, Comparison of North Omaha National Weather Service with Fort Calhoun Station" 14
Table 2.5-2 - "Maximum Recorded and Mean Wind Speeds (MPH)" 15
Table 2.5-3 - "Normal and Extreme Precipitation Amounts (Inches)" 16
Table 2.5-4 - "Normal and Extreme (Maximum) Snow and Ice Pellet Amounts (Inches)" 17
Table 2.5-5 - "Normal and Extreme Temperatures (°F)" 18
Table 2.5-6 - "Monthly and Annual Temperature Normals (°F)" 19
Table 2.5-7 -"Comparative Relative Humidity Values for Eppley Airfield (1964-1990), North Omaha (N.O.) (1984-1990), and Fort Calhoun (1969-1975)" 20
Table 2.5-8 - "Mean Number of Days with Heavy Fog (Visibility 1/4 Mile or Less)" 21
Table 2.5-9 - "Mean Number of Days with Thunderstorms" 21
Table 2.5.10 -"Joint Frequency Vs. Delta-T (Percent) Pasquill Stability Class "A" 22
Table 2.5.11 - "Joint Frequency Vs. Delta-T (Percent) Pasquill Stability Class "B" 23
Table 2.5.12 - "Joint Frequency Vs. Delta-T (Percent) Pasquill Stability Class "C"" 24
Table 2.5.13 - "Joint Frequency Vs. Delta-T (Percent) Pasquill Stability Class "D"" 25
Table 2.5.14 - "Joint Frequency Vs. Delta-T (Percent) Pasquill Stability Class "E"" 26
Table 2.5.15 - "Joint Frequency Vs. Delta-T (Percent) Pasquill Stability Class "F"" 27
Table 2.5.16 - "Joint Frequency Vs. Delta-T (Percent) Pasquill Stability Class "G"" 28
Table 2.5.17 - "Joint Frequency Vs. Delta-T (Percent) Pasquill Stability Class "A" Through "G"" 30
Table 2.5-18 - "Fort Calhoun On-Site Meteorological Measurement Program (Initial Instrumentation and Recording System)" 30
Table 2.5-18 (Continued) 31
Table 2.5-19 - "Fort Calhoun On-site Meteorological Measurement Program (Initial Instrumentation and Recording System)" 32
Table 2.5-20 - "Fort Calhoun On-Site Meteorological Measurements Program (Instrumentation and Recording System)" 33
Table 2.5-21 - "Fort Calhoun 110 Meter Tower Instruments" 36
Table 2.5-22 - "Radials and Minimum Distances from the Containment Structure to the Exclusion Area Boundary" 37
Table 2.5-23 - "Meteorological Analyses of Onsite Data Step-by-Step Description" 38
Table 2.5-24 - "Summary of Percentile Values for Worst Sector Dispersion Factors (x/Q) 1980 and 1981" 39
Table 2.5-25 - "Average Dispersion Factors (x/Q) at the IPZ Outer Boundary for 0-8 Hours 1980 and 1981" 40
Table 2.5-26 - "Average Dispersion Factors (x/Q) at the LPZ Outer Boundary for 8-24 Hours 1980 and 1981" 41
Table 2.5-27 - "Average Dispersion Factors (x/Q) at the LPZ Outer Boundary for 1-4 Days 1980 and 1981" 42
Table 2.5-28 - "Average Dispersion Factors (x/Q) at the LPZ Outer Boundary for 4-30 Days 1980 and 1981" 43
Table 2.5-29 - " Long Term Dispersion Factors For 1982 (Sec/M^3)*" 44
Table 2.5-30 - "Long Term Dispersion Factors For 1983 (Sec/M^3)*" 45

Table 2.5-31 - "Long Term Dispersion Factors For 1984 (Sec/M^3)*" 46
Table 2.5-32 - "Pasquill A Wind Speed (M/S) At 10 M Level" 47
Table 2.5-33 - "Pasquill B Wind Speed (M/S) At 10 M Level" 48
Table 2.5-34 - "Pasquill C Wind Speed (M/S) At 10 M Level" 49
Table 2.5-35 - "Pasquill D Wind Speed (M/S) At 10 M Level" 50
Table 2.5-36 - "Pasquill E Wind Speed (M/S) At 10 M Level" 51
Table 2.5-37 - "Pasquill F Wind Speed (M/S) At 10 M Level" 52
Table 2.5-38 - "Pasquill G Wind Speed (M/S) At 10 M Level" 53
Table 2.5-39 - "All Stability, All DT/DZ Wind Speed (M/S) At 10 M Level" 54
Table 2.5-40 - "Results of Mesodif-II* Trajectory Analysis Using Fort Calhoun Meteorological Tower Data for 1982 Through 1984" 55
Table 2.5-41 - "Pasquill D Wind Speed (M/S) At 10 M Level" 56
Table 2.5-42 - "Pasquill E Wind Speed (M/S) At 10 M Level" 57
Table 2.5-43 - "Pasquill F Wind Speed (M/S) At 10 M Level" 58
Table 2.5-44 - "Pasquill G Wind Speed (M/S) At 10 M Level" 59
Table 2.5-45 - "All Stability, All DT/DZ Wind Speed (M/S) At 10 M Level" 60
Table 2.5-46 - "Results of Mesodif-II* Trajectory Analysis Using Fort Calhoun Meteorological Tower Data for 1982 Through 1984" 61
Table 2.7-1 - "Missouri River Dams" 1
Table 2.7-2 - "Average Missouri River Water Temperatures (°f) at the Metropolitan Utilities District of Omaha Intake (27 Year Average)" 5
Table 2.7-3 - "Analysis of Missouri River Water at the Metropolitan Utilities District of Omaha Intake 1973 Through 1981" 6
Table 2.7-4 - "Test Well Water Analysis 10
Table 2.8-1 - "Population Distribution as of 1990 3
Table 2.9-1 - "Agricultural Land Use in Nebraska of Counties Within 10 miles of the Fort Calhoun Station" 2
Table 2.9-2 - "Agricultural Land Use In Iowa of Counties Within 10 Miles of the Fort Calhoun Station" 4
Table 2.9-3 - "Industry Within 10 Miles of Fort Calhoun Station" 5
Table 2.10-1 - "Gross and Specific Radionuclide Analyses" 1
Table 2.10-2 - "Average Radioactivity of Well and Surface Waters November 1968 - June 1969" 2
Table 2.10-3 - "Food Habits and Radioactivity of Missouri River Fish October 1968 - June 1969" 3
Table 2.10-4 - "Radioactivity in Milk January - March, 1969" 4
Table 2.10-5 - "Radioactivity in Vegetation October, 1968" 5
Table 2.10-6 - "Sample Types and Frequency " 7
Table 2.10-7 - "Radiological Environmental Sampling Locations and Media" 9

List of Figures

The following figures are controlled drawings and can be viewed and printed from the applicable listed aperture cards.

Figure No.	Title	Aperture Card
2.2-1	Aerial Photograph of Site and Environs	36042
2.2-2	Geographical Features Within 60 Miles of Site	36043
2.2-3	Geographical Features Within 30 Miles of Site	36044
2.2-4	Metes and Bounds	36045
2.3-1	Site Topography	36046
2.7-1	Upper Missouri River Basin	36047
2.7-2	Stage Duration Curve	36048
2.7-3	Flow Duration Curve	36049
2.7-4	Missouri River Rating Curve	36050
2.8-1	Actual Population Distribution as of 1980, 0-5 Miles	36051
2.8-2	Actual Population Distribution as of 1980, 5-50 Miles	36052
2.8-3	Projected Population Distribution as of 2010, 0-5 Miles	36053
2.8-4	Projected Population Distribution as of 2010, 5-50 Miles	36054
2.10-1	Environmental Radiological Sampling Points, On Site	36055
2.10-2	Environmental Radiological Sampling Points, 0-5 Miles	36056
2.10-3	Environmental Radiological Sampling Points, 5-20 Miles	36057

2.5 METEOROLOGY

2.5.1 General Climate

The general climate refers to the study of atmospheric sciences, and is a general description of the LONG-TERM (i.e., 50 years +) meteorological observations and conditions of an area.

The general climate is classified as continental, i.e., hotter in summer and colder in winter than in lands near the ocean. Nebraska is located midway between two distinctive climatic zones, the humid east and the dry west[(4)]. Thus, cyclic weather conditions representative of either zone, or combinations of both occur. Changes in weather result from the invasion of large masses of air with dissimilar properties. These air masses tend to get their characteristics from either the warm and humid south-southeast, the warm and dry southwest, the cool and dry north-northwest, or the cold continental polar air of the north[(6)]. The region is also affected by many storms or cyclones (areas of low pressure) which travel across the country, generally from west to east. Thus, periodic and rapid changes in the weather are normal, especially in the winter[(3)].

Annual average precipitation for the region is about 28.50 inches, but annual amounts vary widely from year to year. For example, at Omaha in 1976, the total was 18.37 inches, while in 1965, the total was 44.85 inches. About 75 percent of the precipitation occurs during showers and thunderstorms during April through September[(4)]. Snowfall amounts to about 30.3 inches of snow for an annual average, but total annual amounts vary widely from year to year[(4)].

Reference; Section 2.5.2.1 Wind Direction and Speeds, and Table 2.5-1, Climatological Normals, Comparison of North Omaha Weather Service.

The mean annual temperature for the region is 51.1°F. The January monthly mean is 20.2°F, while that for July is 77.7°F. Relative humidity varies from an average of about 78 percent for the period midnight to noon and about 59 percent from the period noon to midnight. The mean percentage of possible sunshine over the area is about 50 percent in winter and about 75 percent in summer[(4)(5)].

2.5.2 Local Meteorology

The local meteorology, while utilizing the same type of measurements as for the General Climate, refers to atmospheric conditions or the site specific meteorological data at the Fort Calhoun Station.

Data from proximal long-term North Omaha Weather Service have been used to supplement the existing Fort Calhoun Station data in formulating the description of the local meteorology. Refer to Section 2.5.2.1, Wind Direction and Speeds, and Table 2.5-1, Climatologocal Normals, Comparison of North Omaha National Weather Service.

2.5.2.1 Wind Direction and Speeds

Surface wind (20 feet above ground level) data for the North Omaha National Weather Service in Omaha, Nebraska, for 1985 through 1989, were used as a climatological base to compare the normals that are expected at Fort Calhoun Station. On an annual basis, wind from the north is the single most frequent (13.0 percent) wind direction. Cumulatively, the southeast through south-southwest sectors 37.0 percent of the time, and from the northwest through north-northeast sectors 32.2 percent of the time. The least frequent directions are from the southwest through west-northwest and northeast through east-southeast sectors. These frequencies are 15.9 percent and 15.0 percent, respectively. The average wind speed for the total of all observations was 8.7 miles per hour.

A comparison of the climatological normals for the North Omaha National Weather Service and five years of data from the Fort Calhoun Station is shown in Table 2.5-1. The elevation of the North Omaha National Weather Service wind sensor was 20 feet above ground during the five-year period, while the elevation of the Fort Calhoun wind sensor was 10 meters (32.8 feet) above ground. Further discussion on the on-site data for wind direction and speed is presented in Section 2.5.2.6.

The mean and maximum wind speeds recorded at Eppley Airfield and the North Omaha National Weather Service for each month of the year are shown in Table 2.5-2.

2.5.2.2 Precipitation

Monthly and annual normal and extreme precipitation amounts for Eppley Airfield and the North Omaha NWS are presented in Table 2.5-3. Average monthly precipitation follows a seasonal trend, reaching a maximum in May and a minimum in January. The mean number of days with measurable precipitation varies between a 12-day maximum in May to a 5-day minimum in November. The extreme precipitation amounts during 1954 through 1990 have been:

Maximum annual total	44.85 inches (1965)
Minimum annual total	18.37 inches (1976)
Maximum monthly total	14.10 inches (9/65)
Minimum monthly total	Trace (1/86)
Maximum 24-hour total	6.47 inches (1965)

The normals and extremes for amounts of snow and ice pellets (including sleet) for Eppley Airfield and the North Omaha NWS are shown in Table 2.5-4.

2.5.2.3 Temperatures

The monthly temperatures at Eppley Airfield and the North Omaha NWS for the period 1961 through 1990 are shown in Table 2.5-5. This table also shows the record high and record low temperatures recorded at both locations through 1990. Annual extremes have been received at other locations in the Omaha vicinity as follows: a record high of 114°F in July, 1936, and a record low of -32°F in January, 1885[4].

Monthly and annual temperature normals for Eppley Airfield, North Omaha NWS and Blair are presented in Table 2.5-6[4]. These show close agreement. The North Omaha NWS is located approximately 7 miles northwest of Eppley Airfield and 11 miles south-southeast of Fort Calhoun Station. The City of Blair is located approximately 3 miles north-northwest of the Fort Calhoun Station.

2.5.2.4 Relative Humidity

The average relative humidity values for Eppley Airfield, the North Omaha NWS, and the Fort Calhoun Station for four times of the day are shown in Table 2.5-7. The mean number of days with heavy fog (visibility ¼ mile or less) at Eppley Airfield and the North Omaha NWS are shown in Table 2.5-8.

2.5.2.5 Thunderstorms

The mean number of days with thunderstorms at Eppley Airfield and the North Omaha NWS are shown in Table 2.5-9. A maximum frequency of approximately 9.5 thunderstorms occur during the month of June. This decreases to a minimum of approximately 0.1 thunderstorms during the month of January.

2.5.2.6 Atmospheric Stability

Wind direction and speed data are presented in relative frequency distribution (in percent) by stability classes. The data covers the periods of January 1, 1982, through December 31, 1991, and are shown in Tables 2.5-10 through 2.5-17. These tables were prepared from data collected at the Fort Calhoun Station. The wind data were collected by a sensor mounted at an elevation 10 meters above ground. Stability classes were determined from delta-temperature measurements from vertical distances of 110 meters and 10 meters above ground. The tables are self-explanatory except that the calm values have been distributed in the 0.0 mps to 0.4 mps category based on the number of observations in the speed category.

The stability classes used are based on Pasquill's class structure in accordance with Regulatory Guide 1.23 and are as follows:

Pasquill Class	ΔT °C/100m	Description
A	≤ -1.9	Extremely unstable
B	-1.9 <ΔT≤ -1.7	Moderately unstable
C	-1.7 <ΔT≤ -1.5	Slightly unstable
D	-1.5 <ΔT≤ -0.5	Neutral
E	-0.5 <ΔT≤ +1.5	Slightly stable
F	+1.5 <ΔT≤ +4.0	Moderately stable
G	> +4.0	Extremely stable

2.5.2.7 Topographical Description and Its Influence on Site Meteorology

The terrain in the vicinity of Fort Calhoun Station is generally flat from the north, northeast, east and southeast sectors, with an elevation of approximately 1000 feet above mean sea level (msl), for a radius of at least 10 miles. This terrain is generally the flood plain of the Missouri River. Terrain in the remaining sectors, south-southeast through west-northwest show much greater relief from the low lying bluffs, cut by numerous ravines, with elevations of about 1300 feet above msl. These bluffs extend along the western bank of the Missouri River, which runs generally from the northwest to the southeast, and come within about one mile of the Fort Calhoun Station in the south through west-southwest sectors.

Two unusual effects in the site meteorology are: 1) under very light westerly wind flow there is a possibility of weak drainage flow off the bluffs to the west toward the river, and 2) there will possibly be a slowing down of weak winds as air flows across the river from east to west and meets the rising terrain to the west. However, neither of these effects are regarded as significant in their influence on site meteorology and should not, under most synoptic weather types, severely skew the strong measures of covariation (+0.75 to +1.00) which exist between the site and other meteorological stations.

2.5.3 Meteorological Monitoring Program

2.5.3.1 Preoperational and Initial Monitoring Program

The preoperational meteorological program was designed to measure the parameters needed to evaluate the dispersion characteristics of the plant site for the evaluation of the consequences of routine operations and of hypothetical accidental releases of radionuclides to the atmosphere. The on-site data acquisition program was begun in June, 1967, by Omaha Public Power District at the Fort Calhoun Station site. Initially, a climatological station was instrumented with standard-type Weather Bureau quality instruments for recording temperature (thermograph), relative humidity (hygrothermograph), precipitation (rain gauge), barometric pressure (barograph), and wind. The wind system, a Meteorology Research, Inc. (MRI) mechanical weather station (MWS), was installed atop a 40-foot fold tower at a location adjacent to the Missouri River and slightly south-southeast of the reactor centerline. This tower was operable, producing valid data, until June, 1977. In addition, a hilltop station for recording temperature was instrumented during September, 1968. The hilltop thermograph was the same model as that installed at the climatological station and was located on a 310-foot hill about 1 mile southwest of the reactor. The 310-foot elevation difference between the 2 temperature sensors provided interim vertical temperature gradient measurements. The technique of selection of the temperature differences (delta T) which are representative of the various Pasquill-Turner Stability Classes was based on the various temperature gradient parameters developed at NRTS. Table 2.5-18 presents a list of the above instruments including the description, specifications, and installation levels of each sensor.

During 1970, a Rohn 160-foot guyed weather tower was installed on the Fort Calhoun Station plant site which operated, producing valid data, until June, 1977. The weather tower was located more than 1/2-mile northwest of the reactor building. It was located upwind of the prevailing winds over the reactor in order to more accurately measure the winds prior to their passage over the reactor complex towards the greatest concentrations of population in the area. The weather tower was originally instrumented with three aspirated temperature sensors: one at an elevation of approximately 32.8 feet above the plant ventilation discharge duct at 117 feet above ground level (AGL), one at approximately 32.8 feet below the duct outlet, and the third at approximately 6.6 feet AGL. An MRI "vectorvane" wind system was mounted near 115 feet AGL, the above-duct outlet elevation. See Table 2.5-19 for a list of the above sensors which includes the description, specifications, and installation levels of each. The meteorological sensors listed in Table 2.5-18 were continued for backup.

Due to the limited amount of data which was being recovered, Omaha Public Power District began an update and an improvement program of the instrumentation system on the Rohn 160-foot weather tower in late 1973. In January, 1974, the updated system became operational and in full compliance with the Regulatory Guide 1.23. In Table 2.5-20 is shown the updated weather tower system with instruments and sensors listed with specifications and mounted elevations of each. The meteorological parameters sensed on the 160-foot Rohn tower were transmitted to a remote recording system which had hard copy strip charts and was located in the control room. In addition, all parameters from the weather tower were stored digitally through a data logger on discs for computer processing to hourly average values which, in turn, were hard copied to hourly data logs.

2.5.3.2 Permanent Meteorological Monitoring Program

The Fort Calhoun Station has a permanent 110M meteorological tower with appropriate meteorological measurements system. The available instrumentation and the level of redundancy are indicated in Table 2.5-21. Real-time and historical data is available from the plant computer system (ERFCS). This data can be accessed and printed in the control room, Technical Support Center (TSC), and the Emergency Operations Facility (EOF). Additional transmittal of meteorological data on or off-site may be by radio, telephone, computer, or by calling the control room.

Representative backup wind speed and direction data can be obtained from Eppley Airfield. This data may be obtained by telephone communication (voice) or by downloading to a PC. Extensive statistical and climatological studies have been completed to determine meteorological correlations between Epply Airfield and the Fort Calhoun Station. The results of these studies indicate that Eppley Airfield is a reliable and conservative source of backup meteorological data for the Fort Calhoun Station.

2.5.4 Short Term (Accident) Diffusion Estimates

Accidents could result in short-term releases of radioactivity from several possible releasing points (i.e., auxiliary building venting, containment venting, containment leakage, etc.) of plant structures. Atmospheric dispersion factors (x/Q) based on site meteorological data (1980 and 1981) are calculated for various downwind distances for time periods of 1, 8, 16, 72 and 624 hours, corresponding to sequential release, presumed to occur during periods of 0 to 2 hours, 0 to 8 hours, 8 to 24 hours, 1 to 4 days, and 4 to 30 days as prescribed by Regulatory Guide 1.4.[7]

2.5.4.1 The Diffusion Model (8 Hours or Less)

The dispersion factors for ground level releases are calculated from hourly on-site data using the conventional Gaussian Diffusion Model. The downwind centerline dispersion factors are calculated as follows:

$$x/Q = \left[\pi\sigma_y\sigma_z\bar{\mu}\left(1+\frac{cA}{\pi\sigma_y\sigma_z}\right)\right]^{-1}$$

Where:

$\bar{\mu}$	is the hourly average wind speed (m/sec)
c	is the building wake shape factor (0.5)
A	is the minimum cross-sectional area of the reactor containment (1340 m^2)
σ_y	is the horizontal dispersion coefficient of the plume per Pasquill class (meter)
σ_z	is the vertical dispersion coefficient of the plume per Pasquill class (meter)
x/Q	is the dispersion factor (sec/m^3)

The factor $\left(1 + \frac{cA}{\pi\sigma_y\sigma_z}\right)$ in the above equation is the correction term for the wake effect of the containment and is allowed to have a maximum value of 3.0, as prescribed by Regulatory Guide 1.4. It is applied only at distances less than 3000 meters, downwind; at greater distances it is set equal to 1.0. σ_y and σ_z are functions of downwind distance from the effluent source (the reactor containment) and the Pasquill Stability class.

The minimum distance from the reactor containment structure to the exclusion area boundary is 910 meters. Distances to the actual area boundary for each of the sixteen downwind direction sectors are shown in Table 2.5-22. Minimum distance to the outer boundary of the low population zone is 4828 meters.

2.5.4.2 The Diffusion Model (Longer Than 8 Hours)

After the first eight hours, the dispersion factors are represented by a model that accounts for changes in wind direction, and the resultant meandering of the plume. The equation for this model, as presented by Sagendorf[8], is:

$$\chi/Q = 2.032 \sum_j n_j [N X \bar{\mu} \textstyle\sum_{zj}(X)]^{-1}$$

2.032 is $(2/\pi)^{1/2}$ divided by the width in radians of a 22.5° sector;

n_j is the length of time (hours of valid data) weather conditions are observed to be at a given wind direction, and atmospheric stability class, j.

N is the total hours of valid data.

$\bar{\mu}$ is the average of windspeed (meter).

$\sum_{zj}(X)$ is the vertical plume spread with a volumetric correction (see below) for a release within the building wake cavity, at a distance, X for a stability class, j, otherwise $\sum_{zj}(X) = \sigma_{zj}(X)$;

$\sigma_{zj}(X)$ is the vertical dispersion coefficient at a distance, x and for stability class, j (meter)

X is the distance downwind of the source from the reactor containment to the outer boundary of lower population zone (meter);

χ/Q is the dispersion factor (sec/m^3)

The building wake correction factor can be represented by[9]:

$$\textstyle\sum_{zj}(X) = (\sigma^2_{zj}(X) + 0.5D^2/\pi)^{1/2} \geq \sqrt{3}\sigma_{zj}(X)$$

Where:

D is the maximum adjacent building height either up or downwind from the release point;

x, $\sigma_{zj}(X)$, and $\sum_{zj}(X)$, are defined above.

The building wake correction factor is restricted by the condition that:

$$\textstyle\sum_{zj}(X) = \sqrt{3}\sigma_{zj}(X)$$

When:

$$\sigma^2_{zj}(X) + 0.5D^2/\pi\ )^{1/2} > \sqrt{3}\sigma_{zj}(X)$$

2.5.4.3 Hourly Dispersion Factors Based On On-site Data

The hourly values of data for 1980 and 1981 for delta temperature, wind direction, and wind speed were accumulated and used with a minimum exclusion distance of 910 meters from the containment structure and a wake factor of 1340 m^2 to develop the frequency distribution of one-hour dispersion factors. The hourly dispersion factors (χ/Q) for the exclusion area boundary (EAB) were developed using the diffusion model described in subsection 2.5.4.1.

Step-by-step details of the accident diffusion analysis are given in Table 2.5-23. The dispersion factors which are exceeded 5 percent of time, for 1980 and 1981, are 5.69 E-04 sec/m^3 and 5.43 E-04 sec/m^3, respectively. These values are also presented in Table 2.5-30.

2.5.4.4 Dispersion Factors for Periods Up to 30 Days

The average dispersion factors for the outer boundary of the low population zone (LPZ) have been calculated for 0-8 hours, 8-24 hours, 1-4 days, and 4-30 days for each sector. The model for 0-8 hour calculations is described in subsection 2.5.4.1.

The model used for periods longer than 8 hours is the sector spread equation described in subsection 2.5.4.2. Step-by-step details of the accident diffusion analysis are provided in Table 2.5-29.

All dispersion factor averages at each duration were placed in cumulative frequency distributions for each sector. The worst case, the 5 percentile and 50 percentile averages in the worst sector were selected from these frequency distributions for duration interval as summarized in Table 2.5-24. Table 2.5-25 through 2.5-28 show, for each sector and averaging period, the highest relative concentration experienced, the 5 percentile and 50 percentile values for 1980 and 1981.

2.5.5 Long Term (Routing) Diffusion Model

A 3-year period (1982 through 1984) of onsite weather data applicable to routine gaseous releases, using the model described in subsection 2.5.4.2. Table 2.5-23 contains details of the routine release analysis method. Annual average dispersion factors at distances out to 80 km in each sector are listed in Tables 2.5-29 (1982), 2.5-30 (1983), and 2.5-31 (1984). Joint frequency summary tables for the three-year period are provided in Tables 2.5-32 through 2.5-39 for each stability class and a summary of all classes.

A MIDAS (meteorological Information and Dose Assessment System) computer program, referred to as XDCALC was used to compute the annual average χ/Q values completed for each hour of data in the meteorological file.

The meteorological data files for the years 1982, 1983, and 1984 for the Fort Calhoun Weather Tower were procured, quality assured, and found with greater than 90% recovery for the appropriate combinations of weather sensors and levels.

Computer runs, using these data, were made; assuming both ground level and mixed mode releases for each year separately, and for all three years in a fourth run. Three types of dispersion were computed including χ/Q, depleted χ/Q, and deposition. Results of the analyses for each year are summarized in tables 2.5-29 (1982), 2.5-30 (1983), and 2.5-31 (1984) respectively. (These tables included the use of a recirculation factor of 1.25 rather than 4.0 which had been previously used).

Prior to 1982, a conservative factor of 4.0 was used to account for recirculation. The χ/Q values determined using straight-line dispersion models, (e.g., the XDCALC above) were multiplied by this factor. Since it was believed that possibly the factor of 4.0 was too conservative for the Fort Calhoun site, a series of plume trajectory computer runs were made using the MESODIF-II computer program. (This program was developed by Brookhaven National Laboratory under contract with the Nuclear Regulatory Commission and has been widely used for this type of analysis.)

A MESODIF-II run was made for each one-year data period, namely, 1982, 1983, and 1984. Results (shown in Table 2.5-40) showed that the factors were much lower than 4.0. In fact, the highest single downwind direction had a factor in the highest year of 1.23 with other directions showing lower factors. In order to be conservative using these data, a factor of 1.25 was believed to be appropriate for the recirculation factor.

Replacement Sector Average χ/Q values were then calculated for the years 1982, 1983, and 1984 data bases, using the XDCALC computer program. The new recirculation factor of 1.25 was used for all downwind directions. The results are shown in Tables 2.5-29 (1982), 2.5-30 (1983), and 2.5-31 (1984) respectively.

In addition, joint-frequency tables for the three-year period (1982 through 1984) for diffusion conditions at the ten-meter wind level are also supplied. These tables are listed below and are supplied in this text.

Table No.	Pasquil Class	Code
2.5-32	Extremely Unstable	A
2.5-33	Moderately Unstable	B
2.5-34	Slightly Unstable	C
2.5-35	Neutral	D
2.5-36	Slightly Stable	E
2.5.37	Moderately Stable	F
2.5-38	Extremely Stable	G
2.5-39	All Stability	A-G

Table 2.5-39, which includes all three years of data, 1982, 1983, and 1984, shows that 92.56% of all wind and temperature lapse rate (Delta-T) combinations were recovered for the total weather data sample. This meets the required recovery goal of 90.0% required by the Nuclear Regulatory Commission for weather data recovery.

Table 2.5-1 - "Climatological Normals, Comparison of North Omaha National Weather Service with Fort Calhoun Station"

	WIND DIRECTION (PERCENT)		WIND SPEED (MPH)	
Wind Direction	North Omaha National Weather Service (1985-1989)	Fort Calhoun (1985-1989)	North Omaha National Weather Service (1985-1989)	Fort Calhoun (1985-1989)
NNE	3.8	2.6	8.2	4.9
NE	3.3	2.4	6.9	4.6
ENE	3.2	2.4	6.5	4.5
E	3.1	3.1	6.7	5.0
ESE	5.4	5.7	6.8	5.4
SE	7.1	9.0	7.8	6.8
SSE	10.0	10.2	9.9	8.9
S	10.8	10.1	10.4	9.5
SSW	9.1	7.2	9.5	9.3
SW	4.3	3.7	8.5	7.5
WSW	2.4	3.0	7.2	5.5
W	3.9	4.5	7.7	4.4
WNW	5.3	7.6	10.1	4.6
NW	8.6	10.7	12.9	6.6
NNW	6.8	9.4	12.8	6.5
N	13.0	5.5	8.0	5.9
Missing	---	2.9	---	2.9
Average	---	---	8.7	6.3

NOTE: The wind speeds at the North Omaha National Weather Service were recorded 20 feet above ground level, and the wind speeds at Fort Calhoun Station were recorded at 10 meters, above ground level. Data obtained from the Local Climatological Data; see References 4 and 5.

Table 2.5-2 - "Maximum Recorded and Mean Wind Speeds (MPH)"

	EPPLEY AIRFIELD				NORTH OMAHA NWS			
Period	Fastest Wind Speed (1949-1990)	Direction (Degrees)	Year	Mean (1936-1990)	Fastest Wind Speed (1979-1990)	Direction (Degrees)	Year	Mean (1985-1990)
January	57	NW	1938	10.9	41	NW	1978	10.4
February	57	NW	1947	11.1	38	NW	1978	9.6
March	73	NW	1950	12.3	38	NW	1982	10.9
April	65	NW	1937	12.7	46	NW	1982	10.6
May	73	NW	1936	10.9	34	N	1983	8.9
June	72	N	1942	10.1	34	NW	1983	8.4
July	109	N	1936	8.9	46	NW	1980	7.5
August	66	N	1944	8.9	39	NW	1980	7.7
September	47	E	1948	9.5	35	NW	1980	8.4
October	62	NW	1966	9.8	34	NW	1979	8.9
November	56	NW	1951	10.9	38	NW	1982	9.9
December	52	NW	1938	10.7	37	NW	1981	9.9
Year	109	N	1936	10.6	46	NW	1982	9.3

NOTE: The wind speeds at Eppley Airfield were recorded at 70 feet above ground level (agl) until 1974; 20 feet agl since that time. The wind speeds at the North Omaha NWS were recorded at 20 feet agl. Data obtained from the Local Climatological Data; see References 4 and 5.

Table 2.5-3 - "Normal and Extreme Precipitation Amounts (Inches)"

	EPPLEY AIRFIELD (1936-1990)							NORTH OMAHA (1954-1990)						
Period	**Monthly Normal**	**Monthly Maximum**	**Year**	**Monthly Minimum**	**Year**	**24-Hour Maximum (1942-1990)**	**Year**	**Monthly Normal**	**Monthly Maximum**	**Year**	**Monthly Minimum**	**Year**	**24-Hour Maximum (1977-1990)**	**Year**
January	0.77	3.70	1949	Trace	1986	1.52	1967	0.70	1.85	1975	Trace	1986	0.95	1982
February	0.91	2.97	1965	0.09	1981	2.24	1954	0.95	2.86	1965	0.09	1968	0.64	1978
March	1.91	5.96	1973	0.12	1956	1.45	1990	2.00	5.27	1983	0.06	1956	2.04	1982
April	2.94	6.45	1951	0.23	1936	2.56	1938	2.74	7.12	1984	0.15	1962	2.59	1986
May	4.33	10.33	1959	0.56	1948	4.16	1987	4.26	9.09	1959	0.55	1989	3.10	1987
June	4.08	10.81	1947	1.03	1972	3.48	1942	4.21	8.16	1984	0.95	1972	2.77	1988
July	3.62	9.60	1958	0.39	1983	3.37	1958	3.50	9.77	1958	0.29	1975	3.72	1977
August	4.10	10.16	1982	0.61	1984	5.27	1987	4.19	11.77	1960	0.63	1971	3.74	1987
September	3.50	13.75	1965	0.41	1953	6.47	1965	3.36	14.10	1965	0.96	1990	2.77	1989
October	2.09	4.99	1961	Trace	1952	3.13	1968	2.11	5.34	1986	0.06	1958	2.61	1986
November	1.32	4.70	1983	0.03	1976	2.53	1948	1.16	5.11	1983	0.03	1989	2.16	1983
December	0.77	5.42	1984	Trace	1943	3.03	1984	0.76	4.45	1984	0.02	1958	3.10	1984
Year	30.34	13.75	1965	Trace	1986	6.47	1965	29.94	14.10	1965	Trace	1986	3.74	1987

NOTE: Data obtained from the Local Climatological Data; see References 4 and 5.

Table 2.5-4 - "Normal and Extreme (Maximum) Snow and Ice Pellet Amounts (Inches)"

	EPPLEY AIRFIELD (1936-1990)					NORTH OMAHA (1954-1990)				
Period	**Normal**	**Monthly Maximum**	**Year**	**24-Hour Maximum (1942-1990)**	**Year**	**Normal**	**Monthly Maximum**	**Year**	**24-Hour Maximum (1976-1990)**	**Year**
January	7.3	25.7	1936	13.1	1949	7.0	21.5	1975	6.0	1979
February	6.8	25.4	1965	18.3	1965	6.7	23.2	1965	10.0	1978
March	6.6	27.2	1948	13.0	1948	7.2	23.3	1960	13.3	1987
April	0.8	8.6	1945	8.6	1945	1.2	10.3	1983	4.8	1979
May	0.1	2.0	1945	2.0	1945	Trace	0.7	1967	0.0	---
June	Trace	Trace	1990	Trace	1990	0.0	0.0	---	0.0	---
July	0.0	0.0	---	0.0	---	0.0	0.0	---	0.0	---
August	0.0	0.0	---	0.0	---	0.0	0.0	---	0.0	---
September	Trace	Trace	1985	Trace	1985	Trace	0.3	1985	0.3	1985
October	0.3	7.2	1941	7.2	1941	0.4	5.2	1980	5.2	1980
November	2.5	12.0	1957	8.7	1957	3.2	13.9	1957	8.5	1983
December	5.7	19.9	1969	10.2	1969	5.5	19.3	1969	7.5	1984
Year	30.0	27.2	1948	18.3	1965	31.3	23.3	March 1960	13.3	March 1987

NOTE: Data obtained from the Local Climatological Data: see References 4 and 5.

Table 2.5-5 - "Normal and Extreme Temperatures (°F)"

	EPPLEY AIRFIELD							NORTH OMAHA						
	(1961-1990)			(1936-1990)				(1961-1990)			(1954-1990)			
Period	Daily Maximum	Daily Minimum	Monthly Normal	Record High	Year	Record Low	Year	Daily Maximum	Daily Minimum	Monthly Normal	Record High	Year	Record Low	Year
January	31.1	12.7	21.9	69.0	1944	-23.0	1982	29.3	11.1	20.2	66	1981	-22	1982
February	35.8	17.1	26.5	78.0	1972	-21.0	1981	34.3	16.0	25.1	76	1972	-20	1981
March	47.5	27.8	37.7	89.0	1986	-16.0	1948	46.4	26.9	36.7	88	1986	-16	1960
April	62.4	41.1	51.8	97.0	1989	5.0	1975	60.2	38.6	49.4	96	1989	7	1975
May	73.0	52.2	62.6	99.0	1939	27.0	1980	70.6	50.0	60.3	100	1967	25	1967
June	82.5	61.9	72.2	105.0	1953	38.0	1983	81.6	60.9	71.3	104	1988	41	1956
July	87.7	67.1	77.4	114.0	1936	44.0	1972	85.9	66.0	76.0	107	1974	44	1971
August	85.2	64.9	75.1	110.0	1936	43.0	1967	83.8	63.8	73.8	106	1983	44	1986
September	76.9	55.6	66.3	104.0	1939	25.0	1984	74.9	54.3	64.6	103	1955	28	1984
October	65.5	43.7	54.6	96.0	1938	13.0	1972	64.0	42.7	53.4	93	1975	16	1972
November	48.6	29.6	39.1	80.0	1980	-9.0	1964	47.4	28.7	38.1	79	1980	-11	1964
December	35.6	18.4	27.0	72.0	1939	-23.0	1989	33.8	16.4	25.1	66	1976	-25	1989
Year	61.0	41.0	51.0	114.0	1936	-23.0	1989	59.3	39.6	49.5	107	1974	-25	1989

NOTES:

1. Data obtained from the Local Climatological Data; see References 4 and 5.
2. At the time of containment design/construction the lowest recorded temperature at Eppley Airfield was -22.0°F (January 1974).

Table 2.5-6 - "Monthly and Annual Temperature Normals (°F)"

Period	Eppley Airfield (1936-1990)	North Omaha NWS (1954-1990)	Blair (1941-1970)
January	20.2	18.7	20.8
February	27.2	25.3	26.0
March	37.3	35.2	35.2
April	52.2	50.4	50.9
May	63.3	61.7	61.5
June	73.0	71.2	70.8
July	77.7	75.7	75.5
August	75.2	73.5	73.9
September	65.8	64.4	64.3
October	54.5	53.6	54.8
November	39.5	38.0	38.7
December	27.2	25.7	26.6
Year	51.1	49.5	49.9

NOTE: Data obtained from references 4 and 5.

Table 2.5-7 -"Comparative Relative Humidity Values for Eppley Airfield (1964-1990), North Omaha (N.O.) (1984-1990), and Fort Calhoun (1969-1975)"

	0000*			0600*			1200*			1800*			24 Hour Average		
Period	**Eppley**	**N.O.**	**Ft. Calhoun**	**Eppley**	**N.O.**	**Ft. Calhoun**	**Eppley**	**N.O.**	**Ft. Calhoun**	**Eppley**	**N.O.**	**Ft. Calhoun**	**Eppley**	**N.O.**	**Ft. Calhoun**
January	75	70	82	78	74	83	65	60	71	66	60	76	70	66	78
February	76	71	82	79	75	84	63	61	70	63	62	72	70	67	77
March	72	69	80	78	77	84	57	57	66	54	54	65	65	64	74
April	68	65	72	77	75	80	52	51	59	48	46	54	62	60	66
May	72	69	75	80	78	83	54	54	61	51	51	54	64	63	68
June	75	68	75	82	77	83	55	54	59	52	50	52	66	62	67
July	78	75	77	84	83	83	57	60	60	55	57	56	69	69	69
August	80	79	82	86	87	88	59	62	65	58	61	58	71	72	75
September	81	77	82	87	84	88	59	60	65	59	60	60	72	70	74
October	76	69	79	82	78	87	55	55	65	56	56	63	67	65	74
November	76	72	85	81	77	88	62	61	72	65	64	77	71	69	80
December	78	73	84	80	76	87	67	66	74	71	69	80	74	71	81
Year	76	71	80	81	78	85	59	58	66	58	58	64	69	66	74

*Local Standard Time

NOTE: Data obtained from the Local Climatological Data; see References 4 and 5. Fort Calhoun data obtained from PSAR and archived meteorological data files.

Table 2.5-8 - "Mean Number of Days with Heavy Fog (Visibility 1/4 Mile or Less)"

Period	EppleyAirfield (1935-1990)	North Omaha NWS (1975-1990)
January	1.8	1.2
February	1.9	2.1
March	1.4	2.5
April	0.5	0.6
May	0.8	0.8
June	0.4	0.6
July	0.5	0.3
August	1.5	1.1
September	1.4	0.9
October	1.5	1.1
November	1.6	1.9
December	2.1	2.7
Year	15.4	15.8

Table 2.5-9 - "Mean Number of Days with Thunderstorms"

Period	EppleyAirfield (1935-1990)	North Omaha NWS (1975-1990)
January	0.1	0.1
February	0.4	0.4
March	1.5	1.9
April	3.8	3.4
May	7.4	7.7
June	9.4	9.5
July	8.2	8.8
August	7.8	8.1
September	5.3	6.0
October	2.4	2.3
November	0.8	0.7
December	0.2	0.2
Year	47.2	49.0

Data Period 01/01/1982 Through 12/31/1991 Run from Tape Series Tri-Ex
Omaha Public Power District
Fort Calhoun Nuclear Station
Joint Frequency Distribution Wind Direction Vs. Wind Speed in Meters/Sec for
DT100 ≤ -1.9 in Percent Data Used -- WD10, WS10, DT100
Sector Is Wind Direction Not Affected Direction

SECTOR	0.0 TO 0.4	0.5 TO 0.9	1.0 TO 1.4	1.5 TO 1.9	2.0 TO 2.4	2.5 TO 2.9	3.0 TO 3.4	3.5 TO 3.9	4.0 TO 4.4	4.5 TO 4.9	5.0 TO 5.9	6.0 TO 6.9	7.0 TO 7.9	8.0 TO 8.9	9.0 TO INF	TOTAL	UBAR
NNE	0.00	0.00	0.03	0.03	0.03	0.03	0.02	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.14	2.2
NE	0.00	0.01	0.01	0.02	0.02	0.02	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.09	2.4
ENE	0.00	0.00	0.01	0.01	0.01	0.02	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.07	2.9
E	0.00	0.00	0.00	0.01	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.04	2.7
ESE	0.00	0.00	0.00	0.00	0.01	0.01	0.01	0.01	0.00	0.01	0.01	0.00	0.00	0.00	0.00	0.06	3.4
SE	0.00	0.00	0.00	0.01	0.01	0.01	0.01	0.00	0.00	0.01	0.02	0.01	0.00	0.00	0.00	0.08	4.4
SSE	0.00	0.00	0.00	0.00	0.01	0.01	0.01	0.02	0.01	0.02	0.04	0.02	0.01	0.01	0.02	0.18	5.2
S	0.00	0.00	0.00	0.00	0.01	0.01	0.01	0.01	0.02	0.02	0.04	0.02	0.03	0.01	0.02	0.20	5.5
SSW	0.00	0.00	0.00	0.01	0.01	0.02	0.00	0.01	0.02	0.01	0.02	0.03	0.01	0.01	0.00	0.15	4.9
SW	0.00	0.00	0.01	0.01	0.01	0.01	0.02	0.01	0.00	0.01	0.01	0.00	0.00	0.00	0.00	0.09	3.0
WSW	0.00	0.00	0.00	0.01	0.01	0.01	0.01	0.01	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.06	3.1
W	0.00	0.00	0.01	0.03	0.02	0.02	0.03	0.02	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.14	2.7
WNW	0.00	0.00	0.02	0.02	0.04	0.03	0.02	0.02	0.03	0.02	0.02	0.00	0.00	0.00	0.00	0.22	3.2
NW	0.00	0.01	0.02	0.04	0.04	0.04	0.03	0.03	0.03	0.04	0.06	0.03	0.03	0.01	0.00	0.41	3.9
NNW	0.00	0.01	0.02	0.06	0.07	0.07	0.08	0.05	0.05	0.04	0.03	0.00	0.00	0.00	0.00	0.48	3.0
N	0.00	0.01	0.02	0.05	0.04	0.04	0.05	0.02	0.01	0.01	0.02	0.00	0.00	0.00	0.00	0.27	2.7
TOTAL	0.00	0.04	0.15	0.31	0.35	0.36	0.33	0.22	0.18	0.19	0.28	0.11	0.08	0.04	0.04	2.68	3.5

NUMBER OF INVALID OBSERVATIONS = 29
PERCENT OF VALID OBSERVATIONS = 2.7

Table 2.5.10 -"Joint Frequency Vs. Delta-T (Percent) Pasquill Stability Class "A"

Data Period 01/01/1982 Through 12/31/1991 Run From Tape Series Tri-Ex
Omaha Public Power District
Fort Calhoun Nuclear Station
Joint Frequency Distribution Wind Direction Vs. Wind Speed in Meters/sec For
-1.9 < Dt100 ≤ -1.7 in Percent Data Used --- WD10, WS10, DT100
Sector Is Wind Direction Not Affected Direction

SECTOR	0.0 TO 0.4	0.5 TO 0.9	1.0 TO 1.4	1.5 TO 1.9	2.0 TO 2.4	2.5 TO 2.9	3.0 TO 3.4	3.5 TO 3.9	4.0 TO 4.4	4.5 TO 4.9	5.0 TO 5.9	6.0 TO 6.9	7.0 TO 7.9	8.0 TO 8.9	9.0 TO INF	TOTAL	UBAR
NNE	0.00	0.02	0.04	0.05	0.04	0.02	0.02	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.22	2.1
NE	0.00	0.01	0.04	0.04	0.04	0.02	0.02	0.01	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.21	2.5
ENE	0.00	0.01	0.03	0.04	0.03	0.02	0.01	0.00	0.00	0.00	0..0	0.00	0.00	0.00	0.00	0.14	2.2
E	0.00	0.00	0.02	0.03	0.03	0.02	0.01	0.01	0.01	0.00	0.01	0.00	0.00	0.00	0.00	0.14	2.7
ESE	0.00	0.00	0.02	0.02	0.04	0.03	0.03	0.03	0.03	0.01	0.01	0.00	0.00	0.00	0.00	0.22	3.0
SE	0.00	0.01	0.02	0.02	0.03	0.03	0.03	0.02	0.03	0.02	0.05	0.02	0.01	0.00	0.00	0.29	3.7
SSE	0.00	0.01	0.01	0.02	0.02	0.03	0.04	0.04	0.05	0.05	0.11	0.06	0.04	0.04	0.02	0.54	5.1
S	0.00	0.01	0.02	0.03	0.03	0.05	0.04	0.03	0.05	0.05	0.09	0.11	0.05	0.04	0.03	0.63	5.0
SSW	0.00	0.00	0.02	0.03	0.02	0.01	0.03	0.04	0.03	0.03	0.06	0.04	0.03	0.01	0.02	0.37	4.6
SW	0.00	0.00	0.02	0.03	0.03	0.02	0.03	0.02	0.02	0.03	0.03	0.01	0.01	0.00	0.00	0.25	3.6
WSW	0.00	0.00	0.02	0.03	0.02	0.03	0.03	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.16	2.9
W	0.01	0.00	0.03	0.03	0.03	0.02	0.02	0.01	0.01	0.00	0.01	0.00	0.00	0.00	0.00	0.17	2.5
WNW	0.00	0.01	0.03	0.04	0.05	0.03	0.04	0.02	0.01	0.02	0.02	0.01	0.01	0.01	0.00	0.30	2.9
NW	0.01	0.01	0.03	0.05	0.07	0.07	0.07	0.05	0.04	0.06	0.06	0.03	0.01	0.01	0.01	0.58	3.5
NNW	0.00	0.01	0.05	0.05	0.11	0.12	0.10	0.10	0.07	0.05	0.06	0.03	0.01	0.00	0.00	0.76	3.3
N	0.00	0.01	0.05	0.10	0.08	0.08	0.07	0.06	0.02	0.02	0.02	0.01	0.00	0.00	0.01	0.53	2.8
TOTAL	0.02	0.11	0.45	0.61	0.67	0.60	0.59	0.46	0.40	0.37	0.54	0.32	0.17	0.11	0.09	5.51	3.6
NUMBER OF INVALID OBSERVATIONS = 58																	
PERCENT OF VALID OBSERVATIONS = 5.5																	

Table 2.5.11 - "Joint Frequency Vs. Delta-T (Percent) Pasquill Stability Class "B"

Data Period 01/01/1982 Through 12/31/1991 Run From Tape Series Tri-Ex
Omaha Public Power District
Fort Calhoun Nuclear Station
Joint Frequency Distribution Wind Direction Vs. Wind Speed in Meters/Sec For
$-1.7 < DT100 \leq -1.5$ In Percent Data Used -- WD10, WS10, DT100
Sector Is Wind Direction Not Affected Direction

SECTOR	0.0 TO 0.4	0.5 TO 0.9	1.0 TO 1.4	1.5 TO 1.9	2.0 TO 2.4	2.5 TO 2.9	3.0 TO 3.4	3.5 TO 3.9	4.0 TO 4.4	4.5 TO 4.9	5.0 TO 5.9	6.0 TO 6.9	7.0 TO 7.9	8.0 TO 8.9	9.0 TO INF	TOTAL	UBAR
NNE	0.00	0.01	0.03	0.05	0.05	0.05	0.03	0.02	0.02	0.01	0.00	0.00	0.00	0.00	0.00	0.27	2.5
NE	0.00	0.01	0.03	0.04	0.03	0.04	0.02	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.18	2.3
ENE	0.00	0.01	0.03	0.04	0.04	0.03	0.02	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.18	2.2
E	0.00	0.01	0.02	0.05	0.03	0.04	0.02	0.01	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.21	2.6
ESE	0.00	0.01	0.03	0.04	0.04	0.05	0.03	0.02	0.04	0.01	0.02	0.01	0.01	0.00	0.00	0.31	3.1
SE	0.00	0.00	0.02	0.06	0.05	0.05	0.04	0.03	0.04	0.06	0.07	0.03	0.02	0.01	0.00	0.48	3.9
SSE	0.00	0.01	0.02	0.02	0.05	0.06	0.06	0.06	0.06	0.07	0.13	0.09	0.06	0.02	0.02	0.73	4.7
S	0.00	0.00	0.02	0.04	0.05	0.04	0.07	0.06	0.06	0.05	0.13	0.11	0.05	0.03	0.02	0.73	4.7
SSW	0.00	0.01	0.02	0.04	0.03	0.04	0.03	0.03	0.05	0.03	0.07	0.06	0.04	0.02	0.01	0.48	4.4
SW	0.00	0.01	0.02	0.02	0.03	0.04	0.02	0.03	0.02	0.02	0.04	0.02	0.01	0.01	0.00	0.29	3.8
WSW	0.00	0.01	0.01	0.02	0.03	0.02	0.02	0.01	0.01	0.01	0.01	0.01	0.00	0.00	0.00	0.16	3.0
W	0.00	0.01	0.02	0.04	0.04	0.03	0.01	0.02	0.01	0.01	0.01	0.01	0.01	0.00	0.00	0.22	2.8
WNW	0.00	0.01	0.02	0.04	0.04	0.04	0.04	0.03	0.04	0.02	0.02	0.02	0.01	0.00	0.00	0.33	3.2
NW	0.02	0.01	0.04	0.07	0.08	0.12	0.09	0.07	0.08	0.06	0.07	0.05	0.03	0.01	0.00	0.80	3.4
NNW	0.01	0.01	0.04	0.09	0.09	0.12	0.14	0.11	0.10	0.07	0.08	0.04	0.01	0.01	0.00	0.92	3.3
N	0.01	0.02	0.05	0.08	0.09	0.10	0.08	0.06	0.04	0.02	0.02	0.00	0.00	0.00	0.01	0.58	2.9
TOTAL	0.04	0.15	0.42	0.74	0.77	0.87	0.72	0.58	0.58	0.45	0.68	0.45	0.25	0.11	0.06	6.87	3.6

NUMBER OF INVALID OBSERVATIONS = 45
PERCENT OF VALID OBSERVATIONS = 6.9

Table 2.5.12 - "Joint Frequency Vs. Delta-T (Percent) Pasquill Stability Class "C""

Data Period 01/01/1982 Through 12/31/1991 Run From Tape Series Tri-Ex
Omaha Public Power District
Fort Calhoun Nuclear Station
Joint Frequency Distribution Wind Direction Vs. Wind Speed in Meters/Sec for
$-1.5 < DT100 \le -0.5$ in Percent Data Used -- WD10, WS10, DT100
Sector Is Wind Direction Not Affected Direction

SECTOR	0.0 TO 0.4	0.5 TO 0.9	1.0 TO 1.4	1.5 TO 1.9	2.0 TO 2.4	2.5 TO 2.9	3.0 TO 3.4	3.5 TO 3.9	4.0 TO 4.4	4.5 TO 4.9	5.0 TO 5.9	6.0 TO 6.9	7.0 TO 7.9	8.0 TO 8.9	9.0 TO INF	TOTAL	UBAR
NNE	0.01	0.08	0.19	0.27	0.30	0.24	0.18	0.12	0.06	0.04	0.05	0.02	0.01	0.01	0.01	1.59	2.6
NE	0.01	0.08	0.22	0.19	0.22	0.15	0.12	0.09	0.04	0.03	0.02	0.01	0.00	0.00	0.00	1.18	2.3
ENE	0.00	0.09	0.19	0.24	0.20	0.16	0.13	0.10	0.05	0.03	0.04	0.01	0.00	0.00	0.00	1.24	2.4
E	0.00	0.06	0.16	0.25	0.28	0.22	0.15	0.13	0.11	0.08	0.10	0.05	0.02	0.00	0.01	1.62	2.9
ESE	0.01	0.05	0.13	0.22	0.26	0.27	0.25	0.25	0.20	0.14	0.14	0.09	0.04	0.01	0.01	2.07	3.3
SE	0.02	0.05	0.17	0.26	0.30	0.39	0.41	0.38	0.37	0.33	0.42	0.26	0.10	0.06	0.04	3.56	3.8
SSE	0.01	0.04	0.12	0.20	0.25	0.38	0.44	0.46	0.54	0.49	0.81	0.57	0.31	0.15	0.09	4.86	4.5
S	0.01	0.04	0.09	0.14	0.21	0.26	0.32	0.37	0.36	0.42	0.74	0.58	0.32	0.16	0.10	4.12	4.8
SSW	0.01	0.05	0.07	0.13	0.14	0.19	0.21	0.21	0.18	0.17	0.30	0.27	0.16	0.08	0.06	2.23	4.4
SW	0.01	0.05	0.06	0.09	0.11	0.14	0.11	0.09	0.09	0.08	0.10	0.06	0.05	0.02	0.02	1.08	3.7
WSW	0.01	0.05	0.10	0.13	0.13	0.12	0.10	0.07	0.05	0.04	0.05	0.03	0.01	0.01	0.01	0.91	2.8
W	0.03	0.07	0.13	0.16	0.17	0.14	0.11	0.07	0.06	0.03	0.03	0.04	0.02	0.00	0.00	1.06	2.6
WNW	0.05	0.15	0.22	0.22	0.24	0.22	0.18	0.15	0.11	0.09	0.17	0.08	0.04	0.03	0.03	1.98	3.1
NW	0.06	0.15	0.35	0.36	0.47	0.50	0.55	0.49	0.41	0.38	0.48	0.32	0.13	0.05	0.03	4.73	3.5
NNW	0.05	0.18	0.36	0.51	0.68	0.70	0.76	0.72	0.49	0.36	0.41	0.23	0.10	0.02	0.01	5.58	3.2
N	0.02	0.12	0.32	0.39	0.45	0.45	0.38	0.28	0.20	0.10	0.08	0.04	0.02	0.00	0.03	2.88	2.7
TOTAL	0.31	1.31	2.88	3.76	4.41	4.53	4.40	3.98	3.32	2.81	3.94	2.66	1.33	0.60	0.45	40.69	3.6
NUMBER OF INVALID OBSERVATIONS = 330																	
PERCENT OF VALID OBSERVATIONS = 40.7																	

Table 2.5.13 - "Joint Frequency Vs. Delta-T (Percent) Pasquill Stability Class "D""

Data Period 01/01/1982 Through 12/31/1991 Run from Tape Series Tri-Ex
Omaha Public Power District
Fort Calhoun Nuclear Station
Joint Frequency Distribution Wind Direction Vs. Wind Speed in Meters/Sec for
-0.5 < DT100 ≤ +1.5 In Percent Data Used -- WD10, WS10, DT100
Sector Is Wind Direction Not Affected Direction

SECTOR	0.0 TO 0.4	0.5 TO 0.9	1.0 TO 1.4	1.5 TO 1.9	2.0 TO 2.4	2.5 TO 2.9	3.0 TO 3.4	3.5 TO 3.9	4.0 TO 4.4	4.5 TO 4.9	5.0 TO 5.9	6.0 TO 6.9	7.0 TO 7.9	8.0 TO 8.9	9.0 TO INF	TOTAL	UBAR
NNE	0.02	0.10	0.14	0.10	0.07	0.05	0.03	0.02	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.54	1.7
NE	0.02	0.09	0.12	0.09	0.05	0.03	0.03	0.02	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.48	1.8
ENE	0.01	0.08	0.14	0.09	0.05	0.05	0.03	0.02	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.50	1.8
E	0.01	0.11	0.17	0.16	0.11	0.08	0.06	0.05	0.02	0.02	0.04	0.01	0.01	0.01	0.00	0.86	2.3
ESE	0.02	0.15	0.29	0.40	0.29	0.21	0.16	0.09	0.07	0.04	0.05	0.02	0.03	0.02	0.02	1.86	2.5
SE	0.03	0.14	0.39	0.52	0.57	0.56	0.52	0.37	0.25	0.18	0.22	0.10	0.05	0.01	0.01	3.92	2.9
SSE	0.03	0.13	0.15	0.21	0.27	0.43	0.46	0.45	0.40	0.35	0.38	0.19	0.07	0.04	0.03	3.59	3.6
S	0.04	0.11	0.14	0.19	0.20	0.23	0.31	0.32	0.31	0.32	0.49	0.35	0.13	0.05	0.03	3.22	4.0
SSW	0.03	0.12	0.14	0.14	0.11	0.13	0.12	0.11	0.15	0.15	0.28	0.17	0.11	0.06	0.03	1.85	3.9
SW	0.05	0.17	0.11	0.13	0.08	0.09	0.08	0.06	0.05	0.05	0.08	0.06	0.03	0.02	0.01	1.07	2.9
WSW	0.06	0.22	0.17	0.08	0.09	0.08	0.07	0.07	0.06	0.02	0.04	0.01	0.01	0.00	0.00	0.98	2.2
W	0.10	0.42	0.38	0.20	0.14	0.11	0.13	0.09	0.07	0.02	0.01	0.01	0.01	0.01	0.00	1.70	1.8
WNW	0.10	0.64	0.84	0.49	0.31	0.18	0.14	0.10	0.07	0.04	0.05	0.01	0.02	0.01	0.01	3.01	1.8
NW	0.06	0.39	0.61	0.71	0.42	0.34	0.27	0.16	0.12	0.08	0.07	0.05	0.03	0.01	0.02	3.34	2.2
NNW	0.03	0.20	0.30	0.36	0.25	0.21	0.17	0.14	0.07	0.06	0.06	0.04	0.02	0.02	0.04	1.97	2.5
N	0.02	0.14	0.18	0.20	0.14	0.10	0.09	0.05	0.03	0.02	0.02	0.01	0.00	0.00	0.01	1.01	2.1
TOTAL	0.63	3.21	4.27	4.07	3.15	2.88	2.67	2.12	1.70	1.37	1.81	1.03	0.52	0.26	.021	29.90	2.7

NUMBER OF INVALID OBSERVATIONS = 206

PERCENT OF VALID OBSERVATIONS = 29.9

Table 2.5.14 - "Joint Frequency Vs. Delta-T (Percent) Pasquill Stability Class "E""

Data Period 01/01/1982 Through 12/31/1991 Run from Tape Series Tri-Ex
Omaha Public Power District
Fort Calhoun Nuclear Station
Joint Frequency Distribution Wind Direction Vs. Wind Speed in Meters/Sec for
+1.5 < DT100 ≤ +4.0 In Percent Data Used -- WD10, WS10, DT100
Sector Is Wind Direction Not Affected Direction

SECTOR	0.0 TO 0.4	0.5 TO 0.9	1.0 TO 1.4	1.5 TO 1.9	2.0 TO 2.4	2.5 TO 2.9	3.0 TO 3.4	3.5 TO 3.9	4.0 TO 4.4	4.5 TO 4.9	5.0 TO 5.9	6.0 TO 6.9	7.0 TO 7.9	8.0 TO 8.9	9.0 TO INF	TOTAL	UBAR
NNE	0.01	0.04	0.05	0.03	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.15	1.3
NE	0.01	0.06	0.05	0.03	0.00	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.17	1.4
ENE	0.01	0.06	0.06	0.02	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.17	1.2
E	0.01	0.09	0.10	0.05	0.02	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.30	1.3
ESE	0.03	0.14	0.25	0.23	0.16	0.08	0.03	0.02	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.95	1.7
SE	0.05	0.20	0.25	0.24	0.23	0.23	0.10	0.05	0.03	0.01	0.01	0.00	0.00	0.00	0.00	1.40	1.9
SSE	0.03	0.17	0.12	0.13	0.12	0.12	0.08	0.05	0.03	0.01	0.01	0.00	0.01	0.00	0.00	0.88	2.0
S	0.03	0.16	0.13	0.15	0.11	0.15	0.09	0.11	0.07	0.05	0.07	0.03	0.01	0.00	0.00	1.16	2.6
SSW	0.04	0.16	0.10	0.08	0.08	0.05	0.07	0.08	0.08	0.05	0.09	0.07	0.04	0.02	0.01	1.02	3.2
SW	0.04	0.18	0.11	0.05	0.02	0.04	0.05	0.03	0.03	0.03	0.07	0.02	0.02	0.01	0.00	0.70	2.5
WSW	0.08	0.24	0.12	0.06	0.03	0.02	0.03	0.03	0.02	0.02	0.03	0.01	0.00	0.00	0.00	0.69	1.7
W	0.10	0.43	0.27	0.10	0.04	0.03	0.02	0.01	0.01	0.01	0.01	0.01	0.00	0.00	0.00	1.04	1.2
WNW	0.09	0.49	0.51	0.25	0.09	0.03	0.02	0.01	0.00	0.00	0.01	0.00	0.00	0.00	0.00	1.50	1.2
NW	0.03	0.24	0.28	0.15	0.07	0.03	0.02	0.01	0.00	0.01	0.01	0.00	0.00	0.00	0.00	0.85	1.4
NNW	0.01	0.09	0.08	0.04	0.02	0.01	0.01	0.00	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.27	1.4
N	0.02	0.07	0.07	0.04	0.03	0.02	0.00	0.00	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.26	1.5
TOTAL	0.59	2.82	2.55	1.65	1.04	0.85	0.54	0.41	0.28	0.19	0.33	0.14	0.08	0.03	0.01	11.51	1.8

NUMBER OF INVALID OBSERVATIONS = 63
PERCENT OF VALID OBSERVATIONS = 11.5

Table 2.5.15 - "Joint Frequency Vs. Delta-T (Percent) Pasquill Stability Class "F""

Data Period 01/01/1982 Through 12/31/1991 Run from Tape Series Tri-Ex
Omaha Public Power District
Fort Calhoun Nuclear Station
Joint Frequency Distribution Wind Direction Vs. Wind Speed in Meters/Sec for
DT100 > +4.0 In Percent Data Used -- WD10, WS10, DT100
Sector Is Wind Direction Not Affected Direction

SECTOR	0.0 TO 0.4	0.5 TO 0.9	1.0 TO 1.4	1.5 TO 1.9	2.0 TO 2.4	2.5 TO 2.9	3.0 TO 3.4	3.5 TO 3.9	4.0 TO 4.4	4.5 TO 4.9	5.0 TO 5.9	6.0 TO 6.9	7.0 TO 7.9	8.0 TO 8.9	9.0 TO INF	TOTAL	UBAR
NNE	0.00	0.03	0.02	0.01	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.07	1.6
NE	0.01	0.04	0.02	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.08	1.1
ENE	0.01	0.03	0.03	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.07	0.9
E	0.01	0.06	0.05	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.13	1.0
ESE	0.03	0.10	0.11	0.06	0.03	0.02	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.35	1.3
SE	0.02	0.09	0.11	0.05	0.02	0.01	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.31	1.3
SSE	0.02	0.08	0.08	0.04	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.01	0.00	0.00	0.00	0.25	1.4
S	0.02	0.06	0.04	0.01	0.02	0.01	0.02	0.01	0.02	0.01	0.01	0.00	0.00	0.00	0.00	0.23	2.0
SSW	0.02	0.07	0.03	0.02	0.01	0.01	0.01	0.02	0.02	0.01	0.02	0.01	0.01	0.01	0.00	0.27	2.7
SW	0.02	0.05	0.04	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.01	0.00	0.00	0.21	2.6
WSW	0.01	0.07	0.03	0.01	0.01	0.00	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.15	1.6
W	0.03	0.10	0.04	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.20	1.2
WNW	0.03	0.06	0.05	0.02	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.18	1.2
NW	0.01	0.05	0.04	0.01	0.01	0.00	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.13	1.3
NNW	0.01	0.02	0.02	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.01	0.01	0.00	0.00	0.01	0.09	2.7
N	0.01	0.03	0.02	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.01	0.00	0.01	0.00	0.03	0.12	4.3
TOTAL	0.26	0.94	0.73	0.29	0.14	0.10	0.05	0.05	0.07	0.03	0.06	0.04	0.03	0.01	0.04	2.84	1.7

NUMBER OF INVALID OBSERVATIONS = 44
PERCENT OF VALID OBSERVATIONS = 2.8

Table 2.5.16 - "Joint Frequency Vs. Delta-T (Percent) Pasquill Stability Class "G""

Data Period 01/01/1982 Through 12/31/1991 Run from Tape Series Tri-Ex
Omaha Public Power District
Fort Calhoun Nuclear Station
Joint Frequency Distribution Wind Direction Vs. Wind Speed in Meters/Sec for
DT100 = -Inf To + Inf In Percent Data Used -WD10, WS10, DT100
Sector Is Wind Direction Not Affected Direction

SECTOR	0.0 TO 0.4	0.5 TO 0.9	1.0 TO 1.4	1.5 TO 1.9	2.0 TO 2.4	2.5 TO 2.9	3.0 TO 3.4	3.5 TO 3.9	4.0 TO 4.4	4.5 TO 4.9	5.0 TO 5.9	6.0 TO 6.9	7.0 TO 7.9	8.0 TO 8.9	9.0 TO INF	TOTAL	UBAR
NNE	0.04	0.27	0.50	0.53	0.50	0.40	0.27	0.18	0.09	0.06	0.06	0.03	0.02	0.01	0.01	2.97	2.3
NE	0.05	0.29	0.48	0.41	0.35	0.28	0.20	0.13	0.07	0.05	0.05	0.02	0.01	0.00	0.00	2.39	2.1
ENE	0.03	0.27	0.48	0.45	0.33	0.27	0.19	0.14	0.08	0.05	0.06	0.02	0.00	0.00	0.00	2.37	2.1
E	0.03	0.33	0.52	0.57	0.48	0.37	0.26	0.20	0.15	0.12	0.16	0.07	0.02	0.01	0.02	3.31	2.5
ESE	0.09	0.44	0.82	0.97	0.83	0.65	0.52	0.42	0.36	0.20	0.24	0.12	0.07	0.04	0.03	5.80	2.6
SE	0.12	0.50	0.97	1.16	1.21	1.28	1.10	0.85	0.73	0.60	0.77	0.42	0.19	0.08	0.06	10.04	3.1
SSE	0.09	0.43	0.50	0.62	0.74	1.03	1.09	1.07	1.09	0.99	1.48	0.95	0.50	0.27	0.18	11.03	4.0
S	0.10	0.38	0.45	0.57	0.62	0.75	0.85	0.91	0.90	0.93	1.56	1.19	0.59	0.29	0.20	10.29	4.3
SSW	0.10	0.41	0.38	0.45	0.40	0.44	0.49	0.51	0.53	0.45	0.85	0.64	0.39	0.20	0.13	6.37	4.0
SW	0.12	0.45	0.37	0.34	0.29	0.34	0.30	0.25	0.23	0.22	0.33	0.18	0.14	0.07	0.05	3.68	3.2
WSW	0.16	0.59	0.44	0.34	0.32	0.29	0.26	0.20	0.15	0.10	0.14	0.07	0.03	0.01	0.02	3.12	2.3
W	0.27	1.03	0.89	0.57	0.43	0.35	0.32	0.22	0.17	0.08	0.09	0.07	0.03	0.01	0.01	4.54	1.9
WNW	0.28	1.37	1.68	1.07	0.77	0.54	0.46	0.33	0.26	0.20	0.28	0.12	0.07	0.05	0.04	7.52	2.1
NW	0.20	0.86	1.36	1.39	1.15	1.11	1.03	0.81	0.69	0.63	0.76	0.47	0.23	0.09	0.06	10.84	3.0
NNW	0.12	0.53	0.87	1.12	1.22	1.23	1.26	1.12	0.78	0.57	0.65	0.35	0.14	0.05	0.07	10.08	3.1
N	0.07	0.40	0.71	0.86	0.84	0.80	0.67	0.48	0.29	0.17	0.17	0.07	0.03	0.01	0.08	5.65	2.6
TOTAL	1.87	8.55	11.42	11.42	10.48	10.13	9.27	7.82	6.57	5.42	7.65	4.79	2.46	1.19	0.96	100.00	3.1
NUMBER OF INVALID OBSERVATIONS = 2018																	
PERCENT OF VALID OBSERVATIONS = 97.7																	

Table 2.5.17 - "Joint Frequency Vs. Delta-T (Percent) Pasquill Stability Class "A" Through "G""

Table 2.5-18 - "Fort Calhoun On-Site Meteorological Measurement Program (Initial Instrumentation and Recording System)"

Climatological Station - Fold-over 40-foot Tower and Hilltop Station

Meteorological Instrument	Manufacturer	Instrument Model	Sensor Level (feet)[1]	Sensor Specifications
Wind direction, speed and temperature (Mechanical weather station)	Meteorology Research, Inc.	MWS 1071	43.6	Direction Starting threshold: <0.75 mph Delay distance: 4 feet (50 percent recovery) Damping ratio: 0.5-0.6 Range: 0 to 360 degrees Accuracy: ± 1 percent of full scale Speed Starting threshold: <0.75 mph Response distance: 18 feet (63 percent recovery) Flow coefficient: 7.9 ft/rev. Accuracy: ± 2 percent Temperature: ±3 degrees F
Temperature (Thermograph)	Bendix Corporation	W-6	5.7	Bimetal strip, accuracy: ± 2 degrees F Calibrated range: -35 to 110 degrees F
			5.7 Hilltop 310.0	Bimetal strip, accuracy: ± 2 degrees F Calibrated range: -35 to +110 degrees F
Relative Humidity (hygrothermograph)	Belfort Instrument Company	5-594	5.7	Banjo spread human hair, accuracy: ± 4 percent Calibrated range: 0-100 percent Bimetal strip, accuracy: + 2 degrees F Calibrated range: 0-110 degrees F

Table 2.5-18 (Continued)

Climatological Station - Fold-over 40-foot Tower and Hilltop Station

Meteorological Instrument	Manufacturer	Instrument Model	Sensor Level (feet)[1]	Sensor Specifications
Precipitation (Rain gauge)	Belfort Instrument Company	5-780	3.3	Weighing gauge, accuracy: ± 1 percent Calibrated range: 0-12 in. water
Barometric Pressure (Barometer)	Belfort Instrument Company	5-800A	4.7	Bellows, accuracy: ± 0.3 millibars Calibrated range: 28.5 - 31.0 in. mercury

[1]All levels are elevations above ground level (AGL).

Table 2.5-19 - "Fort Calhoun On-site Meteorological Measurement Program (Initial Instrumentation and Recording System)"

Rohn Guyed 160-Foot Weather Tower

Meteorological Instrument	Manufacturer	Instrument Model	Sensor Level (feet)[1]	Sensor Specifications
Wind speed and direction horizontal and vertical (WSI, A1 & SDE & SDA respectively)	Meteorology Research, Inc.	1053 Mark III "Vectorvane"	115	Direction Starting threshold: 0.75 mph Delay distance: 3 feet (50 percent recovery) Damping ratio: ~0.6 Range: azimuth 0-540 degrees ± 1 percent (Accuracy) elevation -60 to +60 degrees ± 2 percent (accuracy) Speed Starting threshold: 0.75 mph Response distance: 3 feet (63 percent recovery) Range: 0-100 mph ± 0.2 mph or 1 percent (whichever is greater)
Temperature	Meteorology Research, Inc.	YSI Thermil-inear No. 44203	84.2	Thermistor accuracy: ± 0.15 degrees C Calibrated range: -30 to 50 degrees C Beckman Whitney Aspirated Shield
Delta Temperature (T1 & T3)	Meteorology Research, Inc.	YSI Thermil-inear No. 1001	6.56 147.5	Thermistor accuracy: ± 0.15 degrees C Calibrated range: -30 to 50 degrees C Beckman Whitney Aspirated Shield Copper/constantan, accuracy ± 0.1 degrees C Threshold 0.1 percent range: 0-100 MV Calibrated range: -1.5 to 3.5 degrees F
Recorders	Leeds & Northup	610xL Multipoint	Reactor Bldg.	

[1]All levels are elevations above ground level (AGL).

Table 2.5-20 - "Fort Calhoun On-Site Meteorological Measurements Program (Instrumentation and Recording System)"

Rohn Guyed 160-foot Weather Tower

Meteorological Instrument	Manufacturer	Instrument Model	Sensor Level (feet)[1]	Sensor Specifications
Wind speed and direction (WDRT1 & WSRT1) (WDRT1 & WSRT2)	Meteorology Research, Inc.	1074 wind system	36.3 117.8	Direction Starting threshold: 0.75 mph Delay distance: 4 feet (50 percent recovery) Damping ratio: 0.5-0.6 Range: 0 degrees -540 degrees ± 1 percent (accuracy) Speed Starting threshold: 0.75 mph Response distance: 18 feet (63 percent recovery) Flow coefficient: 7.9 ft/rev. Range: 0-80 mph Accuracy: ± 0.4 mph
Temperature	Meteorology Research, Inc.	YSI Thermil-inear No. 44203	32.3 5.7	Thermistor accuracy: 0.15 degrees C Calibrated range: -30 to 50 degrees C Beckman Whitney Aspirated Shield
Delta Temperature (DTRT)	Meteorology Research, Inc.	YSI Thermil-inear No. 1001	32.3 147.5	Thermistor accuracy: ± 0.15 degrees C Calibrated range: -5.4 to 5.4 degrees F Beckman Whitney Aspirated Shield
Recorders	Leeds & Northup	AZAR "Speedomax-H" Multipoint	Reactor Bldg.	Copper/constantan, accuracy ± 0.1 degrees C Threshold 0.1 percent Calibrated range: -1.5 to 3.5 degrees F

Table 2.5-20 (Continued)

Climatological Station - Fold-over 40-foot Tower

Meteorological Instrument	Manufacturer	Instrument Model	Sensor Level (feet)[1]	Sensor Specifications
Wind direction, speed and temperature (WSMT & WDMT)	Meteorology Research, Inc.	MWS 1071	43.6	Direction Starting threshold: <0.75 mph Delay distance: 4 feet (50 percent recovery) Damping ratio: 0.5-0.6 Range: 0 degrees -360 degrees ± 1 percent of full scale (accuracy) Speed Starting threshold: <0.75 MPH Response distance: 18 feet (63 percent recovery) Flow coefficient: 7.9 ft/rev. Accuracy: ± 2 percent Temperature: ± 3 degrees F
Temperature	Bendix Corporation	W-6	5.7	Bimetal strip, accuracy: ± 2 degrees F Calibrated range: -35 to +110 degrees F
			5.7 Hilltop 310	Bimetal strip, accuracy: ± 2 degrees F Calibrated range: -35 to +110 degrees F
Relative Humidity	Belfort Instrument Company	5-594	5.7	Banjo spread human hair, accuracy: ± 4 percent Calibrated range: 0-100 percent Bimetal strip, accuracy: ± 2 degrees F Calibrated range: 0-100 degrees F

Table 2.5-20 (Continued)

Climatological Station - Fold-over 40-foot Tower

Meteorological Instrument	Manufacturer	Instrument Model	Sensor Level (feet)[1]	Sensor Specifications
Precipitation	Belfort Instrument Company	5.780	3.3	Weighing gauge, accuracy: ± 1 percent Calibrated range 0-12 in. water
Barometric Pressure	Belfort Instrument Company	5-800A	4.7	Bellows, accuracy: ± 0.3 millibars Calibrated range: 28.5 = 31.0 in. mercury

[1]All levels are elevations above ground level (AGL).

Table 2.5-21 - "Fort Calhoun 110 Meter Tower Instruments"

Meteorological Instrument (or Parameter)	Manufacturer	Instr. Model	Sensor Elevation (Ft.) Instr. Above Ground Level	Qty	Sensor Specification
Wind Direction	Climatronics	F460	1152.6 (@ 60 M) 1037.8 (@ 10M)	2 2	Azimuth 0-540° Starting Threshold .9 mph ACC ±5° Damping Ratio .6 Delay Distance 4 ft.
Wind Speed	Climatronics	F460	1152.6 (@ 60M) 1037.8 (@ 10M)	2 2	Speed Range 0-100 Starting Threshold .9 mph ACC ±.5° mph Response Distance 8 ft.
Ambient Temp.	Climatronics	TS-10	1037.8 (@ 10M)	2	Range -50°C to +15°C ACC ±.1°C
Delta Temp.	Climatronics	Temp/ Temp P/N-100088-2 YSI-703	1037.8-1365.9 (@ 60 M-10M)	2	Range -50°C to +15°C ACC ±.1°C
Sigma Azimuth	Climatronics	IMP-860	1365.9 (@ 60M) 1037.8 (@ 10M) ACC ±1°	1 1	Azimuth Range 0-100° Resolution 1°
Precipitation	Climatronics	100097-1-GO	Ground Level	1	Resolution: 0.25 mm ACC: 10% of Catch

Table 2.5-22 - "Radials and Minimum Distances from the Containment Structure to the Exclusion Area Boundary"

Sector	Radial (Degrees)	Boundary Distance (Meters)
N	000.0	1053.6
NNE	022.5	996.4
NE	045.0	982.1
ENE	067.5	1008.3
E	090.0	1258.4
ESE	112.5	1115.5
SE	135.0	1129.8
SSE	157.5	963.1
S	180.0	910.7
SSW	202.5	915.5
SW	225.0	1015.5
WSW	247.5	1103.6
W	270.0	1096.4
WNW	292.5	1122.6
NW	315.0	1596.5
NNW	337.5	1558.4

Table 2.5-23 - "Meteorological Analyses of Onsite Data Step-by-Step Description"

Hourly (Minimum EAB)	- Calculate hourly average values of χ/Q using centerline invariant wind with wake effects At the single, minimum exclusion area boundary (EAB) arrange values in cumulative frequency distribution - Select 5 percent and 50 percent values
Hourly (Actual EAB)	- The same as hourly, except the calculation is made at the actual exclusion area boundary as selected by the wind direction - Data are arranged in a cumulative frequency distribution - Select 5 percent and 50 percent values
8-Hourly	- The same as hourly, except the calculations are made at the minimum outer boundary of the low population zone in each sector as selected by the wind direction - For each hour, in each sector, average the past 8-hour χ/Q values, using $\chi/Q=0$ when wind direction is not in the sector - Arrange all 8-hourly average values for each sector in a cumulative frequency distribution - Select the worst, the 5 percent and 50 percent values for each sector
16-Hourly	- The same as 8-hourly, except that the sector spread replaces the centerline assumption, the meander factor and wake factor are omitted, and the past 8-hour averages are replaced with past 16-hour averages
Three-Day (72-Hourly)	- The same as 16-hourly, except that the past 16-hour averages are replaced with the past 72-hour averages
26-Day (624-Hourly)	- The same as 16-hourly, except that the past 16-hour averages are replaced by the past 624-hour averages
Average Annual Concentrations	- Calculate average hourly χ/Q values using sector spread equations - At the exclusion zone boundary distance in each sector, and out to 50 miles - Average the χ/Q values for the entire year at each radial distance in each sector ($\chi/Q=0$ if wind is in another sector) - Provide analyses for determining the finite cloud parameters for annual average release calculations

Table 2.5-24 - "Summary of Percentile Values for Worst Sector Dispersion Factors (x/Q) 1980 and 1981"

Duration of Averages Affected	Percentile of all Averages	1980 x/Q(sec/m³)	1980 Sector Affected	1981 x/Q(sec/m³)	1981 Sector
1 Hour	5	5.69 E-04	----	5.43 E-04	----
(EAB)	50	7.69 E-05	----	7.53 E-05	----
8 Hours	Worst	9.56 E-05	E	1.13 E-04	NW
(Worst Sector,	5	1.47 E-05	ESE	1.43 E-05	ESE
LPZ)	50	Zero	----	Zero	----
16 Hours	Worst	9.20 E-06	ESE	7.98 E-06	E
(Worst Sector,	5	2.49 E-06	ESE	2.67 E-06	ESE
LPZ)	50	Zero	----	Zero	----
72 Hours	Worst	3.20 E-06	ESE	3.36 E-06	SE
(Worst Sector,	5	1.73 E-06	ESE	1.63 E-06	ESE
LPZ)	50	3.46 E-07	SE	3.56 E-07	ESE
624 Hours	Worst	1.23 E-06	ESE	3.44 E-06	SSW
(Worst Sector,	5	1.22 E-06	ESE	8.86 E-07	ESE
LPZ)	50	4.01 E-07	SSE	5.09 E-07	ESE

Table 2.5-25 - "Average Dispersion Factors (x/Q) at the IPZ Outer Boundary for 0-8 Hours 1980 and 1981"

0 - 8 Hours
x/Q (sec/m³)

Sector	1980 Worst	1980 5 Percent	1981 Worst	1981 5 Percent
N	6.22 E-05	6.49 E-06	3.88 E-05	4.14 E-06
NNE	6.94 E-05	6.46 E-06	4.96 E-05	3.69 E-06
NE	5.39 E-05	5.84 E-06	8.67 E-05	3.60 E-06
ENE	5.91 E-05	6.32 E-06	1.01 E-04	4.52 E-06
E	9.56 E-05	7.35 E-06	9.48 E-05	7.19 E-06
ESE	8.83 E-05	1.47 E-05	7.85 E-05	1.43 E-05
SE	6.63 E-05	1.25 E-05	7.30 E-05	1.08 E-05
SSE	5.23 E-05	8.07 E-06	4.34 E-05	5.68 E-06
S	2.48 E-05	4.51 E-06	3.47 E-05	3.74 E-06
SSW	2.17 E-05	3.76 E-06	6.94 E-05	2.74 E-06
SW	5.83 E-05	3.19 E-06	7.23 E-05	2.35 E-06
WSW	4.80 E-05	2.63 E-06	6.36 E-05	2.50 E-06
W	3.31 E-05	2.57 E-06	8.67 E-05	3.75 E-06
WNW	7.96 E-05	5.21 E-06	7.19 E-05	7.02 E-06
NW	7.93 E-05	1.24 E-05	1.13 E-04	9.18 E-06
NNW	5.06 E-05	7.17 E-06	6.12 E-05	5.42 E-06

Table 2.5-26 - "Average Dispersion Factors (x/Q) at the LPZ Outer Boundary for 8-24 Hours 1980 and 1981"

	8 - 24 Hours x/Q (sec/m³)			
	1980		1981	
Sector	Worst	5 Percent	Worst	5 Percent
N	5.01 E-06	1.48 E-06	3.30 E-06	9.49 E-07
NNE	5.64 E-06	1.54 E-06	3.94 E-06	8.85 E-07
NE	4.80 E-06	1.42 E-06	6.38 E-06	9.68 E-07
ENE	5.16 E-06	1.37 E-06	7.78 E-06	1.11 E-06
E	7.59 E-06	1.49 E-06	7.98 E-06	1.64 E-06
ESE	9.20 E-06	2.49 E-06	8.80 E-06	2.67 E-06
SE	5.74 E-06	2.11 E-06	6.87 E-06	2.29 E-06
SSE	4.22 E-06	1.82 E-06	4.33 E-06	1.56 E-06
S	2.44 E-06	1.19 E-06	3.13 E-06	9.74 E-07
SSW	2.28 E-06	8.52 E-07	5.42 E-06	7.44 E-07
SW	4.23 E-06	7.86 E-07	5.25 E-06	7.22 E-07
WSW	3.49 E-06	7.31 E-07	4.62 E-06	8.16 E-07
W	3.56 E-06	7.65 E-07	6.29 E-06	9.62 E-07
WNW	5.78 E-06	1.34 E-06	5.95 E-06	1.71 E-06
NW	7.53 E-06	2.10 E-06	8.86 E-06	1.98 E-06
NNW	4.39 E-06	1.48 E-06	5.06 E-06	1.37 E-06

Table 2.5-27 - "Average Dispersion Factors (x/Q) at the LPZ Outer Boundary for 1-4 Days 1980 and 1981"

1 - 4 Days
x/Q (sec/m³)

Sector	1980 Worst	1980 5 Percent	1981 Worst	1981 5 Percent
N	1.98 E-06	8.36 E-07	1.04 E-06	6.56 E-07
NNE	1.37 E-06	1.07 E-06	1.47 E-06	5.89 E-07
NE	1.30 E-06	7.37 E-07	1.60 E-06	5.13 E-07
ENE	1.89 E-06	1.02 E-06	1.95 E-06	7.23 E-07
E	1.86 E-06	9.91 E-07	2.11 E-06	9.76 E-07
ESE	3.20 E-06	1.73 E-06	2.70 E-06	1.63 E-06
SE	2.39 E-06	1.36 E-06	3.36 E-06	1.46 E-06
SSE	2.52 E-06	1.08 E-06	2.53 E-06	9.26 E-07
S	1.33 E-06	6.76 E-07	1.35 E-06	6.21 E-07
SSW	1.24 E-06	5.35 E-07	1.93 E-06	4.74 E-07
SW	1.38 E-06	4.97 E-07	1.55 E-06	6.02 E-07
WSW	1.23 E-06	5.65 E-07	1.96 E-06	6.66 E-07
W	2.15 E-06	5.15 E-07	1.54 E-06	7.96 E-07
WNW	1.59 E-06	7.22 E-07	2.51 E-06	1.18 E-06
NW	1.97 E-06	1.31 E-06	2.60 E-06	1.23 E-06
NNW	1.44 E-06	8.95 E-07	1.39 E-06	9.51 E-07

Table 2.5-28 - "Average Dispersion Factors (x/Q) at the LPZ Outer Boundary for 4-30 Days 1980 and 1981"

8 - 24 Hours
x/Q (sec/m³)

Sector	1980 Worst	1980 5 Percent	1981 Worst	1981 5 Percent
N	5.45 E-07	5.20 E-07	3.72 E-07	3.46 E-07
NNE	8.06 E-07	7.31 E-07	3.10 E-07	2.65 E-07
NE	5.08 E-07	4.99 E-07	3.36 E-07	2.85 E-07
ENE	5.62 E-07	5.37 E-07	4.87 E-07	4.36 E-07
E	4.04 E-07	3.83 E-07	5.52 E-07	4.90 E-07
ESE	1.23 E-06	1.22 E-06	1.00 E-06	8.86 E-07
SE	8.24 E-07	7.58 E-07	8.85 E-07	7.47 E-07
SSE	9.32 E-07	8.32 E-07	6.13 E-07	5.16 E-07
S	3.91 E-07	3.67 E-07	4.55 E-07	4.04 E-07
SSW	4.62 E-07	4.45 E-07	3.44 E-06	3.23 E-07
SW	3.06 E-07	2.79 E-07	3.68 E-07	3.37 E-07
WSW	2.45 E-07	2.33 E-07	2.83 E-07	2.70 E-07
W	4.27 E-07	3.77 E-07	3.93 E-07	3.36 E-07
WNW	5.21 E-07	4.22 E-07	6.51 E-07	5.78 E-07
NW	6.38 E-07	5.68 E-07	7.09 E-07	6.51 E-07
NNW	4.62 E-07	4.27 E-07	5.11 E-07	4.67 E-07

Table 2.5-29 - " Long Term Dispersion Factors For 1982 (Sec/M^3)*"
Long-Term Ground Release CHI/Q = (Summation (2.032/X*WS*Ƶ))/Total Valid Records

Ƶ = Sort ((Ƶ*Ƶ)*(D*D/2PI)) Limited To SQRT(3)*Ƶ

Data Used WS10.WD10.DT100

Wind Direction From	0.5 Mile	1.5 Mile	2.5 Mile	3.5 Mile	4.5 Mile	7.5 Mile	15 Mile	25 Mile	35 Mile	45 Mile
NNE	1.9E-6	3.4E-7	1.5E-7	9.0E-8	6.2E-8	2.9E-8	1.0E-8	4.5E-9	2.8-E9	2.0E-9
NE	1.9E-6	3.5E-7	1.6E-7	9.5E-8	6.6E-8	3.1E-8	1.1E-8	5.0E-9	3.2E-9	2.3E-9
ENE	1.9E-6	3.5E-7	1.6E-7	9.5E-8	6.6E-8	3.1E-8	1.1E-8	4.9E-9	3.1E-9	2.2E-9
E	2.5E-6	4.7E-7	2.2E-7	1.3E-7	9.3E-8	4.4E-8	1.6E-8	7.2E-9	4.6E-9	3.3E-9
ESE	4.4E-6	8.6E-7	4.0E-7	2.4E-7	1.7E-7	7.9E-8	2.9E-8	1.3E-8	8.2E-9	5.9E-9
SE	4.8E-6	9.2E-7	4.2E-7	2.5E-7	1.7E-7	8.1E-8	3.0E-8	1.3E-8	8.4E-9	6.0E-9
SSE	5.1E-6	9.7E-7	4.4E-7	2.6E-7	1.8E-7	8.5E-8	3.1E-8	1.3E-8	8.5E-9	6.1E-9
S	3.9E-6	7.4E-7	3.4E-7	2.0E-7	1.4E-7	6.5E-8	2.4E-8	1.0E-8	6.7E-9	4.8E-9
SSW	3.2E-6	6.4E-7	3.0E-7	1.8E-7	1.2E-7	5.9E-8	2.2E-8	9.7E-9	6.2E-9	4.5E-9
SW	2.8E-6	5.5E-7	2.6E-7	1.5E-7	1.1E-7	5.2E-8	1.9E-8	8.7E-9	5.5E-9	4.0E-9
WSW	3.2E-6	6.3E-7	2.9E-7	1.7E-7	1.2E-7	5.7E-8	2.1E-8	9.4E-9	6.0E-9	4.3E-9
W	5.5E-6	1.1E-6	5.1E-7	3.0E-7	2.1E-7	1.0E-7	3.7E-8	1.6E-8	1.0E-8	7.6E-9
WNW	6.9E-6	1.3E-6	6.1E-7	3.6E-7	2.5E-7	1.2E-7	4.4E-8	1.9E-8	1.2E-8	8.8E-9
NW	6.1E-6	1.1E-6	5.1E-7	2.9E-7	2.0E-7	9.5E-8	3.4E-8	1.4E-8	9.3E-9	6.6E-9
NNW	3.9E-6	7.1E-7	3.2E-7	1.8E-7	1.2E-7	5.7E-8	2.0E-8	8.7E-9	5.4E-9	3.8E-9
N	2.7E-6	4.8E-7	2.1E-7	1.2E-7	8.5E-8	3.9E-8	1.4E-8	5.9E-9	3.7E-9	2.7E-9

*Includes recirculation factor of 1.25 within 10 miles of site.

Table 2.5-30 - "Long Term Dispersion Factors For 1983 (Sec/M^3)*"
Long-Term Ground Release CHI/Q = (Summation (2.032/X*WS*Z))/Total Valid Records

Z = Sort ((Z*Z)*(D*D/2PI)) Limited To SQRT(3)*Z

Data Used WS10.WD10.DT100

Wind Direction From	0.5 Mile	1.5 Mile	2.5 Mile	3.5 Mile	4.5 Mile	7.5 Mile	15 Mile	25 Mile	35 Mile	45 Mile
NNE	1.9E-6	3.4E-7	1.5E-7	8.8E-8	6.1E-8	2.7E-8	9.8E-9	4.2E-9	2.6E-9	1.8E-9
NE	1.8E-6	3.4E-7	1.6E-7	9.4E-8	6.5E-8	3.0E-8	1.1E-8	4.8E-9	3.0E-9	2.1E-9
ENE	1.6E-6	3.1E-7	1.4E-7	8.7E-8	6.0E-8	2.8E-8	1.1E-8	4.4E-9	2.8E-9	2.0E-9
E	2.2E-6	4.3E-7	2.0E-7	1.2E-7	8.5E-8	3.9E-8	1.5E-8	6.5E-9	4.1E-9	2.9E-9
ESE	3.4E-6	6.6E-7	3.1E-7	1.8E-7	1.3E-7	6.0E-8	2.2E-8	9.8E-9	6.2E-9	4.5E-9
SE	6.2E-6	1.2E-6	5.6E-7	3.3E-7	2.3E-7	1.1E-7	4.0E-8	1.7E-8	1.1E-8	8.0E-9
SSE	4.7E-6	9.0E-7	4.1E-7	2.4E-7	1.7E-7	7.9E-8	2.9E-8	1.3E-8	8.1E-9	5.8E-9
S	4.4E-6	8.3E-7	3.8E-7	2.2E-7	1.5E-7	7.3E-8	2.7E-8	1.1E-8	7.4E-9	5.3E-9
SSW	3.0E-6	5.8E-7	2.7E-7	1.6E-7	1.1E-7	5.2E-8	1.9E-8	8.5E-9	5.4E-9	3.9E-9
SW	2.8E-6	5.7E-7	2.7E-7	1.6E-7	1.1E-7	5.3E-8	1.9E-8	8.8E-9	5.6E-9	4.0E-9
WSW	3.3E-6	6.8E-7	3.2E-7	1.9E-7	1.3E-7	6.4E-8	2.4E-8	1.1E-8	6.9E-9	5.0E-9
W	4.9E-6	1.0E-6	4.7E-7	2.8E-7	1.9E-7	9.3E-8	3.5E-8	1.5E-8	9.9E-9	7.2E-9
WNW	6.7E-6	1.3E-6	6.1E-7	3.6E-7	2.5E-7	1.2E-7	4.4E-8	1-.9E8	1.2E-8	8.9E-9
NW	8.7E-6	1.6E-6	7.5E-7	4.4E-7	3.1E-7	1.4E-7	5.2E-8	2.2E-8	1.4E-8	1.0E-8
NNW	6.4E-6	1.1E-6	5.1E-7	2.9E-7	2.0E-7	9.3E-8	3.3E-8	1.4E-8	8.8E-9	6.2E-9
N	3.1E-6	5.6E-7	2.5E-7	1.4E-7	9.8E-8	4.5E-8	1.6E-8	6.8E-9	4.3E-9	3.0E-9

*Includes recirculation factor of 1.25 within 10 miles of site.

Table 2.5-31 - "Long Term Dispersion Factors For 1984 (Sec/M^3)*"

Long-Term Ground Release CHI/Q = (Summation (2.032/X*WS*Ø))/Total Valid Records

Ø = Sort ((Ø* Ø)*(D*D/2PI)) Limited To SQRT(3)*Ø

Data Used WS 10.WD 10.DT100

Wind Direction From	0.5 Mile	1.5 Mile	2.5 Mile	3.5 Mile	4.5 Mile	7.5 Mile	15 Mile	25 Mile	35 Mile	45 Mile
NNE	1.6E-6	2.9E-7	1.3E-7	7.5E-8	5.2E-8	2.4E-8	8.6E-9	3.7E-9	2.3E-9	1.6E-9
NE	1.4E-6	2.6E-7	1.2E-7	7.1E-8	4.9E-8	2.2E-8	8.0E-9	3.5E-9	2.2E-9	1.6E-9
ENE	1.3E-6	2.4E-7	1.1E-7	6.6E-8	4.6E-8	2.1E-8	7.8E-9	3.4E-9	2.1E-9	1.5E-9
E	1.7E-6	3.2E-7	1.5E-7	8.7E-8	6.0E-8	2.8E-8	1.0E-8	4.4E-9	2.8E-9	1.9E-9
ESE	3.9E-6	7.5E-7	3.5E-7	2.1E-7	1.5E-7	6.8E-8	2.5E-8	1.1E-8	7.1E-9	5.1E-9
SE	7.9E-6	1.5E-6	7.1E-7	4.2E-7	2.9E-7	1.4E-7	5.1E-8	2.2E-8	1.4E-8	1.0E-8
SSE	5.4E-6	1.0E-6	4.8E-7	2.8E-7	1.9E-7	9.3E-8	3.4E-8	1.5E-8	9.7E-9	7.1E-9
S	4.4E-6	8.4E-7	3.9E-7	2.3E-7	1.6E-7	7.6E-8	2.8E-8	1.2E-8	7.8E-9	5.6E-9
SSW	3.9E-6	7.7E-7	3.6E-7	2.2E-7	1.5E-7	7.3E-8	2.7E-8	1.2E-8	7.6E-9	5.5E-9
SW	3.6E-6	6.9E-7	3.2E-7	1.9E-7	1.3E-7	6.4E-8	2.4E-8	1.0E-8	6.7E-9	4.8E-9
WSW	3.6E-6	7.2E-7	3.4E-7	2.1E-7	1.4E-7	6.9E-8	2.6E-8	1.1E-8	7.5E-9	5.4E-9
W	4.7E-6	9.5E-7	4.5E-7	2.7E-7	1.9E-7	9.1E-8	3.4E-8	1.5E-8	9.8E-9	7.1E-9
WNW	7.3E-6	1.4E-6	6.9E-7	4.1E7	2.9E7	1.3E-7	5.1E-8	2.3E-8	1.4E-8	1.0E-8
NW	7.1E-6	1.3E-6	6.2E-7	3.6E-7	2.5E-7	1.2E-7	4.4E-8	1.9E-8	1.2E-8	8.8E-9
NNW	5.3E-6	9.2E-7	4.1E-7	2.3E-7	1.6E-7	7.9E-8	2.5E-8	1.1E-8	6.7E-9	4.7E-9
N	2.8E-6	4.7E-7	2.1E-7	1.2E-7	8.4E-8	3.8E-8	1.3E-8	5.7E-9	3.6E-9	2.5E-9

*Includes recirculation factor of 1.25 within 10 miles of site.

Site: Fort Calhoun 05/20/87 12:31
Period of Record 82010101-84123124
Percent of Total
Extremely Unstable (DT/DZ Less Than -1.9 DEG.C/100 M)
Table 2.5-32 - "Pasquill A Wind Speed (M/S) At 10 M Level"

WIND DIR	.22-.50	.51-.75	.76 1.0	1.1-1.5	1.6-2.0	2.1-3.0	3.1-5.0	5.1-7.0	7.1-10.0	10.1-13.0	13.1-18.0	>18	TOT.
N	0.00	0.00	0.00	0.01	0.02	0.02	0.03	0.01	0.00	0.00	0.00	0.00	0.08
NNE	0.01	0.00	0.00	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.03
NE	0.00	0.00	0.01	0.00	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.02
ENE	0.00	0.01	0.00	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.02
E	0.00	0.00	0.00	0.00	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.02
ESE	0.00	0.00	0.00	0.00	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.01
SE	0.00	0.00	0.00	0.00	0.00	0.01	0.00	0.00	0.01	0.00	0.00	0.00	0.01
SSE	0.00	0.00	0.00	0.00	0.00	0.00	0.01	0.01	0.00	0.00	0.00	0.00	0.01
S	0.00	0.00	0.00	0.00	0.00	0.01	0.00	0.01	0.00	0.00	0.00	0.00	0.01
SSW	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.00
SW	0.01	0.00	0.00	0.00	0.00	0.00	0.01	0.01	0.00	0.00	0.00	0.00	0.01
WSW	0.01	0.01	0.00	0.00	0.01	0.00	0.01	0.01	0.00	0.00	0.00	0.00	0.02
W	0.00	0.00	0.00	0.01	0.01	0.02	0.01	0.01	0.00	0.00	0.00	0.00	0.04
WNW	0.00	0.00	0.01	0.00	0.00	0.03	0.02	0.00	0.00	0.00	0.00	0.00	0.05
NW	0.00	0.00	0.00	0.00	0.01	0.02	0.03	0.01	0.00	0.00	0.00	0.00	0.08
NNW	0.00	0.00	0.00	0.01	0.01	0.04	0.06	0.01	0.00	0.00	0.00	0.00	0.11
TOT.	0.01	0.01	0.01	0.02	0.07	0.17	0.16	0.05	0.01	0.00	0.00	0.00	0.51

NUMBER OF CALMS	0
NUMBER OF INVALID HOURS	1955
NUMBER OF VALID HOURS	125
TOTAL HOURS FOR THE PERIOD	26280

Site: Fort Calhoun 05/20/87 12:31
Period of Record 82010101-84123124
Percent of Total
Moderately Unstable (-1.9 < DT DZ < = -1.7 DEG.C/100 M)
Table 2.5-33 - "Pasquill B Wind Speed (M/S) At 10 M Level"

WIND DIR	.22-.50	.51-.75	.76 1.0	1.1-1.5	1.6-2.0	2.1-3.0	3.1-5.0	5.1-7.0	7.1-10.0	10.1-13.0	13.1-18.0	>18	TOT.
N	0.00	0.00	0.01	0.02	0.04	0.18	0.20	0.01	0.00	0.00	0.00	0.00	0.46
NNE	0.00	0.00	0.00	0.01	0.02	0.04	0.03	0.00	0.00	0.00	0.00	0.00	0.11
NE	0.00	0.00	0.00	0.01	0.05	0.05	0.02	0.01	0.00	0.00	0.00	0.00	0.13
ENE	0.00	0.00	0.00	0.01	0.02	0.03	0.01	0.01	0.00	0.00	0.00	0.00	0.07
E	0.00	0.00	0.00	0.01	0.02	0.05	0.02	0.00	0.00	0.00	0.00	0.00	0.09
ESE	0.00	0.00	0.00	0.00	0.01	0.03	0.06	0.00	0.01	0.00	0.00	0.00	0.11
SE	0.00	0.00	0.00	0.00	0.01	0.04	0.05	0.03	0.01	0.00	0.00	0.00	0.14
SSE	0.00	0.00	0.00	0.01	0.00	0.02	0.05	0.07	0.02	0.01	0.00	0.00	0.16
S	0.00	0.01	0.00	0.01	0.02	0.03	0.07	0.12	0.07	0.01	0.00	0.00	0.32
SSW	0.00	0.00	0.00	0.01	0.01	0.02	0.07	0.06	0.03	0.00	0.00	0.00	0.18
SW	0.00	0.00	0.00	0.00	0.01	0.03	0.06	0.04	0.01	0.00	0.00	0.00	0.15
WSW	0.00	0.00	0.00	0.01	0.02	0.04	0.04	0.01	0.01	0.01	0.00	0.00	0.12
W	0.00	0.00	0.00	0.01	0.02	0.05	0.01	0.02	0.00	0.00	0.00	0.00	0.11
WNW	0.00	0.00	0.00	0.01	0.02	0.06	0.02	0.02	0.03	0.00	0.00	0.00	0.15
NW	0.01	0.00	0.00	0.02	0.04	0.16	0.06	0.07	0.01	0.01	0.00	0.00	0.35
NNW	0.00	0.00	0.00	0.02	0.03	0.11	0.21	0.09	0.01	0.00	0.00	0.00	0.45
TOT.	0.01	0.01	0.01	0.12	0.32	0.93	0.97	0.55	0.18	0.02	0.00	0.00	3.10

NUMBER OF CALMS	0
NUMBER OF INVALID HOURS	1955
NUMBER OF VALID HOURS	755
TOTAL HOURS FOR THE PERIOD	26280

Site: Fort Calhoun 05/20/87 12:31
Period of Record 82010101-84123124
Percent of Total
Slightly Unstable (-1.7 <DT/TZ < = -1.5 DEG.C/100 M)
Table 2.5-34 - "Pasquill C Wind Speed (M/S) At 10 M Level"

WIND DIR	.22-.50	. 51-.75	.76 1.0	1.1-1.5	1.6-2.0	2.1-3.0	3.1-5.0	5.1-7.0	7.1-10.0	10.1-13.0	13.1-18.0	>18	TOT.
N	0.01	0.01	0.01	0.04	0.02	0.09	0.10	0.01	0.00	0.00	0.00	0.00	0.28
NNE	0.00	0.00	0.00	0.01	0.01	0.06	0.03	0.00	0.00	0.00	0.00	0.00	0.11
NE	0.00	0.00	0.01	0.00	0.01	0.02	0.02	0.01	0.00	0.00	0.00	0.00	0.07
ENE	0.00	0.00	0.00	0.01	0.01	0.02	0.00	0.01	0.00	0.00	0.00	0.00	0.05
E	0.00	0.00	0.00	0.01	0.02	0.04	0.00	0.00	0.00	0.00	0.00	0.00	0.07
ESE	0.00	0.00	0.00	0.00	0.01	0.03	0.06	0.01	0.01	0.00	0.00	0.00	0.11
SE	0.00	0.00	0.00	0.00	0.01	0.02	0.06	0.05	0.01	0.00	0.00	0.00	0.14
SSE	0.00	0.00	0.00	0.01	0.01	0.04	0.08	0.05	0.05	0.01	0.00	0.00	0.23
S	0.00	0.00	0.00	0.01	0.02	0.03	0.10	0.14	0.06	0.01	0.00	0.00	0.36
SSW	0.00	0.00	0.01	0.01	0.01	0.01	0.05	0.05	0.01	0.00	0.00	0.00	0.14
SW	0.00	0.00	0.00	0.00	0.00	0.03	0.04	0.02	0.01	0.01	0.00	0.00	0.10
WSW	0.00	0.01	0.00	0.01	0.01	0.02	0.01	0.01	0.00	0.01	0.00	0.00	0.05
W	0.00	0.00	0.01	0.01	0.02	0.03	0.03	0.01	0.00	0.00	0.00	0.00	0.12
WNW	0.00	0.01	0.00	0.01	0.01	0.03	0.05	0.02	0.01	0.00	0.00	0.00	0.12
NW	0.00	0.00	0.00	0.01	0.04	0.06	0.05	0.02	0.00	0.00	0.00	0.00	0.18
NNW	0.00	0.01	0.01	0.00	0.03	0.04	0.14	0.04	0.01	0.00	0.00	0.00	0.27
TOT.	0.01	0.02	0.03	0.09	0.23	0.57	0.83	0.43	0.16	0.02	0.00	0.00	2.40

NUMBER OF CALMS	0
NUMBER OF INVALID HOURS	1955
NUMBER OF VALID HOURS	583
TOTAL HOURS FOR THE PERIOD	26280

Site: Fort Calhoun 05/20/87 12:31
Period of Record 82010101-84123124
Percent of Total
Neutral (-1.5 < DT/DZ < = -0.5 DEG.C/100 M)
Table 2.5-35 - "Pasquill D Wind Speed (M/S) At 10 M Level"

WIND DIR	.22-.50	.51-.75	.76 1.0	1.1-1.5	1.6-2.0	2.1-3.0	3.1-5.0	5.1-7.0	7.1-10.0	10.1-13.0	13.1-18.0	>18	TOT.
N	0.01	0.02	0.06	0.32	0.44	1.08	1.02	0.10	0.02	0.00	0.01	0.00	3.08
NNE	0.02	0.01	0.04	0.21	0.32	0.57	0.53	0.04	0.02	0.01	0.00	0.00	1.76
NE	0.01	0.01	0.07	0.17	0.21	0.46	0.39	0.04	0.01	0.00	0.00	0.00	1.36
ENE	0.00	0.01	0.04	0.15	0.22	0.41	0.49	0.09	0.01	0.01	0.00	0.00	1.41
E	0.00	0.01	0.02	0.08	0.25	0.46	0.65	0.17	0.04	0.01	0.00	0.00	1.68
ESE	0.01	0.01	0.02	0.11	0.18	0.69	1.25	0.33	0.07	0.01	0.00	0.00	2.66
SE	0.01	0.02	0.04	0.12	0.21	0.79	2.07	0.85	0.31	0.03	0.00	0.00	4.45
SSE	0.01	0.01	0.05	0.09	0.20	0.65	2.10	1.60	0.56	0.05	0.00	0.00	5.31
S	0.01	0.02	0.03	0.09	0.12	0.52	1.82	1.60	0.60	0.04	0.00	0.00	4.86
SSW	0.01	0.02	0.02	0.04	0.13	0.36	0.99	0.73	0.32	0.01	0.00	0.00	2.63
SW	0.03	0.02	0.03	0.06	0.09	0.34	0.52	0.21	0.07	0.01	0.00	0.00	1.39
WSW	0.01	0.03	0.02	0.05	0.14	0.33	0.32	0.06	0.03	0.02	0.00	0.00	1.02
W	0.00	0.03	0.05	0.14	0.21	0.45	0.29	0.12	0.02	0.00	0.00	0.00	1.32
WNW	0.01	0.03	0.09	0.21	0.31	0.47	0.69	0.30	0.11	0.07	0.00	0.00	2.29
NW	0.05	0.05	0.12	0.35	0.43	1.25	2.24	0.89	0.23	0.01	0.00	0.00	5.61
NNW	0.05	0.02	0.09	0.37	0.55	1.62	3.57	0.90	0.11	0.01	0.01	0.00	7.31
TOT.	0.24	0.31	0.76	2.56	4.04	10.43	18.94	8.04	2.51	0.25	0.01	0.00	48.12

NUMBER OF CALMS	0
NUMBER OF INVALID HOURS	1955
NUMBER OF VALID HOURS	11706
TOTAL HOURS FOR THE PERIOD	26280

Site: Fort Calhoun 05/20/87 12:31
Period of Record 82010101-84123124
Percent of Total
Slightly Stable (-0.5 < DT/DZ < = -1.5 DEG.C/100 M)
Table 2.5-36 - "Pasquill E Wind Speed (M/S) At 10 M Level"

WIND DIR	.22-.50	.51-.75	.76 1.0	1.1-1.5	1.6-2.0	2.1-3.0	3.1-5.0	5.1-7.0	7.1-10.0	10.1-13.0	13.1-18.0	>18	TOT.
N	0.04	0.06	0.09	0.19	0.19	0.23	0.17	0.02	0.01	0.00	0.00	0.00	1.00
NNE	0.02	0.05	0.06	0.14	0.12	0.08	0.09	0.01	0.01	0.00	0.00	0.00	0.58
NE	0.01	0.03	0.07	0.12	0.07	0.09	0.04	0.01	0.00	0.00	0.00	0.00	0.44
ENE	0.02	0.03	0.08	0.12	0.09	0.09	0.09	0.01	0.00	0.00	0.00	0.00	0.52
E	0.00	0.05	0.07	0.16	0.19	0.23	0.18	0.04	0.02	0.00	0.00	0.00	0.94
ESE	0.02	0.04	0.09	0.27	0.46	0.60	0.45	0.11	0.09	0.01	0.00	0.00	2.12
SE	0.04	0.07	0.11	0.38	0.56	1.47	1.73	0.46	0.06	0.01	0.00	0.00	4.90
SSE	0.03	0.07	0.07	0.09	0.19	0.84	2.14	0.74	0.15	0.05	0.00	0.00	4.38
S	0.03	0.04	0.09	0.12	0.13	0.42	1.60	0.93	0.20	0.01	0.00	0.00	3.57
SSW	0.06	0.07	0.07	0.11	0.17	0.24	0.57	0.47	0.12	0.00	0.00	0.00	1.89
SW	0.06	0.08	0.14	0.07	0.14	0.23	0.34	0.14	0.07	0.00	0.00	0.00	1.28
WSW	0.06	0.12	0.17	0.13	0.09	0.24	0.28	0.04	0.00	0.00	0.00	0.00	1.15
W	0.09	0.15	0.31	0.42	0.23	0.33	0.34	0.04	0.01	0.00	0.00	0.00	1.91
WNW	0.12	0.20	0.42	0.90	0.57	0.52	0.48	0.09	0.02	0.00	0.01	0.00	3.34
NW	0.06	0.14	0.30	0.71	0.85	0.83	0.78	0.08	0.06	0.01	0.00	0.00	3.82
NNW	0.04	0.06	0.18	0.30	0.42	0.69	0.55	0.12	0.10	0.01	0.00	0.00	2.49
TOT.	0.69	1.26	2.33	4.23	4.48	7.15	9.83	3.28	0.92	0.09	0.01	0.00	34.32

NUMBER OF CALMS 0
NUMBER OF INVALID HOURS 1955
NUMBER OF VALID HOURS 8349
TOTAL HOURS FOR THE PERIOD 26280

Site: Fort Calhoun 05/20/87 12:31
Period of Record 82010101-84123124
Percent of Total
Moderately Stable (1.5 < = DT/DZ < = -0.5 DEG.C/100 M)
Table 2.5-37 - "Pasquill F Wind Speed (M/S) At 10 M Level"

WIND DIR	.22-.50	.51-.75	.76 1.0	1.1-1.5	1.6-2.0	2.1-3.0	3.1-5.0	5.1-7.0	7.1-10.0	10.1-13.0	13.1-18.0	>18	TOT.
N	0.02	0.02	0.02	0.02	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.08
NNE	0.01	0.01	0.03	0.02	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.09
NE	0.02	0.03	0.03	0.04	0.01	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.14
ENE	0.01	0.01	0.03	0.06	0.01	0.01	0.02	0.00	0.00	0.00	0.00	0.00	0.14
E	0.02	0.03	0.03	0.09	0.05	0.03	0.01	0.00	0.00	0.00	0.00	0.00	0.26
ESE	0.01	0.05	0.08	0.21	0.12	0.18	0.06	0.01	0.00	0.00	0.00	0.00	0.73
SE	0.02	0.07	0.14	0.22	0.19	0.42	0.17	0.01	0.01	0.01	0.00	0.00	1.24
SSE	0.02	0.11	0.08	0.07	0.07	0.18	0.14	0.00	0.00	0.00	0.00	0.00	0.67
S	0.02	0.06	0.09	0.08	0.07	0.18	0.33	0.05	0.00	0.00	0.00	0.00	0.88
SSW	0.03	0.08	0.08	0.07	0.04	0.10	0.33	0.14	0.04	0.00	0.00	0.00	0.90
SW	0.06	0.09	0.08	0.05	0.02	0.03	0.16	0.09	0.02	0.00	0.00	0.00	0.61
WSW	0.07	0.15	0.09	0.07	0.07	0.07	0.14	0.03	0.01	0.00	0.00	0.00	0.70
W	0.08	0.18	0.24	0.23	0.06	0.05	0.03	0.02	0.01	0.00	0.00	0.00	0.89
WNW	0.05	0.12	0.25	0.43	0.21	0.06	0.03	0.02	0.01	0.00	0.00	0.00	1.18
NW	0.07	0.05	0.22	0.22	0.10	0.04	0.02	0.00	0.00	0.00	0.00	0.00	0.73
NNW	0.01	0.02	0.06	0.05	0.02	0.01	0.00	0.01	0.00	0.00	0.00	0.00	0.17
TOT.	0.50	1.07	1.56	1.94	1.05	1.38	1.46	0.36	0.07	0.01	0.00	0.00	9.42

NUMBER OF CALMS	0
NUMBER OF INVALID HOURS	1955
NUMBER OF VALID HOURS	2291
TOTAL HOURS FOR THE PERIOD	26280

Site: Fort Calhoun 05/20/87 12:31
Period of Record 82010101-84123124
Percent of Total
Extremely Stable (DT/DZ Exceeds 4.0 DEG.C/100 M)
Table 2.5-38 - "Pasquill G Wind Speed (M/S) At 10 M Level"

WIND DIR	.22-.50	.51-.75	.76 1.0	1.1-1.5	1.6-2.0	2.1-3.0	3.1-5.0	5.1-7.0	7.1-10.0	10.1-13.0	13.1-18.0	>18	TOT.
N	0.01	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.01	0.00	0.00	0.04
NNE	0.00	0.01	0.01	0.02	0.00	0.00	0.00	0.00	0.00	0.01	0.00	0.00	0.05
NE	0.01	0.01	0.03	0.01	0.02	0.01	0.00	0.01	0.00	0.00	0.00	0.00	0.07
ENE	0.00	0.01	0.04	0.03	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.08
E	0.00	0.02	0.05	0.05	0.01	0.01	0.00	0.00	0.00	0.000	0.00	0.00	0.13
ESE	0.01	0.02	0.04	0.11	0.02	0.06	0.00	0.00	0.00	0.00	0.00	0.00	0.25
SE	0.01	0.02	0.05	0.07	0.07	0.02	0.02	0.00	0.00	0.00	0.00	0.00	0.26
SSE	0.02	0.02	0.05	0.07	0.03	0.02	0.00	0.00	0.00	0.00	0.00	0.00	0.21
S	0.02	0.02	0.04	0.01	0.01	0.02	0.04	0.00	0.00	0.00	0.00	0.00	0.14
SSW	0.02	0.03	0.02	0.01	0.01	0.01	0.06	0.04	0.01	0.00	0.00	0.00	0.19
SW	0.02	0.03	0.02	0.02	0.02	0.02	0.02	0.02	0.04	0.00	0.00	0.00	0.21
WSW	0.02	0.01	0.04	0.02	0.00	0.00	0.01	0.01	0.00	0.00	0.00	0.00	0.11
W	0.03	0.04	0.02	0.02	0.01	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.12
WNW	0.01	0.03	0.02	0.02	0.01	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.11
NW	0.01	0.02	0.04	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.08
NNW	0.01	0.01	0.01	0.01	0.01	0.00	0.00	0.01	0.01	0.01	0.00	0.00	0.06
TOT.	0.17	0.29	0.49	0.50	0.22	0.15	0.16	0.09	0.05	0.01	0.00	0.00	2.12

NUMBER OF CALMS	0
NUMBER OF INVALID HOURS	1955
NUMBER OF VALID HOURS	516
TOTAL HOURS FOR THE PERIOD	26280

Site: Fort Calhoun 05/20/87 12:31
Period of Record 82010101-84123124
Percent of Total
Table 2.5-39 - "All Stability, All DT/DZ Wind Speed (M/S) At 10 M Level"

WIND DIR	.22-.50	.51-.75	.76 1.0	1.1-1.5	1.6-2.0	2.1-3.0	3.1-5.0	5.1-7.0	7.1-10.0	10.1-13.0	13.1-18.0	>18	TOT.
N	0.08	0.11	0.20	0.61	0.72	1.60	1.53	0.14	0.02	0.01	0.01	0.00	5.02
NNE	0.05	0.08	0.14	0.41	0.50	0.77	0.69	0.05	0.03	0.01	0.00	0.00	2.72
NE	0.05	0.07	0.21	0.35	0.37	0.65	0.48	0.08	0.01	0.00	0.00	0.00	2.24
ENE	0.02	0.06	0.18	0.37	0.36	0.56	0.61	0.11	0.01	0.01	0.00	0.00	2.28
E	0.02	0.09	0.17	0.41	0.54	0.82	0.87	0.21	0.06	0.01	0.00	0.00	3.19
ESE	0.04	0.11	0.23	0.69	0.81	1.60	1.88	0.45	0.17	0.01	0.00	0.00	5.99
SE	0.07	0.18	0.33	0.80	1.05	2.76	4.10	1.39	0.41	0.05	0.00	0.00	11.14
SSE	0.08	0.21	0.23	0.33	0.50	1.75	4.51	2.47	0.77	0.11	0.00	0.00	10.97
S	0.08	0.14	0.24	0.30	0.37	1.20	3.95	2.84	0.93	0.06	0.00	0.00	10.14
SSW	0.12	0.20	0.20	0.24	0.36	0.73	2.06	1.48	0.52	0.01	0.00	0.00	5.93
SW	0.17	0.23	0.27	0.21	0.28	0.68	1.15	0.53	0.21	0.01	0.00	0.00	3.75
WSW	0.16	0.32	0.33	0.29	0.34	0.70	0.81	0.14	0.04	0.03	0.00	0.00	3.18
W	0.20	0.40	0.62	0.84	0.56	0.92	0.72	0.21	0.03	0.00	0.00	0.00	4.51
WNW	0.18	0.39	0.79	1.57	1.13	1.16	1.28	0.45	0.18	0.07	0.01	0.00	7.24
NW	0.18	0.25	0.68	1.31	1.47	2.37	3.18	1.07	0.30	0.02	0.00	0.00	10.85
NNW	0.11	0.11	0.35	0.75	1.06	2.51	4.53	1.18	0.22	0.02	0.01	0.00	10.86
TOT.	1.62	2.96	5.18	9.48	10.42	20.78	32.35	12.81	3.89	0.41	0.02	0.00	100.00

NUMBER OF CALMS	0
NUMBER OF INVALID HOURS	1955
NUMBER OF VALID HOURS	24325
TOTAL HOURS FOR THE PERIOD	26280

Table 2.5-40 - "Results of Mesodif-II* Trajectory Analysis Using Fort Calhoun Meteorological Tower Data for 1982 Through 1984"

Direction Toward	Ratio of Recirculation X/Z to Straightline X/Q Values at .75 Miles		
	Ground Level Release		
	1982	1983	1984
N	1.20	1.04	1.05
NNE	1.23	1.11	1.08
NE	1.23	1.13	1.07
ENE	1.14	1.13	1.08
E	1.05	1.07	1.07
ESE	1.07	1.09	1.06
SE	1.03	1.08	1.17
SSE	1.05	1.08	1.08
S	1.08	1.06	1.13
SSW	1.11	1.03	1.07
SW	1.10	1.06	1.04
WSW	1.12	1.05	1.10
W	1.07	1.08	1.23
WNW	1.06	1.13	1.08
NW	1.05	1.05	1.05
NNW	1.01	1.02	1.07

*MESODIF-II is a plume trajectory computer program used in conjunction with analyses to conform with NRC Regulatory Guide 1.111; ref: ORNL RSIC Computer Code Collection, "MESODIF-II - A Variable Trajectory Plume Segment Model to Assess Ground-Level Air Concentrations and Deposition of Routine Effluent Releases from Nuclear Power Facilities," CCC-498.

Site: Fort Calhoun 05/20/87 12:31
Period of Record 82010101-84123124
Percent of Total
Neutral (-1.5 < DT/DZ < = -0.5 DEG.C/100 M)
Table 2.5-41 - "Pasquill D Wind Speed (M/S) At 10 M Level"

WIND DIR	.22-.50	.51-.75	.76 1.0	1.1-1.5	1.6-2.0	2.1-3.0	3.1-5.0	5.1-7.0	7.1-10.0	10.1-13.0	13.1-18.0	>18	TOT.
N	0.01	0.02	0.06	0.32	0.44	1.08	1.02	0.10	0.02	0.00	0.01	0.00	3.08
NNE	0.02	0.01	0.04	0.21	0.32	0.57	0.53	0.04	0.02	0.01	0.00	0.00	1.76
NE	0.01	0.01	0.07	0.17	0.21	0.46	0.39	0.04	0.01	0.00	0.00	0.00	1.36
ENE	0.00	0.01	0.04	0.15	0.22	0.41	0.49	0.09	0.01	0.01	0.00	0.00	1.41
E	0.00	0.01	0.02	0.08	0.25	0.46	0.65	0.17	0.04	0.01	0.00	0.00	1.68
ESE	0.01	0.01	0.02	0.11	0.18	0.69	1.25	0.33	0.07	0.01	0.00	0.00	2.66
SE	0.01	0.02	0.04	0.12	0.21	0.79	2.07	0.85	0.31	0.03	0.00	0.00	4.45
SSE	0.01	0.01	0.05	0.09	0.20	0.65	2.10	1.60	0.56	0.05	0.00	0.00	5.31
S	0.01	0.02	0.03	0.09	0.12	0.52	1.82	1.60	0.60	0.04	0.00	0.00	4.86
SSW	0.01	0.02	0.02	0.04	0.13	0.36	0.99	0.73	0.32	0.01	0.00	0.00	2.63
SW	0.03	0.02	0.03	0.06	0.09	0.34	0.52	0.21	0.07	0.01	0.00	0.00	1.39
WSW	0.01	0.03	0.02	0.05	0.14	0.33	0.32	0.06	0.03	0.02	0.00	0.00	1.02
W	0.00	0.03	0.05	0.14	0.21	0.45	0.29	0.12	0.02	0.00	0.00	0.00	1.32
WNW	0.01	0.03	0.09	0.21	0.31	0.47	0.69	0.30	0.11	0.07	0.00	0.00	2.29
NW	0.05	0.05	0.12	0.35	0.43	1.25	2.24	0.89	0.23	0.01	0.00	0.00	5.61
NNW	0.05	0.02	0.09	0.37	0.55	1.62	3.57	0.90	0.11	0.01	0.01	0.00	7.31
TOT.	0.24	0.31	0.76	2.56	4.04	10.43	18.94	8.04	2.51	0.25	0.01	0.00	48.12

NUMBER OF CALMS	0
NUMBER OF INVALID HOURS	1955
NUMBER OF VALID HOURS	11706
TOTAL HOURS FOR THE PERIOD	26280

Site: Fort Calhoun 05/20/87 12:31
Period of Record 82010101-84123124
Percent of Total
Slightly Stable (-0.5 < DT/DZ < = -1.5 DEG.C/100 M)
Table 2.5-42 - "Pasquill E Wind Speed (M/S) At 10 M Level"

WIND DIR	.22-.50	.51-.75	.76 1.0	1.1-1.5	1.6-2.0	2.1-3.0	3.1-5.0	5.1-7.0	7.1-10.0	10.1-13.0	13.1-18.0	>18	TOT.
N	0.04	0.06	0.09	0.19	0.19	0.23	0.17	0.02	0.01	0.00	0.00	0.00	1.00
NNE	0.02	0.05	0.06	0.14	0.12	0.08	0.09	0.01	0.01	0.00	0.00	0.00	0.58
NE	0.01	0.03	0.07	0.12	0.07	0.09	0.04	0.01	0.00	0.00	0.00	0.00	0.44
ENE	0.02	0.03	0.08	0.12	0.09	0.09	0.09	0.01	0.00	0.00	0.00	0.00	0.52
E	0.00	0.05	0.07	0.16	0.19	0.23	0.18	0.04	0.02	0.00	0.00	0.00	0.94
ESE	0.02	0.04	0.09	0.27	0.46	0.60	0.45	0.11	0.09	0.01	0.00	0.00	2.12
SE	0.04	0.07	0.11	0.38	0.56	1.47	1.73	0.46	0.06	0.01	0.00	0.00	4.90
SSE	0.03	0.07	0.07	0.09	0.19	0.84	2.14	0.74	0.15	0.05	0.00	0.00	4.38
S	0.03	0.04	0.09	0.12	0.13	0.42	1.60	0.93	0.20	0.01	0.00	0.00	3.57
SSW	0.06	0.07	0.07	0.11	0.17	0.24	0.57	0.47	0.12	0.00	0.00	0.00	1.89
SW	0.06	0.08	0.14	0.07	0.14	0.23	0.34	0.14	0.07	0.00	0.00	0.00	1.28
WSW	0.06	0.12	0.17	0.13	0.09	0.24	0.28	0.04	0.00	0.00	0.00	0.00	1.15
W	0.09	0.15	0.31	0.42	0.23	0.33	0.34	0.04	0.01	0.00	0.00	0.00	1.91
WNW	0.12	0.20	0.42	0.90	0.57	0.52	0.48	0.09	0.02	0.00	0.01	0.00	3.34
NW	0.06	0.14	0.30	0.71	0.85	0.83	0.78	0.08	0.06	0.01	0.00	0.00	3.82
NNW	0.04	0.06	0.18	0.30	0.42	0.69	0.55	0.12	0.10	0.01	0.00	0.00	2.49
TOT.	0.69	1.26	2.33	4.23	4.48	7.15	9.83	3.28	0.92	0.09	0.01	0.00	34.32

NUMBER OF CALMS	0
NUMBER OF INVALID HOURS	1955
NUMBER OF VALID HOURS	8349
TOTAL HOURS FOR THE PERIOD	26280

Site: Fort Calhoun 05/20/87 12:31
Period of Record 82010101-84123124
Percent of Total
Moderately Stable (1.5 < = DT/DZ < = -0.5 DEG.C/100 M)
Table 2.5-43 - "Pasquill F Wind Speed (M/S) At 10 M Level"

WIND DIR	.22-.50	.51-.75	.76 1.0	1.1-1.5	1.6-2.0	2.1-3.0	3.1-5.0	5.1-7.0	7.1-10.0	10.1-13.0	13.1-18.0	>18	TOT.
N	0.02	0.02	0.02	0.02	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.08
NNE	0.01	0.01	0.03	0.02	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.09
NE	0.02	0.03	0.03	0.04	0.01	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.14
ENE	0.01	0.01	0.03	0.06	0.01	0.01	0.02	0.00	0.00	0.00	0.00	0.00	0.14
E	0.02	0.03	0.03	0.09	0.05	0.03	0.01	0.00	0.00	0.00	0.00	0.00	0.26
ESE	0.01	0.05	0.08	0.21	0.12	0.18	0.06	0.01	0.00	0.00	0.00	0.00	0.73
SE	0.02	0.07	0.14	0.22	0.19	0.42	0.17	0.01	0.01	0.01	0.00	0.00	1.24
SSE	0.02	0.11	0.08	0.07	0.07	0.18	0.14	0.00	0.00	0.00	0.00	0.00	0.67
S	0.02	0.06	0.09	0.08	0.07	0.18	0.33	0.05	0.00	0.00	0.00	0.00	0.88
SSW	0.03	0.08	0.08	0.07	0.04	0.10	0.33	0.14	0.04	0.00	0.00	0.00	0.90
SW	0.06	0.09	0.08	0.05	0.02	0.03	0.16	0.09	0.02	0.00	0.00	0.00	0.61
WSW	0.07	0.15	0.09	0.07	0.07	0.07	0.14	0.03	0.01	0.00	0.00	0.00	0.70
W	0.08	0.18	0.24	0.23	0.06	0.05	0.03	0.02	0.01	0.00	0.00	0.00	0.89
WNW	0.05	0.12	0.25	0.43	0.21	0.06	0.03	0.02	0.01	0.00	0.00	0.00	1.18
NW	0.07	0.05	0.22	0.22	0.10	0.04	0.02	0.00	0.00	0.00	0.00	0.00	0.73
NNW	0.01	0.02	0.06	0.05	0.02	0.01	0.00	0.01	0.00	0.00	0.00	0.00	0.17
TOT.	0.50	1.07	1.56	1.94	1.05	1.38	1.46	0.36	0.07	0.01	0.00	0.00	9.42

NUMBER OF CALMS	0
NUMBER OF INVALID HOURS	1955
NUMBER OF VALID HOURS	2291
TOTAL HOURS FOR THE PERIOD	26280

Site: Fort Calhoun 05/20/87 12:31
Period of Record 82010101-84123124
Percent of Total
Extremely Stable (DT/DZ Exceeds 4.0 DEG.C/100 M)
Table 2.5-44 - "Pasquill G Wind Speed (M/S) At 10 M Level"

WIND DIR	.22-.50	.51-.75	.76 1.0	1.1-1.5	1.6-2.0	2.1-3.0	3.1-5.0	5.1-7.0	7.1-10.0	10.1-13.0	13.1-18.0	>18	TOT.
N	0.01	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.01	0.00	0.00	0.04
NNE	0.00	0.01	0.01	0.02	0.00	0.00	0.00	0.00	0.00	0.01	0.00	0.00	0.05
NE	0.01	0.01	0.03	0.01	0.02	0.01	0.00	0.01	0.00	0.00	0.00	0.00	0.07
ENE	0.00	0.01	0.04	0.03	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.08
E	0.00	0.02	0.05	0.05	0.01	0.01	0.00	0.00	0.00	0.000	0.00	0.00	0.13
ESE	0.01	0.02	0.04	0.11	0.02	0.06	0.00	0.00	0.00	0.00	0.00	0.00	0.25
SE	0.01	0.02	0.05	0.07	0.07	0.02	0.02	0.00	0.00	0.00	0.00	0.00	0.26
SSE	0.02	0.02	0.05	0.07	0.03	0.02	0.00	0.00	0.00	0.00	0.00	0.00	0.21
S	0.02	0.02	0.04	0.01	0.01	0.02	0.04	0.00	0.00	0.00	0.00	0.00	0.14
SSW	0.02	0.03	0.02	0.01	0.01	0.01	0.06	0.04	0.01	0.00	0.00	0.00	0.19
SW	0.02	0.03	0.02	0.02	0.02	0.02	0.02	0.02	0.04	0.00	0.00	0.00	0.21
WSW	0.02	0.01	0.04	0.02	0.00	0.00	0.01	0.01	0.00	0.00	0.00	0.00	0.11
W	0.03	0.04	0.02	0.02	0.01	0.00	0.00	0.01	0.00	0.00	0.00	0.00	0.12
WNW	0.01	0.03	0.02	0.02	0.01	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.11
NW	0.01	0.02	0.04	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.08
NNW	0.01	0.01	0.01	0.01	0.01	0.00	0.00	0.01	0.01	0.01	0.00	0.00	0.06
TOT.	0.17	0.29	0.49	0.50	0.22	0.15	0.16	0.09	0.05	0.01	0.00	0.00	2.12

NUMBER OF CALMS 0
NUMBER OF INVALID HOURS 1955
NUMBER OF VALID HOURS 516
TOTAL HOURS FOR THE PERIOD 26280

Site: Fort Calhoun 05/20/87 12:31
Period of Record 82010101-84123124
Percent of Total
Table 2.5-45 - "All Stability, All DT/DZ Wind Speed (M/S) At 10 M Level"

WIND DIR	.22-.50	.51-.75	.76 1.0	1.1-1.5	1.6-2.0	2.1-3.0	3.1-5.0	5.1-7.0	7.1-10.0	10.1-13.0	13.1-18.0	>18	TOT.
N	0.08	0.11	0.20	0.61	0.72	1.60	1.53	0.14	0.02	0.01	0.01	0.00	5.02
NNE	0.05	0.08	0.14	0.41	0.50	0.77	0.69	0.05	0.03	0.01	0.00	0.00	2.72
NE	0.05	0.07	0.21	0.35	0.37	0.65	0.48	0.08	0.01	0.00	0.00	0.00	2.24
ENE	0.02	0.06	0.18	0.37	0.36	0.56	0.61	0.11	0.01	0.01	0.00	0.00	2.28
E	0.02	0.09	0.17	0.41	0.54	0.82	0.87	0.21	0.06	0.01	0.00	0.00	3.19
ESE	0.04	0.11	0.23	0.69	0.81	1.60	1.88	0.45	0.17	0.01	0.00	0.00	5.99
SE	0.07	0.18	0.33	0.80	1.05	2.76	4.10	1.39	0.41	0.05	0.00	0.00	11.14
SSE	0.08	0.21	0.23	0.33	0.50	1.75	4.51	2.47	0.77	0.11	0.00	0.00	10.97
S	0.08	0.14	0.24	0.30	0.37	1.20	3.95	2.84	0.93	0.06	0.00	0.00	10.14
SSW	0.12	0.20	0.20	0.24	0.36	0.73	2.06	1.48	0.52	0.01	0.00	0.00	5.93
SW	0.17	0.23	0.27	0.21	0.28	0.68	1.15	0.53	0.21	0.01	0.00	0.00	3.75
WSW	0.16	0.32	0.33	0.29	0.34	0.70	0.81	0.14	0.04	0.03	0.00	0.00	3.18
W	0.20	0.40	0.62	0.84	0.56	0.92	0.72	0.21	0.03	0.00	0.00	0.00	4.51
WNW	0.18	0.39	0.79	1.57	1.13	1.16	1.28	0.45	0.18	0.07	0.01	0.00	7.24
NW	0.18	0.25	0.68	1.31	1.47	2.37	3.18	1.07	0.30	0.02	0.00	0.00	10.85
NNW	0.11	0.11	0.35	0.75	1.06	2.51	4.53	1.18	0.22	0.02	0.01	0.00	10.86
TOT.	1.62	2.96	5.18	9.48	10.42	20.78	32.35	12.81	3.89	0.41	0.02	0.00	100.00

NUMBER OF CALMS	0
NUMBER OF INVALID HOURS	1955
NUMBER OF VALID HOURS	24325
TOTAL HOURS FOR THE PERIOD	26280

Table 2.5-46 - "Results of Mesodif-II* Trajectory Analysis Using Fort Calhoun Meteorological Tower Data for 1982 Through 1984"

Direction Toward	Ratio of Recirculation X/Q to Straightline X/Q Values at .75 Miles		
	Ground Level Release		
	1982	1983	1984
N	1.20	1.04	1.05
NNE	1.23	1.11	1.08
NE	1.23	1.13	1.07
ENE	1.14	1.13	1.08
E	1.05	1.07	1.07
ESE	1.07	1.09	1.06
SE	1.03	1.08	1.17
SSE	1.05	1.08	1.08
S	1.08	1.06	1.13
SSW	1.11	1.03	1.07
SW	1.10	1.06	1.04
WSW	1.12	1.05	1.10
W	1.07	1.08	1.23
WNW	1.06	1.13	1.08
NW	1.05	1.05	1.05
NNW	1.01	1.02	1.07

*MESODIF-II is a plume trajectory computer program used in conjunction with analyses to conform with NRC Regulatory Guide 1.111; ref: ORNL RSIC Computer Code Collection, "MESODIF-II - A Variable Trajectory Plume Segment Model to Assess Ground-Level Air Concentrations and Deposition of Routine Effluent Releases from Nuclear Power Facilities," CCC-498.

2.8 DEMOGRAPHY

The plant site is located on the alluvial plain of the Missouri River in a predominantly agricultural region roughly ten miles north of the Omaha metropolitan area. The distribution of population around the plant site as of 1990 is shown in Table 2.8-1 and Figures 2.8-1 and 2.8-2.

There are no residences within one-half mile of the reactor location. The seven nearest residences are from 3,000 to 4,000 feet distant. These are located generally along Highway 75, the western boundary of the site. There are no schools, hospitals, prisons, or motels/hotels in the immediate vicinity of the site. An industrial Park is located north of the plant property. Industries include a large corn processing facility, agricultural fertilizer storage facilities and various other light industrial plants.

The DeSoto National Wildlife Refuge occupies approximately 7,821 acres east of the plant site. This area is open to the public for day use year round. Visitors to the refuge generally use areas from two to five miles from the plant. Estimates by the U.S. Fish and Wildlife Service place annual usage of the facility at approximately 120,000 for the Visitors Center and 400,000 for the refuge. The expected maximum daily usage of the facility has been placed at 2500 visitors for a Winter weekday and 5000 on a Summer weekend. The Boyer Chute Federal Recreation Area is a day use facility occupying approximately 2000 acres southeast of the plant site. Visitors to the recreation area generally use areas seven to ten miles from the plant. The estimates for annual usage of this facility is approximately 50,000 visitors.

The State of Nebraska operates the Fort Atkinson State Historic Park five and half miles southeast of the plant site. This day use facility is mostly seasonal and estimates place annual usage at 60,000. The State of Iowa maintains Wilson Island State Park with 275 camping spaces south of the DeSoto National Wildlife Refuge and four miles southeast of the plant site. The estimates for usage of this facility range from 500 on winter weekday to 1000 on a summer weekend.

Two private facilities lie to the north of the plant along the Missouri River. The Cottonwood Marina is located approximately four and a half miles from the plant. Estimates place summer weekend usage at 200 people. Timbers at Rivers Edge is a private campground lying directly to the south of Cottonwood Marina and ranging from four to four and a half miles from the plant. The campground has approximately 235 campsites and is open from April to October.

The nearest municipality is the city of Blair, about three miles northwest, with a population of 6,860 per the 1990 census. Dana College in Blair has an average enrollment of about 600 students. Unofficial estimates of growth place the 2000 population of Blair at about 7,307.

Fort Calhoun is about five miles southeast of the plant site. The 1990 census reported a population of 648 in Fort Calhoun and 371 in Kennard Village, about seven miles from the plant site. The 1990 population of Fort Calhoun Township, including the above centers, was 5510. Situated just to the north of the Omaha metropolitan area, unofficial estimates of growth place the 2000 population of Fort Calhoun Township around 7500.

Missouri Valley, Iowa, about 11 miles east, has a 1990 population of 2,888 as compared to the 1980 population of 3,107. In St. Johns Township, of which the city is a part, population has steadily decreased.

The Omaha metropolitan area includes the cities of Omaha and Council Bluffs, Iowa, and the adjoining areas of Douglas, Washington, and Sarpy Counties, Nebraska, and Pottawattamie County, Iowa. The area lies 10 to 25 miles southeast of the site, with the main concentration of population beyond the 15-mile radius. Population studies have been undertaken by the Metropolitan Area Planning Agency. Population information is as follows:

	Omaha City	Metropolitan Area
1960 U.S. Census	301,598	457,873
1970 U.S. Census	346,929	542,646
1980 U.S. Census	313,911	569,614
1990 U.S. Census	335,795	618,262
2000 Estimate	359,204	671,065

Table 2.8-1 - "Population Distribution as of 1990"

SECTOR	DISTANCE FROM REACTOR IN MILES/SQUARE MILES OF SECTOR SEGMENT										TOTALS
	0-1	1-2	2-3	3-4	4-5	5-10	10-20/58	20-30/93	30-40/130	40-50/170	
A	0	0	0	0	0	36	710	948	3,720	1,871	7,285
B	0	0	13	4	4	421	499	1,067	1,578	3,351	6,937
C	0	0	0	0	135	162	703	2,299	2,369	1,151	6,819
D	0	0	0	0	8	2,381	5092	1,009	1,903	7,748	18,141
E	0	0	0	0	0	116	696	2,820	2,615	2,319	8,566
F	0	0	0	0	5	88	893	1,795	3,699	2,307	8,787
G	0	28	9	114	72	745	5,432	12,296	3,151	3,012	24,759
H	21	33	22	90	215	1,278	107,123	141,221	13,637	3,801	277,441
J	0	34	1	41	73	534	163,724	82,464	6,740	4,745	258,356
K	0	12	25	27	83	751	18,269	23,318	4,452	6,335	53,261
L	3	19	54	14	20	223	2,996	4,534	5,111	3,176	16,150
M	0	2	0	322	72	512	2,263	22,334	1,672	1,984	29,161
N	0	33	41	160	40	181	1,184	4,030	2,731	1,693	10,093
P	0	12	417	3,971	1,522	201	1,085	2,060	5,484	2,838	17,600
Q	0	0	195	661	57	163	1,271	740	3,437	2,492	9,016
R	0	0	30	36	13	70	1,085	2,364	1,356	3,087	8,041
TOTALS	24	173	807	5,330	2,318	7,862	313,025	315,299	63,665	51,910	760,413

* Based on 1990 U.S. Census Data
** Rerp Section J may contain more current population figures
Sectors are assigned for each 22.5° segment starting from 11.25° east of north.

The increase from 1990 to 2000 predicted for the city is approximately 9.6 percent and 9.2 percent for the metropolitan area. Included in this area is the city of Council Bluffs, with a population of 54,315 (per 1990 census) and the city of Bellevue, with a population of 30,982 (per 1990 census).

Offutt Air Force Base, about 30 miles southeast of the plant site, has a population of 10,883 and no estimate of the future trend is available. The city of Fremont, which had a population of 23,680 in 1990, is 20 miles west of the plant site. Plattsmouth, about 37 miles south of the plant site, has a 1990 population of 6,412.

The U.S. Census data shows an increase in population in the Omaha metropolitan area and in most of the nearby cities but a decrease in the rural and farm population. While it is probable that the area around the plant site outside of the Omaha metropolitan area will remain largely agricultural and that the population will increase slowly, a general decline of the rural population will continue, reflecting the movement of people into towns and cities. The expansion of the Omaha metropolitan area has been generally south and westward, coinciding with the interstate highway. It is expected that future growth of the metropolitan area will continue south and west and also northwestward. Thus it is probable that the area surrounding the plant site will continue to remain largely agricultural.

A conservative population estimate for the 50-mile radius around the plant site has also been calculated for the year 2010. While certain areas may show more or less growth than projected, it is quite probable that the overall 50-mile radius population in 2010 may exceed 1 million people. The projected population distribution for the 50-mile radius around the site is shown in Figures 2.8-3 and 2.8-4.

2.10 ENVIRONMENTAL RADIATION MONITORING

2.10.1 General

The environmental monitoring program is designed to provide data concerning the types and amount of radioactivity present in the environment of the Fort Calhoun Station. The preoperational program was designed to assess environmental conditions before the arrival of fuel. Subsequent analysis during the operational program is being used to demonstrate that plant operations do not have a significant effect on the environment.

2.10.2 Preoperational Survey Program

The purpose of the preoperational survey program was to determine the base level of existing radioactivity to which future analytical results can be compared; the program extended for four consecutive years. The monitoring program was developed in cooperation with the regulatory agencies of Nebraska and Iowa and the Fish and Wildlife Service of the United States Government Department of the Interior.

Specific radionuclide and/or gross radioactivity analyses were performed on the selected samples. Table 2.10-1 summarizes the types of samples and analyses included in the preoperational program.

Table 2.10-1 - "Gross and Specific Radionuclide Analyses"

	Gross α	Gross β-γ	γ-Spec	Sr-90	H-3	K-40	I-131	Cs-137
Surface Water	X	X	X	X	X			
Well Water	X	X	X	X	X			
Mud and Silt	X	X	X					
Aquatic Biota		X	X	X		X		
Milk		X	X	X		X	X	X
Vegetation	X	X	X	X	X	X		
Air Particulate	X	X	X					
Wildlife				X			X	

2.10.3 Preoperation Survey Results

2.10.3.1 Trial Monitoring Period

The first nine months of the program, starting in September 1968, was a trial period designed to verify the availability of adequate sample types and to select and test analytical procedures.

Results obtained during the trial period were preliminary. The trial period results are included in this report because they describe the background conditions and illustrated the preoperational surveillance program. No significant peaks were evident in any of the gamma scans performed on samples.

Water

Surface water samples were collected at six stations: one at the Desoto National Wildlife Refuge Lake area and five from the Missouri River at sampling stations located above and below the plant site, including the municipal water supplies at Omaha, Nebraska, and Council Bluffs, Iowa.

Well waters were sampled at eleven wells within a four-mile radius of the plant. Table 2.10-2 is a summary of the surface and well water data.

Table 2.10-2 - "Average Radioactivity of Well and Surface Waters November 1968 - June 1969"

	Activity Concentration, pc/liter	
	Well Water (11 Samples)	Surface Water (6 Samples)
Alpha	0.0	0.7
Beta-Gamma	10.9	26.2
Strontium 90	0.1	1.3
Tritium	550	1000

Mud and Silt

Mud and silt samples were taken from the Missouri River downstream of the plant. No alpha radiation was detectable; the analysis for beta-gamma gross activity showed 18 picocuries per kilogram for the mud and silt.

Aquatic Biota

The basis for sampling aquatic biota was formulated from specific recommendations of the Nebraska Game, Forestation and Parks Commission. The fish species selected were chosen because their food habits include organisms within many of the lower trophic levels and because they are important from the standpoint of sport and commercial fishing.

The food habits and radioactivity of the fish samples, which were taken from the Missouri River, are shown in Table 2.10-3.

Table 2.10-3 - "Food Habits and Radioactivity of Missouri River Fish
October 1968 - June 1969"

Specie	Food Habits	(β-γ)-(K-40) nc/kg	K-40 nc/kg	Sr-90 pc/kg
Flathead Catfish #	Fish	3.2	2.6	0.0
Flathead Catfish *	Insects	7.8	10.6	0.0
Channel Catfish #	Fish	3.2	6.7	100.00
Channel Catfish *	Insects	1.6	6.5	0.0
Carp	Omnivorous	8.5	8.4	24.0
Paddlefish	Plankton	---	---	---
Buffalo	Algae & Insects	4.6	9.5	0.0
Shad	Plankton	---	---	---

Greater than 10 inches long
* Less than 10 inches long

The paddlefish is difficult to collect but was included where possible because it feeds exclusively on plankton; the shad and buffalo with food habits similar to the paddlefish are acceptable substitutes. During its lifetime, the flathead catfish remains within approximately one mile of its origin and is therefore, sampled downstream of the plant site. Catfish and carp are the most abundant of the commercial fish varieties.

The Missouri River has a sand bottom which moves with the water flow; therefore, benthos and other bottom organisms are extremely scarce. Joint efforts with the Nebraska Game Commission to obtain sufficient samples for analysis of periphyton have failed; a cooperative study continues as a separate project.

Milk

Milk from large Grade A milk producers in the local milkshed was sampled in cooperation with the Omaha Douglas County Health Department. The dairy herds of these Grade A milk producers are located downwind of the plant site. Radioactivity levels in the milk samples analyzed are shown in Table 2.10-4.

Table 2.10-4 - "Radioactivity in Milk January - March, 1969"

	Fresh Milk		Preserved Milk		
	I-131 pc/1	Cs-137 pc/1	(β-γ)-(K-40) nc/1	K-40 nc/1	Sr-90 pc/gm Calcium
Farm A	0	0	0.53	0.73	1.0
Farm B	0	0	0.81	0.74	1.0
Farm C	0	0	0.71	0.78	0.9

Vegetation

Foods normally consumed by the general population constitute the vegetation samples. Six stations with a total of ten varieties of food were sampled during the 1968 growing season. The variation in analytical results is shown in Table 2.10-5.

Table 2.10-5 - "Radioactivity in Vegetation October, 1968"

	Maximum nc/kg	Minimum nc/kg
Alpha	0.0	0.0
Beta-Gamma minus K-40	14.0	0.3
K-40	39.2	3.2
Sr-90	0.143	0.000
H-3	6	0

Air Particulate

Airborne particulate matter was collected at the plant site on 0.45 micron pore size filters; the filter was removed from the sampler and counted after the radioactivity had decayed for at least seventy-two hours. The air volume passed through the filter was approximately 1,000 cubic feet. None of the 32 samples analyzed showed any indication of alpha activity; the average beta-gamma concentration was 0.26 pc/m^3 with a maximum of 0.78 pc/m^3 and a minimum of 0.08 pc/m^3.

Background radiation readings measured with a Geiger-Mueller survey meter at sixteen stations around the plant site were all in the 0.00-0.02 mr/hr range. Results of the combination film badge-thermoluminescent dosimeters, at eleven stations, were all less than 30 mrem per quarter.

Wildlife

A wild rabbit sample was included to represent wildlife normally consumed in the area. These rabbits are free to wander, but they normally remain in the immediate vicinity. The radioactive content was 20 picocuries of Strontium-90 per gram of calcium in the femur and no iodine-131 was detectable in the thyroid.

2.10.3.2 Preoperational Monitoring Period

Following the trial period, the formal preoperational surveillance monitoring program was started in July, 1969, and continued for three years. This formal preoperational survey was an intensified continuation of the trial period already discussed. The program included soil samples and vegetation which are stored for possible future analyses.

The preoperational program results were documented for future reference and comparison; they defined the pre-operational background levels. Future background conditions may vary due to influences such as fallout from nuclear testing; however, the continuing environmental survey programs will provide adequate data to document changes in the background conditions.

2.10.3.3 Operational Survey Program

The purpose of the operational survey program is to provide public assurance that the Fort Calhoun contribution to naturally existing radioactivity is negligible. The program verifies the effectiveness of the waste disposal systems and radiological safety procedures incorporated in the plant.

Since plant operations began, samples similar to those taken during the preoperational program have continued to be collected routinely. The samples which would show changes in radioactivity first, primarily water and air, are sampled most frequently. Table 2.10-6 shows the types of samples taken, the frequency and number of these samples and the analysis frequency. Figures 2.10-1 thru 2.10-3 show the sampling locations and Figure 2.10-4 shows the types of sampling to be done at each location.

Deviations from the monitoring program may occur concerning sample location sites. If samples are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of equipment or participants ceasing participation in the program, corrective actions will be taken. Alternate samples and/or alternate sample locations will become part of the program as soon as practical in order to satisfy Offsite Dose Calculation Manual (ODCM) requirements. Figures 2.10-1 thru 2.10-4 and Table 2.10-6 will be updated when required to reflect changes in sample locations.

Table 2.10-6 - "Sample Types and Frequency "

Sample Class	Collection Frequency	Analysis Frequency	No. of Samples
Direct Radiation TLD	Quarterly	Quarterly	11
Emergency TLD	Replaced Annually	Site Area and General Emergencies Only	32
Air Monitoring	Weekly	Weekly (Gross Beta and I-131)Quarterly Composite of weekly filters (Gamma Isotopic)	5
Water	Weekly	Monthly (Gamma Isotopic) Analysis) Quarterly (H-3 Analysis)	3
Milk	Semi-monthly during grazing season. (May to October)	Semi-monthly during grazing season. (May to October)	2
Fish	Once per season (May to October)	Once per season (May to October)	5
Sediment	Semi-annual	Semi-annual	1

A land use survey is required to be conducted once every 24 months in order to identify changes in the use of the unrestricted area. As a result of this census, locations other than those presently sampled may be identified as potential higher exposure pathways and will be added to the environmental program. The sample numbers and location may vary from those presented in Table 2.10-6. In addition, other sample classes such as well water and vegetation may be added to the program. However, Table 2.10-6 represents the minimal operating requirements of the sample program.

Table 2.10-7 - "Radiological Environmental Sampling Locations and Media"

Location Number	Location Description	Distance from FCS Reactor Bldg (miles)	Direction (Degrees from north)	Airborne Particulate	Airborne Iodine	TLD	Surface Water	Fresh Milk	Bottom Sediment	Fish	Vegetation
1	Onsite Station No. 1, 110-meter weather tower	0.5	293°	X	X	X					
2	Onsite Station No. 2, adjacent to old plant access road	0.6	208°	X	X	X					
3	Offsite Station No. 3, intersection of Hwy. 75 and farm access road	0.8	145°	X	X	X					
4	Blair OPPD Office	3.0	303°	X	X	X					
5	EOF Building, North Omaha Power Station	17.5	157°	X	X	X					
6	Fort Calhoun City Hall	4.8	149°			X					
7	Fence around intake gate, Desoto Wildlife Refuge	2.0	101°			X					
8	Entrance to Plant Site from Hwy. 75	0.6	180°			X					
9	NW of Plant	1.0	310°			X					
10	WSW of Plant	0.7	250°			X					
11	SE of Plant	0.9	130°			X					
12	Met. Utilities Dist., Florence Treatment Plant North Omaha, NE	17.0	156°				X				
13	West bank Missouri River, downstream from reactor building	0.5	106°				X		X		
14	125' upstream from intake bldg., west bank of river	0.1	345°				X		X(1)		
15	Smith Farm(1)	1.9	133°				X				
16	OPPD Onsite Well(1)	0.1	154°				X				
17	Headquarters Bldg (1) Desoto Wildlife Refuge	3.1	53°				X				

Table 2.10-7 (Continued)

Location Number	Location Description	Distance from FCS Reactor Bldg (miles)	Direction (Degrees from north)	Airborne Particulate	Airborne Iodine	TLD	Well Water	Fresh Milk	Bottom Sediment	Fish	Vegetation
18	Miller Farm(4)	0.8	206°					X			X
19	Flynn Dairy(4)	3.4	310°					X			
20	Mohr Dairy	9.8	187°					x(2)			X
21	Japp Dairy(2)	6.3	219°					X			
22	Fish Sampling Area - Missouri River	R.M. 645.0	-							X	
23	Fish Sampling Area - (5) Missouri River	R.M. 666.0	-							X	
24	Legenhausen Farm(4)	0.7	207°								X
25	Seltz Farm(2)	2.7	168°					X			
26	John Welchert Farm	2.7	138°								X
27	Jerry Welchert Farm	2.0	296°								X
28	Alvin Pechnik Farm(4)	0.9	164°								X
29	E. Ellis Farm	0.7	180°								X(1)
30	Axtell Acreage	0.7	207°								X
31	Hakanson Farm	1.1	205°								X
32	Valley Substation #902	19.5	219°	X	X	X					
33	Bansen Farm(4)	0.7	207°								X
34	W.Jones Farm(6)	0.89	165°								X(3)
35	Onsite Farm Field(1)	0.74	110°								x

NOTES:
(1) Sampling not required for pathway modeling, collections performed for additional information only.
(2) When a milk sample is not available at a location, a broad leaf vegetation sample will be collected at that location as a substitute.
(3) Vegetable/food products sites chosen based on Land Use Survey and calculated doses.
(4) Location currently discontinued. Documented in table for historical reference only.
(5) Location is 21.0 river miels north of plant. Exact location can not be illustrated on map.
(6) Residence at 1.1 mile/155°, Actual garden on farm property at 0.89 miles/165°

Direct Radiation (TLD)

Quarterly TLDs are analyzed for ambient gamma after each respective replacement. Emergency TLDs are replaced annually or would be collected after a Site Area or General Emergency.

Air Monitoring

Air particulate samples are analyzed by gross beta count weekly. A gamma spectral analysis is performed on a quarterly composite of the weekly samples. Iodine cartridges at the air particulate stations are analyzed for I-131 on a weekly basis.

Water

A gamma scan is performed on monthly surface water composites. In addition, quarterly composites of surface water samples undergo analysis for tritium.

Environmental sample analyses are performed to provide compliance with 10 CFR 20 and to differentiate plant releases from natural or other sources of environmental radiation. Local public regulatory agencies who have assisted in the development of the environmental surveillance program are informed of survey results.

During plant operation, waste discharges are analyzed prior to release and are continuously monitored during release. The amount of radioactivity released is documented as a standard plant operating procedure. The environmental surveillance program is an independent survey verifying that the operating procedure for waste releases is effective and plant operations do not have a significant effect on the environment. In the unlikely event of an accidental release, samples will be collected and analyzed at all applicable environmental stations. Additional samples may also be obtained to better evaluate the magnitude of such a release.

Milk

During the time the cows are on pasture (May to October) samples of milk are collected semi-monthly within a five mile radius of the plant and analyzed for radioiodine content, calculated as I-131. Analyses are accomplished within eight days of sampling. The milk samples are also analyzed by gamma scan.

Fish

Samples of fish are collected once per season (May to October). *The fish selected for analysis are:

Category	Species	Size	Basic Food Habits
1	Flathead Catfish (juvenile)	<10" length	Carnivorous (insects)
	or Goldeye (adult)	N/A	
	or Buffalo (adult)	N/A	
	or Freshwater Drum (adult)	N/A	
2	Flathead Catfish (adult)	>10" length	Carnivorous (fish)
3	Carp (adult)	N/A	Omnivorous (insects, plants, fish)
4	Gizzard Shad (adult)	N/A	Planktonic/ Carnivorous (drift, single cell organisms, insects)

*Species obtained may vary if desired fish are unavailable or are considered a rare or protected species.
A Gamma scan analysis is performed on fish samples.

Sediment

Gamma scan analysis is performed on these samples semi-annually.

NT

SECTION 3

REACTOR

Table of Contents

3. REACTOR . . . 1
3.1 SUMMARY OF DESCRIPTION . . . 1
3.2 DESIGN BASES . . . 1
3.2.1 Performance Objectives . . . 1
3.2.2 Design Objectives . . . 1
3.2.3 Design Criteria and Limits . . . 2
3.2.3.1 Nuclear Criteria and Limits . . . 2
3.2.3.2 Reactivity Control Criteria and Limits . . . 3
3.2.3.3 Thermal and Hydraulic Criteria and Limits . . . 3
3.2.3.4 Mechanical Design Criteria and Limits . . . 4
3.3 REACTOR AND CORE FUEL CYCLE CONSIDERATIONS . . . 1
3.4 NUCLEAR DESIGN AND EVALUATION . . . 1
3.4.1 Reactivity and Control Requirements . . . 5
3.4.1.1 Doppler Defect and Moderator Temperature Defect . . . 7
3.4.1.2 Axial Flux Redistribution . . . 8
3.4.1.3 Moderator Voids . . . 8
3.4.1.4 CEA Power Dependent Insertion Limit (PDIL) . . . 8
3.4.1.5 Reactivity Worth Allowances . . . 8
3.4.1.6 Shutdown Margin and Safeguards Allowance . . . 8
3.4.2 Reactivity Coefficients . . . 9
3.4.2.1 Moderator Temperature Coefficient . . . 9
3.4.2.2 Moderator Pressure Coefficient . . . 10
3.4.2.3 Moderator Void Coefficient . . . 11
3.4.2.4 Fuel Temperature Coefficient . . . 11
3.4.2.5 Power Coefficient . . . 11
3.4.3 Control Element Assembly Worths . . . 12
3.4.4 Reactivity Insertion Rates . . . 13
3.4.5 Power Distribution . . . 14
3.4.5.1 Malpositioned CEA's . . . 14
3.4.6 Pressure Vessel Fluence . . . 15
3.4.7 Nuclear Evaluation . . . 17
3.4.7.1 Nuclear Design Methods . . . 17
3.4.7.2 Comparisons With Experiments . . . 18
3.4.8 Reactor Stability . . . 28
3.4.8.1 Method of Analysis . . . 28
3.4.8.2 Radial Mode Oscillations . . . 29
3.4.8.3 Azimuthal Mode Oscillations . . . 30
3.4.8.4 Axial Mode Oscillations . . . 30
3.4.8.5 Detection of Xenon Oscillations . . . 31
3.4.8.6 Control of Xenon Oscillations . . . 33
3.4.8.7 Xenon Oscillation Operating Experience . . . 33

3.5 CYCLE 6 CORE POWER UPRATE TO 1500 MWt 1
3.5.1 Plant Parameter Variations 3
3.5.2 Hot Channel Factors 4
3.5.2.1 Description of Hot Channel Factors 4
3.5.2.2 Summary of Hot Channel Factors 6
3.5.3 Coolant Flow 6
3.5.3.1 Total Coolant Flow Rate and Bypass Flow 6
3.5.3.2 Pressure Drop 7
3.5.3.3 Partial Flow Loop Operations 8
3.5.4 Subchannel MDNBR Analysis 8
3.5.5 Departure From Nucleate Boiling 9
3.5.5.1 Design Approach to Departure From Nucleate Boiling 9
3.5.6 Thermal and Hydraulic Evaluation 11
3.5.6.1 Analytical Models 11
3.5.6.2 Statistical Analysis of Hot Channel Factors 11
3.6 THERMAL AND HYDRAULIC DESIGN EVALUATION FOR LIMITING TRANSIENTS AND SETPOINT ANALYSIS 1
3.6.1 General 1
3.6.2 Coolant Flow 2
3.6.2.1 Total Coolant Flow Rate and Bypass Flow 2
3.6.2.2 Inlet Flow Distribution 3
3.6.2.3 Exit Pressure Distribution 3
3.6.2.4 Partial Flow Loop Operations 3
3.6.3 Peak Linear Heat Rate 4
3.6.4 Peak Linear Heat Rate Protection 4
3.6.5 Thermal Margin Analysis 5
3.6.5.1 Engineering Factors 5
3.6.6 Departure From Nucleate Boiling 6
3.6.7 Vapor Fraction 7
3.6.8 Thermal and Hydraulic Evaluation 7
3.6.10 Flow Stability 8
3.7 MECHANICAL DESIGN AND EVALUATION 1
3.7.1 Reactor Internals 6
3.7.1.1 Core Support Assembly 6
3.7.1.2 Core Support Barrel 9
3.7.1.3 Core Support Plate and Support Columns 9
3.7.1.4 Thermal Shield 10
3.7.1.5 Core Shroud Plates and Centering Plates 10
3.7.1.6 Flow Skirt 11
3.7.1.7 Upper Guide Structure Assembly 11
3.7.2 Control Element Drive Mechanism 13
3.7.2.1 CEDM Pressure Housing 14
3.7.2.2 Rack and Pinion Assembly 16

3.7.2.3 Motor Drive Package 17
3.7.2.4 Position Readout Equipment 17
3.7.2.5 Control Element Assembly Disconnect 18
3.7.2.6 CEDM Evaluation 18
3.7.3 Core Mechanical Design 18
3.7.3.1 Fuel Assembly 19
3.7.3.2 Fuel Rods 20
3.7.3.3 Poison Rods 22
3.7.3.4 Part Length Poison Rods 22
3.7.3.5 Hafnium Flux Suppression Rods 23
3.7.3.6 Clad Evaluation 23
3.7.3.7 Control Element Assembly 27
3.7.3.8 Control Element Assembly Evaluation 28
3.7.4 Vibration Analysis and Monitoring 29
3.8 FUEL PERFORMANCE DURING ANTICIPATED TRANSIENTS AFTER LONG BURNUP 1
3.8.1 Fuel Design and Analysis 1
3.8.2 Summary of Fuels Irradiation Information 4
3.8.2.1 Public Information 5
3.8.2.2 Operating Fuel Experience 5
3.8.2.3 Fuel Irradiation Programs 6
3.8.2.4 Fuel Surveillance Programs 12
3.8.3 Summary 12
3.9 SPECIFIC REFERENCES 1
3.10 GENERAL REFERENCES 1
3.10.1 Prediction of Departure from Nucleate Boiling for an Axially Non-Uniform Heat Flux Distribution, Tong, L. S., Journal of Nuclear Energy, 21, 241 to 248 (1967). 1
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3.10.16 OPPD-NA-8302-P, "Neutronics Design Methods and Verification." 2

List of Tables

Table 3.2-1 - "Primary Stress Limits for Critical Reactor Vessel Internal Structures" 5
Table 3.4-1 - "Nuclear Design Parameters 2
Table 3.4-2 - "Effective Multiplication Factors and Reactivity Under Various Conditions" (No Control Element Assemblies or Dissolved Boron, Beginning-of-Cycle Cycle 20 Core) 5
Table 3.4-3 -"Dissolved Boron Requirements For Criticality (Control Element Assemblies Fully Withdrawn, Beginning-Of-Cycle 20 Core) 6
Table 3.4-4 - "CEA Reactivity Allowances, %Δρ" 7
Table 3.4-5 - "Moderator Temperature Coefficients" 10
Table 3.4-6 - "Calculated CEA Worths, %Δρ (@ 532°F)" 12
Table 3.4-7 - "Worth of CEA Groups, %Δρ (@ BOC 532°F)" 13
Table 3.4-8 - "Fuel Rod Description" 19
Table 3.4-9 - "Results of Analysis of Critical and Subcritical UO_2 Systems" 20
Table 3.4-10 - "Results of Analysis of $Pu0_2$-U02 Fueled Lattices" 21
Table 3.4-11 - "BOC HZP Critical Boron Concentration (ARO)" 22
Table 3.4-12 - "Inventory Change Comparison" 23
Table 3.4-13 - "Comparison of Predicted and Measured Power Coefficients" 24
Table 3.4-14 - "Comparison of Predicted and Measured Moderator Temperature Coefficients" 26
Table 3.5-1 - "Thermal and Hydraulic Parameters" 2
Table 3.5-2 - "Plant Parameters for Thermal and Hydraulic Design, Steady State" 3
Table 3.5-3 - "Summary of Hot Channel Factors" 6
Table 3.5-4 - "Reactor Pressure Drops" 8
Table 3.5-5 - "Probability Distribution, DNB Limits" 11
Table 3.6-1 - "Thermal and Hydraulic Parameters for Safety Analyses" 2
Table 3.7-1 - "Mechanical Design Parameters" 2
Table 3.7-2 - "Comparison of Areas in the Core Support Structure Which Experience the Highest Fluence With Ductility at End of Life" 8
Table 3.8-1 - "Tensile Test Results on Irradiated Saxton Core III Cladding" Fluence (<1 Mev) 4.7×10^{21} n/cm^2 (Estimated) 2

List of Figures

The following figures are controlled drawings and can be viewed and printed from the applicable listed aperture card.

Figure No.	Title	Aperture Card
3.1-1	Reactor Arrangement	36414
3-1.2	Reactor Core Cross Section	36415
3.4-1	Assy Avg Burnup Distribution, Beginning of Cycle	36416
3.4-2	Assy Avg Burnup Distribution, End of Cycle (15,500 MWD/MTU)	36417
3.4-3	Power Dependent, Insertion Limit	36418
3.4-4	Position of Fuel Assemblies and CEA Groups	36419
3.4-5	Core Radial Power Distribution, Beginning of Cycle ARO	36420
3.4-6	Core Radial Power Distribution at 1500 MWD/MTU Burnup ARO	36421
3.4-7	Core Radial Power Distribution, End of Cycle (15,500 MWD/MTU) ARO	36422
3-4.9	Comparison of Calculated and Measured Plutonium-to-Uranium Mass Ratio in the Asymptotic Neutron Spectrum for Yankee	36424
3.4-10	Plutonium Isotopic Composition vs. Fuel Depletion in the Asymptotic Spectrum for Yankee	36425
3.4-11	Reactivity Difference Between Fundamental and Excited States of a Bare Cylindrical Reactor	36426
3.4-12	Thermal Neutron Flux at the Center of the Core vs. Time	36427
3.4-13	Damping Coeff vs. Reactivity Difference Between Fundamental and Excited State	36428
3.4-14	End of Life Oscillations with Doppler Feedback, Full Power (2 Hour Time Steps)	36429
3.4-15	Split Detector Response to Axial Power Profiles in the Core	36430
3.6-3	Two Phase Flow Regime Transitions Stable Flow Limits	36433
3.6-4	Linear Heat Rate SAFDL Protection by APD Trip	36431
3.7-1	Upper Guide Structure Assembly	36441
3.7-2	Control Element Drive Mechanism	36442
3.8-1	Fuel Rod Circumferential Strain vs. Temperature	36449
3.8-2	Design Curve for Cyclic Strain Usage of Zircaloy-4 at 700 Degrees F	36450

3. REACTOR

3.1 SUMMARY OF DESCRIPTION

The reactor is of the pressurized water type, using two reactor coolant loops. A vertical cross section of the reactor is shown in Figure 3.1-1. The reactor core is composed of 133 fuel assemblies and 49 control element assemblies (CEA's). The fuel assemblies are arranged to approximate a right circular cylinder with an equivalent diameter of 106.5 inches and an active fuel length of 128 inches. The fuel assembly, which provides for 176 fuel rod positions, (14 x 14 array) consists of 5 guide tubes attached to spacer grids and is closed at the top and bottom by end fittings. The fuel rods are retained in an open framework, restrained laterally by Zircaloy, ZIRLO, or alloy 718 spacer grids. The guide tubes each displace four fuel rod positions and provide channels that guide the CEA's over their entire length of travel. In-core instrumentation is routed into the fuel assemblies through the upper head of the reactor vessel. Figure 3.1-2 shows the reactor core cross section and dimensional relations between fuel assemblies, fuel rods and CEA guide tubes.

The fuel is low enrichment UO_2 in the form of ceramic pellets and is encapsulated in prepressurized helium filled Zircaloy or ZIRLO tubes, which form a hermetic enclosure.

The reactor coolant enters the inlet nozzles of the reactor vessel, flows downward between the reactor vessel wall and the core barrel, and passes through the flow skirt and lower core barrel section, where the flow distribution is equalized, and into the lower plenum. The coolant then flows upward through the core removing heat from the fuel rods. The heated coolant enters the core outlet region where the coolant flows around the outside of control element assembly shroud tubes to the reactor vessel outlet nozzles. The control element assembly shroud tubes protect the individual neutron absorber elements of the CEA's from the effects of coolant cross flow above the core.

The reactor internals, which support and orient the fuel assemblies, control elements assemblies, and in-core instrumentation, also guide the reactor coolant through the reactor vessel. The internals absorb the static and dynamic loads and transmit the loads to the reactor vessel flange, and they will safely perform their functions during normal operating, upset and emergency conditions. The internals are designed to safely withstand the forces due to the deadweight, handling, pressure differentials, flow impingement, temperature differentials, vibration and seismic acceleration.

The design of the reactor internals limits deflection where such limits are required by function. The stress values of all structural members under normal operating and expected transient conditions are not greater than those established by Section III of the ASME Pressure Vessel Code. The effect of neutron irradiation on the material utilized is included in the design evaluation. The effect of accident loadings on the internals is included in the design analysis.

Reactivity control is provided by two independent systems: namely, the Control Element Drive System and the Chemical and Volume Control System. The Control Element Drive System controls short term reactivity changes and is used for rapid shutdown. The Chemical and Volume Control System is used to compensate for long-term reactivity changes and can make the reactor subcritical without the benefit of the Control Element Drive System. The design of the core and the Reactor Protective System prevents fuel damage limits from being exceeded for any single malfunction in either of the reactivity control systems.

The CEA's consist of five Inconel tubes, 0.948 inch in diameter, containing boron carbide pellets with silver-indium-cadmium slugs in the tips to reduce clad strain. Four tubes are assembled in a square array around the central fifth tube. The tubes are joined by a spider at the upper end. The hub of the spider couples the CEA to the drive assembly. The CEA's are actuated by rack and pinion control element drive mechanisms (CEDM's) mounted on the reactor vessel head. Four full length CEA's are non-trippable and are required by the Technical Specifications to remain in an essentially fully withdrawn position during power operation. These four CEA's were originally designed as part-length CEAs to be used for axial power distribution control during power level changes.

Control element assemblies are moved in groups to satisfy the requirements of shutdown, power level changes and operational maneuvering. The maximum reactivity worth of the CEA's and the associated reactivity addition rate are limited by system design to prevent rapid large reactivity increases. The design restraints are such that reactivity increases will not result in the violation of the fuel damage limits, rupture of the reactor coolant pressure boundary, or physical disruption of the core or other internals in such a way as to impair the effectiveness of emergency core cooling.

Boric acid dissolved in the coolant is used as a neutron absorber to provide long term reactivity control. In the event it becomes necessary to reduce the boric acid concentration required at the beginning-of-cycle operating conditions in order to reduce the algebraic magnitude of the moderator temperature coefficient of the core, appropriate neutron absorber material (poison) will be provided in certain reload fuel assemblies.

The nuclear design of the core will assure that, in the power operating range, the combined response of all reactivity coefficients to an increase in reactor thermal power yields a net decrease in reactivity. Core monitoring and administrative controls on the plant will result in power distributions during normal operation such that the Reactor Protective System will prevent both the fuel temperature and the departure from nucleate boiling ratio (DNBR) from exceeding acceptable values for postulated accidents and anticipated transients.

The details of the reactor and core design are discussed in the following subsections of this Section 3 of the USAR. The design bases are described in Section 3.2, and reactor core and fuel cycle considerations are discussed in Section 3.3. The design and evaluation of the nuclear, thermal hydraulic, and mechanical characteristics of the reactor are described in Sections 3.4, 3.5, and 3.6, respectively, and the corresponding summary lists of significant core parameters are presented in Tables 3.4-1, 3.5-1 and 3.6-1, respectively.

3.2 DESIGN BASES

3.2.1 Performance Objectives

The initial full-power thermal rating of the core was 1420 MWt, which corresponds to a gross electrical output of 481 MWe. Although the plant was designed for a full-power rating of 1500 MWt, the initial license application and the first five fuel cycles of operation were at this lower power rating of 1420 MWt. On August 15, 1980, Fort Calhoun Station was issued a license amendment (Amendment No. 50) to allow operation at a steady state full rated power level of 1500 MWt, and the safety analysis described in this USAR was performed for a full rated power level of 1500 MWt.

3.2.2 Design Objectives

During normal operating conditions and anticipated transients, the reactor core, together with its control systems and the reactor protective system, is designed to function over its lifetime to prevent fuel damage based upon application of conservative limits for excessive fuel temperature, cladding strain, and cladding stress as specified in Section 3.2.3.

The combined response of all reactivity feedback mechanisms to an increase in reactor thermal power is a net decrease in reactivity. The combined effect of all reactivity coefficients in conjunction with the reactor control system provides stable reactor operation. If power oscillations do occur, their magnitude will be such that the fuel damage limits are not exceeded.

The maximum reactivity worth of the CEA's and the associated reactivity addition rate are limited by core, CEA and control element drive system designs to prevent rapid, large reactivity increases. Such reactivity increases are precluded in order to avoid violation of the fuel damage limits, rupture of the reactor coolant pressure boundary, or disruption of the core or other internals sufficient to impair the effectiveness of emergency cooling.

3.2.3 Design Criteria and Limits

3.2.3.1 Nuclear Criteria and Limits

The design of the core is based upon the following nuclear criteria and limitations:

a. The local fuel pellet burnup limit is determined by material and mechanical design rather than nuclear considerations. The conservatism of the resulting limit is confirmed by actual irradiation of demonstration fuel assemblies to the corresponding limit specified for that particular fuel design.

b. The combined response of all reactivity coefficients to an increase in reactor thermal power yields a net decrease in reactivity.

c. As noted in Section 3.1, CEA's are moved in groups to satisfy the requirements of shutdown, power level changes and operational maneuvering. The control systems are designed to produce power distributions that are within the acceptable limits on the overall nuclear heat flux factor (F^N_q) and departure from nucleate boiling ratio (DNBR) limits. The reactor protective system and the Technical Specification Limiting Conditions for Operation assure that these limits are not exceeded.

d. Axial power distributions are manually controlled by trippable full length CEA's, using information provided by the out-of-core detectors.

e. The melting point of the UO_2 fuel shall not be reached during normal operation and anticipated transients.

3.2.3.2 Reactivity Control Criteria and Limits

The control system and operating procedures provide for adequate control of the core reactivity and power distributions such that the following are met:

a. Sufficient CEA's are withdrawn to provide an adequate shutdown reactivity margin following a reactor trip.

b. The shutdown margin is maintained with the highest worth CEA assumed stuck in its fully withdrawn position.

c. The chemical and volume control system is capable of adding boric acid to the reactor coolant at a rate sufficient to maintain the shutdown margin during a reactor coolant system cooldown at the design rate following a reactor trip.

3.2.3.3 Thermal and Hydraulic Criteria and Limits

The principal criterion for the thermal and hydraulic design is to avoid thermally induced fuel damage during normal operation and anticipated transients. It is recognized that there is a small probability of limited fuel damage in certain postulated events as discussed in Section 14.

The following corollary thermal and hydraulic design bases are established, but violation of either does not necessarily result in fuel damage:

a. There shall be a high confidence level that DNB is avoided during normal operation and anticipated transients.

b. For Limiting Safety System Settings (LSSSs), Limiting Conditions of Operation (LCOs) and certain transients (Section 3.6), the minimum DNBR must be greater than or equal to the minimum DNBR safety limit. For the HTP correlation (reference 3-3), this limit is 1.141. If appliable, a 2% mixed-core penalty is applied (Reference 3-5). For the CE-1 correlation, this limit is 1.18 for SCU applications and 1.15 for determinstic analyses (References 3-53, 3-54, 3-55, and 3-6).

The reactor protective system and the reactor control system provide for automatic reactor trip or corrective actions before these design limits are violated.

Reactor internal flow passages and fuel coolant channels are designed to prevent hydraulic instabilities. Flow maldistributions are limited by design to be compatible with the specified thermal design criteria.

3.2.3.4 Mechanical Design Criteria and Limits

The reactor internals are designed to safely perform their functions during steady state conditions and normal operating transients. The internals can safely withstand the forces due to deadweight, handling, system pressure, flow-inducted pressure drop, flow impingement, temperature differential, shock, and vibration. The structural components satisfy stress values given in Section III of the ASME Boiler and Pressure Vessel Code.

The following limitations on stresses or deformations are employed to assure the capability exists for a safe and orderly shutdown in the event of earthquake and major loss-of-coolant accident loading conditions. For reactor vessel internal structures, the stress criteria are given in Table 3.2-1. The intent of the limits in this table can be described as follows:

a. Under design loadings plus design earthquake forces, (see Appendix F) the critical reactor vessel internal structures are designed in accordance with the stress criteria established in Section III of the ASME Boiler and Pressure Vessel Code, Article 4.

b. Under normal operating loadings plus maximum hypothetical earthquake forces, the design criteria permit a small amount of local yielding.

c. Under normal operating loadings plus coolant pipe rupture loadings plus maximum hypothetical earthquake forces, permanent deformations are permitted by the design criteria.

In the loading combinations listed in Table 3.2-1, the earthquake forces include both horizontal and vertical seismic excitations acting simultaneously.

To properly perform their functions, the critical reactor internal structures are designed to satisfy the additional deflection limits described below, in addition to the stress limits given in Table 3.2-1.

Under loading combinations (a) and (b) of Table 3.2-1, deflections are limited, so that the CEA's can function and adequate core cooling is maintained. Under loading combination (c) of Table 3.2-1, the deflection design criteria depend on the size of the piping break. If the equivalent diameter of the pipe break is no larger than the largest line connected to the main reactor coolant lines, deflections are limited, so that the core is held in place, the CEA's function normally, and adequate core cooling is maintained. Those deflections which would influence CEA movement are limited to less than two-thirds of the deflection required to prevent CEA function. For pipe breaks larger than the above, the criteria are that the fuel is held in place in a manner permitting core cooling and that adequate coolant flow passages are maintained. Further, although not required for shutdown, all CEA's will be insertable. For the larger break sizes, critical components which meet the stress criteria of Table 3.2-1 are also restrained from buckling by further limiting the stress levels to two-thirds of the stress level calculated to produce buckling.

Table 3.2-1 - "Primary Stress Limits for Critical Reactor Vessel Internal Structures"

a.	Design Loadings Plus Design Earthquake	$P_m \leq S_m$
	Forces	$P_B + P_L \leq 1.5\ S_m$
b.	Normal Operating Loadings Plus Maximum	$P_m \leq S_D$
	Hypothetical Earthquake Forces	$P_B \leq 1.5 \left[1 - \frac{(Pm)^2}{S_D} \right] S_D$

c. Normal Operating Loadings Plus Maximum Hypothetical Earthquake Forces Plus Pipe Rupture Loadings

$$P_m \leq S_L$$

$$P_B \leq 1.5 \left[1 - \frac{(Pm)^2}{S_L} \right] S_L$$

where:

P_L, P_m, P_B, S_m, S_y are defined in the ASME Boiler and Pressure Vessel Code, Section III, Article 4

S_u = Minimum tensile strength of material at temperature

$S_L = S_y + (1/3)(S_u - S_y)$

S_D = Design Stresses = 1.2 S_m

Fuel Assemblies

The fuel assemblies are designed to maintain their structural integrity under steady state and transient operating conditions, as well as under normal handling, shipping, and refueling loads. The design takes into account differential thermal expansion of fuel rods, thermal bowing of fuel rods and CEA guide tubes, irradiation effects, and wear of all components. Mechanical tolerances and clearances have been established on the basis of the functional requirements of the components. All components including welds are highly resistant to the corrosive action of the reactor environment.

The fuel rod design takes into account external pressure, differential expansion of the fuel and clad, fuel swelling, clad creep, fission and other gas releases, thermal stress, pressure and temperature cycling and flow-induced vibrations.

Control Element Assemblies (CEA's)

The CEA's are designed to maintain their structural integrity both under all steady state and transient operating conditions, and under handling, shipping and refueling loads. Thermal distortion, mechanical tolerances, vibration and wear are all taken into account in the design. Clearances and corresponding fuel assembly alignment are established so that the possible stackup of mechanical tolerances and thermal distortion would not result in frictional forces that could prevent reliable operation of the system. The structural criteria are based on limiting the maximum stress intensity to those values specified in Section III of the ASME Boiler and Pressure Vessel Code.

The control element drive mechanisms (CEDM's) are capable of actuating the CEA's under steady state and transient operating conditions and during hypothetical seismic occurrences. For pipe rupture accident loads, the CEDM's are designed to support and maintain the position of the CEA's in the core and to be capable of actuating them when these loads have diminished.

The speed at which the CEA's are inserted or withdrawn from the core is consistent with the reactivity change requirements during reactor operation. For conditions that require a rapid shutdown of the reactor, the CEDM clutches release to allow the CEA's and the connecting CEDM components to drop by gravity into the core. The reactivity is reduced during such a CEA drop at a rate sufficient to prevent violation of fuel damage limits.

The CEDM pressure housings are an extension of the reactor vessel, providing a part of the reactor coolant boundary, and are therefore, designed to meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels. Pressure and thermal transients as well as steady state loadings were considered in the design analysis.

3.4 NUCLEAR DESIGN AND EVALUATION

This section summarizes the nuclear characteristics of the design and discusses the design parameters which are of significance to the performance of the core in normal transient and steady state operational conditions. A discussion of the nuclear design methods employed and comparisons with experiments which support the use of these methods is included.

The numerical values presented are based on the Cycle 20 core. The analysis performed for this fuel cycle shows that all necessary requirements for safe operation have been met. Table 3.4-1 shows a summary of the nuclear design parameters for the fuel cycle. Figures 3.4-1 and 3.4-2 show the assembly average burnup distributions for both the beginning and end of Cycle 20, respectively.

Table 3.4-1 - "Nuclear Design Parameters"

Performance Characteristics

Fuel Management	20-Batch, Mixed Central Zone
Average Cycle Burnup, MWD/MTU	15,630 (Coast Down)

Nominal Central Zone U-235 Enrichment, w/o (initial)

T1	4.16
T3	4.16
T7	3.76
T8	3.75
X1	3.65
X2	4.40
X3	4.40
X4	4.40
X5	4.40
X6	4.40
Y1	0.27
Y2	3.87
Y3	4.16
Y4	4.06
Y5	4.23
Y6	4.21
Y7	4.19
Y8	4.20
Y9	4.18
Y10	3.54

$H_20/U0_2$ Volume Ratio, Unit Cell (Cold) 1.66

Table 3.4-1 (Continued)

<u>Control Characteristics</u>	
k_{eff}, Beginning-of-Cycle, No Control Element Assemblies	
Cold (68°F)	1.18
Hot (532°F), Zero Power	1.14
Hot, Equilibrium Xe, Full Power	1.09
Number of Control Element Assemblies (CEA's)	
Full Length	45
Non-trippable	4
<u>Total CEA Worth, %Δρ</u>	
Beginning-of-Cycle (w/o Group N)	
Hot Zero Power (532°F)	7.73
End-of-Cycle (w/o Group N)	
Hot Zero Power (532°F)	8.82
<u>Dissolved Boron</u>	
Dissolved Boron Content for Criticality, ppm, (CEAs withdrawn, BOC)	
Cold (68°F), ppm	1582
Hot (532°F), Zero Power, ppm	1575
Hot (570°F), Equilibrium Xe, Full Power, ppm	1086
Dissolved Boron Content for Refueling, ppm	See COLR (TDB-VI)
Boron Worth, ppm/%Δρ	
Hot (570°F)	-130.4
Cold (68°F)	-95.6

Table 3.4-1 (Continued)

Reactivity Coefficients (CEA's Withdrawn)

Moderator Temperature Coefficient,	
α_{mod}, $\Delta\rho/°F$	
Hot, Full Power (570°F)	
Beginning-of-Cycle	-0.62×10^{-4}
End-of-Cycle	-3.00×10^{-4}
Hot Zero Power (532°F)	
Beginning-of-Cycle	$+0.01 \times 10^{-4}$
Fuel Temperature Coefficient,	
α_{fuel}, $\Delta\rho/°F$	
Hot, Zero Power (532°F, BOC)	-2.15×10^{-5}
Full Power (1080°F, BOC)	-1.67×10^{-5}
Moderator Void Coefficient,	
α_{void}, $\Delta\rho/\%$ Void	
Hot, Operating (572°F)	
Beginning-of-Cycle	-0.18×10^{-3}
End-of-Cycle	-1.46×10^{-3}
Moderator Pressure Coefficient,	
$\alpha\rho$, $\Delta\rho$/psi	
Hot, Operating (572°F)	
Beginning-of-Cycle	$+0.4 \times 10^{-6}$
End-of-Cycle	$+2.1 \times 10^{-6}$

3.4.1 Reactivity and Control Requirements

Table 3.4-2 lists the effective multiplication factors (k_{eff}) and reactivity (ρ) under various conditions.

Table 3.4-2 - "Effective Multiplication Factors and Reactivity Under Various Conditions"
(No Control Element Assemblies or Dissolved Boron, Beginning-of-Cycle Cycle 20 Core)

	k_{eff}	ρ
Cold (68°F), ARO	1.18	0.166
Hot (532°F), Zero Power	1.14	0.127
Hot, Full Power, Equilibrium Xe, ARO	1.09	0.083

The maximum excess reactivity ρ is 16.6 percent for the cold, unborated core at beginning of cycle. The reactivity decrease from zero to full power is due to the change in fuel temperature which causes Doppler broadening of the U-238 resonances and the change in moderator temperature coefficient which becomes more negative (due to a lower reactor coolant system boron concentration).

Reactivity control in the reactor is accomplished by adjusting both the position of the CEAs and the concentration of boric acid dissolved in the reactor coolant system. The CEAs permit rapid changes in reactivity, as required for reactor trip and to compensate for changes in moderator and fuel temperature and void formation associated with changes in power level. There are 45 standard and four full length non-trippable CEAs. The standard CEAs are used for shutdown and for regulation. The CEAs designated as shutdown CEAs are divided into two separately controlled groups; those designated as regulating CEAs are divided into four groups. During power operation, the shutdown groups are fully withdrawn while the position of the regulating groups is adjusted to meet reactivity and power distribution requirements. All CEAs except the full length non-trippable CEAs drop to a fully inserted position upon reactor trip.

Adjustment of the boric acid concentration is used to control the relatively slow reactivity changes associated with plant heatup and cooldown, fuel burnup, and certain xenon variations. Also, additional boric acid is used to provide a large shutdown margin for refueling. The use of boric acid dissolved in the reactor coolant makes it possible to maintain most of the CEAs in a withdrawn position during full power operation, thus minimizing the distortions in power distribution. Table 3.4-3 lists the concentrations of natural boron required to maintain the core critical under various conditions, assuming all control element assemblies are fully withdrawn.

Table 3.4-3 -"Dissolved Boron Requirements For Criticality (Control Element Assemblies Fully Withdrawn, Beginning-Of-Cycle 20 Core)"

	Natural Boron, ppm
Cold (68°F)	1582
Hot (532°F), Zero Power, Clean	1575
Hot, Full Power, Equilibrium Xe	1086
Refueling	See COLR (TDB-VI)

The refueling boron concentration (without considering the worth of the CEAs) provides a reactivity shutdown of approximately 5 percent for the cold condition. The refueling concentration is specified as an equivalent ppm boric acid (H_3B0_3) in the coolant in the COLR, which is approximately 10 percent of the solubility limit at refueling temperatures. After a normal shutdown or reactor trip, boric acid may be injected into the reactor coolant system to compensate for reactivity increases due to normal cooldown and xenon decay. Although the boric acid system reduces reactivity relatively slowly, the rate of reduction is more than sufficient to maintain the shutdown margin against the effects of normal cooldown and xenon decay.

Sufficient worth is available in the regulating CEAs to compensate for the rapid changes in reactivity associated with power level changes. In addition, these CEAs may be used for partial control of xenon transients and minor variations in moderator temperature and boron concentration. Table 3.4-4 summarizes the hot zero power control rod worths at both beginning and end-of-cycle. The reactivity variations are described in Sections 3.4.1.1 through 3.4.1.6. The worth of all CEAs, including shutdown CEAs, covers the reactivity variations and provides adequate shutdown with the most reactive CEA stuck in the fully withdrawn position. Margin is provided between the calculated CEA worth and the reactivity variations to account for uncertainties in the calculations. Only the 45 full length trippable CEAs are considered in Table 3.4-4.

Table 3.4-4 - "CEA Reactivity Allowances, %Δρ"

Control Rod Worth (%Δρ)	BOC	EOC (16,000 MWD/MTU)
	HZP	HZP
Total Full Length Rod Worth	7.73	8.82
Stuck Rod Worth	0.96	1.36
Total Minus Stuck Rod Worth	6.77	7.46
PDIL Rod Insertion	1.12	1.19
Gross PDIL Scram Worth	5.65	6.27
Uncertainty (6.52%)	0.37	0.41
Net PDIL Rod Worth	5.28	5.86
Required Shutdown Margin	4.0	4.0
Excess Margin	1.28	1.86

3.4.1.1 Doppler Defect and Moderator Temperature Defect

The increase in reactivity associated with the change from full to zero power from both the Doppler effect in U-238 and the moderator temperature effect is 1.1%Δρ at beginning-of-cycle and 2.1%Δρ at end-of-cycle. This change in reactivity is compensated by CEA movement.

3.4.1.2 Axial Flux Redistribution

A change in reactivity occurs due to axial flux redistribution over a cycle as a result of the localized burn-out and redistribution of Xenon. This is conservatively estimated to be 0.2%$\Delta\rho$ at beginning-of-cycle and 0.4%$\Delta\rho$ at end-of-cycle.

3.4.1.3 Moderator Voids

A change in reactivity results from the formation of voids due to local boiling. The average void content in this core is very small and is estimated to be one-fourth of 1 percent at full power. As with the moderator temperature effect, the maximum increase in reactivity from full to zero power occurs at end-of-cycle when the least amount of dissolved boron is present. The maximum reactivity variation due to one-fourth of 1 percent voids is conservatively estimated to be 0.1 percent $\Delta\rho$.

3.4.1.4 CEA Power Dependent Insertion Limit (PDIL)

The PDIL rod insertion is a measure of the rod worth associated with the permissible CEA configurations (see Figure 3.4-3) for both hot zero power and hot full power.

3.4.1.5 Reactivity Worth Allowances

An allowance is made in the reactivity worth of the CEA's to compensate for variations in xenon, dissolved boron concentration, and moderator temperature. When the CEA's reach the limits imposed on CEA motion, additional reactivity changes will be made by changing the boron concentration.

3.4.1.6 Shutdown Margin and Safeguards Allowance

An allowance of 4.0 percent $\Delta\rho$ at both the beginning-of-cycle (BOC) and at the end-of-cycle (EOC), respectively, has been made for the shutdown margin and safeguards allowances at hot, zero power conditions with the most reactive CEA stuck in the withdrawn position.

3.4.2 Reactivity Coefficients

The factors which contribute to the reactivity of a reactor, such as the thermal utilization, resonance escape probability, and nonleakage probabilities, are dependent upon certain parameters, such as moderator temperature and pressure and fuel temperature. Reactivity coefficients, denoted by α, relate changes in the core reactivity to variations in these parameters.

3.4.2.1 Moderator Temperature Coefficient

The reactivity worth of 1086 ppm of boron (amount of reactivity needed to maintain the reactor just critical at BOC full power conditions) increases from 8.33%$\Delta\rho$ to 8.79%$\Delta\rho$ as the moderator temperature decreases from operating to zero power temperature. The interaction of these temperature effects (along with the temperature coefficient of the unborated core) results in a net moderator temperature coefficient of reactivity, α_{mod}, at operating temperature which ranges from strongly negative to slightly positive, depending on the moderator temperature, the soluble boron content, and the fuel burnup.

In a core which is controlled by chemical shim dissolved in the moderator, there are two factors which cause the moderator temperature coefficient to become less negative as the fraction of reactivity controlled by the dissolved boron increases (i.e., at higher boron concentrations). First, an increase in moderator temperature reduces the effective density of the chemical poison and hardens the thermal neutron spectrum, thereby decreasing neutron absorption in the boron. Secondly, the effective reactivity worth of a solid poison such as the CEA's increases as the moderator temperature increases; thus, since there are fewer solid poison control elements than would be required in a reactor without chemical shim, the magnitude of this effect is reduced.

The calculated moderator temperature coefficient for various core conditions is given in Table 3.4-5. As shown in the table, the most positive value occurs at the beginning of cycle (HZP) when the dissolved boron content is at its maximum.

Table 3.4-5 - "Moderator Temperature Coefficients"

Conditions	(α_{mod}, $\Delta\rho$/°F)
Beginning of Cycle	
Hot, Full Power, CEAs Out	-0.62×10^{-4}
Hot, Zero Power, CEAs Out	$+0.01 \times 10^{-4}$
End of Cycle	
Hot, Full Power, CEAs Out, Zero ppm	-3.00×10^{-4}

The moderator coefficient becomes more negative with burnup, due mainly to the reduction in the dissolved boron content with burnup. The effects of plutonium and fission products are small when compared to the above; however, the buildup of xenon supplies a positive contribution to the coefficient for a constant boron concentration. Equilibrium xenon raises α_{mod} by 0.05×10^{-4}. However, when the dissolved boron concentration is reduced by the reactivity equivalent of xenon, the α_{mod} becomes more negative by 0.3×10^{-4} per °F.

The change in moderator temperature coefficient as a function of boron concentration is linear, being $+0.20 \times 10^{-4}$ per 100 ppm soluble boron.

3.4.2.2 Moderator Pressure Coefficient

The moderator pressure coefficient, α_p, is the change in reactivity per unit change in reactor coolant system pressure. Since an increase in pressure increases the water density, the pressure coefficient is opposite in sign to the temperature coefficient. The reactivity effect of increasing the pressure is reduced in the presence of dissolved boron because an increase in water density adds boron to the core. The calculated pressure coefficients for the beginning and end of the first cycle at full power were $+0.4 \times 10^{-6}$ $\Delta\rho$/psi and $+2.1 \times 10^{-6}$ $\Delta\rho$/psi, respectively.

3.4.2.3 Moderator Void Coefficient

During full power operation, some local boiling occurs resulting in a predicted average void fraction in the moderator of about one-fourth of 1 percent. Changes in reactivity are associated with the appearance of these voids in the moderator and are reflected in the void coefficient of reactivity, α_{void}. The presence of boron has a positive effect on the coefficient since an increase in voids results in a reduction in the boron content in the core. The calculated values at BOC and EOC are -0.18×10^{-3} $\Delta\rho$/% void and -1.46×10^{-3} $\Delta\rho$/% void, respectively.

3.4.2.4 Fuel Temperature Coefficient

The fuel temperature coefficient, α_{fuel} (commonly called the Doppler coefficient), reflects the change of core reactivity with fuel temperature. The effect may be broken into two parts, the thermal and the epithermal (Doppler) contributions. The thermal contribution is due to hardening of the spectrum as the temperature increases. The epithermal contribution is the temperature dependence of the resonance escape probability, which in turn is physically due to Doppler broadening of the resonances in U-238.

The variation in fuel coefficient over the fuel cycle is small. The hot full power coefficient is -1.67×10^{-5} $\Delta\rho/°F$ at BOC and -1.73×10^{-5} $\Delta\rho/°F$ at EOC.

3.4.2.5 Power Coefficient

The power coefficient, α_{power}, is the change in core reactivity per unit change in core power level. All of the previously mentioned coefficients contribute to the α_{power}, but only the moderator temperature coefficient and the fuel temperature coefficient are significant due to the relative magnitudes. To determine the change in reactivity with power, it is necessary to know the change in the weighted average fuel temperature with power. However, the determination of average fuel pellet temperatures is extremely complex. An "effective fuel temperature" may be defined as that temperature which gives the correct fuel temperature and power coefficients when used in a standard design calculation. The method used is contained in Reference 3-1.

This correlation, which is a function of moderator temperature, fuel burnup and local power, is incorporated into the standard design calculations.

3.4.3 Control Element Assembly Worths

Figure 3.4-4 is a schematic of one quadrant of the core cross section, showing the location and the groupings of the 45 trippable and 4 full length non-trippable CEA's. The total worth available from the trippable CEA's and the worth with the highest worth CEA stuck out are given in Table 3.4-6 for beginning and end-of-cycle.

Table 3.4-6 - "Calculated CEA Worths, %Δρ (@ 532°F)"

	Beginning-of-Cycle	End-of-Cycle
All 45 Standard CEA's Inserted	7.73	8.82
44 CEA's Inserted; Highest Worth CEA Stuck Out	6.77	7.46

Table 3.4-7 gives the worth of each group of CEA's relative to the full power condition. These worths are from full out to full in. The CEA withdrawal procedure, meeting the minimum requirements of the PDIL, as shown in Figure 3.4-3 is as follows:

a. With the reactor subcritical, Shutdown Group A is fully withdrawn and then Shutdown Group B is withdrawn;

b. Regulating Groups 1 and 2 are fully withdrawn and Group 3 is withdrawn to at least 20% to take the core critical. Adjustments in dissolved boron concentration are made to maintain Group 3 above the PDIL;

c. Withdrawal of Groups 3 and 4 is made sequentially with the prescribed overlaps and within the specified range until the desired power level and power distribution is achieved.

Table 3.4-7 - "Worth of CEA Groups, %Δρ (@ BOC 532°F)"

When Sequenced as Listed Above*	Worth(%Δρ)
Shutdown CEA's	
Group A	1.85
Group B	2.73
Regulating CEA's	
Group 1	0.90
Group 2	1.49
Group 3	0.75
Group 4	0.48

* The worth listed assumes that Group N is withdrawn after Group B in the sequence.

Adherence to the relationship of power to CEA insertion ensures that acceptable peaking factors are maintained within the bounds assumed for the LSSS and LCO shutdown margin is maintained, and that the potential consequences of a CEA ejection accident are limited to acceptable levels. Operation with the CEA's inserted beyond the PDIL is prevented by the rod block system.

3.4.4 Reactivity Insertion Rates

The maximum rate of reactivity insertion of the regulating groups at full power is 0.02%Δρ/sec. Analyses of CEA withdrawal incidents (Section 14.2) show that no core thermal limits would be exceeded for rates considerably in excess of the above values.

The maximum rate of reactivity insertion due to boron removal by operation of the chemical and volume control system is covered by the low end of the CEA insertion rate spectrum. Adequate time is available to take corrective measures as described in the analysis of the boron dilution incident in Section 14.3.

3.4.5 Power Distribution

The power distribution in the core, and in particular the peak heat flux and enthalpy rise, is of major importance in determining core thermal margin. The maximum expected peaking factors for Cycle 20 are 1.696 for F_r and 2.317 for F_q^T. The COLR limit for F_r^T is 1.732.

The behavior of the gross radial power distribution in the unrodded core through the 20th burnup cycle is shown in Figures 3.4-5, 3.4-6, and 3.4-7. The trend of the overall radial power distribution is to start at the center of the core at BOC, and remain toward the center through EOC.

CEA's are used to a minimum extent and in configurations that will result in a combined radial and axial peaking factor which is within the design limits stated above.

3.4.5.1 Malpositioned CEA's

The two worst cases of a malpositioned CEA were evaluated with respect to permissible operating modes and current administrative guidelines. The two cases evaluated are:

a. The worst case of a CEA left in the core and

b. Insertion of the CEA bank permissible at full power with one CEA left out.

Worst Case of a CEA Left in the Core

Startup or operation with the most reactive CEA left in the core would result in a large distortion in the radial power distribution and consequently excessive peaking. It is not necessary to analyze this event, because the rod block system will prevent its occurrence. This system prevents rod group motion with one CEA position deviating from the groups position by more than 12 inches. Group motion will stop before the 12 inch deviation limit is reached, preventing leaving one rod in and the rest of the group withdrawn. This limit is referenced in Technical Specification 2.10.2(4). In addition, the CEA position sensing system provides alarms from the synchros for deviations of four and eight inches.

If a CEA was not coupled to the rack and pinion drive system, which could occur after refueling, rack movement could indicate no deviation. This condition is assured not to exist by measuring the coupled and uncoupled weights of the letdown tool used in the CEDM coupling procedure. The weight difference between the coupled and uncoupled weights of 55 pounds for a single CEA and 110 pounds for a dual CEA proves that the CEAs are properly coupled. The administrative requirement for performing this test ensures that the CEA's are coupled, eliminating this concern.

Even if a rod was totally inserted with all others withdrawn, the inserted rod would be detected by the incore monitoring system which would cause alarms due to excessive flux peaking in detector locations across the core from the rodded and depressed neutron flux area. A detectable change in flux tilt would also be detected by both the incore and excore detector systems.

Insertion of the CEA Bank Permissible at Full Power
With One CEA Left Out

Insertion of Regulating Group 4 with one of the CEA's left out is prevented from occurring by the rod block circuitry. As in the case of leaving one CEA inserted, group motion will automatically be stopped and prevented prior to the rod-group deviation exceeding 12 inches. In addition, deviation alarms at four and eight inches provide warning of the asymmetry. Therefore, due to the rod block circuitry preventing the occurrence of this condition, no further analysis is required.

3.4.6 Pressure Vessel Fluence

The design of the reactor internals and of the water annulus between the active core and vessel wall is such that for NSSS operations at 1500 MWt and an 80 percent plant capacity factor, the integrated fast neutron fluence at the vessel/clad interface (E>1 MeV) is 3.54×10^{19} n/cm^2 over the 40-year design life of the vessel and corresponds to a fluence of 2.55×10^{19} n/cm^2 at the critical reactor vessel beltline weld (Reference 3-68). The fluence was determined using the threshold detector analysis for the surveillance capsule removed at the end of Cycle 7 (Reference 3-2). The SAND-II and ANISN computer codes were used to calculate the fast fluence at the reactor vessel clad interface.

The SAND-II computer code is used to calculate a neutron flux spectrum from the measured activities of the flux monitors. SAND-II requires an initial flux spectrum estimate; this is calculated using ANISN. The measured activities must be adjusted before they can be put into SAND-II. The various steps of the procedure are described below.

The measured activities must be decay corrected to reactor shutdown. Before being used by SAND-II, the foil activities must be converted to saturated activity with units of disintegrations per second per target atom (dps/a).

For U-238 fission product activities, the required SAND input has dimensions of fissions per second per U-238 atom (fps/a). This is obtained by dividing the saturated activity by the fractional fission yield of the fission product whose activity was measured.

The uranium foil is shielded with cadmium to prevent thermal fissioning in any U-235 impurities. However, the cadmium cover does not prevent fast fissioning in U-235. Therefore, an unshielded uranium foil is included in the flux monitor set. The activity of the unshielded foil can be used to determine the amount of fissioning in the shielded uranium foil caused by U-235. As a result of this calculation, the U-238 fission rate was determined to be 75% of the shielded uranium foil activity.

SAND-II requires an initial estimate of the neutron flux spectrum. This initial estimate was calculated using ANISN, a one-dimensional discrete ordinate code.

The peak fluence at the vessel/clad interface was determined to be 8.8 x 10^{18} n/cm^2 at the end of Cycle 7. To credit the azimuthal flux distribution and low radial leakage symmetric core loading patterns initiated in Cycle 8, conservative flux reduction factors were determined from DOT 4.3 azimuthal flux distribution plots.

No credit was assumed for axial flux distribution or the asymmetric core loading pattern of Cycle 10. Based on these assumptions the peak end of life reactor vessel/clad interface fluence was determined to be 3.54 x 10^{19} n/cm^2. The corresponding fluence to the critical reactor vessel beltline weld material is 2.55 x 10^{19} n/cm^2.

The SAND-II code will give fluxes that are accurate to within ±10% to ±30% if the errors in the measured activities are within similar limits. The 2-sigma uncertainties in the measured activities were less than ±12%. Therefore, it is estimated that the uncertainty in the measured fluence at the surveillance capsule location is ±20% to ±30%. The extrapolated fluence in the vessel will be slightly higher and is estimated to be ±30%.

3.4.7 Nuclear Evaluation

3.4.7.1 Nuclear Design Methods

The nuclear analysis design package developed for use in the design of low enrichment PWR cores is based on a combination of multigroup spectrum calculations, over which cross sections are appropriately averaged to obtain few group constants, and few group, one- two- and three-dimensional diffusion theory calculations of integral and differential reactivity effects and power distributions. The multigroup calculations include spatial effects in those portions of the neutron energy spectrum where volume homogenization is inappropriate, e.g., the thermal neutron energy range. The majority of the calculations are performed with the aid of computer programs embodying analytical procedures and fundamental nuclear data consistent with the current state-of-the-art.

The current design methods involve the use of OPPD metholology and Framatome ANP Richland, Inc. methodology. For transient and setpoint analyses performed by Framatome ANP RIchland, Inc., nuclear design data is provided to bo consistent with their approved methodology. This process has been NRC approved (Reference 3-66). The current list of approved methods are specified in the COLR.

OPPD has incorporated the Studsvik Scandpower, Inc. CMS software package into their nuclear design metholology. With the CMS software package, CASMO-3 is used for generating cross-sections and STIMULATE-3 is used for generating cycle depletions, power distributions, reactivity coefficients, etc. (References 3.10.15 and 3.10.16).

3.4.7.2 Comparisons With Experiments

Reactivity

The Combustion Engineering nuclear design package has been checked against a variety of critical and subcritical experiments. Table 3.4-8 summarizes the properties of the fuel rods employed in the lattices analyzed; Tables 3.4-9 and 3.4-10 summarize certain pertinent characteristics of the lattice and the eigenvalues calculated with the design package.

Table 3.4-8 - "Fuel Rod Description"

Laboratory	Clad OD (in.)	Clad Thickness (in.)	Clad Mat.	Fuel Pellet OD (in.)	Fuel Density (gm/cc)	Fuel Enrichment w/o U-235	w/o PuO_2
B&W	0.4755	0.016	SS 304	0.4440	9.46	4.020	0
B&W	0.4748	0.032	A1 6061	0.4054	10.24	2.459	0
Yankee	0.3383	0.0161	SS 304	0.3000	10.18	2.700	0
Winfrith	0.4301	0.01051	SS 304	0.3984	10.44	3.003	0
Brookhaven	0.499	0.02743	SS 304	0.4441	9.30	3.006	0
Bettis	0.453	0.028	A1	0.3830	10.53	1.311	0
Hanford	0.426	0.027	Zr-2	0.372	9.646*	0.22	1.50
Battelle N. W. Westinghouse	0.568	0.030	Zr-4	0.508	9.869*	0.72	2.20

* effective fuel density

Table 3.4-9 - "Results of Analysis of Critical and Subcritical UO_2 Systems"

	Lattice	Pitch w/o U-235	(in.)	Boron $H_2O/UO2$	(ppm)	K_{eff}	Ref
B&W-1273	1	4.020	0.595	1.137	0	0.9998	3-7
	2	4.020	0.595	1.137	3390	1.0018	3-7
	3	4.020	0.571	0.956	0	0.9963	3-7
	4	2.459	0.595	1.371	0	1.0009	3-7
	5	2.459	0.595	1.371	1675	1.0016	3-7
B&W-3467	6	2.459	0.644	1.846	0	1.0004	3-8
	7	2.459	0.644	1.846	864	1.0014	3-8
	8	2.459	0.644	1.846	1536	0.9997	3-8
Yankee	9	2.700	0.405	1.048	0	0.9965	3-9
	10	2.700	0.435	1.405	0	0.9979	3-9
	11	2.700	0.470	1.853	0	0.9990	3-9
	12	2.700	0.493	2.166	0	1.0004	3-10
Winfrith	13 (20°C)	3.003	0.520	1.001	0	0.9987	3-11
	14 (80°C)	3.003	0.520	1.001	0	0.9977	3-11
	15	3.003	0.735	3.164	0	1.0009	3-11
	16	3.003	0.492	0.779	0	0.9992	3-11
Bettis	17	1.311	0.6133=	1.429	0	0.9963	3-12
	18	1.311	0.6133=	1.429	0	0.9963	3-12
	19	1.311	0.6133=	1.429	0	0.9970	3-12
	20	1.311	0.6504=	1.781	0	0.9962	3-12
	21	1.311	0.6504=	1.781	0	0.9975	3-12
	22	1.311	0.7110=	2.401	0	0.9968	3-12
	23	1.311	0.7110=	2.401	0	0.9975	3-12
BNL(a)	24	3.006	0.6767=	1.319	0	0.9997	3-13
	25	3.006	0.6767=	1.319	1363	0.9932	3-13
	26	3.006	0.7163=	1.632	0	0.9964	3-13
	27	3.006	0.7163=	1.632	470	0.9950	3-13
	28	3.006	0.7163=	1.632	992	0.9931	3-13
	29	3.006	0.7163=	1.632	1345	0.9940	3-13
	30	3.006	0.7706=	2.091	0	0.9981	3-13
	31	3.006	0.7706=	2.091	1141	0.9931	3-13
B&W(a)	32 (66°F)	4.020	0.595	1.137	0	1.0046	3-14
	33 (103°F)	4.020	0.595	1.137	0	1.0036	3-14
	34 (203°F)	4.020	0.595	1.137	0	1.0003	3-14
	35 (308°F)	4.020	0.595	1.137	0	0.9992	3-14
	36 (406°F)	4.020	0.595	1.137	0	1.0010	3-14

= Triangular Pitch
(a) Subcritical Measurements

Table 3.4-10 - "Results of Analysis of PuO_2-U02 Fueled Lattices"

		Pitch		Boron			
Lattice	w/o U-235	w/o Pu-02	(in.)	H_2O/Fuel	(ppm)	K_{eff}	Ref
Hanford	0.22	1.50	0.55=	1.099	0	1.0027	3-15
			0.60=	1.557	0	1.0056	3-15
			0.71=	2.705	0	1.0108	3-15
			0.80=	3.788	0	1.0094	3-15
BNWL	0.72	2.2(1)	0.85=	1.837	0	1.0056	3-16
			0.93=	2.445	0	1.0099	3-16
WCAP	0.72	2.2(1)	0.69	1.099	0	0.9994	3-17
			0.75	1.525	0	1.0058	3-17
			0.69	1.099	261	0.9998	3-17
			0.9758	3.448	261	1.0122	3-17
			0.69	1.099	526	1.0005	3-17
			0.9758	3.448	526	1.0099	3-17
BNWL	0.72	2.2(2)	0.93=	2.445	0	1.0112	3-16
			1.05=	3.461	0	1.0068	3-16
BNWL	0.72	2.2(3)	0.85=	1.837	0	1.0113	3-16
			0.93=	2.445	0	1.0123	3-16
WCAP	0.72	2.2(3)	0.9758	3.448	0	1.0206	3-16

= Triangular Pitch
(1) 7.654 w/o Pu-240 in Pu
(2) 16.54 w/o Pu-240 in Pu
(3) 23.503 w/o Pu-240 in Pu

The average eigenvalue for the critical uranium lattices in Table 3.4-9 (numbers 1 through 23) is 0.9987 $\pm$ 0.0019 and for the mixed oxide lattices of Table 3.4-10 the corresponding number is 1.00799 $\pm$ .0053. The UO_2 experiments cover a wide range of core dimensions, boron concentrations, temperature, enrichment, water-to-fuel ratios, and clad materials, thus giving confidence in the validity of the design package to predict beginning-of-life fuel properties with an acceptable accuracy. The analysis of the mixed oxided lattices exhibits larger deviations than for the UO_2 lattices; this result is not surprising in view of the limited amount of data compared with UO_2 systems, the relatively large experimental bucklings, and uncertainties in the same.

The rods-out, beginning-of-cycle, cold and hot zero power reactivities of the Obrigheim (Reference 3-13) and Connecticut Yankee (Reference 3-14) reactors were also calculated to demonstrate the validity of the model in large multiregion cores.

The results are summarized here:

	Reactor	Temperature	Boron (ppm)	K_{eff}
(a)	Obrigheim	cold	1727	0.9964
		hot	1962	0.9989
(b)	Connecticut Yankee	260°F	2040	1.0025
		560°F	2305	1.0002

Table 3.4-11 summarizes the predicted and measured values of the BOC hot zero power critical boron concentrations, with all-rods-out (ARO), for Cycles 1 through 19. This table shows excellent agreement between the two values for each cycle.

Table 3.4-11 - "BOC HZP Critical Boron Concentration (ARO)"

Cycle	Predicted C_B (ppm)	Measured C_B (ppm)
1	911	933
2	1248	1240
3	964	996
4	1023	1027
5	1235	1242
6	1230	1230
7	1240	1241
8	1239	1240
9	1520	1518
10	1481	1474
11	1502	1502
12	1510	1507
13	1560	1568
14	1201	1182
15	1392	1411
16	1534	1546
17	1592	1621
18	1581	1608
19	1512	1552

Depletion Calculation

Over 50 spent fuel samples from Yankee Core I were subjected to isotopic and radio-chemical analyses which were performed in the Tracerlab Laboratory at Richmond, California and by the Vallecitos Atomic Laboratory of the General Electric Company (Reference 3-15). Depletion calculations were performed on the Yankee core for comparison with the above measurements. Figure 3.4-9 compares measured and calculated values of the Pu/U mass ratio versus exposure, and Figure 3.4-10 shows a comparison for the relative isotopic composition of plutonium as a function of fractional U-235 depletion.

The inventory changes for the 74 fuel assemblies from Yankee Core I are compared with measured results (Reference 3-16) in Table 3.4-12; the calculations were carried out using both one-dimensional and three- dimensional (RZ) representations.

Table 3.4-12 - "Inventory Change Comparison"

	U-235 Dep. (kg)	Total Pu. (kg)	Fiss. Pu. (kg)	Fissile Consumption (g/MWd)
NFS Meas.	171.0±4.7	91.1±1.0	80.27±0.88	0.535±0.028
1-D	170.8	91.0	80.88	0.530
3-D (RZ)	169.0	89.9	79.48	0.528

Doppler and Power Coefficient

The Doppler coefficient of reactivity is due to Doppler broadening of the U-238 resonances with increasing fuel temperature. The power coefficient of reactivity is the change in reactivity associated with the Doppler and moderator coefficients as a function of power. The fuel temperature used to calculate the Doppler coefficient as a function of the core average power level and coolant temperature is determined on the basis of the Reference 3-1 model. Table 3.4-13 shows a comparison between the predicted and measured power coefficients, beginning with Cycles 2 through 19. All of the pairs of measured and predicted values show good agreement.

Table 3.4-13 - "Comparison of Predicted and Measured Power Coefficients"

Cycle	Burnup MWD/MTU	Percent of Rated Power	Critical Boron Concentration	Predicted Power Coefficient ($\Delta\rho$/% Power)	Measured Power Coefficient ($\Delta\rho$/% Power)
2	10877	46(1)	104	-1.70×10^{-4}	-1.95×10^{-4}
3	157	46(1)	720	-1.60×10^{-4}	-1.47×10^{-4}
3	1513	90(1)	535	-1.20×10^{-4}	-1.12×10^{-4}
3	4183	90(1)	309	-1.26×10^{-4}	-1.31×10^{-4}
3	7208	90(1)	62	-1.74×10^{-4}	-1.48×10^{-4}
4	267	92(1)	690	-1.06×10^{-4}	-1.04×10^{-4}
4	4690	94(1)	288	-1.31×10^{-4}	-1.12×10^{-4}
4	8027	95(1)	44	-1.52×10^{-4}	-1.10×10^{-4}
5	426	93(1)	876	-0.65×10^{-4}	-1.05×10^{-4}
5	6815	94(1)	296	-1.33×10^{-4}	-1.25×10^{-4}
6	400	95(1)	848	-1.18×10^{-4}	-1.11×10^{-4}
6	6467	96(2)	307	-1.53×10^{-4}	-1.45×10^{-4}
7	450	96(2)	817	-1.20×10^{-4}	-0.98×10^{-4}
7	6900	95(2)	283	-1.45×10^{-4}	-1.30×10^{-4}
7	7800	95(2)	192	-1.52×10^{-4}	-1.57×10^{-4}
8	459	79(2)	817	-1.18×10^{-4}	-1.18×10^{-4}
8	6150	95(2)	292	-1.41×10^{-4}	-1.70×10^{-4}
9	420	95(2)	1036	-0.86×10^{-4}	-1.64×10^{-4}
9	9663	96(2)	300	-1.35×10^{-4}	-1.57×10^{-4}
10	583	95(2)	1017	-0.90×10^{-4}	-1.24×10^{-4}
10	9261	95(2)	302	-1.39×10^{-4}	-1.40×10^{-4}
11	433	93(2)	1073	-0.91×10^{-4}	-0.95×10^{-4}
11	9765	95(2)	301	-1.41×10^{-4}	-1.52×10^{-4}
12	425	93(2)	1050	-0.95×10^{-4}	-1.42×10^{-4}
12	9691	95(2)	309	-1.43×10^{-4}	-1.63×10^{-4}
13	373	94(2)	1113	-0.91×10^{-4}	-1.26×10^{-4}
13	10694	95(2)	325	-1.52×10^{-4}	-1.51×10^{-4}
14	355	90(2)	768	-1.25×10^{-4}	-1.55×10^{-4}
14	10559	94(2)	319	-1.73×10^{-4}	-1.77×10^{-4}
15	340	95(2)	948	-1.05×10^{-4}	-1.57×10^{-4}
15	10180	94(2)	303	-1.48×10^{-4}	-2.13×10^{-4}
16	400	95(2)	1088	-0.95×10^{-4}	-1.54×10^{-4}
16	11,350	95(2)	315	-1.46×10^{-4}	-2.08×10^{-4}
17	237	92(2)	1104	-1.06×10^{-4}	-1.70×10^{-4}
17	10,550	95(2)	311	-1.53×10^{-4}	-2.27×10^{-4}
18	420	95(2)	1102	-0.99×10^{-4}	-1.66×10^{-4}
18	11,806	97(2)	287	-1.40×10^{-4}	-2.14×10^{-4}
19	368	92(2)	1033	-0.97×10^{-4}	-1.47×10^{-4}

Moderator Temperature Coefficient

The moderator temperature coefficient (MTC) for Fort Calhoun has been measured at both BOC and EOC (approximate), for full power conditions, beginning with Cycle 1. Table 3.4-14 summarizes the measurements and predictions which show good agreement for all cycles.

Power Distributions

Comparisons between predicted and measured power distributions using the design methodology of Combustion Engineering are contained in Reference 3-17.

Table 3.4-14 - "Comparison of Predicted and Measured Moderator Temperature Coefficients"

	BOC				EOC			
Cycle	Percent of Rated Power	Critical Boron Concentration (ppm)	Predicted MTC (Δρ/°F)	Measured MTC (Δρ/°F)	Percent of Rated	Critical Boron Concentration (ppm)	Predicted MTC (Δρ/°F)	Measured MTC (Δρ/°F)
1	-	-	-	-	75(1)	239	-1.02×10^{-4}	-0.98×10^{-4}
2	69(1)	927	-0.27×10^{-4}	-0.28×10^{-4}	46(1)	104	-1.66×10^{-4}	-1.62×10^{-4}
3	46(1)	720	-1.04×10^{-4}	-0.41×10^{-4}	90(1)	62	-2.04×10^{-4}	-1.65×10^{-4}
4	92(1)	690	-0.66×10^{-4}	-0.42×10^{-4}	95(1)	44	-2.16×10^{-4}	-1.41×10^{-4}
5	93(1)	876	-0.64×10^{-4}	-0.19×10^{-4}	94(1)	296	-1.33×10^{-4}	-0.97×10^{-4}
6	95(1)	848	-0.61×10^{-4}	-0.34×10^{-4}	96(2)	307	-1.79×10^{-4}	-1.38×10^{-4}
7	96(2)	817	-0.61×10^{-4}	-0.40×10^{-4}	95(2)(3)	192	-1.79×10^{-4}	-1.79×10^{-4}
8	80(2)	817	-0.84×10^{-4}	-0.74×10^{-4}	95(2)	292	-1.66×10^{-4}	-1.76×10^{-4}

(1) Full Rated Power = 1420 MWt
(2) Full Rated Power = 1500 Mwt
(3) Predicted

Table 3.4-14 - "Comparison of Predicted and Measured Moderator Temperature Coefficients" (continued)

	BOC				EOC			
Cycle	Percent of Rated Power	Critical Boron Concentration (ppm)	Predicted MTC ($\Delta\rho/°F$)	Measured MTC ($\Delta\rho/°F$)	Percent of Rated	Critical Boron Concentration (ppm)	Predicted MTC ($\Delta\rho/°F$)	Measured MTC ($\Delta\rho/°F$)
9	95[2]	1036	-0.32×10^{-4}	-0.29×10^{-4}	96[2]	300	-1.53×10^{-4}	-1.37×10^{-4}
10	95[2]	1017	-0.41×10^{-4}	-0.36×10^{-4}	95[2]	302	-1.63×10^{-4}	-1.41×10^{-4}
11	93[2]	1073	-0.39×10^{-4}	-0.31×10^{-4}	95[2]	301	-1.65×10^{-4}	-1.47×10^{-4}
12	93[2]	1050	-0.39×10^{-4}	-0.41×10^{-4}	95[2]	309	-1.68×10^{-4}	-1.66×10^{-4}
13	94[2]	1113	-0.27×10^{-4}	-0.39×10^{-4}	95[2]	325	-1.71×10^{-4}	-1.58×10^{-4}
14	90[2]	768	-0.81×10^{-4}	-0.74×10^{-4}	94[2]	319	-1.80×10^{-4}	-1.67×10^{-4}
15	95[2]	939	$-0.57 \text{ X } 10^{-4}$	$-0.68 \text{ X } 10^{-4}$	94[2]	303	-1.77×10^{-4}	-1.89×10^{-4}
16	95[2]	1088	-0.44×10^{-4}	-0.55×10^{-4}	95[2]	315	-1.804×10^{-4}	-2.08×10^{-4}
17	92[2]	1104	-0.66×10^{-4}	-0.79×10^{-4}	95[2]	311	-2.15×10^{-4}	-2.23×10^{-4}
18	95[2]	1102	-0.63×10^{-4}	-0.67×10^{-4}	97[2]	287	-2.15×10^{-4}	-2.08×10^{-4}
19	92[2]	1033	-0.74×10^{-4}	-0.83×10^{-4}	-	-	-	-

(1) Full Rated Power = 1420 MWt
(2) Full Rated Power = 1500 Mwt
(3) Predicted

3.4.8 Reactor Stability

Xenon stability analyses on the Fort Calhoun core indicate that any radial and azimuthal xenon oscillations induced in the core will be damped, but that the core could exhibit instabilities with respect to axial xenon oscillations during certain portions of the burnup cycle, in the absence of appropriate control action. Before discussing the methods of analysis employed to obtain these predictions, it is appropriate to reiterate several important aspects of the xenon oscillation problem.

a. The time scale on which the oscillations occur is long, and any induced oscillations typically exhibit a period of 30 to 50 hours;

b. Xenon oscillations are detectable as discussed below;

c. As long as the initial power peak associated with the perturbation initiating the oscillation is acceptable, the operator has time in the order of from hours to days to decide upon and to take appropriate remedial action prior to the time when allowable peaking factors would be exceeded.

3.4.8.1 Method of Analysis

The classic method for assessing spatial xenon oscillations is that developed by Randall and St. John (Reference 3-18) which consists of expanding small perturbations of the flux and xenon concentrations about equilibrium values in eigenfunctions of the system with equilibrium xenon present. While the Randall-St. John technique is correct only for a uniform unreflected system, its use of the separations between the eigenvalues of the various excited states of the system and the eigenvalue of the fundamental state is helpful in directing attention to which of the various excited states are the most likely to occur. As indicated in Figure 3.4-11, the first axial mode, which has the minimum eigenvalue separation from fundamental mode, is the most likely to occur, and the higher modes would have, on the basis of this simple theory, the indicated relative likelihoods of occurrence.

However, it is necessary to extend this simpler linear analysis to treat cores which are non-uniform because of fuel zoning, depletion and CEA patterns, for example. Such extensions have been worked out and are reported in References 3-19 and 3-20. In this extension, the eigenvalue separations between the excited state of interest and the fundamental are computed numerically for symmetrical flux shapes. For nonsymmetrical flux shapes, the eigenvalue separation can usually be obtained indirectly from the dominance ratio $\lambda 1/\lambda 0$, computed during the iteration cycle of the machine spatial calculation.

In making the analysis, numerical space-time calculations are performed in the required number of spatial dimensions for the various modes as checkpoints for the predictions of the extended Randall-St. John treatment described above.

3.4.8.2 Radial Mode Oscillations

From the remote position of the first radial excited eigenvalue in Figure 3.4-11 (over 4 percent in λ), it is expected that such oscillations would be rapidly damped even in a core whose power was flattened for example, by enrichment zoning. To confirm that this mode is extremely stable, a space-time calculation was run for a reflected, zone core 11 feet in diameter without including the damping effects of the negative power coefficient. The initial perturbation was a poison worth 0.4 percent in reactivity placed in the central 20 percent of the core for 1 hour. Following removal of the perturbation, the resulting oscillation was followed in 4-hour time steps for a period of 80 hours. As shown in Figure 3.4-12, the resulting oscillation died out very rapidly with a damping factor of about -0.06 per hour. If this damping coefficient is corrected for a finite time mesh by the formula in Reference 3-21, it would become even more strongly convergent. On this basis, it is concluded that radial oscillations are highly unlikely.

This conclusion is of particular significance because it means that there is no type of oscillation where the inner portions of the core act independently of the peripheral portions of the core whose behavior is most closely followed by the out-of-core flux detectors. As will be noted later, primary reliance is placed on these for the detection of any xenon oscillations.

3.4.8.3 Azimuthal Mode Oscillations

Azimuthal oscillations in an unreflected uniform reactor are less likely than axial mode oscillations as indicated in Figure 3.4-11. The situation is quite different in a radially power-flattened reflected core even at beginning of cycle, as shown in Figure 3.4-13. Here, the eigenvalue separations for the actual core are predicted by the modified Randall-St. John treatment and include the effects of power flattening. On the basis of this information, it appears that the azimuthal mode is the most easily excited at beginning of life even though the axial mode becomes the most unstable later.

With reference to Figure 3.4-13, it is indicated that the eigenvalue separation between the first azimuthal harmonic and the fundamental is about 1.2 percent in λ. Although the axial oscillations were found to be relatively insensitive to the moderator temperature feedback because of the constant power condition, the azimuthal modes should be stabilized appreciably by the negative moderator coefficient. Furthermore, the Doppler coefficient applicable to the Fort Calhoun reactor is calculated to be approximately -1.35×10^{-3} $\Delta\rho$/kW-ft^{-1}, which is more than enough to ensure stability of all the azimuthal modes.

3.4.8.4 Axial Mode Oscillations

As checkpoints for the predictions of the modified Randall-St John approach, numerical spatial time calculations have been performed for the axial case at both beginning and end of cycle. The fuel and poison distributions were obtained by depletion with soluble boron control so that, although the power distribution was strongly flattened, it was still symmetric about the core midplane. Spatial Doppler feedback was included in these calculations. In Figure 3.4-14 the time variation of the thermal neutron flux is shown for two points along the core axis near end of life with Doppler feedback. The initial perturbation used to excite the oscillations was a 20 percent insertion into the top of the reactor of a 1.5 percent reactivity CEA bank for 1 hour. As is indicated, the damping factor for this case was about +0.02 per hour. When corrected for finite time mesh by the methods of Reference 3-21, however, the damping factor is approximately +0.05.

When this damping factor is plotted on Figure 3.4-13 at the appropriate eigenvalue separation for this mode at end of cycle, it is apparent that good agreement is obtained with the modified Randall-St. John prediction.

At beginning of cycle, the space-time calculations indicated a positive damping coefficient of about +0.04 per hour in the absence of spatial Doppler feedback, and a negative damping coefficient of -0.05 per hour results with a power coefficient of -1.35×10^{-3} $\Delta\rho$/kW-ft^{-1}. Again, these space-time results are in excellent agreement with the predictions of the modified Randall-St. John technique.

Calculations performed with both Doppler and moderator feedback have resulted in damping factors which were essentially the same as those obtained with Doppler feedback alone. This result suggests that the constant power condition which applies to the axial oscillations results in a very weak moderator feedback since the moderator density is fixed at the top and bottom of the core and only the density distribution in between can change.

For the estimated Doppler coefficient of -1.35×10^{-3} $\Delta\rho$/kW-ft^{-1} (see Section 3.4.2.4) it can be seen from Figure 3.4-12 that the damping factor toward end of the burnup cycle is positive; thus within the uncertainties in predicting power coefficients and uncertainties in the analysis, there is a possibility of unstable axial xenon oscillations in the absence of any control action. These oscillations, however, are sufficiently slow, (doubling time of 14 hours with a damping factor of +0.05 hr^{-1}, detected as outlined below), that there would be sufficient time to institute corrective action.

3.4.8.5 Detection of Xenon Oscillations

Primary reliance for the detection of any xenon oscillations is placed on the out-of-core flux monitoring instrumentation, one channel of which per quadrant is an axially split ionization detector. As indicated earlier, oscillations in modes such as the radial, which would allow the center of the core to behave independently from the peripheral portions of the core, are highly unlikely and this lends support to reliance on the out-of-core detectors for this purpose.

Furthermore, as an example of the ability of the axially split out-of-core detectors to respond to axial flux tilts (i.e., axial shape index (ASI)) in the core, Figure 3.4-15 indicates the ratio of the lower half of the axially split detector signal to the signal from the upper half for two different power distributions; one is axially symmetric, the other contains a strong contribution from the first axial harmonic and has a peaking factor of about 1.8. In the latter case, the signal from the lower half of the detector is 50 percent higher than that from the upper half.

Considering that the primary response of these detectors will be to the power in the peripheral fuel assemblies, but noting that the lower modes of any induced oscillations will affect the power shapes in these peripheral assemblies, it has been concluded that any flux tilts can be observed and identified by the use of out-of-core instrumentation to provide data upon which appropriate remedial action can be based.

The incore or core average ASI, Y_I is related to the excore detector ASI, Y_E, by the Shape Annealing Factor (SAF) in the following equation.

$$Y_I = SAF * Y_E + bias$$

The SAF is a function of excore detector geometry and is determined by reactor physics testing. The incore detectors (see Section 3.6.1) were utilized to compute the core average ASI, the excore detector signals were used to calculate excore ASI and the equation was solved for the SAF during either a power change or a controlled axial xenon oscillation. The core average ASI is computed by the incore detectors during normal operation. This value is compared to Y_I computed by the excore detectors on a periodic basis and if the difference between the two exceeds a prescribed limit the split excore detectors are calibrated to assure that the "correct" value of Y_I is calculated.

3.4.8.6 Control of Xenon Oscillations

The split detectors of the power range safety and control channels are used to calculate the Axial Shape Index, ASI, which is defined as the ratio of the difference and sum of the signals from the lower and upper detectors respectively. Three separate limits have been established for allowable ASI as a function of reactor power. These functions allow the axial peaks to increase as reactor power decreases. The first and most restrictive of these limits is the maintenance of the ASI around an Equilibrium Shape Index (ESI). The ESI is defined as the ASI when the core is at a constant power level with an equilibrium xenon concentration and all CEA's removed from the core. The operator is to maintain the ASI within a given band using the Bank 4 CEA's for fuel performance considerations.

The second limit is a Limiting Condition for Operation (LCO) based either on DNBR or peak linear heat generation rate. Since the peak linear heat generation rate is usually monitored by the incore detectors the DNBR LCO defines the ASI limit during normal operation. The Technical Specifications state that if the ASI exceeds the DNBR LCO, it is to be restored to within the limits in two hours or take the reactor to less than 15% of rated power in the next eight hours. The operator is to utilize the Bank 4 CEA's to maintain the ASI within limits.

The last and least restrictive limit on ASI as a function of power is the Reactor Protective System Axial Power Distribution protection channels. Each independent channel compares the observed ASI with the ASI limit. A trip is initiated on two out of four logic if the ASI exceeds the limit. The limit is derived through consideration of the DNBR and the peak linear heat generation rate for various ASI's.

3.4.8.7 Xenon Oscillation Operating Experience

Section 3.4.8.4 discusses the theoretical possibility of unstable axial xenon oscillations in the absence of any control action. During the operation of the Fort Calhoun Station from August, 1973 through December, 2000, no unstable axial xenon oscillations have been observed. Near end of cycle stable axial xenon oscillations with slightly positive damping factors have been observed. Half cycle damping techniques utilizing Bank 4 CEA's have been successfully utilized to control these oscillations.

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3.5 CYCLE 6 CORE POWER UPRATE TO 1500 MWt

Thermal And Hydraulic Design And Evaluation Of Selected Transients

The thermal and hydraulic design of the reactor has as its primary objective the assurance that the core can meet normal steady state and transient performance requirements without exceeding thermal and hydraulic design limits. This section is a historical discussion of the thermal and hydraulic characteristics that relate those transients that were analyzed when the core power rating was changed from 1420 MWt to 1500 MWt. The transients included in this analysis can be found in Section 14 and are as follows:

14.9 Loss of Load
Loss of Load to Both Steam Generators

14.10 Malfunctions of the Feedwater System

14.11 Excess Load

The thermal and hydraulic design was based on a limiting minimum departure from nucleate boiling ratio of 1.3 as calculated using the W-3 correlation. To ensure that this limit is not exceeded, the reactor protective system is designed to trip the reactor before this condition can be achieved.

This section also discusses the fuel pellet performance characteristics that relate the reactor performance to the margin to design limits. The fuel pellet performance design limit ensures that fuel pellet centerline melt does not occur. The fuel centerline melt design criterion is based on maintaining the peak linear heat rate below a prescribed limit of 21 kw/ft. To ensure that this limit is not exceeded, the reactor protective system is designed to trip the reactor before this condition can be achieved.

A summary of the historical thermal and hydraulic parameters for the core power uprate to 1500 MWt is presented in Table 3.5-1.

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Table 3.5-1 - "Thermal and Hydraulic Parameters"

General Characteristics at Full Power	
Total Heat Output, MW	1500
Btu/hr	5120×10^6
Heat Generated in Fuel, Fraction	0.975
Pressurizer Pressure	
Nominal, psia	2,100
Minimum in Steady State, psia	2,075
Maximum in Steady State, psia	2,150
Nominal Coolant Inlet Temperature, °F	545
Design Inlet Temperature, Steady State, °F	547
Nominal Vessel Outlet Temperature, °F	596.5
Nominal Core Bulk Outlet Temperature, °F	599.5
Total Reactor Coolant Flow, lb/hr	71.7×10^6
Coolant Flow Through Core, lb/hr	68.5×10^6
Hydraulic Diameter Nominal Channel, ft	0.0436
Average Mass Velocity, lb/hr-ft^2	2.16×10^6
Average Coolant Velocity In-Core, ft/sec	12.7
Core Average Heat Flux, Btu/hr-ft^2	177,530
Total Heat Transfer Area, ft^2	28,840
Average Linear Heat Rate of Rod, kW/ft	6.01
Design Overpower, %	112
Average Core Enthalpy Rise, 100% Power, Btu/lb	72.6
Limiting Assembly Peaking	
Engineering Heat Flux Factor	1.03
Planar Radial Peaking Factor	1.60
Axial Peaking Factor	1.52
Total Nuclear Peaking Factor	2.50
Enthalpy Rise Factors, Nominal Coolant Conditions, Hot Channel	
Heat Input Factors	
Engineering Factor on Hot Channel Heat Input	1.03
Flow Factors	
Inlet Plenum Maldistribution	1.05
Total Flow Factor	1.05

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3.5.1 Plant Parameter Variations

Normal reactor operation includes both the nominal steady state design conditions and variations from these conditions during expected operating transients. Instrument and control errors are taken into account in the analysis of transients by setting the initial conditions at the most adverse values within the steady state operating envelope. Delays between parameter changes, trip signals and initiation of CEA movement are made a part of the transient calculations. Values of plant parameters are shown in Table 3.5-2 for the nominal, steady state design and reactor trip conditions.

Table 3.5-2 - "Plant Parameters for Thermal and Hydraulic Design, Steady State"

	Nominal	Design (Steady State)	Reactor Trip Condition
Pressure, psia	2100	2150 Max 2075 Min	2400 Max 1750 Min
Vessel Inlet Temperature, °F	545	547	--
Vessel Outlet Temperature, °F	596.5	599.5	--
Flow Rate, lb/hr x 10^6	71.7	71.7	66.7 Min
Reactor Power, %	100	100	112 Max

The plant parameters for reactor trip conditions as shown in Table 3.5-2 are based on the automatic protection set point being at the adverse value while the other plant parameters are at the nominal value. The maximum overpressure trip setpoint is 2400 psia. The minimum pressure at which a thermal margin trip will be actuated is 1750 psia. The maximum vessel outlet temperature when an overpower trip occurs (at 112 percent power), is 605°F for an inlet temperature of 547°F. The minimum flow rate at which a low flow trip occurs is 93 percent and the maximum overpower trip setpoint is 112 percent. All of the above trip setpoints are discussed in more detail in Section 7.2.

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3.5.2 Hot Channel Factors

3.5.2.1 Description of Hot Channel Factors

The heat flux hot channel factor is the ratio of maximum heat flux in the core to the average heat flux, and the enthalpy rise hot channel factor is the ratio of enthalpy rise in the hot channel to the core average enthalpy rise. Each of these factors is customarily divided into subfactors to account for specific physical effects. A subfactor is identified as a nuclear or as an engineering factor.

Engineering factors account for physical differences between the hot channel and a nominal channel, other than those differences due to nuclear effects. The engineering hot channel factors can be further classified as statistical or nonstatistical factors. Statistical factors are those that result from the effects of manufacturing tolerances on heat flux or enthalpy rise. They are termed statistical factors because manufacturing tolerances are randomly distributed about a mean value. It is assumed that the functional combination of tolerance data into a subfactor results in a normally distributed value for the subfactor. This assumption is reasonable for the small tolerance deviations in fuel assemblies. Nonstatistical engineering factors are those that are due to known physical effects that can be measured or calculated.

Nuclear Power Factor

The nuclear heat flux factor relates the peak heat flux in the core to the core average heat flux. It is the maximum value of the product of the nuclear enthalpy factor, the rod-to-channel factor and the axial peaking factor. A design value of 2.50 is established for this factor. The core average heat flux is reduced by 2.5 percent from that obtained from total core power and total heat transfer area to account for heat generated in the moderator.

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Engineering Heat Flux Factor

The effect on local heat flux of deviations from nominal design dimensions and specifications is accounted for by the engineering heat flux factor. Design variables that contribute to this factor are fuel density, fuel enrichment, pellet diameter, and clad outside diameter. These variables may be combined statistically to obtain the engineering heat flux factor. A design value of 1.03 is used for the engineering heat flux factor.

Engineering Enthalpy Rise Factor

The engineering enthalpy rise factor accounts for the effects of deviations in fuel fabrication from nominal dimensions or specifications on the enthalpy rise in the hot channel. Tolerance deviations (averaged over the length of the four fuel rods that enclose the hot channel) for fuel density, fuel enrichment, pellet diameter, and clad outside diameter, contribute to this factor.

The engineering enthalpy rise factor accounts for increased heat input resulting from higher-than-nominal U-235 content. Because of the difficulty in evaluating average pellet tolerance variations for groups of four fuel rods, the enthalpy rise factor is conservatively assumed to be equal to the engineering heat flux factor.

Inlet Flow Distribution Factor

The inlet flow distribution factor accounts for the effects of nonuniform flow at the core inlet on the hot channel enthalpy rise. The latest hydraulic analysis was based on the value of 1.05. This value is conservative with respect to the value of 1.03 which was derived from flow model tests using a one-fourth scale model of the reactor. Details of the flow model program are given in Section 1.4.6.

The evaluation of the core thermal-hydraulic performances was based on the minimum flow to the highest powered assembly. This, in effect reduces the flow in the hot region of the core by 5% and increases the flow to the cold region of the core by 5%.

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3.5.2.2 Summary of Hot Channel Factors

Table 3.5-3 presents a summary of the hot channel factors for nominal, design, and hot channel conditions for the core power uprate to 1500 MW_t.

Table 3.5-3 - "Summary of Hot Channel Factors"

	Nominal	Design	Hot Channel
Heat Flux Factors			
Nuclear Heat Flux Factor	2.43	2.50	2.50
Engineering Heat Flux Factor	1.0	1.0	1.03
Total Heat Flux Factor	2.43	2.50	2.58
Enthalpy Rise Factor			
Engineering Enthalpy Rise Factor	1.0	1.0	1.03
Inlet Flow Distribution Factor	1.05	1.05	1.05
Total Enthalpy Rise Factor at Nominal Conditions	1.05	1.05	1.08

3.5.3 Coolant Flow

3.5.3.1 Total Coolant Flow Rate and Bypass Flow

The minimum total coolant flow rate at full power is 71.7 x 10^6 lb/hr. The coolant flow path can be traced in Figure 3.1-1. Coolant enters the four inlet nozzles and flows into the annular plenum between the reactor vessel and the core support barrel. It then flows down on both sides of the thermal shield and through the flow skirt to the plenum below the core lower support structure. Pressure losses in the skirt and lower support structure help to even out the inlet flow distribution to the core. The coolant passes through the openings in the lower core plate and flows axially upward through the fuel assemblies. A portion flows through the lower core plate and into the guide tubes in the fuel assemblies. Flow limiting devices have been incorporated in the guide tubes of fuel assemblies without CEA's to limit bypass flow when these fuel assemblies are placed under spare CEA locations.

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After passing through the core, the coolant flows into the region outside the control element assembly shrouds. From this region the coolant flows across the control element assembly shrouds and passes out through the outlet sleeves on the core barrel to the outlet nozzles.

The principal core bypass routes are direct inlet-to-outlet coolant flow at the joint between the core support barrel sleeve and the outlet nozzle and the flow in the reflector region in excess of that required for cooling. The design limits the total guide tube flow and core bypass to a maximum of 3.2×10^6 lb/hr, yielding a core flow rate of 68.5×10^6 lb/hr. Some internal leakage occurs within the core and is included in the 68.5×10^6 lb/hr flow rate.

The coolant required to cool the control elements flows in the annulus between the control element and the guide tube and then into the region outside the control element assembly shrouds. A similar but smaller leakage will occur at the upper end of those guide tubes without control elements.

3.5.3.2 Pressure Drop

At the design flow rate of 71.7×10^6 lb per hour and an inlet temperature of 534.6°F, the best estimate of irrecoverable pressure loss from inlet to outlet nozzles is 23.4 psi. Table 3.5-4 is a tabulation of the pressure drops and velocities for various segments along the inlet-to-outlet nozzle flow path.

These individual pressure drops were obtained using measured loss coefficients from the one-fourth scale airflow model of the Fort Calhoun reactor with appropriate Reynolds number corrections where necessary (see Section 1.4.6). The upper limit overall pressure drop, considering experimental uncertainties and adverse tolerances in the as-built reactor, is 29.1 psi for design flow rate and design inlet temperature of 547°F.

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Table 3.5-4 - "Reactor Pressure Drops"

	Velocity (ft/sec)	Pressure Drop (psi)
Inlet Nozzle and 90° Turn	33.2	4.3
Thermal Shield	24.2	2.1
Lower Plenum	11.8	4.4
Core	12.7	7.0
Core Outlet to Outlet Nozzle	40.7	5.6
Total		23.4

3.5.3.3 Partial Flow Loop Operations

There are two steam generators and four reactor coolant pumps which give rise to six possible configurations for operation. At present the Fort Calhoun Nuclear Power Station is only licensed for the normal four pump configuration. In the future, the unit may be licensed for part loop pump configurations.

3.5.4 Subchannel MDNBR Analysis

The basic aims of the subchannel analysis are to evaluate the enthalpy rise in the MDNBR limiting subchannel and to predict the available margin to conditioning which would result in a departure from nucleate boiling (DNB). The subchannel MDNBR analysis resembles the core flow analysis in considering the lateral mixing of coolant between subchannels which results from diversion crossflow and turbulent mixing. Such flow mixing between adjacent subchannels reduces the radial enthalpy gradient across the assembly.

In addition to the flow penalties due to differences in assembly pressure loss coefficients, the subchannel thermal analysis considers the effects of hot channel factors for heat flux and enthalpy, factors which arise from nuclear effects and engineering uncertainties. The individual factors included in the subchannel analysis are:

- Fuel fabrication tolerance (on rod pitch, and rod diameter) which can result in reduced subchannel flow.

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- fabrication tolerances on pellet diameter, density, and enrichment which account for the variation in the quantity of fissionable material in the fuel pellet.

- Inlet flow maldistribution which results in reduction in flow to the hot assembly.

- Flow mixing which accounts for momentum and enthalpy interchange between parallel and laterally open subchannels.

- Heat flux penalties resulting from fuel densification, i.e., increase in linear heat generation rate due to a decrease in active fuel rod length.

The W-3 DNB correlation, with correction factors for both unheated subchannel boundaries and a nonuniform axial heat flux profile (Ref. 3-22 and 3-56), was used to predict the margin to DNB. Reference 3-27 provides a detailed justification for using the W-3 correlation. Local subchannel fluid conditions are predicted with the XCOBRA IIIC (3-23) computer code.

3.5.5 Departure From Nucleate Boiling

3.5.5.1 Design Approach to Departure From Nucleate Boiling

The margin to departure from nucleate boiling (DNB) at any point in the core is expressed in terms of the departure from nucleate boiling ratio (DNBR). The DNBR is defined as the ratio of the heat flux required to produce departure from nucleate boiling at specific local coolant conditions to the actual local heat flux. At some point in the core the DNBR is a minimum and it is at this point that the margin to DNB for the core is evaluated. The following items are important in determining the core margin to DNB:

a) The coolant inlet conditions;
b) The power level;
c) The nuclear power distribution;
d) The analytical methods utilized to predict local coolant conditions;
e) The correlation used to predict DNB heat flux.

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The conventional approach for evaluating the margin to DNB concentrates on the most limiting location in the core and does not consider the DNBR of the core taken as a whole. Alternatively, typical distributions of DNBR for a larger group of channels can be calculated to show the number of rods which may approach the DNB limit.

Because of the uncertainties associated with predicting DNB there is a finite probability that if a channel is operated at a specified DNB ratio greater than one based on a particular correlation, it will be at or above its DNB heat flux. Therefore, the proper interpretation of DNB ratio is that it is a measure of the probability that DNB would occur in the particular design situation to which the DNB correlation is applied. This interpretation assumes, of course, that all operating parameters are known precisely and that the probability being evaluated is only that associated with the correlation. It is customary to establish the relationship between DNB ratio and probability of DNB statistically evaluating the scatter between actual values of DNB heat flux, as measured experimentally for many test geometries and operating conditions, and the corresponding values that are predicted by the correlation. Uncertainties associated with prediction of the operating conditions in the channel are subject to separate statistical interpretation. The approach used in design is to select core operating conditions and analytical methods in such a way that there is a very small probability that the actual hot channel coolant conditions are more severe than the calculated conditions used as input to the DNB correlation.

The W-3 DNB correlation presented in Reference 1 is used for the Fort Calhoun design. The probability that the DNB heat flux has been exceeded for several values of the DNB ratio, according to Reference 3-24 is shown in Table 3.5-5.

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Table 3.5-5 - "Probability Distribution, DNB Limits"

DNB Ratio	Probability That DNB Heat Flux Has Been Exceeded
2.5	0.0000085
2.0	0.00018
1.75	0.001
1.50	0.01
1.30	0.05

3.5.6 Thermal and Hydraulic Evaluation

3.5.6.1 Analytical Models

The XCOBRA IIIC (Ref. 3-23) computer program provides both steady state and transient calculation capabilities while including the effects of cross flow mixing between fuel assemblies. XCOBRA computes flow and enthalpy distributions on a subchannel basis. For subchannel analysis the "hot" channel and its nearest neighbors are modeled explicitly. The balance of the "hot" assembly is "lumped" as one channel, and the balance of the symmetric section of the core is represented by a single "lumped" channel. Each channel is then axially nodalized for more detail.

For core flow distribution analysis, the core is nodalized such that each radial node represents no more than one fuel assembly and each assembly is represented by multiple axial nodes. In this way the calculations of the core flow distribution include:
(1) differences in assembly hydraulic resistance, (2) localized flow leakage in assemblies, and (3) crossflow between the hot assembly and its neighbors.

3.5.6.2 Statistical Analysis of Hot Channel Factors

Random variations from nominal values in enrichment, pellet density, pellet diameter, and clad diameter affect the hot channel factors for heat flux. Hot channel heat input and rod diameter contribute to the flow factor. Estimation of these factors is based on inspection data on "as-manufactured" fuel assemblies.

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The factors contributing to the engineering heat flux factor are pellet density, pellet diameter, pellet enrichment and clad diameter. The design value is 1.03. For conservatism, the hot channel heat input factor is set equal to the same value.

3.6 THERMAL AND HYDRAULIC DESIGN EVALUATION FOR LIMITING TRANSIENTS AND SETPOINT ANALYSIS

3.6.1 General

The thermal and hydraulic design of the reactor has as its primary objective the assurance that the core can meet normal steady state and transient performance requirements without exceeding thermal and hydraulic design limits. This section, therefore, discusses the thermal and hydraulic characteristics that relate reactor performance to the margin of design limits.

The following transients are re-evaluated for validity, or reanalyzed, on a cycle-by-cycle basis. The criteria of this section was applied to the setpoint analysis and the following transients:

14.2 CEA Withdrawal Incident
14.3 Boron Dilution Incident
14.4 CEA Drop Incident
14.6 Loss of Coolant Flow
-Loss of Coolant Flow
-Seized Rotor
14.11 Excess Load Increase
14.12 Main Steam Line Break Accident
14.13 CEA Ejection Accident
14.15 Loss of Coolant Accident
14.22 Reactor Coolant System Depressurization Incident

The thermal and hydraulic design is based on a minimum DNBR safety limit, which must not be exceeded (i.e., the minimum DNBR must be greater than or equal to the minimum DNBR safety limit). For the HTP correlation, this safety limit is 1.141 (Reference 3-3). If applicable, a 2% mixed core penalty is applied (Reference 3-5). For the CE-1 correlation, this limit is 1.18 (Reference 3-53, 3-54, 3-55, and 3-6). To ensure that this limit is not exceeded, the reactor protective system is designed to trip the reactor before this condition is reached.

This section also discusses the fuel pellet performance characteristics that relate the reactor performance to the margin to design limits. The fuel pellet performance design limit ensures that fuel pellet centerline melt does not occur. The fuel centerline melt design criterion is based on maintaining the peak linear heat rate below prescribed limit of 22 kW/ft. To ensure that this limit is not exceeded, the reactor protective system is designed to trip the reactor before this condition can be achieved.

A summary of thermal and hydraulic parameters used for limiting safety analyses is presented in Table 3.6-1.

Table 3.6-1 - "Thermal and Hydraulic Parameters for Safety Analyses"

General Characteristics at Full Power

Total Heat Output, MWt	1500
Heat Generated in Fuel, Fraction (i.e., fuel and cladding)	0.975
Pressurizer Pressure	
Nominal, psia	2,100
Minimum in Steady State, psia	2,075
Maximum in Steady State, psia	2,150
Design Inlet Temperature, Steady State, °F	545
Total Reactor Coolant Volumetric Flow, gpm	206,000
Core Bypass Flow, %	4.54
Design Overpower, %	112

3.6.2 Coolant Flow

3.6.2.1 Total Coolant Flow Rate and Bypass Flow

The credited total coolant flow rate at full power is 206,000 gpm. The coolant flow path can be traced in Figure 3.1-1. Coolant enters the four inlet nozzles and flows into the annular plenum between the reactor vessel and the core support barrel. It then flows down on both sides of the thermal shield and through the flow skirt to the plenum below the core lower support structure. Pressure losses in the skirt and lower support structure help to even out the inlet flow distribution to the core. The coolant passes through the openings in the lower flow plate and flows axially upward through the fuel assemblies. A portion flows through the lower core plate and into the guide tubes in the fuel assemblies. Flow limiting devices have been placed in the guide tubes of fuel assemblies located under spare CEDM locations to prevent asymmetric bypass flows. After passing through the core, the coolant flows into the region outside the control element shrouds. From this region the coolant flows across the control element assembly shrouds and passes out through the outlet sleeves on the core barrel to the outlet nozzles.

The principal core bypass routes are direct inlet-to-outlet coolant flow at the joint between the core support barrel sleeve and the outlet nozzle and the flow in the reflector region in excess of that required for cooling. The total core bypass flow is $\leq$4.54% of the total RCS loop flow.

The coolant required to cool the control elements flows in the annulus between the control element and the guide tube and then into the region outside the control element assembly shrouds. A similar but smaller leakage will occur at the upper end of those guide tubes without control elements.

3.6.2.2 Inlet Flow Distribution

The Cycle 20 core is a mixed core design, which includes Framatome ANP Richland, Inc. and Westinghouse fuel assemblies. The inlet flow distribution was revised to account for the difference in inlet pressure losses between the different fuel designs.

3.6.2.3 Exit Pressure Distribution

The exit pressure distributions were derived in conjunction with the inlet flow distribution. For each inlet flow distribution there is a corresponding exit pressure distribution. They are also input directly into the hydraulic code. The exit pressures allow the code to more accurately predict the delta pressure across each assembly. This in turn leads to more accurate modeling of the enthalpy rise in each assembly.

3.6.2.4 Partial Flow Loop Operations

There are two steam generators and four reactor coolant pumps which give rise to six possible configurations of operation. At present the Fort Calhoun Nuclear Station is only licensed for the normal four-pump configurations.

One three pump configuration was analyzed (Ref. 3-69). This analysis assumed flow in three loops and no flow (forward or reverse) in the fourth loop. Inlet flow distribution and its associated exit pressure for the three pump configuration served as input to the thermal-hydraulics code. The results were then used to evaluate the Seized Rotor Incident (Section 14.6) and its impact on thermal margin degradation.

3.6.3 Peak Linear Heat Rate

The peak linear heat rate (PLHR), in the limiting fuel pin in the core shall not exceed that corresponding to the onset of fuel centerline melt. This fuel melt limit was calculated 22.77 Kw/ft for Westinghouse fuel and 24.8 Kw/ft for Framatome ANP Richland, Inc. fuel (References 3-73 and 3-74).

3.6.4 Peak Linear Heat Rate Protection

The axial power distribution (APD) trip is provided to ensure that excessive axial peaking will not cause fuel damage. The APD trip performs the following two functions:

- It provides a reactor trip before the peak kW/ft exceeds the power to fuel centerline melt value of the fuel (22 kW/ft), by working in combination with the variable high power trip, rod block system and the LCO's shown in Figure 3.6-4.
- It provides a reactor trip before the axial power distribution becomes more severe than that assumed to exist by the thermal margin/low pressure trip.

The maximum radial power peak that can occur for power levels up to the variable high power trip limit (in the event of a design basis AOO) is factored into the axial power distribution LSSS. The radial power peak that is allowed at any steady-state or transient core power is specified through the Power Dependent Insertion Limit (PDIL). The rod block system assures that no single electrical component failure in the control element drive system (other than a dropped CEA) can result in CEA group insertion in violation of the PDIL. The rod block system also controls CEA group sequencing, deviation and overlap. A penalty is factored into the LSSS to allow the existence of a 3% azimuthal tilt in core power.

3.6.5 Thermal Margin Analysis

The basic objective of thermal margin analysis is to identify the combinations of steady state operating conditions which satisfy the Specified Acceptable Fuel Design Limit on minimum DNBR. To meet this objective, calculations are performed over a range of operating parameters to determine the power levels that would be required to reach the DNBR design limit.

These calculated powers to DNB or overpower margins are represented in curves which are used to assess and quantify available thermal margin. These curves, which constitute the Thermal Margin information, provide the upper limits on core power over the specified range of operating conditions. This information is used to establish the DNB related Limiting Safety System Settings (LSSS) (Core Operating Limits Report (COLR) Figure 1) and Limiting Conditions for Operation (LCO) (COLR Figure 5).

3.6.5.1 Engineering Factors

Engineering Heat Flux Factor

The effect on local heat flux of deviations from nominal design dimensions and specifications is accounted for by the engineering heat flux factor. Design variables that contribute to this factor are fuel density, fuel enrichment, pellet diameter, and clad outside diameter. These variables may be combined statistically to obtain the engineering heat flux factor. A design value of 1.03 is used for the engineering heat flux factor.

Engineering Enthalpy Rise Factor

The engineering enthalpy rise factor accounts for the effects of deviations in fuel fabrication from nominal dimensions or specifications on the enthalpy rise in the hot channel. Tolerance deviations (averaged over the length of the four fuel rods that enclose the hot channel) for fuel density, fuel enrichment, pellet diameter, and clad outside diameter, contribute to this factor.

The engineering enthalpy rise factor accounts for increased heat input resulting from higher-than-nominal U-235 content. Because of the difficulty in evaluating average pellet tolerance variations for groups of four fuel rods, the enthalpy rise factor is conservatively assumed to be equal to the engineering heat flux factor.

Fuel Rod Bowing Effects

Fuel rod bowing effects on DNB margin for the Framatome ANP fuel were evaluated in Reference 3-64. The Cycle 20 core is comprised of both Framatome ANP and Westinghouse fuel.

3.6.6 Departure From Nucleate Boiling

The margin to departure from nucleate boiling (DNB) at any point in the core is expressed in terms of the departure from nucleate boiling ratio (DNBR). The DNBR is defined as the ratio of the heat flux required to produce departure from nucleate boiling at specific local coolant conditions to the actual local heat flux. At some point in the core the DNBR is a minimum and it is at this point that the margin to DNB for the core is evaluated.

Because of the uncertainties associated with predicting DNB there is a finite probability that if a channel is operated at a specified DNB ratio greater than one based on a particular correlation, it will be at or above its DNB heat flux. Therefore, the proper interpretation of DNB ratio is that it is a measure of the probability that DNB would occur in the particular design situation to which the DNB correlation is applied. It is customary to establish the relationship between DNB ratio and probability of DNB by statistically evaluating the scatter between actual values of DNB heat flux, as measured experimentally for many test geometries and operating conditions, and the corresponding values that are predicted by the correlation. Uncertainties associated with prediction of the operating conditions in the channel are subject to separate statistical interpretation. The approach used in design is to select core operating conditions and analytical methods in such a way that there is a very small probability that the actual hot channel coolant conditions are more severe than the calculated conditions used as input to the DNB correlation.

The HTP DNB correlation (Reference 3-3) is used for the Fort Calhoun design, although other DNB correlations (such as CE-1 and W-3) may have been used in the analysis of non-limiting events.

3.6.7 Vapor Fraction

The high operating pressure of the reactor minimizes vapor formation. A calculation, assuming 2 percent overpower, maximum inlet temperature and design coolant flow rate, shows the core vapor fraction is less than 0.1 percent. A conservative value of 0.25 percent is assumed in assessing the effect of voids on reactivity (Section 3.4.1.3).

To avoid the possibility of departure from nucleate boiling as the result of local flow oscillations, a conservative limit has been established to prevent flow instabilities. The limits to assure stable flow are based on avoiding flow regime changes in the hot channel that could affect the flow-pressure drop characteristics so as to cause an instability. Figure 3.6-3 shows flow regimes and regions of stable flow as a function of local mass flow rate and void fraction based on the data in Reference 3-25.

An automatic reactor shutdown (thermal margin trip) will occur before the flow instability limit is reached; thus departure from nucleate boiling resulting from flow oscillations is prevented.

3.6.8 Thermal and Hydraulic Evaluation

The TORC (Reference 3-56), CETOP (References 3-57, 3-58, 3-59, 3-60, 3-61, and 3-63), or XCOBRA-IIIC (Reference 3-70) computer code is used to calculate the thermal-hydraulic and DNB performance of the DNB-limiting assembly in the core. The TORC and CETOP codes use the CE-1 critical heat flux correlation (Reference 3-53, 3-54, and 3-55). The XCOBRA-IIIC code uses the HTP critical heat flux correlation (Reference 3-3).

The TORC and XCOBRA-IIIC models consist of thermal-hydraulic models of the core (with separate representations of each fuel assembly in the core or quarter core) that is linked to thermal-hydraulic models of the limiting fuel assembly (with separate representations of each fuel rod and flow channel within the assembly). These computer codes solve the conservation equations for a 3-dimensional representation of the open lattice core to determine local coolant conditions at all points within the core. The calculations include the calculation of the minimum DNBR.

The CETOP code differs from the TORC code in that the enthalpy transport coefficients are used to improve modeling of coolant conditions in the vicinity of the hot sub-channel and in that more rapid equation-solving routines are used. The CETOP models are tuned to always give conservative minimum DNBR results relative to the detailed TORC models. The CETOP code is used only because it reduces computer time significantly; no margin gain is realized.

Depending on the setpoint methodology used, either the CETOP or the XCOBRA-IIIC code is used in the setpoint analyses.

3.6.9 Fuel Temperature Conditions

Framatome ANP Richland, Inc. evaluates the fuel rod design using the RODEX2 code (References 3-71 and 3-72). This code is a quasi steady-state code that determines the pellet temperatures, cladding temperatures, cladding corrosion, fission gas release, and cladding creepdown and swelling as a function of the exposure.

The inlet coolant conditions are used as the starting boundary conditions. The code then uses finite elements over defined time increments to determine the rod behavior. The projected limiting power histories for many different rods (i.e., the limiting first cycle, second cycle, third cycle, and high burnup cycle) are evaluated. The code capabilities include gadolinia bearing and fuel multiple axial gadolinia and enrichment columns in a fuel rod.

The output of the RODEX2 is used to evaluate compliance with the approved generic fuel design criteria and to provide rod input for other evaluations, such as the large break LOCA. The RODEX2 code has been generically reviewed and accepted by the NRC for fuel rod analyses.

3.6.10 Flow Stability

Flow oscillations of significant amplitude may be sustained in some channels when heat is added to two-phase flow in parallel channels. This possibility results from two conditions that exist within the core:

a. The pressure drop flow characteristics with two-phase flow are such that large changes in flow can occur for small changes in pressure drop;

b. With parallel channels, the flow has an alternate path.

The flow regimes may be classed as separated or homogeneous. Homogeneous flow is bubbly or froth flow. Separate flow is annular or slug. Reference 3-30 describes these flow regimes in detail. For homogeneous flow, the channel pressure drop continuously increases with increasing flow rate or increasing vapor fraction. A change in the flow regime to separated flow results in a change in the flow characteristics and flow oscillations in the parallel channels are then possible. Figure 3.6-3 shows flow regimes as a function of mass flow rate and void fraction based on the data of Reference 3-25. In general, increasing void fraction results in a transition to an annular type flow and decreasing void fraction results in a transition to slug flow. A comparison of this flow regime map with data of observed flow regimes reported in References 3-26, 3-27, 3-28 and 3-29 has been made to verify the effect of variation of such parameters as channel length, diameter and pressure and to check the consistency of the data. Good agreement was obtained, and the limits for stable flow as shown in Figure 3.6-3 are considered a reasonable and conservative representation of flow regime changes for core hot channel conditions.

The limit to ensure flow stability as applied to core conditions is conservative since the "openness" of the channels to crossflow tends to damp any flow oscillations. This is explained by the basic requirement that in order for flow oscillations to occur, a feedback effect (from the channel outlet to the channel inlet region) on channel flow and pressure loss is necessary. Crossflow tends to damp the feedback effect and tends to make the open channel array stable even when parallel closed channels would not be stable. This conclusion is supported by the observation of the absence of DNB conditions in the open array experiments reported in Reference 3-30. The experimental results of Reference 3-25 to 3-29 are all for closed channels.

3.7 MECHANICAL DESIGN AND EVALUATION

The reactor core and internals are shown in Figure 3.1-1. A cross section of the reactor core and internals is shown in Figure 3.1-2. Mechanical design features of the reactor internals, the control element drive mechanisms and the reactor core are described below. Mechanical design parameters are listed in Table 3.7-1.

Table 3.7-1 - "Mechanical Design Parameters"

Fuel Assemblies

Type	Number of assemblies	Number of Fuel rods per assembly	Number of Fuel Displacing Poison Rods per Assembly	Number of Fuel Rods with Intergral Poison per Assembly	Spacers per Assembly	Weight of Contained Uranium per Assembly, kg	Spacer Material
T1	12	176	N/A	0	9	375	Zircaloy-4 & Inconel
T3	12	176	N/A	48*	9	374	Zircaloy-4 & Inconel
T7	12	176	N/A	48*	9	374	Zircaloy-4 & Inconel
T8	4	176	N/A	64*	9	375	Zircaloy-4 & Inconel
X1	4	176	N/A	84*	9	371	Inconel
X2	8	176	N/A	0	9	375	Inconel
X3	8	176	N/A	48*	9	373	Inconel
X4	4	176	N/A	56*	9	373	Inconel
X5	4	176	N/A	64*	9	372	Inconel
X6	12	176	N/A	84*	9	371	Inconel
Y1	4	176	N/A	0	9	375	Zircaloy-4 & Inconel
Y2	8	176	N/A	16**	9	372	Zircaloy-4 & Inconel
Y3	10	176	N/A	12**	9	373	Zircaloy-4 & Inconel
Y4	8	176	N/A	16**	9	372	Zircaloy-4 & Inconel
Y5	4	176	N/A	0	9	375	Zircaloy-4 & Inconel
Y6	4	176	N/A	4**	9	375	Zircaloy-4 & Inconel
Y7	2	176	N/A	8**	9	374	Zircaloy-4 & Inconel
Y8	4	176	N/A	4**	9	374	Zircaloy-4 & Inconel
Y9	8	176	N/A	8**	9	374	Zircaloy-4 & Inconel
Y10	1	176	N/A	12**	9	373	Zircaloy-4 & Inconel

* Westinghouse assemblies have IFBA as an integral poison.

** Framatome ANP assemblies have gadolinia as an integral poison.

Table 3.7-1 (Cont'd)

Fuel Rod	<u>Y</u>	<u>X</u>	<u>T</u>
Fuel Material (Sintered Pellets)	UO_2	UO_2	UO_2
Pellet Diameter, inches	0.377	0.3765	0.3765
Pellet Length, inches	0.435*	0.452*	0.452
Pellet Density, g/cc	10.45**	10.41	10.41
Clad Material	Zircaloy-4	ZIRLO	Zircaloy-4
Clad ID, inches	0.384	0.384	0.384
Clad OD, inches (nominal)	0.440	0.440	0.440
Active Length, inches	128	128	128
Total Length Between End Plates, inches	137.22	138.15	138.15
Maximum Allowable Fuel Rod Average Burnup (MWD/MTU)	62,000	60,000	60,000

* Axial Blanket Pellets are 0.545 inches in length.

** The pellet density varies based on the loading of gadolinia.

Table 3.7-1 (Cont'd)

Control Element Assemblies (CEA's)

Type	Original Design*	Revised Design	Improved Design
CEA Description	Five(5) non-reconstitutable Full-Length Poison Rods Attached to a Spider	Five(5) non-reconstitutable Full-Length Poison Rods Attached to a Spider	Five(5) reconstitutable Full Length Poison Rods Attached to a Spider
Poison Material	B_4C Pellets	B_4C Pellets Ag-In-Cd Tips	B_4C Pellets Ag-In-Cd Tips
Sheath Material	Inconel 625	Inconel 625	Inconel 625
Number of CEAs in the Core	0	8	41
Poison Length	125"	125"	125"
Overall Length	148"	148"	152"
Corner Element Pitch	4.64"	4.64"	4.64"
Pre-Irradiated CEA Rod Diameter	0.95"	0.95"	0.95"
CEA Weight	75± 1 lbs.	75± 1 lbs.	75± 1 lbs.
Total Operating CEA weight	260 lbs.	260 lbs.	260 lbs.

* All original design CEAs discharged. Data for information only.

Table 3.7-1 (Cont'd)

Core Arrangement

Number of Fuel Assemblies in Core, Total	133
Number of CEAs	49
Number of Active Fuel Rods (Design)	23,405*
CEA Pitch, min, inches	11.57
Spacing Between Fuel Assemblies, Fuel Rod Surface to Surface, inches	0.198
Spacing, Outer Fuel Rod Surface to Core Shroud, inches	0.179
Hydraulic Diameter, Nominal Channel, Feet	0.04442
Total Flow Area (Excluding Guide tubes), sq ft	32.84
Total Core Area, sq ft	62
Core Equivalent Diameter, inches	106.448
Core Circumscribed Diameter, inches	116.484
Core Volume, liters	18,726
Total Heat Transfer Area, sq ft	28,758*

* Excludes 3 stainless steel rods

3.7.1 Reactor Internals

The reactor internals are designed to support and orient the reactor core fuel assemblies and control element assemblies, absorb the CEA dynamic loads and transmit these and other loads to the reactor vessel flange, provide a passageway for the reactor coolant, and support in-core instrumentation.

The internals are designed to safely perform their functions during all steady state conditions and during normal operating transients. The internals are designed to safely withstand the forces due to deadweight, handling, system pressure, flow impingement, temperature differential, shock and vibration. All reactor components are considered Class 1 for seismic design. The reactor internals design limits deflection where required by function. The structural components satisfy stress values given in Section III of the ASME Boiler and Pressure Vessel Code. Certain components have been subjected to a fatigue analysis. Where appropriate, the effect of neutron irradiation on the materials concerned is included in the design evaluation.

The components of the reactor internals are divided into three major parts consisting of the core support barrel (including the lower core support structure, the core shroud and the thermal shield), the upper guide structure (including the CEA shrouds and the in-core instrumentation guide tubes) and the flow skirt. These components are shown in Figure 3.1-1. The in-core instrumentation is described in Sections 7.5.1 and 7.5.2.

3.7.1.1 Core Support Assembly

The major support member of the reactor internals is the core support assembly. This assembled structure consists of the core support barrel, the core support plate and the support columns, the core shroud, the thermal shield, the core support barrel to pressure vessel snubbers and the core support barrel to upper guide structure guide pins. The major material for the assembly is Type 304 stainless steel.

The core support assembly is supported at its upper flange from a ledge in the reactor vessel flange. The lower end is restrained in its lateral movement by six core support barrel-to-pressure vessel snubbers. Within the core support barrel are axial shroud plates which are attached to the core support barrel wall by horizontal former plates and to the core support plate by anchor blocks. The core support plate is positioned within the barrel at the lower end and is supported both by a ledge in the core support barrel and by 44 columns. The core support plate provides support and orientation for the fuel assemblies. Also within the core support barrel just below the nozzles are four guide pins which align and prevent excessive motion of the lower end of the guide structure relative to the core support barrel during operation. The thermal shield is affixed to the outside of the core support barrel.

The effect of neutron irradiation on the core support structure will be a reduction in the ductility of the structures in the areas of highest fluence. In the design of the structures in these areas, the deflections and resultant strains were determined and compared with the estimated ductility values at end-of-service life in order to ensure adequacy of the design. Table 3.7-2 shows this comparison for certain areas in the core support structure which experience high fluence.

Table 3.7-2 - "Comparison of Areas in the Core Support Structure Which Experience the Highest Fluence With Ductility at End of Life"

COMPONENT	LOCATION	FLUENCE Cm^2	CALCULATED STRAIN %	UNIFORM ELONGATION % AT OPERAT. TEMP.
Core Barrel	Opposite Center of Core	1.30×10^{21}	.01	2.5
Core Support Columns	Top of Column	7.50×10^{21}	.04	0.5
Core Barrel	Upper Flange	$< 10^{20}$	.08	> 23%
Upper Guide Structure	Grid Beams	$< 10^{20}$	.08	> 23%
Lower Support Beams	Top of Beam	$< 10^{20}$	.08	> 23%

3.7.1.2 Core Support Barrel

The core support barrel net weight (325,000 pounds) consists of the entire core and other internals. It is a right circular cylinder with a nominal inside diameter of 120-5/8 inches and a minimum wall thickness in the weld preparation area of 1 inch. It is suspended by a 4-inch thick flange from a ledge on the pressure vessel. The core support barrel in turn supports the core support plate upon which the fuel assemblies rest. Press fitted into the flange of the core support barrel are four alignment keys located 90 degrees apart. The reactor vessel, closure head and upper guide structure assembly flanges are slotted in locations corresponding to the alignment key locations to provide proper alignment between these components in the vessel flange region.

Since the core support barrel is 26 feet long and is supported only at its upper end, it is possible that coolant flow could induce vibrations in the structure. Therefore, amplitude limiting devices, or snubbers, are installed near the bottom outside end of the core support barrel. The snubbers consist of six equally spaced double lugs around the circumference and are the grooves of the "tongue-and-groove" assembly; the pressure vessel lugs are the tongues. Minimizing the clearance between the two mating pieces limits the amplitude of any vibration. At assembly, as the internals are lowered into the vessel, the pressure vessel tongues engage the core support grooves in an axial direction. With this design, the internals may be viewed as a beam with supports at the furthest extremities. Radial and axial expansions of the core support barrel are accommodated, but lateral movement of the core support barrel is restricted by this design. The pressure vessel tongues have bolted, lock welded Inconel X shims and the core support barrel grooves are hardfaced with Stellite to minimize wear.

3.7.1.3 Core Support Plate and Support Columns

The core support plate is a 120-inch diameter, 2-inch thick, Type 304 stainless steel plate into which the necessary flow distributor holes for the fuel assemblies have been machined. Fuel assembly locating holes (four for each assembly) are also machined into this plate.

Columns and support beams are placed between this plate and the bottom of the core support barrel in order to provide stiffness to this plate and transmit the core load to the bottom of the core support barrel.

3.7.1.4 Thermal Shield

The 3-inch thick, Type 304 stainless steel thermal shield is a cylindrical structure which reduces the neutron flux and radiation heating in the reactor vessel wall to an acceptable level. At the upper end, the shield is supported by eight equally spaced lugs on the outer periphery of the core support barrel. A 0.005-inch gap between the thermal shield and the lower portion of the lug is provided to permit assembly of the core support barrel and the thermal shield. The lower end of the thermal shield is positioned radially utilizing 16 equally placed positioning pins which pass through the shield and butt against the core support barrel.

3.7.1.5 Core Shroud Plates and Centering Plates

The core shroud provides an envelope for the perimeter of the core and limits the amounts of coolant bypass flow. The shroud consists of rectangular plates 5/8-inch thick, 142-3/8 inches long and of varying widths. The bottom edges of these plates are fastened to the core support plate by use of anchor blocks.

The critical gap between the outside of the peripheral fuel assemblies and the shroud plates is maintained by eight tiers of centering plates attached to the shroud plates and centered during initial assembly by adjusting bushings located in the core support barrel. The overall core shroud assembly, including the rectangular plates, the centering plates, and the anchor blocks, is a bolted and lock welded assembly. In locations where mechanical connections are used, bolts and pins are designed with respect to shear, binding and bearing stresses. All bolts and pins are lock welded. In addition, all bolts (bodies and heads) are designed to be captured in the event of fracture; the bolt heads are trapped by lock bars or lock welds, and the bodies are trapped by the use of non-thru holes or by incomplete tapping of thru holes. Holes are provided in the core support plate to allow coolant to flow upward between the core shroud and the core support barrel, thereby minimizing thermal stresses in the shroud plates and eliminating stagnant pockets.

3.7.1.6 Flow Skirt

The Inconel flow skirt is a perforated (2-1/4-in. diameter holes) right circular cylinder, reinforced at the top and bottom with stiffening rings. The flow skirt is used to reduce inequalities in core inlet flow distributions and to prevent formation of large vortices in the lower plenum. The skirt provides a nearly equalized pressure distribution across the bottom of the core support barrel. The skirt is fastened to the pressure vessel lower head by nine equally spaced welds.

3.7.1.7 Upper Guide Structure Assembly

This assembly (Figure 3.7-1) consists of a plate, 41 control element assembly shrouds (two of which have been modified to act as Heated Junction Thermocouple Probe holders), a fuel assembly alignment plate and a ring shim. The upper guide structure aligns and laterally supports the upper end of the fuel assemblies, maintains the CEA spacing, prevents fuel assemblies from being lifted out of position during a severe accident condition and protects the CEA's from the effect of coolant crossflow in the upper plenum. It also supports the in-core instrumentation guide tubing. The upper guide structure is handled as one unit during installation and refueling.

The upper end of the assembly is a flanged grid structure consisting of a grid array of 24-inch deep beams. The grid is encircled by a 24-inch deep cylinder with a 3-inch thick plate welded to the cylinder. The periphery of the plate contains four accurately machined and located alignment keyways, equally spaced at 90-degree intervals, which engage the core barrel alignment keys. The reactor vessel closure head flange is slotted to engage the upper ends of the alignment keys in the core barrel. This system of keys and slots provides an accurate means of aligning the core with the closure head. The grid aligns and supports the upper end of the CEA shrouds.

The control element assembly shrouds extend from the fuel assembly alignment plate to an elevation about 8 inches above the support plate. There are 29 single-type shrouds. These consist of centrifugally cast cylindrical upper sections welded to cast bottom sections, which are shaped to provide flow passages for the coolant passing through the alignment plate while shrouding the CEA's from crossflow. There are also 12 dual-type shrouds which in configuration consist of two single-type shrouds connected by a rectangular section shaped to accommodate the dual control element assemblies. The shrouds are bolted to the fuel assembly alignment plate. At the upper guide structure support plate, the single shrouds are connected to the plate by spanner nuts which permit axial adjustment. The spanner nuts are torqued in place and lockwelded. The dual shrouds are attached to the upper plate by welding.

The fuel assembly alignment plate is designed to align the upper ends of the fuel assemblies and to support and align the lower ends of the CEA shrouds. Precision machined and located holes in the fuel assembly alignment plate align the fuel assemblies. The fuel assembly alignment plate also has four equally spaced slots on its outer edge which engage with Stellite hardfaced pins protruding out from the core support barrel to prevent lateral motion of the upper guide structure assembly during operation. Since the weight of a fuel assembly under all normal operating conditions is greater than the flow lifting force, it is not necessary for the upper guide structure assembly to hold down the core. However, the assembly would capture the core and limit upward movement in the event of an accident.

A ring shim bears on the flange at the top of the assembly to resist axial upward movement of the upper guide structure assembly and to accommodate axial differential thermal expansions between the core barrel flange, upperguide structure flange and pressure vessel flange support edge and head flange recess.

The upper guide structure assembly also supports the in-core instrument guide tubes. The tubes are conduits which protect the in-core instruments and guide them during removal and insertion operations.

3.7.2 Control Element Drive Mechanism

The control element drive mechanism (CEDM) drives the CEA within the reactor core and indicates the position of the CEA with respect to the core. The speed at which the CEA is inserted or withdrawn from the core is consistent with the reactivity change requirements during reactor operation. For conditions that require a rapid shutdown of the reactor, the CEDM drive releases to allow the CEA and the supporting CEDM components to drop into the core by gravity. The reactivity is reduced during such a drop at a rate sufficient to control the core under any operating transient or accident condition. Since CEA speed is a direct function of drive motor power supply frequency, which is limited by the transmission frequency control system to 60 cycles/sec, CEA speed limiting features are not needed on Fort Calhoun and none are included as such.

The CEA is decelerated at the end of the drop by the CEDM which supports the CEA in the fully inserted position.

There are 37 CEDM's mounted on flanged nozzles on top of the reactor vessel closure head, located directly over the CEA's in the reactor core. Each CEDM is connected to a CEA by a locked coupling. The weight of the CEA's and CEDM's is carried by the vessel head. In order to provide lateral stability, particularly in resisting horizontal earthquake forces, the CEDM's are supported in the horizontal direction by a seismic support structure which is a cylindrical structure surrounding the CEDM's and attached to the reactor vessel head. This structure restricts bending deflection so as to limit stresses to allowable values in the lower housing and nozzle areas. Air is drawn through the structure for cooling (see Section 9.10).

The CEDM is designed to handle dual or single CEA's. The total stroke of the drive is 128 inches. The speed of the drive is 46 inches per minute. The time from receiving a trip signal to 90 percent of the fully inserted position of the CEA is less than 2-1/2 seconds under operating conditions. The CEA is allowed to accelerate to about 11 ft/sec and is decelerated to a stop at the end of the stroke.

The CEDM is of the vertical rack and pinion type with the drive shaft running parallel to the rack and driving the pinion gear through a set of bevel gears. The design of the drive is shown in Figure 3.7-2. The trippable CEA is driven by an electric motor operating through a gear reducer and a magnetic clutch. By de-energizing the magnetic clutch, the CEA drops into the reactor under the influence of gravity. The magnetic clutch incorporates an anti-reversing device which prevents upward CEA movement when the clutch is deenergized. The non-trippable CEA is driven by a CEDM that has been modified by replacing the magnetic clutch with a solid shaft assembly to eliminate the trip function so that the non-trippable CEA maintains its position during a reactor trip. Otherwise, this CEDM is the same as those attached to the other CEA's. The drive shaft penetration through the pressure housing is closed by means of a face-type rotating seal. The rack is connected to the CEA by means of a rack extension containing an external collet-type coupling which expands and locks into a mating shouldered bore on top of the CEA.

The rack extension is connected to the rack through a tie bolt by means of a nut and locking device at the upper end of the rack. A small diameter closure located at the top of the pressure housing provides tool access to this nut for releasing the CEA from the CEDM. The rack is guided at its upper end by a section having an enlarged diameter which operates in a tube extending the full length of the CEA travel. The final cushioning at the end of a CEA drop is provided by the dashpot action of the enlarged diameter of the rack entering a reduced diameter in the guide tube.

3.7.2.1 CEDM Pressure Housing

The pressure housing consists of a lower and an upper section joined near the top of the drive by means of a threaded autoclave type closure. The pressure housing design and fabrication conforms to the requirements of the ASME Boiler and Pressure Vessel Code, Section III, for Class A vessels. The housing is designed for steady state conditions as well as all anticipated pressure and thermal transients.

The lower housing section is a stainless steel tubular section welded to an eccentric reducer and flange piece at the lower end. This flange fits the nozzle flange provided on the reactor vessel closure head and is seal welded to it by an omega-type seal. Once seal welded and bolted into place, the lower pressure housing need not be removed since all servicing of the drive is performed from the top of this housing. The upper part of the lower housing is machined to form the autoclave-type closure and is provided with a recessed gasket surface for a gasket.

The upper part of the pressure housing has a flange which mates with the lower housing autoclave-type closure, a cavity which contains the drive rotating seal, and a tubular housing extension with a small flange closure which provides access for attaching and detaching the CEA. The shaft seal is a face-type rotating seal with mating surfaces of Graphitar and tungsten carbide. The two parts of the seal are fitted with O-rings to prevent leakage around the seal. The O-rings are static seals. A cooling jacket surrounds the seal area to maintain the temperature of the seal and O-rings at an acceptable level. This cooling water is from the component cooling system (see Section 9.7). A seal leak-off line is connected to the upper housing. A thermocouple in the seal leak-off connection at the upper housing monitors leak-off water temperature which provides an indication of seal leakage.

There are four spare CEDM penetrations in the reactor vessel head. Two have spare CEA pressure housings attached. These are used to house the Heated Junction Thermocouples. Each HJTC assembly consists of a Graylock adapter hub with a threaded autoclave-type closure that conforms to the requirement of the ASME Boiler and Pressure Vessel Code, Section III, for Class A vessels. Each adapter hub utilizes a seal plug to form a pressure boundary about each Heated Junction Thermocouple assembly. The other two spare penetrations are blind flanged.

3.7.2.2 Rack and Pinion Assembly

The rack and pinion assembly is an integrated unit which fits into the lower pressure housing and couples to the motor drive package through the upper pressure housing. This unit carries the bevel gears which transmit torque from the vertical drive shaft to the pinion gear. The vertical drive shaft has couplings at both ends and may be lifted out when the upper pressure housing is removed. Ball bearings are provided for supporting the bevel gears and the pinion gear. The rack engages the pinion, and is held in proper engagement with the pinion by the backup rollers which carry the load due to gear tooth reactions. The gear assembly is attached to a stainless steel tube supported by the upper part of the pressure housing. This tube also carries and positions the guide tube which surrounds the rack. The rack is a tube with gear teeth on one side of its outer surface and a flat on the opposite side which forms a contact surface for guide rollers. The upper end of the rack is fitted with an enlarged section which runs in the guide tube and provides lateral support for the upper end of the rack. It also acts as a piston in controlling water flow in the lower guide tube dashpot. The top section also carries a permanent magnet which is used to operate a reed switch position indicator outside the pressure housing. The guide tube is connected at its upper end to the support tube. The support for the guide tube contains an energy absorber at the top end of the tube which deforms to limit the stresses on the CEA, in case the mechanism is tripped without water in the dashpot. If such a "dry trip" should occur, the mechanism and CEA would not be damaged; however, it would be necessary to disassemble the CEDM and replace the energy absorber.

3.7.2.3 Motor Drive Package

Power to operate the drive is supplied by a fractional horsepower, 120-V, single-phase, 60-Hz motor. The output is coupled to the vertical drive shaft through a magnetic clutch and an anti-reverse clutch operating in parallel. When the magnetic clutch is energized, the drive motor is connected to the main shaft and can drive the CEA either up or down. When de-energized, the magnetic clutch separates and the CEA drops due to its own weight. The anti-reverse clutch prevents rotation of the drive in the up direction and holds the CEA inposition against upward forces. The action is completely mechanical and does not rely on any outside source of power. The motor, brake, clutches, position indicator and limit switches are all mounted on a common frame for maintaining position and alignment. This entire drive package is assembled and checked as a unit and can be removed and replaced without disturbing the other parts of the mechanism. The frame for the drive package is provided with a flange which is bolted to a flange on the upper pressure housing for positioning the drive assembly. The electrical connections are located on the top of the drive package and are readily accessible.

3.7.2.4 Position Readout Equipment

Two independent position readout systems are provided for indicating the position of the CEA. One (primary system) is a synchro transmitter geared to the main drive shaft with readout provided by synchro receivers connected to the transmitter. The other (secondary system) position indicator consists of a series of reed switches built into a subassembly which is fastened to the outside of the CEDM along the pressure housing. The permanent magnet built into the top of the rack actuates the reed switches one at a time as it passes by them. A resistor network in conjunction with these switches controls the readout to indicate position. Limit switches located in the motor drive package are geared to the drive shaft and are used to provide indication of CEA position at certain predetermined points. These switches are used in the CEDM control system. The systems are described in Section 7.5.

3.7.2.5 Control Element Assembly Disconnect

The CEA is connected to the CEDM by means of an extension shaft with an internal collet-type coupling at its lower end. A tie rod connects the extension shaft to the rack. In order to disengage the CEA from the drive, the flange access closure at the top of the CEDM is removed. A tool is then inserted through this opening and, with the CEA in the full down position, the tool is used to release the nut locking device and to unscrew the nut on top of the tie rod. By turning another handle on the tool, the tie rod is rotated about a quarter turn and lifted about 2 inches to unlock the collet coupling and disengage the CEDM from the CEA.

3.7.2.6 CEDM Evaluation

Development models of internal and external drive components, subassemblies of the CEDM, as well as a complete model CEDM have undergone accelerated life tests under reactor conditions and have demonstrated that the CEDM fulfills all drive, trip and endurance requirements (see Section 1.4.4).

In addition to these development tests, a prototype CEDM with a simulated reactor core module was accelerated-life tested in an autoclave under reactor conditions to provide the overall adequacy of the CEDM during its design life. Each CEDM was tested at design pressure to prove its functional adequacy.

3.7.3 Core Mechanical Design

The core approximates a right circular cylinder with an equivalent diameter of 106.4 inches and an active fuel height of 128 inches. It is made up of Zircaloy-4 or ZIRLO clad fuel rods containing approximately 49 metric tons of slightly enriched uranium in the form of sintered $U0_2$ pellets. The fuel rods are grouped into 133 assemblies. Information in the following sections is based on fuel designs by Westinghouse (W), CE, and Framatome ANP Richland, Inc. fuel. Failed fuel rods are replaced with stainless steel rods per engineering analysis justified in nuclear engineering instructions.

Short term reactivity control is provided by 49 control element assemblies. Four of the CEA's are non-trippable. The CEA's are guided within the core by the guide tubes which are integral parts of the fuel assemblies.

3.7.3.1 Fuel Assembly

The 133 fuel assemblies consist of 176 fuel rods, five guide tubes, nine fuel spacer grids, and upper and lower end fittings. The structural frame of the assembly consists of the guide tubes, spacer grids and end fittings. The five guide tubes are attached to the end fittings. The spacer grids contained in the assemblies are attached to the guide tubes.

The lower end fitting is a cast structure of 304 stainless steel. It is machined to accept alignment pins, which fit in corresponding holes in the core support plate. The alignment pins provide lateral alignment of the lower end of the fuel assembly. The length of the alignment pin engagement ensures that the spacing between fuel assemblies will not be altered even during postulated accident conditions when a fuel assembly is lifted into contact with the upper guide structure. The Westinghouse lower end fitting contains flow holes and holes for positioning the fuel rods and guide tubes. The flow holes on the Westinghouse flowplate are of small diameter to limit the size of debris particles flowing into the fuel array. The Framatome ANP lower end fitting is the FUELGUARD ™ design. This design has flow vanes, which trap debris of a size able to be lodged in the fuel assembly. This reduces the potential for debris related fuel clad failures.

The fuel assembly upper end fitting is a cast structure of 304 stainless steel. It serves as an attachment for the guide tubes and as the lifting fixture. The pin-shaped protrusions serve as guide pins and mate with precision-drilled holes in the alignment plate to provide the alignment of the upper ends of the fuel assembly.

The fuel rod spacer grids maintain the fuel rod pitch over the full length of the fuel rods. The grids in the Westinghouse assemblies are fabricated from Inconel, Zircaloy-4, or ZIRLO strips interlocked in an egg crate fashion and welded together. For the Westinghouse batch X assemblies, the top, bottom, and middle grids are fabricated from Inconel. For the Westinghouse batch T assemblies, the top and bottom grids are fabricated from Inconel, while the middle grids are fabricated from Zircaloy-4. Each batch T and X fuel rod is supported by two support dimples. The spacer grids on the Framatome ANP Richland, Inc. batch Y assemblies are the high thermal performance (HTP) design. The eight upper grids are fabricated from Zircaloy-4 and the bottom grid is fabricated from Alloy-718. The fuel rods are supported by castillations in the grid strips. These castillations provide line contact between the rod and the grid, thus improving the flow induced vibration fretting resistance. Analysis of loss coefficients for Westinghouse and Framatome ANP Richland, Inc. type grids have been performed (Reference 3-62 and 3-67).

3.7.3.2 Fuel Rods

The fuel rods consist of UO_2 pellets, a compression spring and spacer discs, all encapsulated within a Zircaloy-4 or ZIRLO tube. The Westinghouse UO_2 pellets have a nominal density of 10.41 g/cc and the Framatome ANP Richland, Inc pellets have a nominal density of 10.45 g/cc. The pellets are dished at both ends to accommodate the effects of thermal expansion and swelling.

The fuel cladding is slightly cold worked Zircaloy-4 or ZIRLO tubing. The cold nominal diametral gap between the pellet and clad ID for the Westinghouse design is 0.0075 inches, and has been set taking into account clad stresses and strains and transfer of heat from the pellets. The Westinghouse design has a compression spring located at the top of the fuel pellet column is of 302 stainless steel. The Framatome ANP Richland, Inc. design has an Alloy-718 spring. This compression spring maintains the column in its proper position during handling and shipping. It also provides support for the clad in the plenum region to prevent local collapse. The adequacy of the spring to perform its functions has been demonstrated in a series of long term creep collapse tests with plenum clad temperatures above those expected in the reactor.

There is one spacer at the top of the pellet stack in Westinghouse fuel rods and Framatome ANP Richland, Inc. fuel does not include a spacer. The upper spacer on the Westinghouse prevents UO_2 chips from entering the plenum region.

The plenum above the pellet column provides space for axial thermal expansion of the fuel column and for expansion of fission gas. The rod designs have been evaluated to ensure that the maximum end-of-life internal pressure will be less than the design limit.

Each fuel rod is internally pressurized with helium. The internal pressurization with helium improves the thermal conductance between the fuel pellets and the cladding, resulting in a decrease in fuel temperature with an attendant reduction in the release of fission products and an increase in the margins between operating temperatures and allowable thermal limits. In addition, by reducing the differential pressure across the clad, internal pressurization affords a substantial reduction in the adverse effects of fuel-clad interaction.

Fuel rods containing axial blankets have a reduced U-235 enrichment in the top and bottom ends of each fuel rod. Since normal power production in the fuel rod axial ends is significantly lower than in the central portion of a fuel rod, the reduced enrichment does not negatively impact rod power. In addition, axial blankets reduce the total U-235 inventory at discharge. The combination of these effects improves uranium utilization, which translates into direct fuel cycle cost savings.

3.7.3.3 Poison Rods

In some of the fuel assemblies, poison rods are included to make the beginning of cycle moderator coefficient more negative. The poison rods are mechanically similar to fuel rods except that they contain a poison (neutron absorber) integrated with the fuel. For the Westinghouse fuel, the poison pellets are fuel pellets coated with ZrB_2, which is a burnable poison that does not displace fuel from the rod (Integral Fuel Burnable Absorber-IFBA). The pellet diameter is 0.353 inches. The clad thickness for the poison rod is 0.032 inches. The outside diameter of all poison rods is identical to those of the fuel rods. Some of the axial blankets in the Westinghouse IFBA fuel rods are comprised of annular pellets. The annular pellet design increases the fuel rod plenum volume, thereby reducing the maximum rod internal pressure during operation. The annular axial blankets also provide an economic benefit in reduced fuel costs. The Framatome ANP Richland, Inc. fuel design uses gadolinia (Gd_2O_3) as an integral burner absorber. The Gd_2O_3 loading varies up to 8 %. The enriched UO_2 pellets containing Gd_2O_3 are 107 inch column with about 11 inch blankets. The pellet and cladding configuration for the poison rod is identical to the fuel rod.

3.7.3.4 Part Length Poison Rods

The part length poison rods (PLPR), when used, reside in guide tubes of selected peripheral assemblies. Due to space limitations, the rods can not be attached to a spider section. Therefore, each individual rod has a gripper section at the upper end for handling purposes. The gripper section is also spring loaded to provide a holddown function when the upper guide structure is installed. The holddown load is transmitted over the entire length of the rod, causing the tip of the rod to be pressed against the bottom of the guide tube.

The poison rod cladding is fabricated of 0.04 inch wall Inconel 625, 0.948 inches in outer diameter. The central fifty percent of the rod in the core region contains B_4C pellets 0.860 inch in diameter, while a column of Alumina (AL203) pellets, 0.851 inches in length, encloses each end of the B_4C pellet stack. The initial atmosphere within the poison rods is unpressurized helium. Based on an anticipated burn-up of boron and considering swelling of B_4C and a clad strain limit of one percent, the PLPRs have an estimated lifetime of about ten years. The PLPR's were removed at the end of Cycle 10 and are currently stored in the spent fuel pool.

3.7.3.5 Hafnium Flux Suppression Rods

The Hafnium Flux Suppression Rods, when used, reside in guide tubes of selected peripheral assemblies. Due to space limitations, the rods can not be attached to a spider section. Therefore, each individual rod has a gripper section at the upper end for handling purposes. The gripper section is also spring loaded to provide a holddown function when the upper guide structure is installed. The holddown load is transmitted over the entire length of the rod, causing the tip of the rod to be pressed against the bottom of the guide tube.

The poison rods are composed of unclad hafnium 0.948 inches in outer diameter over the active length of the core. The absence of cladding prevents the localized hydriding and swelling of the rods. An oxide layer forms on the unclad hafnium during exposure to the primary coolant and is an effective barrier to hydrogen absorption. This avoids the concern for the localized hydriding phenomenon in this design extending the expected life of the rods to 15 years.

3.7.3.6 Clad Evaluation

The following information was part of the original fuel design effort and not necessarily reflective of current fuel designs.

The fuel rod cladding is designed to satisfy the design limits given in Section 3.2.3. The effects of irradiation of UO_2 and Zircaloy-4 or ZIRLO have been considered in the design calculations. The predicted effects of anticipated transients have also been considered in the design process.

The design bases are conservative and the calculations used to demonstrate their compliance are conducted for limiting cases using limiting assumptions.

A series of transverse and torsional deflection and thermal bow tests has been performed on a 12 x 12 fuel assembly to provide experimental support to the analytical effort in defining the structural action of a fuel assembly. The information gained from these tests has been used in the design of fuel assemblies and of lifting fixtures and shipping containers.

These tests show that the fuel assembly is sufficiently flexible to accommodate alignment tolerances, has adequate structural stability for reactor operation and maintains its as-fabricated dimensions during handling. All handling from a horizontal to vertical position was performed using an auxiliary support structure.

Based upon the thermal tests, the maximum thermal bow expected under adverse temperature conditions is 7 mils. This amount of thermal bowing has no significant effect on CEA operation.

Clad stress-strain behavior is based upon a triaxial stress analysis which includes the effect of creep. The loads considered are those due to fuel thermal and fission growth, fission gas pressure and external coolant pressure.

The fuel thermal and fission growth was calculated considering the fuel as a solid rod with unrestrained thermal expansion and a volumetric growth rate of 0.16 percent for 10^{20} fissions/cm^3 (Reference 3-31), an average clad temperature of 688°F and a linear heat rate of 17.6 kW/ft. The fission gas pressure was calculated for a 31.5 percent fission gas release which was derived from the data of Hoffmann and Coplin (Reference 3-32) considering the change in plenum volume due to thermal expansion and growth of the rod.

The analysis is based upon an incremental approach which divides the 3-year fuel life span into discrete time intervals and evaluates the clad stress and strain, including the effect of creep, during these intervals. The relation between the incremental creep and the actual stress state is expressed by the Prandtl-Reuss formula (Reference 3-33). The basis for creep is given by the von Mises criterion, (Reference 3-39) and the relation between creep rate and generalized stress is that given by Scott (Reference 3-34). A rapidly convergent iterative technique is employed to solve the resulting nonlinear equations.

For the nominal pellet-to-clad gap, at about 1000 hours after the beginning of life, the fuel has expanded to completely fill the fuel/clad gap and to restore the clad to a circular shape after its initial creep onto the fuel. The fuel is subsequently assumed to swell unrestrained with the clad following. Based upon this conservative assumption, the final strain after 3-years' service is 0.42 percent; that is, for average fuel-to-clad gap at peak power density the strain criterion is satisfied without credit for fuel strain under load.

For the most adverse initial condition, i.e., minimum clad ID, maximum pellet OD coincident with the point of maximum power density which is assumed to be sustained over lifetime, application of the unrestrained fuel growth model in a computed strain at end of life of about 0.91 percent. However, it has been shown (Reference 3-35, 3-36 and 3-37) that the effect of restraint from the exterior cooler regions of the fuel pellet, the clad and the external pressure results in a significant limitation on radial swelling with corresponding flow of pellet material into the dish provided. The assessment of this effect, using the methods of Reference 3-36, gives an upper limit strain for these adverse conditions of 0.73 percent.

These analyses have been conducted throughout with design beginning-of-life power density, although it is known that in fuel in its third burnup cycle, the local power density will be substantially below these values. Thus, the local power density increase which might be associated with overpower transients near end of fuel life has been conservatively considered. The maximum linear heat rating for the first core is 17.6 kW/ft at BOL (actual heat generated in fuel is 97.5 percent of the total heat generated in the core); therefore, actual peak linear heat rate is less than 17.6 kW/ft, and the maximum heat rating near EOC is estimated to be 14.9 kW/ft, resulting in a BOL/EOC ratio of 1.18. This is greater than the value of 1.12 for the ratio of maximum transient to steady state heat ratings. Thus, utilization of beginning-of-life power densities in these calculations for end-of-life transients has provided considerable margin.

Studies by Notley et al (References 3-36 and 3-37), in which 27 fuel elements were irradiated without failure, reported measured clad strains up to 3.33 percent. In a series of experimental element irradiations, Westinghouse (Reference 3-35) reported strain values at failure for Zr-4 clad fuel elements of 0.78 percent to 2.6 percent, depending on the fuel properties assumed. Also, Lustman (Reference 3-38) has noted that failures in-pile have occurred at strain values between 0.5 percent and 1.0 percent. However, these results are based on relatively low Zr-4 cladding temperatures as compared to current large commercial PWR's. It is known (Reference 3-39) that permissible strain values for zircaloy increase above 650°F. The average Zr-4 cladding temperatures of about 688°F in the Fort Calhoun reactor should result in increased ductility and thus higher strain limit to failure.

Westinghouse has conducted similar studies for the 14 x 14 fuel supplied for the Fort Calhoun Unit One reactor. A detailed discussion of this work can be found in WCAP-12977 (Reference 3-22).

Framatome ANP RICHLAND, Inc., also performed a mechanical design evaluation of the Cycle 20 fuel design. This evaluation used the NRC approved mechanical analysis codes and methodology to demonstrate compliance with the NRC approved design criteria.

The Cycle 20 rod performance was calculated using the RODEX2 code (References 3-71 and 3-72), The pellet dimensions, fission gas release, pellet temperatures, cladding creepdown and creepout, cladding temperature, and corrosion were determined over the lifetime of the fuel. The power histories were used to support the Technical Specification power limits. These analyses demonstrate margin to the design limits throughout the lifetime of the fuel.

3.7.3.7 Control Element Assembly

Each CEA is composed of five (5) full length control rods (i.e., fingers) assembled in a square array (i.e., spider), one (1) finger in the center capture tube (i.e., hub) and four (4) fingers at each corner of the spider. The overall length of each finger is approximately 150 inches and the active length is approximately the lower 125 inches.

The active length contains neutron absorbers (i.e., poison) in the Inconel tube and each finger is sealed by a welded end cap at the bottom and by a welded end fitting at the top. The upper 25 inches of nonactive length contains a gas expansion space to limit the maximum tube stress due to internal pressure buildup by the helium gas and moisture release from the B_4C pellets during the reactor operation. All CEA's in the core include an Ag-In-Cd pallet tip to minimize the strains due to swelling induced by high neutron exposure during reactor operation. The general description of the CEA's in the core is shown in Table 3.7-1.

The hub of the spider couples the CEA to the CEDM through the extension shaft and rack. A dashpot is provided in the CEDM to slow down the CEA in the last part of the insertion following a reactor trip as described in Section 3.7-1.

There are a total of 49 CEA's in the core and these are driven by 37 CEDM's on the reactor vessel head. Twenty-one CEDM's drive trippable single CEA's, twelve CEDM's drive trippable dual CEA's and four CEDM's drive non-trippable single CEA's. The dual CEA is made up of two single CEA's connected to separate grippers and carried by an extension shaft. The arrangement of the CEA's in the core is shown in Figure 3.4-1.

The CEA's in the core are grouped into seven different groups to provide flexible operational maneuvering. Groups A and B are shutdown groups; Groups 1, 2, 3 and 4 are regulating groups to achieve reactivity control and Group N is an additional group.

The CEA's can be controlled by the group or by the individual single/dual CEA's. Except for Group N, all of the CEA's will be automatically inserted following a reactor trip. Group N can be inserted manually following a reactor trip.

3.7.3.8 Control Element Assembly Evaluation

Several parallel experimental efforts (see Section 1.4) were pursued to assure that the CEA's will function under a wide variety of adverse conditions that could be encountered under normal and abnormal operation.

The results of the cold water tests indicate that the CEA's will operate satisfactorily and achieve acceptable scram times. Various hydraulic and friction forces developed in scramming a CEA have been ascertained. Measured drop times are less than those assumed in the accident analyses.

Additional cold water flow tests were conducted on CEA's to determine the effects on drop time of guide tube mechanical and thermal bow, core pressure drop, misalignments of all applicable components, guide structure clearance variations above the core, and CEA-to-guide tube clearance variation within the fuel assembly. These test conditions were more severe than the worst accumulations of tolerances and expected operating conditions.

Burst and collapse were also performed on the poison rod cladding. Reactor operating conditions for periods up to 10,000 hours were simulated using tubing which contained defects in the wall and tubes filled with B_4C pellets that have been water logged.

Full size and weight prototype CEA's were installed in a high temperature, high pressure test facility designed to simulate pressurized water reactor coolant conditions. It is an isothermal system with capabilities of operating at temperatures up to 625°F and pressures up to 2550 psig.

Hot flow tests exposing the fuel assemblies, CEA's and a CEDM to long term reactor conditions were also conducted. The main purpose of these tests was to proof test the CEA design. CEA scram time and operational characteristics, and wear and corrosion of CEA poison tube and guide tubes were evaluated.

3.7.4 Vibration Analysis and Monitoring

Design analyses were performed to verify the structural integrity of the Fort Calhoun reactor internals and fuel assemblies. Emphasis was placed on the dynamic analysis of those components which are particularly critical and vulnerable to vibratory excitation. Thermal shields on reactors built prior to Fort Calhoun Station experienced some vibrational problems; however, for the Fort Calhoun reactor, a more reliable design was achieved by using a top vs. a bottom support design, which eliminates a free edge in the flow path; increasing the number of supports to provide a stiffer structure; and using an all welded shield to eliminate local flexibilities and relative motion at bolted joints. Operating data have shown that the thermal shield is stable on its support system when exposed to the axial annular flow.

The response of the fuel assemblies to mechanical and flow excitation was evaluated. The calculated response (amplitude and frequency) of the fundamental mode of vibration of the core support barrel was used as the mechanical excitation of the fuel assemblies. The calculated fuel assembly response was then used to assure that test conditions of fuel assemblies were more severe than expected operating conditions. Vibration analysis of the fuel assemblies demonstrates that the most likely modes of vibration do not coincide in frequency with known excitations.

A digital vibration-loose parts monitoring system as described in Reference 3-40 was installed following Cycle 14. This system was designed to provide monitoring, recording, and analysis for vibration and/or loose parts on the primary coolant loop major components and neutron flux related motion of the core and its components. The system contains accelerometers placed on each of the following locations: lower reactor vessel (2), reactor vessel flange (2), steam generator primary manway (1 on each generator), steam generator secondary handhole (1 on each generator), and steam generator secondary manway (1 on each generator). Six channels of neutron flux signals generated from the excore detectors are routed to the monitoring system for measuring core internals vibration.

The core support barrel, the support structure for the core, was initially analyzed to provide assurance that this major structure does not exhibit excessive vibrations. Vibration analysis of the barrel based on inlet flow impingement forces and turbulent flow were performed to demonstrate that the anticipated RMS response of the barrel would be low. Spectral analyses of plant operations using the vibration-loose parts monitoring system have shown that although core barrel motion is present, it is within acceptable limits.

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