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# Duke Power Company
# OCONEE NUCLEAR STATION
# FINAL SAFETY ANALYSIS REPORT

Revised to Include: 1995 Update
Effective Date of Contents: December 31, 1995

# CHAPTER 1. INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

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# 1.2 GENERAL PLANT DESCRIPTION

## 1.2.1 SITE CHARACTERISTICS

The site is characterized by a one-mile exclusion radius; remoteness from population centers; sound, hard rock foundation for structures; freedom from flooding; an abundant supply of cooling water; an on-site hydroelectric station capable of supplying ample emergency power; and favorable conditions of hydrology, geology, seismology, and meteorology.

## 1.2.2 STATION DESCRIPTION

### 1.2.2.1 General Arrangement

The general arrangement of the major equipment and structures is shown in Figure 1-2 through Figure 1-9.

### 1.2.2.2 Nuclear Steam Supply System

Each Nuclear Steam Supply System consists of a pressurized water reactor and a two-loop Reactor Coolant System. The mechanical, thermal-hydraulic, and nuclear design of the reactor core is similar to other systems operating or under construction.

The reactor core is composed of uranium dioxide pellets enclosed in Zircaloy tubes with welded end plugs. The tubes are supported in assemblies by spacer grid assemblies and the upper and lower end fitting assemblies. The reactor core is initially loaded in three regions of different enrichments. The control rod assemblies consist of clusters of stainless steel clad Ag-In-Cd absorber rods and guide tubes located within the fuel assembly.

The two steam generators are vertical, straight tube units producing super-heated steam at constant pressure. With the once-through design, natural circulation flow is adequate to remove full decay heat without the use of reactor coolant pumps. Thus, with total loss of pumps, departure from nucleate boiling will not occur in the core.

An electrically heated pressurizer establishes and maintains the reactor coolant pressure and provides a surge chamber and a water reserve to accommodate reactor coolant volume changes during operation.

The reactor coolant pumps (two in each loop) are vertical, single speed, centrifugal units equipped with controlled leakage shaft seals.

### 1.2.2.3 Containment System

The prestressed, post-tensioned, steel lined, concrete Reactor Building is designed to withstand the maximum internal pressure resulting from an analysis of a spectrum of Reactor Coolant System leaks.

Isolation valves are provided on fluid piping penetrating the Reactor Building to provide containment integrity when required. These valves are actuated automatically by signals received from the Engineered Safeguards Protective System.

5 All electrical and fluid penetrations with the exception of those penetrations listed in Section 6.5.1.2,
5 "System Design" on page 6-55 are grouped in a penetration room. Any leakage that might occur from any of these penetrations (except the noted lines) will be filtered and exhausted through a unit vent. Access hatches are provided with double seals, and the volume between the seals is piped to the penetration room. Provision is made to leak test all the access hatch closures.

### 1.2.2.4 Engineered Safeguards Systems

Engineered Safeguards Systems reduce the potential radiation dose to the general public from the Maximum Hypothetical Accident to less than the guideline values of 10CFR100. Automatic isolation of Reactor Building fluid penetrations that are not required for limiting the consequences of the accident reduces potential leakage paths. Long term potential releases following the accident are reduced by rapidly decreasing the Reactor Building pressure to near atmospheric, thereby reducing the driving potential for fission product escape.

In addition, the Engineered Safeguards System provides ample core cooling following the worst postulated loss-of-coolant accident. This is accomplished by large capacity, Injection Core Flooding Systems. These systems, coupled with the thermal, hydraulic, and blowdown characteristics of the reactors, reliably prevent metal-water reactions.

Each reactor unit has the following engineered safeguards equipment, with the normal operating mode of each as indicated:

1. High Pressure Injection System - a portion is used in normal reactor operation.
2. Low Pressure Injection System - operates for shutdown cooling.
3. Core flooding tanks - normally ready for operation.
4. Reactor Building Spray System - normally shutdown.
5. Reactor Building emergency coolers - operate for Reactor Building cooling during normal operation.
6. Penetration Room Ventilation System - test operation during normal operation.
7. Reactor Building Isolation System - normally ready for operation and testable.

2 The Engineered Safeguards Systems are independent for each unit. Table 1-2 lists the major equipment in each system.

### 1.2.2.5 Unit Control

The reactor is controlled by control rod movement and regulation of the boric acid concentration in the reactor coolant. Between 15 percent and 100 percent full power the Integrated Control System maintains constant average reactor coolant temperature. Constant steam pressure is maintained over the full power range.

The Reactor Protective System and the Engineered Safeguards System automatically initiate appropriate action whenever the parameters monitored by these systems reach pre-established set-points. These systems act to trip the reactor, provide core cooling, close isolation valves, and initiate the operation of standby systems as required.

### 1.2.2.6 Electrical System and Emergency Power

Each of the three nuclear units at Oconee have up to six available sources of electrical power:

1. Eight 230 kV transmission lines from three directions and three 500 kV transmission lines from three directions serve Oconee.
2. The other two nuclear units.
3. The Central Switchyard or the Lee Steam Station Combustion Turbines via the 100 kV transmission line (capable of being separated from other system loads).
4. One of the quick-starting on-site Keowee Hydroelectric 87,500 KVA Generating Units connected to Oconee by an underground 13.8 kV cable.
5. The other Keowee Hydroelectric Generating Unit connected to Oconee by an overhead 230 kV transmission line.

Oconee has multiple redundant buses and tie buses supplying power to loads, instruments, and controls. The engineered safeguards for each unit are generally arranged on a three-component basis and supplied from three separate auxiliary power buses, each of which can be supplied from any of the six principal sources of power.

The sources of power and associated electrical equipment will insure safe functioning of the station and its engineered safeguards.

### 1.2.2.7 Steam and Power Conversion System

The Steam and Power Conversion System for each unit is designed to remove heat energy from the reactor coolant in the two steam generators and convert it to electrical energy. The closed feedwater cycle will condense the steam and heat feedwater for return to the steam generators.

### 1.2.2.8 Fuel Handling and Storage

Both new and spent fuel are stored in the spent fuel pool and transferred to and from the Reactor Building via the fuel transfer tubes. One spent fuel pool is shared between Oconee 1 and 2, and a separate spent fuel pool is provided for Oconee 3. The system is designed to minimize the possibility of mishandling or maloperations that could cause fuel assembly damage or potential fission product release,
1 or both. In addition to the spent fuel pools, the Independent Spent Fuel Storage Installation (ISFSI) is
1 available, at Oconee, to provide long-term storage of irradiated fuel assemblies. Refer to the ISFSI Final
1 Safety Analysis Report for further details.

### 1.2.2.9 Radioactive Waste Control

Gaseous Waste Disposal Systems collect, holdup as necessary, filter, monitor, release, and record the gaseous effluent from the station. Liquid Waste Disposal Systems provide for collection, storage, treatment, monitoring, disposal, and recording of liquid wastes. Solid radioactive wastes are stored, packaged, and shipped off-site. Greater than originally anticipated gas and liquid waste volumes lead Duke Power to build an interim radwaste facility. This facility consists of liquid processing equipment, volume reduction equipment and associated auxiliary systems. This radwaste facility provides greatly enhanced capacity and greater overall decontamination factors. The volume and types of radioactive effluents which needed to be processed at Oconee increased beyond the capabilities of the interim facility due to more stringent environmental release limits and disposal criteria. A Radwaste Facility has been added to handle these increased volumes and types of wastes. The systems which comprise the facility are Resin Recovery,

Liquid Processing and Recycle, and Waste Solidification. The facility is capable of processing and packaging for burial these types of waste in optimal fashion.

### 1.2.2.10 Standby Shutdown Facility (SSF)

2 The Standby Shutdown Facility provides capability to shutdown the nuclear reactors from outside the
2 control room in the event of a fire, flood, or sabotage-related emergency. It provides additional
2 "defense-in-depth" by serving as a backup to safety-related systems. The SSF has the capability of
2 maintaining hot shutdown conditions in all three units for approximately three days following a loss of
2 normal AC power. It is designed to maintain reactor coolant system (RCS) inventory, maintain RCS
2 pressure, remove decay heat, and maintain shutdown margin.

## 1.3 COMPARISON TABLES

The important design and operating characteristics of the Nuclear Steam Supply Systems for the Oconee Nuclear Station are summarized in Table 1-3.

The more significant design revisions made to the units since the Preliminary Safety Analysis Report are listed below:

Control Rod Drives

Each of the units utilizes sealed roller-nut and leadscrew type control rod drives rather than shaft seal rack and pinion drives.

Fuel Assembly

The fuel assembly utilizes Inconel spacer grids supported by the control rod guide tubes and center instrument tube rather than stainless steel grids supported by an external stainless steel perforated can. All fuel rods except those in the low burnup region of Core 1, Oconee 1, are internally prepressurized with helium to minimize clad fatigue due to power and pressure cycling.

Axial Power Shaping Rod Assemblies

Eight of the 69 control rods contain neutron absorber for a portion of their length to aid in controlling Xenon oscillations.

Burnable Poison Rod Assembly

Burnable poison rod assemblies are used in current cycles to reduce the magnitude of the beginning of life positive moderator temperature coefficient.

Emergency Core Cooling System

The Emergency Core Cooling System piping and equipment have been arranged to meet the intent of Criterion 44.

Reactor Building Coolers

The Reactor Building coolers are designed to serve both the normal and emergency building cooling functions.

In-Core Instrumentation Readout

Auxiliary readout of selected in-core detectors is included as a part of the control room instrumentation.

Control Rod Withdrawal Stop and Trip Signals

4 A wide range high start-up rate signal prevents further withdrawal of control rods. Reactor Trip during startup is furnished by a high power range flux level setting.

Emergency Core Cooling System Actuation

The Emergency Core Cooling System is actuated on high Reactor Building pressure as well as low Reactor Coolant System pressure.

## 1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

Duke Power Company, through its corporate organization, is responsible for the design, purchasing, construction, and operation of Oconee, a practice successfully followed for all of the Company's major generating facilities now in service or planned.

Duke contracted with B&W to design, manufacture, and deliver to the site three complete Nuclear Steam Supply Systems and fuel. In addition, B&W supplied technical direction of erection; and consultation for initial fuel loading, testing, and initial startup of the complete Nuclear Steam Supply System with coordination, scheduling, and administrative direction by Duke.

The Bechtel Corporation was retained by Duke as a general consultant to provide such engineering assistance as needed during the design and construction of the station. Layout, engineering, and design of the Reactor Buildings were assigned to Bechtel.

Duke retained Pittsburgh Testing Laboratory for shop inspection of valves and piping as required. As consultants on seismology and meteorology, the firm of Dames & Moore was retained. Duke also retained Mr. William V. Conn from Atlanta, Georgia, for geology studies and the Law Engineering Testing Company for subsurface investigations under the direction of Dr. George F. Sowers.

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# CHAPTER 2. SITE CHARACTERISTICS

# 2.1 GEOGRAPHY AND DEMOGRAPHY

## 2.1.1 SITE LOCATION AND DESCRIPTION

### 2.1.1.1 Specification Of Location

Oconee Nuclear Station is located in eastern Oconee County, South Carolina, approximately 8 miles northeast of Seneca, South Carolina at latitude 34°-47′-38.2″N and longitude 82°-53′-55.4″W. Duke Power Company's Lake Keowee occupies the area immediately north and west of the site. The Corps of Engineer's Hartwell Reservoir is south of the site. Duke's Lake Jocassee lies approximately 11 miles to the north. Figure 2-1 shows the site location with respect to neighboring states and counties within 50 miles. Figure 2-2 shows the relationship of the site with Lakes Keowee and Hartwell and the topography within 5 miles. Figure 2-3 shows the general geographical and topographical features within 50 miles of the site.

### 2.1.1.2 Site Area Map

Figure 2-4 shows the site layout, property lines, and other structures within the site area. There are no industrial, commercial, institutional, recreational or residential structures within the site boundary.

5 Located within 1 mile of the station center are the Visitors Center, the Keowee Hydroelectric Station, and
5 the Crescent Resources (Keowee Division) office complex and appurtenances. All of these facilities are
5 Duke properties. Old Pickens Church and Cemetery, an historic property which is not in use, occupies a small property to the east of the station.

The exclusion area is defined as a 1 mile radius from the station center.

### 2.1.1.3 Boundaries for Establishing Effluent Release Limits

The boundary for establishing gaseous effluent release limit is the exclusion area. The exclusion area is defined as a 1 mile radius from the station center. The boundary for liquid effluent is a 154 ft. wide by 216 ft. long area at the Keowee Dam extending from the face of the powerhouse to the crest of the tailrace. This area lies within the 1 mile radius for establishing gaseous effluent limits. The exclusion area boundary and the site boundary fences for the liquid effluents are shown in Figure 2-5.

Access to the station property is through gates controlled by security guards. That area outside the station security fence is under control of security personnel.

## 2.1.2 EXCLUSION AREA AUTHORITY AND CONTROL

### 2.1.2.1 Authority

All the property within the 1 mile radius exclusion area is owned in fee, including mineral rights, by Duke except for the small rural church plot belonging to Old Pickens Church, rights-of-way for existing highways and approximately 9.8 acres of U. S. Government property involved with Hartwell Reservoir.

The Hartwell property is either a portion of the Hartwell Reservoir or subject to flooding and not suitable for other uses. Duke has obtained from the owners of the church plot and from the United States the right to restrict activities on these properties and to evacuate them of all persons at any time without prior

notice if, in its opinion, such evacuation is necessary or desirable in the interest of public health and safety.

The property which is within the exclusion area and which is not owned by Duke is shown on Figure 2-4.

#### 2.1.2.2 Control of Activities Unrelated to Plant Operation

5 Unrelated activities are limited to the highways through the Exclusion Area, Duke's Visitor Center,
5 Cresent Resources, recreation on the lakes, and the Old Pickens Church and Cemetery which are
5 historical landmarks and will not be used for regular services. The only commercial enterprises within the
5 Exclusion Area will be Duke's Keowee Hydroelectric Station, Crescent Resources and the Oconee
5 Nuclear Station.

#### 2.1.2.3 Arrangements for Traffic Control

Arrangements have been made with the South Carolina State Highway Department to control and limit traffic on public highways in the Exclusion Area should it become necessary in the interest of public health and safety.

### 2.1.3 POPULATION DISTRIBUTION

The 1970 population distribution is based on the 1970 census. The 2010 population projection is a linear extrapolation of the 1910-1960 long term trend adjusted upward to anticipate lake proximity developments extending out as much as 20 miles from the site, particularly in the NW and NNW sectors.

Figure 2-6 shows the location and population of population centers within 100 miles of Oconee. The largest city, Knoxville, Tennessee, located 97 miles northwest of the site, had a 1970 population of 174,587. The nearest population center is Anderson, South Carolina with a 1970 population of 27,556.

#### 2.1.3.1 Population Within 10 Miles

Table 2-1 gives the 1970 population distribution within 10 miles of Oconee. The projected population
2 for 2010 are shown on Table 2-2. The 1990 population distribution is shown in Section J of the Oconee
2 Nuclear Site Emergency Plan.

#### 2.1.3.2 Population Between 10 and 50 Miles

Table 2-3 and Table 2-4 show the 1970 and projected 2010 population distribution. Figure 2-6 shows population centers within 100 miles of the site.

#### 2.1.3.3 Transient Population

2 It is expected that Lake Keowee's 300 mile shoreline will be fully developed by 2015 at which time the estimated transient population will be 36,000. This estimate is based on development of lakeside lots, public access areas, and expanded commercial activities to take advantage of expanded recreational opportunities. There will not be any cottages within the Exclusion Area.

The estimated transient population within the low population boundary is 2000 for 1970 and 19000 for 2010.

The visitors center, located on Duke Property just north of the plant and within the Exclusion Area, was host to 510,000 people during its first 25 months of operation.

There are no industries within 5 miles of the site therefore no industrial transients.

### 2.1.3.4 Low Population Zone

The actual permanent population within the low population boundary (6 miles from site) is 3620 for 1970 and estimated to be 8900 for 2010.

### 2.1.3.5 Population Center

The nearest population center is Anderson, South Carolina, located approximately 21 miles to the south southeast of the plant (Figure 2-6).

### 2.1.3.6 Population Density

Table 2-5 and Table 2-6 tabulate the population density to 50 miles for 1970 and projected density for 2010.

# 2.2 NEARBY INDUSTRIAL, TRANSPORTATION, AND MILITARY FACILITIES

## 2.2.1 LOCATION AND ROUTES

Figure 2-3 shows the transportation routes within 5 miles of Oconee. There are no oil or gas pipelines
1 within 5 miles of the site, except that natural gas distribution pipelines are located approximately 3.5 miles
1 from the site in the direction of Six Mile.

## 2.2.2 DESCRIPTIONS

### 2.2.2.1 Description of Facilities

There are no industrial or military facilities or activities within 5 miles of Oconee.

### 2.2.2.2 Description of Products and Materials

The highways passing through the 1 mile radius exclusion area are local roads with infrequent trucking of hazardous chemicals and explosives since the general area is nonindustrial.

Only small amounts of chlorine are stored on-site since chlorine is not used for condenser cleaning at Oconee. No individual container contains more than 150 pounds of chlorine. The chlorine is used for
5 waste treatment, with three to five 150 pound containers typically being in use. The maximum total number of containers on hand at any time is approximately ten.

### 2.2.2.3 Pipelines

1 There are no pipelines within 5 miles of Oconee, except for natural gas distribution pipelines located
1 approximately 3.5 miles from the site in the direction of Six Mile. The lines, which run parallel to
1 highway 183, are considered high pressure with an operating pressure of approximately 400 psi.

## 2.2.3 EVALUATION OF POTENTIAL ACCIDENTS

### 2.2.3.1 Determination of Design Basis Events

#### 2.2.3.1.1 Explosions

An incident involving fire, chemicals or explosives at the closest point along the highway would be more than 1000 feet from the Reactor and Auxiliary Buildings. We believe that fire or chemical reactions at this distance would not affect plant operation. The blast pressure (Reference 1 on page 2-9) from a truck loaded with 40,000 pounds (Reference 2 on page 2-9) of TNT at this distance would be less than the design tornado loading on the structures.

0 #### 2.2.3.1.2 Deleted per 1990 Update

#### 2.2.3.1.3 Toxic Chemicals

If a highway incident should result in the release of toxic gases, the gases under most circumstances would either move in a direction away from the plant or be sufficiently dispersed by the time they reach the plant that they would not interfere with the safe operation of the plant. But if adverse environmental conditions should make it necessary, the plant could safely be operated or shut down from the control room. The control room is an enclosed area which can be isolated from the outside environment. Portable breathing equipment is also provided to allow access to areas outside the control room.

Only small quantities of chlorine are stored on-site since chlorine is not used for condenser cleaning at Oconee. No individual container on the site contains more than 150 pounds of chlorine. The chlorine is used for sanitary waste treatment, with three to five 150-pound containers typically being in use, and the maximum total number of containers on hand at any time is approximately ten. It is unlikely that leaks from these small chlorine containers could result in dangerous concentrations in the control room, but the control room can be isolated from the outside environment if necessary and portable breathing equipment, suitable for protection against chlorine, is also provided.

54.4% hydrazine is also stored on-site in 55 gallon drums and 350 gallon containers. Hydrazine is used to maintain feedwater chemistry during power operation and steam generator wet layup chemistry during outages, with three 350 gallon containers (one for each unit) typically being in use. 54.4% hydrazine, diluted to 10%, is also used, as needed to reduce reactor coolant system dissolved oxygen concentrations during unit startups. The maximum combined total number of containers (350/55 gal) on-site at any time is approximately ten. It is unlikely that leaks from these hydrazine containers could result in dangerous concentrations in the control room. In addition, the control room can be isolated from the outside environment and portable breathing equipment is also available.

#### 2.2.3.1.4 Fires

Liquid material spills would follow the pattern of roadside drainage toward Lake Keowee and Keowee River. On the event flammable material should reach the cooling water intake structure and burn, the cooling water pumps and related equipment would likely not be affected, but the operation of these pumps is not required for plant safety, and the most serious consequence would be a plant shutdown due to lack of condenser cooling water.

### 2.2.3.2 Effects of Design Basis Events

No design basis events have been identified in Section 2.2.3.1, "Determination of Design Basis Events" on page 2-7.

## 2.2.4 REFERENCES

1. Effects of Impact and Explosion, AD 221 586, National Defense Research Committee, Vol. 1, 1946.
2. Interstate Commerce Commission and Department of Transportation Regulations of Maximum Truck Limit.

# 2.3 METEOROLOGY

Meteorology is evaluated for use in structural design and in consideration of environmental safeguards for gaseous releases. The following paragraphs summarize the atmospheric characteristics pertinent to these design bases.

## 2.3.1 REGIONAL CLIMATOLOGY

### 2.3.1.1 General Climate

In addition to synoptic features that are modified in the crossing and descent of the Appalachian Mountains, the mountains cause channeling of surface winds. As a result, the prevailing wind direction is bimodal, with maximum frequencies in the sectors north-northeast to east-northeast and southwest to west.

### 2.3.1.2 Regional Meteorological Conditions for Design and Operating Bases

In general, the threat of tropical storms in the fall months of the year (and sometimes in other months) is present almost every year. Table 2-7 indicates the frequency of occurrences of conditions which produce some effect on the weather at the nuclear plant site. In the 95 years of record shown, 164 storms of tropical origin affected the area in some manner. There were only 11 years in the 95 in which no storms affecting the area occurred. There were six years where more than twice the average number of storms occurred.

Despite the fact that so many storms have influenced the area, no hurricane conditions which would include damaging winds of major proportions have ever been reported, so far as is known. Normally, by the time a tropical cyclone has passed onto the continent to the nuclear site area, winds have always been reduced below hurricane strength. However, major problems have been encountered with rainfall amounts generally four to five inches within a 24-hour period and occasionally up to nine to ten inches. Stations within a 50-mile radius of the nuclear site have reported up to double the latter amount but normally over more than a single 24-hour period (References 1 on page 2-34, 2 on page 2-34, 3 on page 2-34, and 4 on page 2-34).

Tornado events are rather rare and cover extremely small areas. In order to provide for more than a superficial estimate, it was decided to ascertain the frequency of tornadoes for Oconee County in South Carolina as well as those which occurred in the peripheral counties in Georgia, South Carolina, and North Carolina. Accordingly, records were examined for the following counties:

In Georgia: Rabun, Habersham, Stephens, Franklin, and Hart

In South Carolina: Oconee, Pickens, and Anderson

In North Carolina: Macon, Jackson, Transylvania, and Henderson

(References 5 on page 2-34, 6 on page 2-34, 7 on page 2-34, 8 on page 2-34, 9 on page 2-34, and 10 on page 2-34 were consulted.)

The records revealed that five tornadoes have occurred in Oconee County and 17 the peripheral counties in the 50-year period from 1916 through 1965. These storms, however, were only those which had tracks long enough to plot. In order to gain a more realistic figure, the overall statistics showed that each of these figures should be multiplied by 2.5 yielding 55 tornadoes in the 12-county area in the 50-year period.

This is considered a reasonable estimate of those tornadoes which reached the ground. Funnel clouds not reaching the ground have also been observed but are not included in the above statistics. Tornadoes reach their maximum frequency during the spring months of the year and normally are more likely in April and May at the site.

The values above indicate only 13 tornadoes in Oconee County in the 50-year period and the relative incidence of tornadoes proximal to the site area is small.

Table 2-8 indicates the mean number of thunderstorm days which are encountered in the plant site vicinity. A thunderstorm day is defined as a day in which thunder is heard at any time in the 24-hour period. Past experience indicates that increasing the thunderstorm day statistic by 10 to 15 percent will provide a reasonable estimate of the frequency of actual thunderstorms in the area.

The site is located in a region characterized by a generally high frequency of low wind speeds and calm conditions. These characteristics lead to a relatively high forecast of high-pollution-potential days as shown in Figure 2-7. The duration and frequency of calm and near-calm conditions for three nearly locations are tabulated by season in Table 2-9.

## 2.3.2 LOCAL METEOROLOGY

### 2.3.2.1 Normal and Extreme Values Of Meteorological Parameters

Table 2-10 illustrates the overall wind direction and speed statistics for a five-year period at Greenville, South Carolina. In general, the NE sector and the WSW sector (22.5 degree sectors) dominate the flow over the site area. The NNE, NE, and ENE sectors account for 30.7 percent of all winds while the SW, WSW, and W sectors account for 25.2 percent. These sectors combined then account for 55.9 percent of all winds at the Greenville, South Carolina airport station. This dominance is important as it continues to appear in all wind statistics in varying degrees as the study progresses. Apparently, the main reason for this dominance is the nearby Appalachian Mountain range which causes surface winds to channel toward these directions whenever the opportunity affords itself. The wind rose is schematically shown in Figure 2-8.

Winds of three knots or less occurred 17.4 percent of the time at Greenville. Winds greater than ten knots appear to favor the prevailing directions. (One knot = 0.515 meters per second.)

Table 2-11 illustrates the diurnal variation of wind speeds at various hours of the day. Lighter winds dominate the nighttime hours while the strongest winds tended to occur in the afternoon. The statistics illustrate the typical diurnal pattern of wind speeds.

Table 2-12 shows the frequency of calms and near-calm (winds equal to or less than one knot) conditions at three locations. Calm conditions occur on the average some 332 hours per year or about 4.0 percent of the time. Of these calms, 93.4 percent last less than six hours. Wind speeds equal to or less than one knot occur 4.21 percent of the time and of these conditions 93.5 percent last less than six hours. (The prolonged calm condition shown on Table 2-12 in the 36 - 41 hour winter block was investigated. The observation was made at Charlotte, North Carolina immediately after the anemometer had been moved from a building top level to the ground. Thus one can ignore this as a statistic applicable to the discussion.)

Reference 14 on page 2-34 indicates that winds can be expected to reach a highest speed in excess of 50 miles per hour in any month of the year as an estimate of maximum winds to be encountered. Fourteen years of record for Greenville, South Carolina Municipal Airport indicate that 50 miles per hour has been exceeded at least once for every month of the year except September where it was 47. Two months of the

year showed values of 70 and 79 miles per hour, the former in January 1948, and the latter in October 1946. Clemson, South Carolina records (Reference 15b on page 2-34) indicate that the highest one-minute wind speed was 73 miles per hour in June of 1948.

Table 2-13 and Table 2-14 illustrate the percentage distribution of annual winds at Athens, Georgia as observed at 0630 Eastern Standard Time. These statistics are derived from an analysis of the Adiabatic Chart records of the Athens, Georgia Rawinsonde data. The period of record is December 1, 1959 through November 30, 1961. The data have been analyzed and documented in Reference 16 on page 2-35. The wind roses are schematically shown in Figure 2-9 and Figure 2-10.

The winds observed over Athens, Georgia are probably more representative than those from any other Rawinsonde station near the site. Note that the height above ground in each table is variable. This is because the winds are normally transmitted at standard pressure levels in the atmosphere rather than at fixed heights.

Table 2-13 indicates that the wind sectors which dominate the flow at around 1000 feet above terrain are the NE, ENE, and E sectors (24.54 percent) and the W, WNW, and NW sectors (33.43 percent). These sectors combined account for about 58 percent of all winds. Compared to the surface winds, there has been a shift of dominance from the northeast sector to the northwest - more westerly flow. Calms occurred only 12 times.

Table 2-14 portrays wind conditions 2300-2800 feet above the ground. At this level, wind sector dominance has shifted to westerly flow. In fact, the WSW, W, and WNW sectors account for 33.6 percent of all winds, whereas the SW, WSW, W, WNW, and NW sectors account for 49.8 percent of all winds. Calms occurred less than eight times in the total period of record.

The combination of the surface and upper winds indicates that in the layer between the ground and about 3000 feet, there is likely to be considerable wind shear. As a matter of interest, the change in wind direction with height was examined in a previous study (Reference 16 on page 2-35) as a function of the lapse rate in the lower 50 meters of the atmosphere. The results for the two-year period of record are shown in Table 2-15. Note that the directional shear for stable conditions is from 50 to 100 percent greater than for unstable conditions. This favors slightly greater diffusive properties at the site than is calculated with a single wind direction prevailing throughout the diffusion period during a stable condition, particularly if any significant depth of atmosphere is taken into account.

Figure 2-11 represents cumulative probability of wind directional persistence at Greenville, South Carolina, for winds observed annually. Curve A represents the duration of persistence for a single sector wind direction, i.e., from the northeast, or from the southwest. Note that about 70 percent of all wind directions persist for only one hour. About 94 percent persist for three hours or less, etc.

Curve B indicates the persistence of a single wind direction plus or minus one additional direction on either side of the prime direction, i.e., northeast plus north-northeast and east-northeast (67.5 degrees). Curve B shows that 93 percent of all winds persist five hours or less under these conditions. Curve C indicates the persistence of a single wind direction plus and minus two additional directions on either side of the prime direction (112.5 degrees). About 90 percent of all wind directions persist for ten hours or less.

The above wind persistence statistics are derived for all wind directions, including calms. Directional persistence statistics are also calculated. However, the statistics for a single wind sector essentially show similar results to Curve A. Table 2-16 reveals persistence values by direction. Two values are shown for each of the two seasons, the average value P, and the root-mean-square value RMSP. The merit of the

RMSP values is that these are reasonable approximations of the 65 to 70 percent frequency of occurrence level. In other words, 65 to 70 percent of all persistence values were less than the RMSP figures.

The remaining two columns in each case are those specific events when the wind condition persisted 24 hours or more. (1-41 means one case of 41 hours duration.)

Table 2-16 deals with wind directions for a single 67.5 degree sector (or single sector plus and minus one sector). Table 2-17 deals with single wind directions for a single 112.5 degree sector (or single sector plus and minus two sectors).

Table 2-16 reveals that the most persistent winds come from the prevailing directions as might be expected. Table 2-17 shows a more confused pattern in general, but again shows prevailing wind dominance.

The nearest station of long-term surface temperature is that of Clemson, South Carolina where some 68 years of record are available. The means and extremes shown in Table 2-18 for minimum temperatures are all on the cooler side than records available from the Greenville WBAS, South Carolina weather station and are regarded as more representative of the nuclear site area. The References for these records are listed as 15a on page 2-34 through 15f on page 2-35.

Clemson, South Carolina records are also used to gain estimates of rainfall statistics. Some 71 years of record are available as shown in Table 2-19. Again References 15a on page 2-34 through 15f on page 2-35 are used as source material. Considerable fluctuation in precipitation from month to month and from year to year is experienced from the normals shown in Table 2-19. From a brief examination of Reference 14 on page 2-34, it can be postulated that the normal annual precipitation for the site area is actually about ten percent higher than at Clemson. It is interesting to note that the maximum rainfall occurrences in short periods of time have all been associated with proximal tropical storms or their aftermath. However, severe thunderstorms can produce similar amounts of rainfall in the same periods of time.

By dividing the wind directional frequency for heavy precipitation intensity by the total precipitation wind directional frequency for each direction, directions which are more likely to produce heavy precipitation can be determined. Those directions which produce frequencies greater than the average are north through west and southeast plus south-southeast. These are directions which dominate the showery weather regimes at the site, particularly the thundershowers.

Precipitation occurs only 9.8 percent of all hours of the year.

Statistics related to wind directions and speeds while precipitation is falling are shown in Table 2-20. The most frequent wind sectors are NNE, NE, and ENE which account for 52 percent of all precipitation winds. The table is set up in terms of precipitation intensities. Precipitation rates determine these intensities and are normally classed as light, moderate and heavy. Approximately 90 percent of all precipitation at Greenville, South Carolina during this five-year period was light, seven percent was moderate, and about three percent was heavy. The precipitation wind rose is schematically shown in Figure 2-12.

Comparison with all of the surface wind data in Table 2-10 shows that with winds from the southwest through west to north (the mountain exposure side), precipitation occurs about five percent of the time, while all other directions experience twice this percentage.

In 1961, Pasquill (Reference 18 on page 2-35) suggested that a relationship might be established which would be useful for estimating the frequency of various wind-temperature lapse rate conditions for a given area. The inputs were:

Time of Day

Cloud Cover

Surface Wind Speed

The wind speed that was used was that observed at ten meters above the ground. Essentially his classification system identified six categories of stability regimes. These have come to be known as Pasquill categories. These are:

| Pasquill Categories | Stability Class |
|---|---|
| A | Extremely Unstable |
| B | Unstable |
| C | Slightly Unstable |
| D | Neutral |
| E | Stable |
| F | Extremely Stable |

Although Pasquill suggested the initial classification scheme, it remained merely a scheme until Turner (Reference 19 on page 2-35) quantified it into a reasonably rigorous method. The technique is amenable for use with standard United States Weather Bureau hourly weather observations which are readily available at the National Climatic Center at Asheville, North Carolina, for certain specific United States Weather Bureau weather stations - namely those which observe the weather 24 hours per day throughout the year.

The closest station to the site which maintains such records is Greenville, South Carolina. Data was procured for the Greenville WBAS, South Carolina location (References 20 on page 2-35 and 21 on page 2-35) and the classification of the hourly weather records into Pasquill categories was accomplished for the two-year period of records selected for analysis.

The Pasquill categories selected follow:

| Pasquill Category | Stability |
|---|---|
| A-B | Unstable |
| C | Slightly Unstable |
| D | Neutral |
| E | Stable |
| F | Extremely Stable |

The period of record was December 1, 1959 through November 30, 1961. The results of these classifications are shown in Table 2-21 and Table 2-22. A wind direction rose for Pasquill E and F conditions is shown on Figure 2-13.

Table 2-21 shows the percentage frequency of occurrence of the Pasquill categories and their associated mean wind speeds by direction. All values in the percentage columns are in terms of percent of total observations. Column 1 deals with the Pasquill C category, Column 2 with the Pasquill D category, Column 3 with the Pasquill E and F categories, while Column 4 deals with the Pasquill F category alone. All winds are in knots. Total percentages by categories are also shown.

Table 2-22 completes the Pasquill classification effort. Column 5 deals with Pasquill A-B, unstable categories or "Lapse" conditions. Column 6 deals with a category which normally falls under Pasquill A-B but does not if a stack is used to vent at the site. Column 6 indicates the percentage frequency of fumigation from a stack release. Fumigation is typical of the early portion of the day between sunrise and roughly ten AM.

Column 7 shows the results of combining all wind data. Note particularly the dominance of northeasterly and west-southwesterly flow in the sample data. Column 8 shows the results of a much larger sample of data taken for the entire five-year period, 1959-1963, (Reference 12 on page 2-34).

The frequency of wind directions for the limited sample shown in Column 7 is correlated with the much larger sample shown in Column 8. The correlation coefficient is +0.987, showing that the limited sample indeed possesses a very high agreement with the much larger sample.

Work completed over a period of years has produced a useful relationship which was applied to the nuclear site area in mountain-plain relationships. It is found (Reference 22 on page 2-35) that with terrain differences of greater than about 200 feet, the minimum or early morning temperature observed on hilltops is fairly representative of the free air temperature at the same altitude above proximal valley locations. Thus, it is possible to obtain estimates of the frequency of temperature inversions by comparing hilltop minimum temperatures with valley floor minimum temperatures. Subsequent tower measurements in the same valley location indicate that this postulation, indeed, possesses considerable merit in assessing the strength and frequency of the low-level temperature inversions. Examination of climatic records (Reference 23 on page 2-35) for South Carolina indicates that some estimate of temperature inversion frequency might be possible through a comparison of daily minimum temperatures from Paris Mountain Fire Tower, located seven miles north of Greenville, South Carolina, at an altitude of 2047 feet and Clemson, South Carolina, at an altitude of 850 feet.

Limited data permitted the analysis of some 602 days representing the four seasons of the year for the two-year period of December 1, 1959 through November 30, 1961. It is possible to examine the daily minimum temperature difference (Paris Mountain Fire Tower minus Clemson) for these days and compare these differences with Pasquill Stability classes as observed from hourly weather observations at Greenville, South Carolina, on the same days at hours near dawn. Table 2-23 shows the results. The table essentially shows that, in general, the Pasquill classes do match the proper average temperature differences.

Combined Pasquill E and F conditions logged for the entire two-year period from Greenville, South Carolina, for the dawn hour revealed the following frequency of inversions by season:

| | Frequency of Pasquill E and F Conditions (Inversions) | | | |
|---|---|---|---|---|
| | Winter | Spring | Summer | Fall |
| Two years of Dawn-Hour Records at Greenville, South Carolina | 43.96% | 56.52% | 65.58% | 60.56% |
| 602 Days of Paris Mountain-Clemson Records | 49.14% | 54.30% | 67.41% | 53.10% |

As a result of the above, it appears that the estimates shown by the Pasquill Stability classes are reasonable estimates for inversion data at and near the proposed nuclear site.

STAR Processing of Greenville-Spartanburg Airport is shown for the period January, 1975 - December, 1975 in Table 2-24. The five-year period of January, 1968 - December, 1972 is shown in Table 2-25. The STAR program gives annual joint frequency distributions of wind speed and wind direction by atmospheric stability. These tables will be used to judge the representativeness of a year of onsite data with regard to long-term conditions (e.g., five-year period) as described in Section 2.3.3, "Onsite Meteorological Measurements Programs."

### 2.3.2.2 Potential Influence Of the Plant and its Facilities on Local Meteorology

Several modifications to the local climatology occur as site development progresses. The initial clearing and leveling of land at the specific site location produces an increase in drainage potential of light winds within the site boundary.

The addition of the large bodies of water has three effects on meteorology. First, it lessens ground frictional effects and tends to increase the wind speeds, most noticeably under light wind conditions. Second, the large bodies of water increase the humidity by about ten percent in the area and tend to decrease the frequency of Pasquill F and to increase the frequency of Pasquill E conditions. Third, the creation of a major lake area in the vicinity of the nuclear plant serves to increase the precipitation approximately an additional five to ten percent.

The heat load on the lake, due to the operation of the nuclear plant, results in additional local fogging during some days of the year, although the area beyond the lake that is affected is not expected to be large. The increase of temperature of the lake results in the evaporation of about 32 million gallons of additional water per day from the reservoir into the atmosphere.

The incremental offset in the diffusion climatology due to heated water discharge should be in the direction of improvement, but is not of a magnitude to warrant special emphasis. The effect of warmer surface waters in the vicinity of the discharge increases the speed change of air flow from land to water and decreases the change of wind range for such trajectories (Reference 24 on page 2-35). In regard to further modification of low-level stability, additional enhancement is tempered, to some extent, from effects of the relatively large deep reservoir. A conservative assessment would assume some improvement, but minimal impact on the total climate.

Figure 2-2 shows a detailed topography, as modified by the plant, to 5 mi. Figure 2-3 shows the general topography within a 50 mi. radius of the plant.

Figure 2-14 through Figure 2-23 show plots of the maximum elevation versus distance from the center of the plant in each of the sixteen 22.5 degree compass point sectors radiating from the plant to a distance of ten miles.

## 2.3.3 ONSITE METEOROLOGICAL MEASUREMENTS PROGRAMS

0 ### 2.3.3.1 Early Meteorological Studies (1966-1975)

Onsite meteorological measurements used in diffusion analyses were conducted for various time periods and measurement locations. These time periods include October 19, 1966 through January 23, 1967, June 19, 1967 through May 31, 1968, March 15, 1970 through March 14, 1972, and January, 1975 through December, 1975. Data for the period June 19, 1968 through June 19, 1969 is discussed in relation to the valley drainage model in Section 2.3.4, "Short-Term Diffusion Estimates" on page 2-22.

The evaluations of two comprehensive meteorological surveys conducted on-site confirm that the meteorological characteristics of the site are favorable for the Oconee Nuclear Station.

The first survey, started in mid-October 1966 and extended until late October 1967, was a study of near-ground diffusion climatology. Wind data were continuously recorded by a Packard Bell Electronics Corporation WS-101 system mounted on a 14 meter pole located near mid-site (see Figure 2-24). Temperature gradients were determined by thermographs located in standard United States Weather Bureau Cotton Region Instrument Shelters stationed on the site at varying terrain elevations. A standard recording precipitation gage with wind shield was installed near the base of the 14 meter pole. After initial installation and check-out, the instrumentation was checked twice weekly until performance of the instrumentation was verified. Thereafter, a weekly check was maintained. A total of 8619 hourly observations were considered valid out of a possible 9058 or about 95 percent. The results of this study established the frequency of wind conditions with varying lapse rates near ground. The results are shown below:

1. Frequency of temperature inversions of total hourly observations was 24 percent.
2. Direction of predominating inversion wind was north (Figure 2-25).
3. Inversion wind speed average was 1.40 meters per second.
4. The minimum average standard deviation of inversion winds in any sector for the one year averages 6.6 degrees.

Wind roses presenting near-ground data, Figure 2-25, Figure 2-26, and Figure 2-27 compared to Greenville-Spartanburg, South Carolina Airport data (Figure 2-8 and Figure 2-13), reflect wind reorientation by nearby mountain ranges and some channeling by the river valley.

The second survey was started in June of 1967, using the permanent station equipment at the time, to establish meteorological parameters related to elevated (vent) releases. Reference to Figure 2-24 illustrates the arrangement of meteorological instrumentation required to initiate this study. Investigations of winds and atmospheric stability were made at vent effluent levels by wind and temperature gradient measuring systems mounted on the 46 meter tower. In addition to tower meteorological instrumentation, a standard weather instrument shelter containing a thermograph and a mercury-in-glass dry bulb thermometer, for comparison was set up near the tower base. A standard recording precipitation gage with wind shield was also installed nearby.

A brief summary of data through the first year (June 19, 1967 through May 31, 1968) shows the following:

1. The average wind speed recorded by the anemometer at elevation 1028 ft (232 ft above plant yard level) was 6.5 miles per hour or about 3 meters per second for all conditions, and about 2 meters per second during inversions.
2. The dominant all-wind direction was northerly which accounts for 10.98 percent of all observations (Figure 2-28 and Figure 2-29).
3. The average standard deviation associated with winds less than 1 meter per second was about 22 degrees. As expected, the standard deviations decreased generally as wind speeds increased.
4. A frequency of inversions of approximately 40 percent was found for the one year of tower data compared to 24 percent for near-ground observations. Although the two periods of observations are not chronologically identical, one would expect the inversion duration time to be less near-ground due to more rapid inversion "burn-off"; however, it is also noteworthy that the frequency of inversions for the Greenville-Spartanburg Airport for Pasquill-Turner computations also increased for the year during tower observations compared to near-ground observation period.

5. The maximum amount of rain was from the northeast where during the year 7.09 inches of rain fell in an aggregate of 71 hours (Figure 2-30).

### 2.3.3.2 Continuous Meteorological Data Collection

Meteorological data has been taken continuously onsite since June 23, 1967. Meteorological measurements include wind direction and speed, horizontal wind direction fluctuation, temperature, and vertical temperature gradient. The current relative position of instruments with respect to station yard is noted in Figure 2-5. Relative elevations of both surface levels and instrument levels are depicted in Figure 2-24.

1967-1984

Until April 9, 1984, wind measurements were made with the Packard Bell Model W/S 1O1B series wind direction-speed system, with starting thresholds of 0.7 and 0.6 miles per hour for direction and speed respectively. Temperature and delta temperature measurements were made with the Leeds and Northrup 8100 Series 100 ohm resistance temperature device with Packard Bell Model 327 thermal radiation shields. Wind direction and speed were recorded in the control room on Esterline Angus Model A 601 C strip chart recorders with a system accuracy of ± 5.4 degrees for direction and ± 0.45 miles per hour for speed. Temperature and delta temperature were recorded on the Leeds and Northrup Speedomax W recorder with a system accuracy of ± 1F for temperature (at 10m level) and ± 0.5F for delta temperature (46m level referenced to the 10m level). During the period June 23, 1967 to February 24, 1977, delta temperature was measured between the 46m level and the 1.5m level. Rainfall was measured with the Belfort Weighing Rain Gauge, Model 5-780 with an accuracy of ±0.03 in. and ±0.06 in. for zero to six and six to twelve inch totals, respectively.

The location of the 46m wind measuring sensors on the microwave tower (Figure 2-5), was appropriate for estimating wind direction and speed for vent releases. Inasmuch as low level flow direction could not be adequately represented by a 10m sensor due to 20m trees near the tower base, all low level input was derived from sensors atop the tower. Wind speed was adjusted by a power law relationship in accordance with the evaluation in Reference 25 on page 2-35. The location of the 46m meteorological system was taken as reasonably representative of topography in the vicinity of the plant with respect to wind direction and vertical temperature gradient. The surface immediately below the tower was characterized as a grassy area.

The effect on vertical temperature gradient from the positioning at 1.5m was to introduce some uncertainty where partially elevated releases were concerned. Consequently, the lower level was moved to 10m on February 24, 1977. The following is offered as the limits in uncertainty in delta temperature measured at 1.5m. A bias toward very unstable lapse rates during the day is seen by the occurrence of intense lapse conditions in the existing data. It was suspect, however, after observing daytime stability Class A rates at other Duke Power lake sites, in that the total number of Class A conditions would not change appreciably with the lower sensor at 10 meters. The effect of the 20 meter trees on unstable lapse rates should not have been significant. These trees were not sufficiently dense to constitute a canopy, and the effect could be disregarded during well mixed conditions. The bias towards more stable profiles at night did not readily appear in the strength of inversions typical at the site. This condition was not unexpected since the 20m trees would provide radiative exchange, to some extent, tending to sustain relatively warmer temperatures near the ground. Assuming the effect of the trees was to shift the temperature profile below 20 meters toward a less stable rate, the measured gradient with the 1.5m sensor could be slightly less stable or slightly more stable than a gradient measured with a 10m sensor. No pronounced bias toward anomalously stable conditions is expected in the pre-February 24, 1977 data.

Table 2-26 depicts joint frequencies of wind direction and speed by stability class for the period March 15, 1970 through March 14, 1972. Stability is defined in terms of vertical temperature gradient and
0 indexed as follows, for the period:

| Stability Class | Vertical Temperature Gradient Class Interval | |
|---|---|---|
| G | > | + 3.3F in 145 ft. |
| F | + | 1.3 to + 3.3F in 145 ft. |
| E | - | 0.5 to + 1.2F in 145 ft. |
| D | - | 1.2 to - 1.6F in 145 ft. |
| B-C | - | 1.5 to - 1.3F in 145 ft. |
| A | < | - 1.5F in 145 ft. |

Table 2-27 is a display of the joint frequency of wind direction and speed by atmospheric stability type for both low-level and high-level wind summaries for the period, January, 1975 through December, 1975. Comparison of Table 2-27 with Greenville-Spartanburg Airport data (Table 2-24 and Table 2-25) forms the basis for judging the representativeness of data for this time period with regard to long-term conditions (e.g., five year period). Consideration of wind speed by stability type for the two periods shows a lower wind speed in general for the period January, 1975 through December, 1975; the occurrence of calms and winds less than 4 knots are up about four percentage points from 23 percent for the period January, 1968 through December, 1972. A slight shift in stability is noted for the period January, 1975 through
0 December, 1975; intermediately stable and unstable Classes E, F, and C, respectively, decreased while
0 strongly stable and unstable Classes G, A, and B increased. Minor changes in wind direction frequencies are also noted for the period January, 1975 through December, 1975; prevailing wind sectors north,
0 northeast, south, southwest, and south-southwest increased their frequency, at the expense of the other sectors. On balance, the period is taken as reasonably representative of long-term conditions in the vicinity of the site.

0 1984-1988

0 Since April 17, 1984, operational measurements have consisted of near real-time digital outputs in addition
0 to the previously described analog system. An entirely new set of instrumentation was installed at this
0 time, including the measurement of dew point at the 10 m level. A supplemental low-level wind system at
0 10 m level provides input for emergency dose assessments (see Figure 2-5 and Figure 2-24). The type of
0 rain gauge was changed to a tipping bucket rain gauge, and was relocated near the supplemental wind
0 system, as well.

Instrument specifications for operational measurements are:

1. Wind Direction
   a. Manufacturer Teledyne Geotech
   b. Time-averaged digital accuracy ± 3 degrees of azimuth
   c. Time-averaged analog accuracy ± 6 degrees of azimuth
   d. Starting threshold 0.3m/sec at 10 degrees initial deflection
   e. Damping ratio 0.4 at 10 degrees initial deflection
   f. Distance constant 1.1m
2. Wind Speed
   a. Manufacturer Teledyne Geotech

0 b. Time-averaged digital accuracy ± 0.27 m/sec for speeds < 27 m/sec

0 c. Time-averaged analog accuracy ± 0.40 m/sec for speeds < 27 m/sec

0 d. Starting threshold 0.45 m/sec

e. Distance constant 1.5m

3. Temperature

a. Manufacturer Teledyne Geotech

b. Time-averaged digital accuracy ± 0.3 degrees C

0 c. Time-averaged analog accuracy ± 0.5 degrees C

4. Delta Temperature

a. Manufacturer Teledyne Geotech

0 b. Time-averaged digital accuracy ± 0.10 degrees C

0 c. Time-averaged analog accuracy ± 0.15 degrees C

5. Dew Point

a. Manufacturer General Eastern

b. Time-averaged digital accuracy ± 0.4 degrees C

0 c. Time-averaged analog accuracy ± 0.6 degrees C

6. Precipitation

a. Manufacturer Teledyne Geotech

b. Digital accuracy ± 6% of total accumulation at 15 cm/hr

0 c. Analog accuracy ± 9% of total accumulation at 15 cm/hr

d. Resolution 0.25mm

0 1988-Present

0 The meteorological tower was relocated to approximately 1750 ft. northwest of its original location at the
0 microwave tower. Instrumentation signals are processed digitally, transmitted via buried cable to the
0 plant, and then processed back to analog for use by the chart recorders and the plant OAC. Signal
0 processing is necessary because of the distance between the meterological tower and the plant. Relocating the meterological tower became necessary due to the erection of the new Administration Building near the microwave tower. The building's close proximity to the tower would have significantly influenced air flow near the tower. The relative position of the new tower is shown in Figure 2-4 and Figure 2-5 and the instrumentation elevations relative to the plant are given in Figure 2-24.

0 The new 60 meter high meteorological tower began operation on April 23, 1988, with wind speed and
0 direction measured at the 10m and 60m levels and delta temperature measured between these intervals.
0 The dew point temperature system was not reinstalled, since no regulatory requirements for this parameter
0 at Oconee Nuclear Station. Instrument specifications are the same as those given in the 1984-1988 listing,
0 with the exception of discontinued dewpoint measurements. Both upper and lower wind direction sensors
0 for the northwest tower were upgraded from potentiometric sensors to resolver sensors. This improved
0 performance and reliability. The wind direction sensor for the supplemental tower at Keowee River was
0 upgraded June 22, 1990.

Because of the change in distance between temperature sensors (50m) for measuring ΔT, the stability classifications are defined by new delta temperature ranges as given below:

| Stability Class | Delta Temperature Range (°C) |
|---|---|
| A | < - 0.95 |
| B | -0.95 to -0.86 |
| C | -0.85 to -0.76 |
| D | -0.75 to -0.26 |
| E | -0.25 to +0.74 |
| F | +0.75 to +2.0 |
| G | > +2.0 |

Near real-time digital outputs of meteorological measurements are summarized for end-to-end 15 minute periods for use in a near real-time puff-advection model to calculate offsite dose during potential radiological emergencies. The Operator Aid Computer (OAC) system computes the 15 minute quantities from a sampling interval of 60 seconds. It calculates 15 minute average values for high and low level wind direction and speed; 15 minute averages are also calculated for delta temperature and ambient temperature. Total water equivalence is computed for precipitation. All 15 minute values are stored with a 24 hour recall. Permanent archiving of data from the digital system is made by combining the 15 minute quantities into one hour values.

Weekly equipment calibration and maintenance checks are performed in the field for all parameters, as specified by station procedure IP-O-B-1601-03 (Duke Power Company Oconee Nuclear Station Meteorological Equipment Checks). Semiannual calibration checks are performed as per associated station procedures, listed below.

| Parameter | Procedure |
|---|---|
| Precipitation Channel | IP-O-B-1601-008 |
| Air Temperature and Delta-Temperature Channel | IP-O-B-1601-014 |
| Wind Speed Channel | IP-O-B-1601-011 |
| Wind Direction Channel | IP-O-B-1601-012 |

## 2.3.4 SHORT-TERM DIFFUSION ESTIMATES

### 2.3.4.1 Objectives

Conservative and realistic estimates of atmospheric dilution factors at the site boundary or exclusion area boundary and at the outer boundary of the low population zone are provided in this subsection for various time periods to 30 days. Various periods of onsite and offsite data are used in the different studies conducted and are noted in the text where appropriate.

### 2.3.4.2 Calculations

Reference 26 on page 2-35 indicates that the appropriate equation to use for calculating the two-hour site boundary relative concentration is:

$$\frac{X}{Q} = \frac{1}{\bar{u}(\pi\sigma_y\sigma_z + cA)}$$

In this equation $\sigma_y$ and $\sigma_z$ are the standard deviations of the cloud concentration in the horizontal and vertical directions, respectively. These values are normally determined from on-site observations. In lieu thereof, it is permissible to use graphical values as shown in Reference 27 on page 2-35. The $\sigma_y$ $\sigma_z$ values are those which are appropriate for the one mile (1610 meters) exclusion radius of the site.

Normal assumptions to be used with this equation are:

1. Moderate temperature inversion - Pasquill F Conditions prevail.
2. Unidirectional wind for two consecutive hours.
3. Average wind speed ($\bar{u}$) is one meter per second.
4. Building shape factor (c) is between 0.5 and 2.0.
5. Building cross-section (A) is in square meters.

Each of the entry values to the equation is discussed below.

Pasquill F conditions occur frequently at the site. Their overall frequency has been documented at 24 percent in an earlier section of this report. It is estimated that this frequency will diminish to about 12 percent when all lakes in the vicinity of the nuclear plant are full. The frequency of Pasquill F conditions is expected to diminish, while Pasquill E conditions will increase from a current eight percent to about 14 percent of all observations. Thus, there is about a 50-50 chance, once the site is completed, that an inversion condition will be either Pasquill F or E.

The assumption of the unidirectional wind for two hours was examined. Neglecting calms, in a sample of 547 hours of Pasquill F conditions, only 68 cases were found where winds persisted from the same direction for two hours. Thus, it appears that this assumption is conservative.

The average wind speed ($\bar{u}$) observed under Pasquill F conditions (neglecting calms) was found to be 1.9 meters per second for the Greenville area. It is recommended that this wind speed be used for on-site wind speed estimates.

The building shape factor (c) was assumed to be equal to 1.0.

The cross-sectional areas of the buildings are shown in Figure 2-31. The minimum total building cross-section is 5180 square meters, while the front view area is 6792 square meters. The minimum building complex cross-section will be oriented in such a manner as to take advantage of increased flow due to site air drainage patterns, although no credit is taken for this in the analysis.

The values for entry into the equation are:

| | | | | | |
|---|---|---|---|---|---|
| $\bar{u}$ | = | 1.9 mps | | | |
| $\sigma_y$ | = | 60 m | c | = | 1.0 |
| $\sigma_z$ | = | 20 m | A | = | 5180 m$^2$, and |
| $\frac{X}{Q}$ | = | $5.9 \times 10^{-5}$ | | | |

An investigation was conducted to determine the most pessimistic theoretical 24-hour period at the site.

Thirty months of data from Greenville, South Carolina were scanned and those days where the average wind speeds for the entire day were approximately two meters per second or less were studied in detail. Thirty-seven cases were documented. Each hour of each day was classified according to the Pasquill method and a composite was derived which shows the poorest diffusion condition observed for each hour of the day during the 37 cases examined. The composite conditions are shown in Table 2-28.

Examination of Table 2-28 indicates that the poorest composite diffusion day would be to start at 1700 hours and maintain a Pasquill F condition for 16 consecutive hours, then one hour of Pasquill E, and finally seven hours of Pasquill D. This could be referred to as the most pessimistic theoretical 24-hour day for diffusion. (Meteorologically, this type of day would be difficult to achieve since cloud cover would be required to arrive immediately after dawn. Normally, if low cloud cover forms, it indicates that moisture sufficient to raise the probability of fog to very high values must have existed. In which case, fog would have been expected earlier, and some relaxation of the F and E criteria for the early morning hours would be realized).

This condition (as shown in Table 2-28) was not observed. It merely serves to document what might be termed a poorest possible diffusion day. This day is recommended for use in diffusion calculations.

A further examination of the 37 poorest diffusion days was conducted to ascertain whether a high frequency of winds from a single direction was representative of the low wind speed cases. The following summary shows the results for the light wind speed days, (less than or equal to 2.0 meters per second for the entire day).

| Percentage of Winds From a Single 22.5° Sector During a 24-Hour Period | Probability of Single Day Occurrence (Based on 30 Months of Data) |
|---|---|
| 33.33% | .00218 |
| 29.17% | .00436 |
| 25.00% | .0109 |
| 20.83% | .0229 |
| 16.67% | .0360 |
| 12.50% | .0393 |
| 8.33% | .0404 |

It is recommended that a value of 20.83% be used as typical of the most frequent single sector percentage. As a result of review of the data, it was determined that a light wind condition provided a frequency of about 20% of winds from a single direction and that as the speed of the wind rose to higher values, the frequency from a single direction increased.

It was noted that often the fluctuation of the wind during a light wind day would be extremely erratic, such that even though the wind blew from a single direction for several hours of the 24, these were not necessarily consecutive hours.

Four days in December, 1959, were examined which experienced high average wind speeds. The following summary shows the frequency of a single wind direction for strong wind days based on this extremely limited sample.

| Date | Average Speed (m/sec) | Maximum Wind Direction Percentage |
|---|---|---|
| 12/7/59 | 6.85 | 54.17 |
| 12/24/59 | 7.00 | 67.00 |
| 12/27/59 | 6.49 | 41.70 |
| 12/28/59 | 6.39 | 67.00 |

As a result of the review of mean wind speeds for a day versus percentage frequency of wind from a single wind direction, the following estimates were derived for various Pasquill conditions:

| Pasquill Classes | Mean Wind Speed (mps) | % of Wind From a Single Direction |
|---|---|---|
| A | 2 | 16% |
| B | 3 | 25% |
| C | 5 | 58% |
| D | 4 | 50% |
| E | 3 | 35% |
| F | 2 | 21% |

Upon completion of the analysis of a few well-separated months of data, it was learned that conditions expressed in Table 2-10, Table 2-21, and Table 2-22 were representative of wind and stability classes for the nuclear site area for 1 to 30 day or longer term releases.

Table 2-21 justifies the selection of the frequencies and wind speeds for various Pasquill classes as follows:

| | | |
|---|---|---|
| 25% | Pasquill F | $\bar{u}$ = 2.0 mps |
| 10% | Pasquill E | $\bar{u}$ = 3.0 mps |
| 20% | Pasquill D | $\bar{u}$ = 4.0 mps |

In order to introduce further conservatism in the long-term model, it was decided to class all other conditions as Pasquill C and thus the remaining group became:

| | | |
|---|---|---|
| 45% | Pasquill C | $\bar{u}$ = 5.0 mps |

Further, examination of Table 2-10 data bears out that the maximum frequency of a single wind direction for a long-term condition is about 15 percent. Table 2-22 also shows this to be true. (This can be determined by adding the percentage frequencies for all Pasquill classes as shown in Column 7.)

Dispersion factors (X/Q, seconds $m^{-3}$) as shown in Table 2-29 are to be used for accident (10CFR100) and routine operational (10CFR20) analyses. Dispersion factors for elevated releases are based on analysis of on-site meteorological data. The factors given for ground releases were negotiated through discussions with the AEC/DRL staff during the early summer of 1970. These discussions were related to the additional meteorological studies in support of the 0 to 2 hour Valley Drainage Model presented later in this subsection. During the negotiations, Duke agreed to reduce the Reactor building design leakage rate from 0.5 percent by volume in 24 hours to 0.25 percent by volume and increase the atmospheric dispersion factors for ground releases. It was agreed to depart from the dispersion factors for ground releases as submitted previously and supported by the Near Ground Study and the Valley Drainage Model. The accepted ground release dispersion factor at the exclusion area boundary (one mile) for

Oconee Units 1, 2, and 3 is 1.16 x $10^{-4}$ for the 0-2 hour analysis. Long term dispersion factors are also given in Table 2-29.

The meteorological conditions basis for elevated release dispersion factors are presented below.

0-2 hours: Pasquill F with a wind speed of 1.0 meters per second with plume confined to "Sigma Y" dimensions.

2-24 hours: Pasquill F with a wind of 1.0 meters per second with plume confined to one sector.

1-7 days: Pasquill F 50 percent of the time, 1 meter per second wind speed. Pasquill D 50 percent of the time, 4 meters per second wind speed. Wind confined to one sector 30 percent of the time.

7-30 days: Pasquill F 35 percent of the time, 1 meter per second wind speed. Pasquill E 5 percent of the time, 2 meters per second wind speed. Wind confined to one section 12.5 percent of the time.

Table 2-30 shows the appropriate dispersion factors to be used during various release conditions.

For ground releases, actual on-site data results in greater dispersion than was calculated by the valley drainage model. For example, ground-release X/Q for the 30-day model at 6 1/2 miles from site, near boundary of low population zone, is 4.34 x $10^{-7}$ seconds $m^{-3}$ from on-site data, but is 1.17 x $10^{-6}$ seconds $m^{-3}$ from the valley drainage concept. Nevertheless, dispersions of ground level releases are based on the valley drainage concept in the safety analyses for Oconee.

Estimates of atmospheric dispersion of radioactive effluents employ a Gaussian straight-line trajectory model. The data of 1975 used in Section 2.3.3, "Onsite Meteorological Measurements Programs" on page 2-17 is applied as a data base for these estimates. Joint recovery of wind speed, direction, and stability data was 86 percent for the period.

The calculational grid contains 504 receptors. Seventy-two receptors are located at five degree intervals on each of seven radii from the Exclusion Area Boundary to a distance of five miles from the nearest reactor vent.

The model calculates hourly relative concentration (X/Q) values at each receptor for each hour of the period. These values are accumulated, then averaged to obtain the field of annual average X/Q values.

Releases from the 60 meter vent stacks are considered partially elevated and partially ground level releases. The fraction of the plume material which remains elevated depends on the ratio of exit velocity to wind speed at release height. This fraction has been calculated from equations 7 and 8 of Regulatory Guide 1.111.

Plume height for elevated releases is calculated from equation 4 of Regulatory Guide 1.111. Stack downwash is determined from equation 5 of the same reference. Plume rise is computed from the exit velocity (20 m $sec^{-1}$), stack diameter (1.8 m) and annual mean wind speed at vent height (3 m $sec^{-1}$) according to Reference 28 on page 2-35. The effect of terrain on effective plume height is included according to Reference 29 on page 2-35. If all heights are referenced to plant grade, $h_e$ is the effective plume height without terrain correction, and $h_t$ is the height of the terrain feature: then the corrected plume height is $h_e - h_t/2$. An exception noted is that plume height is constrained to remain between $h_e$ and $h_e/2$. The $h_t$ values represent the highest terrain in the vicinity of the receptor within the 22.5° sector.

The equation employed for each hourly X/Q calculation for the ground release portion is

$$(X/Q)_g = \frac{F_g}{u_1(\pi\sigma_y\sigma_z + cA)} \exp\left[\frac{-y_1^{\ 2}}{2(\sigma_y^{\ 2} + cA/\pi)}\right]$$

The equation employed for the elevated portion is

$$(X/Q)_e = \frac{F_e}{u_2\pi\sigma_y\sigma_z}\left[\exp\left(\frac{-y_2^{\ 2}}{2\sigma_y^{\ 2}}\right) + \exp\left(\frac{-H^2}{2\sigma_z^{\ 2}}\right)\right]$$

$F_g$ and $F_e$ are the fractions of the plume which are ground level and elevated respectively.

$u_1$and $u_2$ are the low level and high level average wind speeds respectively (m/sec.). A minimum value of .447 m/sec is assumed.

cA is the mixing zone for the aerodynamically entrained effluent. It is one half the cross-sectional area of the adjacent containment structure normal to the wind, that is 1150 $m^2$.

$Y_1$ and $Y_2$ are the lateral distances of the receptor from the wind direction vectors $u_1$ and $u_2$ respectively.

H is the plume height considering all corrections as discussed above (m).

$\sigma_y$ and $\sigma_z$ are the crosswind and vertical plume standard deviations (m) which are functions of atmospheric stability and distance downwind. Stability categories are determined by vertical temperature gradient according to Regulatory Guide 1.23. Standard deviation values are consistent with Reference 19 on page 2-35.

The factor $(\pi\sigma_y\sigma_z + cA)$ is a measure of plume spread. This factor is restricted to be no greater than ( $3\pi\sigma_y\sigma_z$) as recommended in Regulatory Guide 1.111.

The $(X/Q)_g$ values are modified to account for plume depletion by dry deposition. The method employed is as recommended in Regulatory Guide 1.111.

The X/Q value at each receptor for each hour is the sum of the elevated contribution and the ground level contribution. Successive hourly values are calculated to crosswind distances of $\pm$ 20 degrees from observed wind directions. Points in the computational grid beyond $\pm$ 20 degrees for any one hour are assumed at zero relative concentration for that hour.

Regulatory Guide 1.111 suggests the use of a correction factor to adjust the computed X/Q values. The Oconee station is located in a river valley which does induce some channelling and valley drainage wind; therefore, the river valley correction factors of the above reference are applicable. Although the derivation of these factors is not presented in the Guide, we understand that they are a result of a limited comparison of a Gaussian straight-line X/Q projection and variable trajectory model X/Q projecton for a hypothetical valley site where all winds are parallel to the valley axis. Also, recirculation of effluent with a time scale of about 24 hours is the most probable cause of the different X/Q values. We suggest that a significant percentage of winds not along the valley axis at Oconee and the relatively short duration of higher activity effluent releases would result in lower correction factors or no correction at the Oconee station. Since we have no evidence at this time to confirm or quantify the above hypotheses, we have applied the indicated correction factors for river valley sites, and present the resulting X/Q values as conservative estimates.

The diffusion model used for this study differs from the recommendations of Regulatory Guide 1.111. The principal differences from the Guide are as follows:

1. X/Q values are calculated at 5° intervals instead of averaged over 22.5° sectors;
2. X/Q values are accumulated from a chronological record of meteorological data instead of employing the joint frequency distribution developed from the meteorological data; and
3. For the purpose of achieving realistic X/Q estimates, a less conservative terrain correction is employed.

Because the onsite winds are recorded to the nearest 5° direction, the model effectively assumes that the plume centerline impacts some radial line of receptors at each hour. This assumption is slightly more conservative than the sector average approach. The use of a time series of meteorological data would be no different from the use of a well formulated frequency distribution of the same data. Finally, the terrain correction prohibits impaction of the plume centerline onto terrain features, but does simulate the approach of the plume toward hills as they are forced over or around the obstruction.

Values for dry deposition ($m^{-2}$) are calculated according to Regulatory Guide 1.111. These D/Q values account for the terrain correction factors considered above. Also they consider the fractional breakdown of elevated and ground level plume contributions to D/Q in the same manner as the X/Q values above. Wind direction, speed, and stability frequencies for these calculations were obtained from a joint frequency distribution of hourly onsite meteorology for the period of record.

All X/Q and D/Q values at specific receptors were interpolated from isopleth fields generated using the above mentioned receptor grid.

Values of X/Q, adjusted for dry deposition, are shown for selected receptors in Table 2-31. Relative deposition values, depicted in Table 2-32, are computed for the same set of receptors. X/Q values, which do not allow for removal processes, are presented in Table 2-33.

For the 0 to 2 hour accident relative concentration, X/Q, a value of $7.41 \times 10^{-5}$ was submitted based on the valley drainage concept. Additional meteorological studies have been performed subsequent to this submittal which give evidence that the valley drainage model is conservative. These studies show a X/Q value of $6.12 \times 10^{-5}$ as being descriptive of the 0 to 2 hour accident relative concentration; therefore, the relative concentration value of $7.41 \times 10^{-5}$ will not be changed. The following is a description of additional meteorological studies supporting this conclusion.

The site dispersion characteristics were investigated with five instruments (Figure 2-32) indicating and recording wind direction and speed, two of which were elevated. During these studies, vertical temperature gradients were measured at two locations. Fifteen $SF_6$ (Sulfur Hexafluoride) gas-tracer experiments were conducted under poor diffusion conditions, during periods with a temperature inversion.

The 0 to 2 hour accident relative concentration was recalculated using the equation $X/Q = (\bar{u}\pi\sigma_y\sigma_z)^{-1}$. Wind speed was obtained from the microwave tower instrument. Standard deviations of the lateral concentration distribution (Sigma Y) were computed from Pasquill assignments for standard deviations of the horizontal wind azimuth (Sigma Theta). Standard deviation of the horizontal wind was derived from wind range on the microwave tower instrument. Standard deviations of the vertical concentration distribution (Sigma Z) were determined by vertical temperature gradients for the following class intervals.

| Pasquill Categories | Vertical Temp. Gradient Class Intervals | |
|---|---|---|
| F | > | 2.0F in 150 feet |
| E | | 2.0 to 0.1F in 150 ft. |
| D | | 0.0 to - 1.4F in 150 ft. |
| C | - | 1.5 to - 2.9F in 150 ft. |
| B | - | 3.0 to - 4.5F in 150 ft. |
| A | < | -4.5F in 150 ft. |

Pasquill assignments for Sigma Z were again made for categories A, B, and C; however, for D, E, and F gas-tracer test values, were substituted. Test Sigma Y values, although larger than Pasquill values, were not used because analysis for given stabilities and wind speeds showed horizontal dispersion too directionally dependent. It is noteworthy that Sigma Y was computed and used without a building effect term. Gas-tracer test results implied that Pasquill Sigma Z values for D, E, and F were too low. A reasonable representation for standard deviation of the vertical concentration distribution was sought for these class intervals, and based on test results, redefined as follows:

Pasquill F = 40 meters, Pasquill E = 50 meters,
Pasquill D = 50 meters.

A relative concentration calculation was made for each pair of valid consecutive observations from the microwave tower wind and temperature data. Relative concentration was computed as the average of the two one hour concentrations, if in successive hours there was an overlap in plume widths defined as 4.30 Sigma Y. Relative concentration was computed from the highest one hour concentration averaged with ten percent of the lowest one hour concentration, if successive hours showed no overlap as above, but did give an overlap of wind range sectors. Finally, relative concentration was computed from the highest one hour concentration averaged with 0, if successive hours showed no overlap of wind range sectors. A relative concentration frequency distribution was determined for the period June 1, 1968 to May 31, 1969 (Table 2-34). A hand calculation check on the relative concentration program ascertained its validity.

Wind speed for each hour was read as the average speed in the preceding 30 minute period. Wind speeds less than or equal to 0.9 miles per hour were read as 1.0 miles per hour. Wind range read for each hour also covered the preceding 30 minute period. Vertical temperature differentials read for each hour covered a period of 30 minutes before and after the hour. Further, vertical temperature differentials for each hour were read: (a) as highest value if all readings positive, (b) as highest value if both positive and negative readings occurred the same hour, (c) as 0 if both 0 and negative readings occurred the same hour, and (d) as the lowest value if all readings were negative during the same hour.

Data from the five wind instruments were evaluated simultaneously and classified into five flow patterns. Comparisons were made of flow patterns during gas-tracer test (January 15, 1970 to March 11, 1970) with those during temperature inversions from available data of an earlier period (October 13, 1969 to November 23, 1969). The most frequent test flow pattern was also the most frequent configuration during the earlier period. All five patterns occurred in both periods.

Sample calculation at 1 mile (1609 meters):

$$X/Q = (\bar{u}\pi\sigma_y\sigma_z)^{-1}$$

Input Parameters:

$\bar{u}$ = 2.5 meters per second

wind range = 15°

vertical temperature differential = 3°F in 150 feet

$\sigma_\theta$ = 15/6 = 2.5°

$\sigma_y$ = 57 meters

$\sigma_z$ = 40 meters

X/Q = 1/(3.1416)(2.5)(57)(40)

X/Q = $5.58 \times 10^{-5}$ seconds per meter$^3$

The balance used in gas release rates, "Q" was calibrated by certified weights. Calibration of gas chromatograph was performed by the instrument manufacturer prior to testing. The instrument was also checked for calibration drift with a known gas mixture certified by an independent analytical laboratory.

The following is an outline of the $SF_6$ field procedures:

1. Personnel required
   a. One test supervisor
   b. One release man
   c. One vehicle driver
   d. Two instrument operators
2. Test equipment
   a. Vehicle with test mount
   b. Utility vehicle
   c. Balance
   d. Gas bottle filled with $SF_6$
   e. $SF_6$ detector with recorder
   f. Flow meter
   g. Wooden stakes
   h. Weather instruments as noted
   a. Steel tape measure
3. Limitations and precautions

   Prevent sampling vehicle and its occupants from being contaminated with $SF_6$.
4. Prerequisite weather conditions
   a. Temperature inversion should exist as determined from the temperature gradient on the microwave tower.
   b. The test should not be run if fog or precipitation is occurring.
5. Data required
   a. Locations and non-zero readings of background points (Figure 2-33)

b. Locations, readings, and times of sampling points (Figure 2-34 and Figure 2-35)

c. Release rate (Figure 2-35)

d. Weather data (Figure 2-35)

e. Plume size data (Figure 2-35)

6. Test procedure

a. Check the weather instruments on the microwave tower and record the data on Figure 2-35.

b. Set up release equipment about 200 feet east of auxiliary boiler vent between the turbine and the switchyard.

c. Take background readings at points marked on Figure 2-33. Note any indications of $SF_6$ on Figure 2-33.

d. Prepare a smoky fire near the $SF_6$ release point and observe the general plume drift.

e. When the background check has been made start the release of $SF_6$. Record data on Figure 2-34 every 20 minutes with release rate of four (4) to six (6) grams per minute.

f. Begin sampling for $SF_6$ around the site perimeter downwind from the release point. Number each sample point consecutively; however, if two (2) samples are taken at one point, only one (1) number needs to be used.

g. When $SF_6$ gas is found, mark the point with a numbered wooden stake. By choosing appropriate sampling points attempt to define the width of any plume found.

h. Continue sampling around the release point during the test period.

i. Read the instruments at the utility pole weather station, near the Keowee River, at least once during the test (Figure 2-35).

j. Continue to sample and record data until good plume definition can be established.

k. After sampling has ended, record the distances between sample stakes.

The procedures for the study analysis are summarized below:

1. Note each pass through a detection area and approximate time of the pass. Place data points marking positions where $SF_6$ is detected in a sequential space order (not time).
2. From map of area, determine the average distance from the source to the detection stations.
3. Convert the source strength, Q, to micrograms per second from the release rate data.
4. Convert the detector scale readings to micrograms per cubic meter.
5. Estimate the average wind speed from surface instrumentation, and when applicable, microwave tower winds.
6. Utilize computer program to fit a Gaussian curve to the spatially ordered data points.
7. Find the first and second moment arms of the distribution of concentration. From the first moment arm, note the center line position; from the second moment arm, note the variance of the horizontal dispersion of the concentration.
8. Take positive square root of the variance to get a standard deviation in the horizontal, Sigma Y.
9. Obtain center line concentration by $X = A\sigma_y - 1\ (2\pi)^{-1/2}$ where A is the area under the distribution curve.

10. Calculate the standard deviation in the vertical, Sigma Z, by $\sigma_z = Q(\pi \bar{u} \sigma_y X)^{-1}$ which is applicable for a ground release.
11. Determine the stability category by the temperature differential on the microwave tower.
12. Using graphs of Sigma Y and Sigma Z as functions of stability and distance from a source, locate test values.
13. Following the curvature of the Pasquill curves for the stability found in Number 11 above, read Sigma Y and Sigma Z values for one (1) mile from the graph.
14. Compute the center line values X/Q at one (1) mile by $X/Q = [\pi \bar{u} \sigma_y (1 \text{ mi.}) \sigma_z (1 \text{ mi.})]^{-1}$.

Results of the gas-tracer experiment are shown in Table 2-35.

A 1.4 wind speed correction factor for the period June 1968 to September 1969 may be warranted based on a calibration check made October 1, 1969, and comparative wind speed data at Greenville-Spartanburg and Oconee. A relative concentration frequency distribution was determined with a 1.4 wind speed correction factor for the period June 1, 1968 to May 31, 1969, (Table 2-36). No wind speed correction was factored into the 0 to 2 hour accident relative concentration value of $6.12 \times 10^{-5}$.

Table 2-37 displays comparative wind speed data for Greenville-Spartanburg and Oconee from June, 1968 to January, 1970. Comparisons were made at 13:00 EST for wind speeds equal to or greater than eight knots at Greenville-Spartanburg.

Supplemental data is presented and includes an all occurrence annual wind rose, a Pasquill F annual wind rose, a Pasquill E annual wind rose, a Pasquill A, B, C, and D annual wind rose, a relative concentration frequency distribution based on single hour calculations, and $SF_6$ field data. This material is presented in Table 2-38, and Figure 2-36 and Figure 2-37.

To assess the effects of topography on short-term diffusion estimates, terrain profiles were plotted for the 16 principal points of the compass within the 0.5 mile radius. Maximum and minimum elevations were recorded for each of the eight principal lines drawn to gain an estimate of potential drainage wind flow. The results are shown below:

| Orientation | Maximum Height Upstream | Minimum Height Downstream | Difference |
|---|---|---|---|
| From N to S | 870 feet | 740 feet | 130 feet |
| From NNW to SSE | 880 | 710 | 170 |
| From NW to SE | 827 | 690 | 137 |
| From WNW to ESE | 872 | 680 | 192 |
| From W to E | 910 | 670 | 240 |
| From WSW to ENE | 817 | 700 | 117 |
| From SW to NE | 917 | 750 | 167 |
| From SSW to NNE | 862 | 760 | 102 |

All of the eight lines pass through the central site area, i.e., from one-half mile north through the site center to one-half mile south. In general, the results show that the drainage of wind would be toward the east within the site exclusion radius.

Within the 3.0 mile radius - USGS topographic maps permit estimates of the overall drainage possibilities out to a three-mile radius. Figure 2-38 shows the results of a gross assessment of the terrain. The terrain at elevations equal to or less than 800 feet is shaded to more readily portray the potential drainage wind

area. It is important to note that this approximate plot assumes that all proposed lakes are full in the final configuration as proposed for this area. Note that, although drainage to the east and east-south-east is shown for the central site area, the terrain modifies the drainage flow direction to that following the Keowee River.

## 2.3.5 LONG-TERM DIFFUSION ESTIMATES

### 2.3.5.1 Objectives

The adequacy of onsite meteorological data in terms of long-term diffusion estimates is presented in this subsection. The discussion of long-term diffusion factors is presented in Section 2.3.4, "Short-Term Diffusion Estimates" on page 2-22 for continuity purposes.

### 2.3.5.2 Calculations

Examination of the joint frequency of wind direction and speed by atmospheric stability class reveals a preponderance of air flow movement down the Keowee River valley axis at Oconee. This is taken as symptomatic of the occurrence of gravity induced flows during stable atmospheric conditions when winds are observed in this direction. In the absence of a straight walled river valley in the vicinity of Oconee, interactions of gravity flows on a smaller scale with the more general gravity flow down the Keowee River valley are postulated for flows near the surface. An indication of near surface flow during these conditions cannot be ascertained by a simple measurement of wind direction at the surface.

Considering the above, tower data at Oconee has been analyzed and can be shown representative of long-term diffusion conditions at the site. For the X/Q and D/Q models employed, meteorological and effluent exit conditions as given above result in only about 2 percent of total radioactivity released at ground level. Some portion of this 2 percent would occur during synoptic flows, and thus would be adequately represented by tower data. Consequently, annual doses can be represented by X/Q and D/Q estimates with wind direction inputs from tower data.

For other than gravity flow conditions, air flow trajectories can be assumed to be adequately represented by straight line flow on all time and distance scales to a distance of five miles. For the relatively undulating terrain surrounding Oconee, the measurement of wind speed and delta temperature from the meteorological tower is viewed as characteristic of prevailing conditions at the site.

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# 2.4 HYDROLOGIC ENGINEERING

## 2.4.1 HYDROLOGIC DESCRIPTION

### 2.4.1.1 Site and Facilities

The location and description of Oconee presented in Chapter 1, "Introduction and General Description of Plant" on page 1.0 and Chapter 2, "Site Characteristics" on page 2-1 include reference to figures showing the general arrangement, layout and relevant elevations of the station. Yard grade is 796 ft. mean sea level (msl). The mezzanine floor elevation in the Turbine, Auxiliary, and Service Buildings is 796.5 it. (msl). Exterior accesses to these buildings are at elevation 796.5 ft. (msl).

All of the man-made dikes and dams forming the Keowee Reservoir rise to an elevation of 815 ft. msl including the intake channel dike. The crest of the submerged weir in the intake canal is at elevation 770 ft. msl.

Changes to the natural drainage of the original site are shown on Figure 2-4.

### 2.4.1.2 Hydrosphere

The main hydrologic features influencing the plant are the Jocassee and Keowee Reservoirs. Lake Jocassee was created in 1973 with the construction of the Jocassee Dam on the Keowee River. The lake provides pump storage capacity to the reversible turbine-generators of the Jocassee Hydroelectric Station, located approximately 11 miles north of the plant. At full pond, elevation 1100 ft. msl, Lake Jocassee has a surface area of 7565 Ac, a shoreline of approximately 75 mi, a volume of 1,160,298 Ac-ft., and a total drainage area of about 148 sq mi.

Lake Keowee was created in 1971 with the construction of the Keowee Dam on the Keowee River and the Little River Dam on the Little River. Its primary purpose is to provide cooling water for the plant and water to turn the turbines of the Keowee Hydroelectric Station. At full pond, elevation 800 ft. msl, Lake Keowee has a surface area of 18,372 Ac, a shoreline of approximately 300 mi, a volume of 955,586 Ac-ft., and a total drainage area of about 439 sq mi. The Jocassee and Keowee Reservoirs and the hydroelectric stations located at these reservoirs are owned and operated by Duke.

The area presently provides for a few raw water users. The City of Greenville and the Town of Seneca take their raw water supplies from Lake Keowee. The Town of Anderson, the Town of Clemson, the Town of Pendleton, Clemson University, and several industrial plants take their raw water supplies from Hartwell Reservoir.

Greenville's raw water intake is located approximately 2 miles north of the plant on Lake Keowee. Seneca's raw water intake is located approximately 7 miles south of the plant on the Little River Arm of Lake Keowee. Anderson raw water intake is located approximately 40 river miles downstream of the Keowee tailrace and also supplies Pendleton, Clemson and Clemson University.

The existing raw water intakes for Greenville, Seneca, and Anderson are shown and located relative to the site on Figure 2-39.

## 2.4.2 FLOODS

### 2.4.2.1 Flood History

Since Oconee is located near the ridgeline between the Keowee and Little River valleys, or more than 100 ft. above the maximum known flood in either valley, the records of past floods are not directly applicable to siting considerations.

### 2.4.2.2 Flood Design Consideration

In accordance with sound engineering practice, records of past floods as well as meteorological records and statistical procedures have been applied in studies of floods through the Keowee and Jocassee Reservoirs as a basis for spillway and freeboard design.

The spillway capacities for Lake Keowee and Jocassee were selected in accordance with the empirical expression for design discharge:

$$Q = C\sqrt{DA}$$

Where Q = peak discharge in cfs
D A = drainage area in square miles
C = 5000, a runoff constant judged to be characteristic of the drainage area

The following tabulation gives pertinent data on this design flood flow:

| Lake Keowee[1] | Lake Jocassee | |
|---|---|---|
| 439 | 148 | Drainage area at damsite, sq mi |
| 25,200 (Newry Gage D A 455 sq mi) | 21,000 (Jocassee Gage D A 148 sq mi) | Maximum recorded flow at nearby USGS gages, cfs |
| 8-13-40 | 10-4-64 | Data of maximum flow |
| 1939-1961 | 1950-1965 | Period of record |
| 105,000 | 61,000 | Spillway design discharge, cfs |
| 800 | 1,110 | Full pond elevation |
| 815 | 1,125 | Crest of dam elevation |
| 0 | 0 | Surcharge on full pond for design discharge |
| 4 | 2 | Number of spillway gates |
| 38 ft. x 35 ft. | 40 ft. x 32 ft. | Size of spillway gates |
| | | Discharge capacity, cfs |
| 107,200 | 45,700 | Spillway |
| – | 16,500 (2 units of 4) | Dependable flood flow through units |
| 107,200 | 62,200 | Total discharge capacity, cfs |

[1] Little River and Keowee River Arms

The above discharge capacities assume no surcharge above normal full pond level. Statistical analyses have shown design reservoir inflows for both Lake Keowee and Lake Jocassee equal to respective design discharge capacities outlined above to have recurrence intervals less frequent than once in 10,000 years.

The maximum wave height and wave run-up have been calculated for Lake Keowee and Lake Jocassee by the Sverdrup-Munk formulae. The results of these calculations are as follows:

| Wave Height | Wave Run-Up | Maximum Fetch | Lake |
|---|---|---|---|
| 3.70 ft. | 7.85 ft. | 8 miles | Keowee (Keowee River Arm) |
| 3.02 ft. | 6.42 ft. | 4 miles | Jocassee |
| 3.02 ft. | 6.42 ft. | 4 miles | Keowee (Little River Arm) |

The wave height and wave run-up figures are vertical measurements above full pond elevations as tabulated above.

Studies were also made to evaluate effects on reservoirs and spillways of maximum hypothetical precipitation occurring over the entire respective drainage areas. This rainfall was estimated to be 26.6 inches within a 48 hour period. Unit hydrographs were prepared based on a distribution in time of the storms of October 4-6, 1964, for Jocassee and August 13-15, 1940, for Keowee. Results are summarized as follows:

| Keowee | Jocassee | |
|---|---|---|
| 147,800 | 70,500 | Maximum spillway discharge, cfs |
| 808.0 | 1114.6 | Maximum reservoir elevation |
| 7.0 ft. | 10.4 ft. | Freeboard below top of dam |

While spillway capacities at Keowee and Jocassee have been designed to pass the design flood with no surcharge on full pond, the dams and other hydraulic structures have been designed with adequate freeboard and structural safety factors to safely accommodate the effects of maximum hypothetical precipitation. Because of the time-lag characteristics of the runoff hydrograph after a storm, it is not considered credible that the maximum reservoir elevation due to maximum hypothetical precipitation would occur simultaneously with winds causing maximum wave heights and run-ups.

The maximum Keowee tailwater level during hydro operation has been calculated to be elevation 672.0 ft. (msl), which is 124 ft. below the nuclear station yard elevation 796.0 ft. (msl).

The maximum discharge calculated, due to hydro operating, is expected to be 19,800 cfs. The minimum discharge calculated with no units operating, is expected to be 30 cfs.

In summary, the above results of flood studies show that Lakes Keowee and Jocassee are designed with adequate margins to contain and control floods which pose no risk to the nuclear site.

## 2.4.3 PROBABLE MAXIMUM FLOOD ON STREAMS AND RIVERS

### 2.4.3.1 Probable Maximum Precipitation

See Section 2.4.2.2, "Flood Design Consideration" on page 2-38.

0 **2.4.3.2 Deleted per 1990 Update**

### 2.4.3.3 Runoff and Stream Course Models

See Section 2.4.2.2, "Flood Design Consideration" on page 2-38.

### 2.4.3.4 Probable Maximum Flood Flow

See Section 2.4.2.2, "Flood Design Consideration" on page 2-38.

0 **2.4.3.5 Deleted per 1990 Update**

### 2.4.3.6 Coincident Wind Wave Activity

See Section 2.4.2.2, "Flood Design Consideration" on page 2-38.

## 2.4.4 POTENTIAL DAM FAILURES, SEISMICALLY INDUCED

Duke has designed the Keowee Dam, Little River Dam, Jocassee Dam, Intake Canal Dike, and the Intake Canal Submerged Weir based on sound Civil Engineering methods and criteria. These designs have been reviewed by a board of consultants and reviewed and approved by the Federal Power Commission in accordance with the license issued by that agency. The Keowee Dam, Little River Dam, Jocassee Dam, Intake Canal Dike, and the Intake Canal Submerged Weir have also been designed to have an adequate factor of safety under the same conditions of seismic loading as used for design of Oconee.

The construction, maintenance, and inspection of the dams are consistent with their functions as major hydro projects. The safety of such structures is the major objective of Duke's designers and builders, with or without the presence of the nuclear station.

0

0 **2.4.5 DELETED PER 1990 UPDATE**

0 **2.4.6 DELETED PER 1990 UPDATE**

0 **2.4.7 DELETED PER 1990 UPDATE**

0 **2.4.8 DELETED PER 1990 UPDATE**

0 **2.4.9 DELETED PER 1990 UPDATE**

## 2.4.10 FLOODING PROTECTION REQUIREMENTS

See Section "Water Level (Flood) Design" on page 3.4.

## 2.4.11 LOW WATER CONSIDERATIONS

0 **2.4.11.1 Deleted per 1990 Update**

0 **2.4.11.2 Deleted per 1990 Update**

0 **2.4.11.3 Deleted per 1990 Update**

0 **2.4.11.4 Deleted per 1990 Update**

0 **2.4.11.5 Deleted per 1990 Update**

### 2.4.11.6 Heat Sink Dependability Requirements

Oconee has four sources of water for shutdown and cooldown. These sources are: (1) water from Lake Keowee via the intake canal using the circulating water pumps; (2) gravity flow through the circulating water system; (3) water trapped between the submerged weir in the intake canal and the intake structure in the event of a loss of Lake Keowee and; (4) 8,825,000 gallons of water trapped in the plants Circulating Water System with appropriate valving, pumping and recirculation as a backup in the event of the loss of all external water supplies.

## 0 2.4.12 DELETED PER 1990 UPDATE

## 2.4.13 GROUNDWATER

### 2.4.13.1 Description and Onsite Use

#### 2.4.13.1.1 Regional Groundwater Conditions

The Oconee site lies within the drainage area of the Little and Keowee Rivers which flow southerly into the Seneca River and subsequently discharge into the main drainage course of the Savannah River. The average annual rainfall at the site area is approximately 53 inches.

The deposits of the Little and Keowee drainage basin are generally of low permeability which result in nearly total runoff to the two rivers and their numerous tributary creeks. Runoff occurs soon after precipitation, particularly during the spring and summer months when the soil percolation rates are exceeded by the short term but higher yielding rainfall periods. The area is characterized by youthful narrow streams and creeks which discharge into the mature Little and Keowee Rivers.

Throughout the area, groundwater occurs at shallow depths within the saprolite (residual soil which is a weathering product of the underlying parent rock) soil mantle overlying the metamorphic and igneous rock complex (Reference 1 on page 2-46). Refer to Section 2.5, "Geology, Seismology, and Geotechnical Engineering" on page 2-47. This saprolite soil, which ranges in thickness from a few feet to over 100 feet, is the aquifer for most of the groundwater supply. Wells are shallow and few exceed a total depth of 100 feet. Depths to water commonly range from 5 to 40 feet below the land surface. Seasonal fluctuation is wholly dependent of the rainfall and the magnitude of change may vary considerably from well to well due to the limited areas of available recharge. Average fluctuation is about 3 to 5 feet. Both surface water and groundwater in this area are of low mineral content and generally of good quality for all uses.

To determine the general groundwater environment surrounding the proposed site, groundwater levels were established in numerous domestic wells and exploratory drill holes within a four-mile radius. Additional data was obtained from interviews with local residents regarding specific wells and discussions with State and Federal personnel. The results of the groundwater level survey are shown on Figure 2-40. The results demonstrate that local subsurface drainage generally travels down the topographic slopes within the more permeable saprolite soil zones toward the nearby surface creek or stream. Gross drainage is southward to the Little and Keowee Rivers which act as a base for the gradient.

Because the topography and thickness of the residual soil, overlying bedrock control the hydraulic gradient throughout the area, and further, the relief is highly variable within short distances, it is not possible to assign a meaningful average gradient for the 15 square mile area surveyed. In all small areas studied within the four-mile radius, the groundwater hydraulic gradient is steep and conforms to the topographic slope. Water released on the surface will percolate downward and move toward the main drainage channels at an estimated rate of 150 to 250 feet per year.

The gradient throughout the area represents the upper surface of unconfined groundwater and therefore is subject to atmospheric conditions. Confined groundwater occurs only locally as evidenced by the existence of isolated springs and a few exploratory drill holes which encountered artesian conditions. These examples do not reflect general conditions covering large areas but merely represent isolated local strata within the saprolite soil which contain water under a semi-perched condition and/or permeable strata overlain by impermeable clay lenses which have been breached by erosion at its exit and recharged short distances upslope by vertical percolation.

The site area is on a moderately sloping, northwest trending topographic ridge which forms a drainage divide between the Little and Keowee Rivers located approximately 0.5 mile to the west and east, respectively. Groundwater levels at the site, measured during the 1966 drilling program and subsequently in four piezometer holes drilled for pre-construction monitoring purposes, ranged from elevation 792 ft. (msl) to 696 ft. (msl). The slope of this apparently free water surface is predominantly southeasterly toward the Keowee River and its tributary drainage channels. An average hydraulic gradient to the southeast of approximately 8.0 percent was plotted along a line of measured wells. This closely conforms to the existing topography as expected. Refer to Figure 2-41 for measured water levels and typical water table profile.

Field permeability tests conducted during the 1966 exploratory program within the saprolite soil yielded values ranging from 100 to 250 feet per year. Refer to Section 2.4.13.2.2, "Program of Investigation" on page 2-44. The permeability tests were performed in holes of varying depths to determine if the zoned typed weathering of the saprolite soil affects vertical permeability. Based on the test results, inspection of nearby road cuts, and a study of the exploratory drill logs, it is tentatively concluded that the surficial saprolite possesses lower permeability values than that found in the deeper strata. This correlates with the general profile of the saprolite in that the later stages of weathering produce a soil having a higher clay content than the more coarse-grained silty sand sediments below. This natural process of weathering results in the formation of a partial barrier to downward movement of surface water.

#### 2.4.13.1.2 Groundwater Quality

The surface water and groundwater of the area is generally of good quality (Reference 2 on page 2-46). Of the wells surveyed, none were noted where water treatment is being conducted. Temperature of well water measured ranged from a low of 46 to a high of 59 degrees. The majority of readings were from 50 to 53 degrees Fahrenheit.

Water contains different kinds and amounts of mineral constituents. Temperature, pressure and length of time water is in contact with various rock types and soils determine the type and amount of mineral constituents present. Because ground waters are in intimate contact with the host rocks for longer periods of time, they have a more uniform and concentrated mineral content than surface waters. The mineral content of natural surface waters in the Piedmont Province is low due to the relative insolubility of the granitic, gneissic, and schistose host rocks and the reduced contact time caused by rapid runoff in the mountainous areas.

Tabulated below are the surface water constituents reported in parts per million from the Keowee River near Jocassee, South Carolina. The water sample was taken and analyzed by the U.S. Geological Survey, Water Resources Division in June 1965.

| | | | |
|---|---|---|---|
| Silica ($SiO_2$) | 7.8 | Carbonate ($CO_3$) | 0.0 |
| Iron (Fe) | 0.01 | Bicarbonate ($HCO_3$) | 7.0 |
| Calcium (Ca) | 1.0 | Sulfate ($SO_4$) | 1.0 |
| Magnesium (Mg) | 0.1 | Chloride (Cl) | 0.6 |
| Sodium (Na) | 1.2 | Fluoride (F) | 0.1 |
| Potassium (K) | 0.4 | Nitrate ($NO_3$) | 0.1 |
| Dissolved Solids | 15.0 | Phosphate ($PO_4$) | 0.0 |
| Hardness as $CaCO_3$ | 3.0 | | |
| pH | 6.6 | | |
| Specific Conductance | 13.0 | | |

At present, no water quality data is available of groundwater within the area surrounding the Oconee site. Selected representative groundwater and surface water samples will be analyzed in the near future with continued periodic analysis during the environmental surveillance program. Tests will include complete chemical analysis; and gross Beta and Gamma content to establish radioactive background count. It is expected that groundwater analyses will indicate slightly higher relative ion concentrations than that of nearby surface waters but still in the range to meet requirements for classification as good quality water.

Soil surveys conducted by the U.S. Department of Agriculture in cooperation with the South Carolina Agricultural Experiment Station assign pH values of between 5.0 and 6.0 for the Hayesville and Cecil soil series which are present at the site area (Reference 3 on page 2-46). Surface water samples taken from the Keowee River within one mile of the site have a pH of 6.5 to 7.0. It is expected groundwater at the site has a pH ranging between 5.5 and 6.0.

The cation exchange potential can be evaluated by knowing the SAR (Sodium Absorption Ratio), saturation extract values, and the pH of the soil. Two samples of saprolite soil were obtained from drill holes used in determining field permeability values and tested for Sodium Absorption Ratio (SAR). The results are tabulated as follows:

| Sample No. | pH | Saturation Extract Values Milligram-equivalent per 100 grains of soil | | | | SAR |
|---|---|---|---|---|---|---|
| | | Cond. (mhos) | Calcium | Magnesium | Sodium | |
| 1 | 5.8 | 5 | 0.015 | 0.000 | 0.0108 | 0.122 |
| 2 | 5.7 | 7 | 0.010 | 0.000 | 0.0166 | 0.235 |

Considering the amount of soil that is available is so great, it is evident that many times the amount of strontium and/or cesium contained in the waste could be absorbed. Further, the distribution coefficient for ion exchange of radionuclides with the sediments is dependent on the pH of the water in the formation (Reference 4 on page 2-46). The distribution coefficient is a ratio of the reaction of these radionuclides that are absorbed on the soil and the fraction remaining in solution. It is expected that the soils surrounding Oconee have a ratio in the range of 80 to 150, and consequently a substantially lower average velocity for any radionuclide to that of natural water will result.

The estimated maximum rate of movement of water through the soils is about 0.75 feet per day. Using this rate in relation with the above distribution coefficient, bulk density and porosity of the soil, and ratio of the weight of soil to volume of groundwater it indicates the radionuclide velocity will be about .0015 that of groundwater. Using a safety factor of five for variance in flow and competition for exchangeable sodium ions, it would require more than 1000 years for strontium or cesium ions to migrate a distance of one-half mile. In summary, the movement would be so extremely slow that the saprolite soil is an effective natural barrier to the migration of radionuclides.

## 2.4.13.2 Sources

### 2.4.13.2.1 Groundwater Users

The completed field survey of approximately 30 wells determined that groundwater usage is almost entirely from the permeable zones within the saprolite with only minor amounts obtained from the underlying fractured bedrock. Yields from these shallow wells are low, generally less than 5 gpm, and are used to supply domestic water for homes and irrigation of lawns, gardens, and limited amounts for livestock. With only a few exceptions, the wells are hand dug, equipped with bucket lift and/or jet pump, 2 and 40 to 60 feet deep. At present, there is no industrial demand for groundwater within the area.

### 2.4.13.2.2 Program of Investigation

Permeability tests were performed in borings to determine permeabilities of the soil underlying the site. The tests were run according to the Bureau of Reclamations Field Permeability Tests, Designation E-19. Figure 2-42 shows the arrangement of the field test equipment along with a brief description of the procedure used in determining the soil permeability test results. Test results are from 5 borings as presented in Table 2-93. The formulae used in the calculations of the k valves are shown in Figure 2-43.

### 2.4.13.2.3 Groundwater Conditions Due to Keowee Reservoir

As previously discussed, the groundwater levels at the site range from elevation 792 ft. (msl) to below elevation 696 ft. (msl). The Keowee Reservoir will operate with a maximum pool elevation of 800 ft. (msl). This will result in raising the surface water elevation to that datum on the northern and western portions of land adjoining Oconee. It will also raise the existing groundwater table for those local areas bordering the reservoir where presently the ground water surface is below elevation 800.0 ft (msl). The 2 reservoir will materially contribute in establishing a potentially larger recharge area and where it affects the groundwater will result in a more stable hydraulic gradient with less seasonal fluctuation than presently exists.

Preliminary studies indicate that Keowee Reservoir will create the following groundwater conditions at Oconee.

1. Groundwater should continue to migrate downslope through the saprolite soil on a slightly steeper gradient in a southeasterly direction toward the Keowee River base datum.
2. There are two topographic divides which will separate the nuclear station from the nearby reservoir: (1) a one-half mile wide north-south stretch of terrain west of the site, and (2) a narrow 500 foot wide ridge north of the site. Recent groundwater measurements in drill hole K-12, located atop the northern ridge, show water table conditions exist at about elevation 810 ft. (msl).
3. It is unknown if the saprolite soil existing beneath those topographic ridges provide a hydraulic connection between the nuclear plant and the reservoir. However, it is probable that there will be avenues of slow seepage whereby percolating water may locally raise the groundwater surface at the plant to an elevation approaching elevation 800 ft. (msl). A drainage system will be provided to control all seepage encountered.
4. There should be no reversal of groundwater movement at the site, and all water will percolate downward and away from the plant area.
5. The construction of Keowee Dam and Reservoir will not create adverse groundwater conditions at the plant site.
6. Infiltration of domestic wells, located beyond the proposed one-mile exclusion radius, by surface water from the site should not be possible under the existing or future groundwater conditions imposed by Keowee Reservoir.

0 **2.4.13.3 Deleted per 1990 Update**

0 **2.4.13.4 Deleted per 1990 Update**

**2.4.13.5 Design Bases for Subsurface Hydrostatic Loading**

See Section 2.4.13.2.3, "Groundwater Conditions Due to Keowee Reservoir" on page 2-44.

## 2.4.14 REFERENCES

1. *Geologic Notes*, Division of Geology, State Devel opment Board, Vol. 7, No. 5, September-October 1963.
2. *Chemical Character of Surface Waters of South Carolina* , South Carolina State Development Board, (Bulletin No. 16C) 1962.
3. *Soil Survey - Oconee County, South Carolina*, United States Department of Agriculture, Series 1958, No. 25, February 1963.
4. *Storage of Radioactive Wastes in Basement Rock Beneath the Savannah River Plant*, DP-844 Waste Disposal and Processing (TID-4500, 28th Ed.), March 1964.

# 2.5 GEOLOGY, SEISMOLOGY, AND GEOTECHNICAL ENGINEERING

## 2.5.1 BASIC GEOLOGIC AND SEISMIC INFORMATION

Geologic and seismic investigative studies for Oconee Nuclear Station include the following:

1. a review of the available geological and seismological literature pertaining to the region;
2. a geological reconnaissance of the site, performed primarily for the purpose of evaluating the possibility of active faulting in the area;
3. geophysical explorations and laboratory tests to provide parameters for evaluating the response of foundation materials to earthquake ground motion;
4. an evaluation of the seismic history to aid in the selection of the design earthquake that the station might experience; and
5. The development and recommendation of aseismic design parameters for the proposed structures.

The geologic field work at the site started concurrently with the drilling. The site reconnaissance is a continuation of the geologic field work done for the Keowee Dam. Local outcrops, though scarce, are examined and the rock types, joint and foliation orientation noted.

The 21 borings completed at the Oconee Nuclear Site, supplemented by information from the nearby Keowee Hydro Site borings, have been sufficient for a determination of the geologic structure and petrography.

The structures are founded on normal Piedmont granite gneisses. The construction characteristics of the residual soils overlying the rock are known and present no problems in design or construction. The rock underlying the site, below surface weathering, is hard and structurally sound and contains no defects which would influence the design of heavy structures.

The southeastern Piedmont rocks are highly stable seismologically, and the Oconee Nuclear Site should be one of the nation's most inactive areas with respect to earthquake activity.

### 2.5.1.1 Regional Geology

The regional structure is typical of the southern Piedmont and Blue Ridge. The region was subjected to compression in the northwest-southeast direction which produced a complex assortment of more or less parallel folds whose axes lie in a northeast-southwest direction. The Blue Ridge uplift was the climax of the folding, and it was accompanied by major faulting, along a line stretching northeast through Atlanta and Gainesville, Georgia and across South Carolina, 11 miles northwest of the site. This has been termed the Brevard Fault.

The age of these uplifts has not been agreed on by geologists. The consensus of geologic opinion seems to require a period of severe deformation followed by at least one additional period of less severity. Probably all occurred during the Paleozoic Era, but it has been suggested that the last major uplift was as late as the Triassic (180 million years ago) when the Coastal Plain to the east was downwarped. A number of investigators have maintained that the major deformative movements occurred at least 225 million years ago. However, all the resulting stresses have not yet been fully dissipated.

There is no evidence of any displacement along these faults during either historic times or during the Geologic Recent Era as indicated in displacements in the residual soils that blanket the region. While the well known Brevard Fault passes 11 miles northwest of the site, there is no indication of a major fault in the immediate vicinity of the site. Furthermore, the major faults of the region are ancient and dormant, except for minor adjustments at considerable depth. Therefore, there is no indication of any structural hazard to foundations.

The site is underlain by crystalline rocks which are a part of the southeastern Piedmont physiographic province. This northeastward - trending belt of ancient metamorphic rocks extends northward from Alabama east of the Appalachians, and in South Carolina crosses the State from the Fall Line on the east to the Blue Ridge and Appalachian Mountains on the west. These rocks are generally recognized as being divided into four northeast-southwest trending belts in the Carolinas. From southeast to northwest they are the Carolina slate belt, Charlotte belt, Kings Mountain belt, and Inner Piedmont belt. The Oconee Nuclear Site is in the western, or Inner Piedmont Belt.

The Piedmont metamorphic rocks of the site were formed under many different combinations of pressure and temperature, and represent a complex succession of geologic events. The formerly accepted concept that the Piedmont consists only of the deep, worn-down roots of ancient mountains now seems untenable. The older theory that the rocks were exclusively of igneous origin is being replaced by the proposition that they represent highly metamorphosed sediments which have been folded, faulted, and injected to result in one of the most complex geologic environments in the world. It can be said with certainty, however, that these rocks represent some of the oldest on the continent. The new techniques of dating by radioactive decay have placed the age of the metamorphic episodes that produced these rocks as occurring from 1,100 my (million years) to 260 my ago. The successive northeastward trending bands of rocks vary greatly in lithology from granitic types to highly basic classifications, with gneisses and schists being the predominant classifications petrographically. In summary, the regional geology of the Oconee Nuclear Site can be accepted as typical of the southeastern Piedmont - narrow belts of metamorphic rocks trending northeast, with the foliation dipping generally to the southeast. The regional geologic map is shown in Figure 2-44.

## 2.5.1.2 Site Geology

### 2.5.1.2.1 Geologic History, Physiography, and Lithography

The rock present at this site is metamorphic. It is believed to be Precambrian in age; thus, it was formed over 600 million years ago. The complete history of this region is quite complex and has not been fully unravelled. However, it is the consensus of the geologic opinion that the formation consisted of thick strata of sedimentary rocks which were later downwarped and altered by heat and pressure. This first rock formed is termed the country rock.

More than one episode of regional metamorphism transformed the rock into metasediments with accompanying injection and mobilization by plastic flow.

Since the formation of the country rock, most of the mass has been altered or replaced by injection of granite gneiss, biotite hornblende gneiss, and one or possibly more pegmatite dikes.

It is not definite which is the younger: the granite gneiss injection or the biotite hornblende gneiss injection. The limited evidence points to the granite gneiss as the younger of the two.

The pegmatite dikes are the youngest rock known at this site. One such dike is exposed in the road cut on the east side of the state highway passing through the site. It clearly shows the pegmatite cutting through the older rocks, and thus, demonstrates that it is the youngest.

Regional metamorphism, folding, and some minor faulting occurred concurrently much of this early time.

This site is located within the Inner Piedmont Belt, at this locality the westernmost component of the Piedmont Physiographic Province. The topography of the area is undulating to rolling; the surface elevations ranging from about 700 feet to 900 feet. The region is moderately well dissected with rounded hilltops, representing a mature regional development. The area is well drained by several intermittent streams flowing away from the center of the site in a radial pattern. The general station area is shown on the maps in Figure 2-45, Figure 2-46, Figure 2-2, and Figure 2-4.

The local geology of the Oconee Nuclear Site is typical of the southeastern Inner Piedmont Belt. The foundation rock is biotite and hornblende gneiss, striking generally northeast, with the foliation dipping southeast. The rock is overlain by residual soils, which vary from silty clays at the surface, where the rock decomposition has completed its cycle, to partially weathered rock, and finally to sound rock.

The strike of the foliation planes or bands of mineral segregation is north 6 degrees to 15 degrees east with an average dip of 22 degrees to 28 degrees to the southeast. However, due to the local folding or warping at this site, minor variations in the strike and dip of the foliation will occur within the site.

It is almost inevitable that when minor compression folding of this nature occurs, some minor shear displacements will result. We noted only one such displacement. In boring NA-20, at depth of about 79.6 feet below the ground surface, a shear displacement of about one-half inch was recorded. This should not be considered uncommon where hard rock or possibly slightly plastic rock has been folded. While the rock is being folded, minute cracks in the rock develop. The acting compressive forces then cause slight shifts or displacements in the rock resulting in a more relaxed state. The shear displacement noted in boring NA-20, was completely healed or recemented. There is no evidence noted of any recent displacements.

There have been periods of erosion and perhaps even continuous erosion since the close of the Paleozoic Era. The rock now encountered at this site represents the deeper portions of the original metamorphic complex.

The rock encountered at this site is of three main types; light to medium gray granite gneiss, light gray to black biotite hornblende gneiss and white quartz pegmatite with local concentrations of mica, both muscovite and biotite varieties.

The dominate rock type at this site is the light to medium gray granite gneiss. This rock type is generally moderately hard and hard below the initial soft layers encountered in the rock surface. Joints in this rock are brown iron stained in the upper softer layers, but in the deeper harder rock, the joints are not stained. This helps illustrate that the jointing at this site does not control the weathering or decomposition of the rock.

The second most abundant rock type is the biotite hornblende gneiss. The rock is generally weathered or softer to a greater depth than the granite gneiss. This is probably due to the higher percentage of biotite mica. Biotite mica is a potassium magnesium-iron aluminum silicate. The iron content of the biotite mica causes the rate of decomposition to accelerate. However, generally at the deeper portions of the borings, the biotite hornblende gneiss hardness increases to moderately hard or harder. Only a few thin soft layers were noted in this rock in the deeper portion of the borings.

A few layers of hard quartz pegmatite with local concentrations of mica were recorded. The thickness of the pegmatite layers are generally less than three feet. These pegmatite layers are dikes. A dike is a sheetlike body of igneous rock that fills a fissure in the older rock which it encountered while in a molten condition. There is an exposure of mica-quartz pegmatite dike on the east side of the state road cut

passing through this project. This dike exposure is about 3.5 feet wide, but due to the lack of knowledge of orientation of the dike, the exact width cannot be computed. The quartz pegmatite encountered in the borings probably represent other smaller dikes of the same material. These dikes are of hard, sound and durable material and should cause no concern to construction or foundation requirements.

#### 2.5.1.2.2 Rock Weathering

Where heavily banded with dark biotite and hornblende the rock is weaker than in its lighter colored portions, since the highly foliated biotite will split along the foliations, and is also more subject to weathering and consequent rock decay. The borings indicate that even after apparently sound rock has been reached local bands or zones of biotite - usually less than a foot thick - may be soft and weathered to considerable depths.

Rock weathering at the Oconee Nuclear Site is about normal for Piedmont biotite gneisses. While highly variable, the normal range of depth before sound rock is reached is 30 to 50 feet. Although the weathering is deep, the resulting residual materials - clays, silts, and weathered rock - are structurally strong, and are used for the foundations of moderately loaded structures.

#### 2.5.1.2.3 Jointing

The rock at this site is moderately jointed. All of the visible rock outcrops were studied in attempting to determine the correct orientation of the joint patterns. Some moderately good rock outcrops were found and several joint pattern orientations measured. While studying and logging the rock cores, all of the joint dips were recorded. The dips of the joint patterns recorded in the rock cores were associated with the dips measured in the rock outcrops.

The rock has apparently not been subjected to stresses causing high concentrations of joints. The core borings indicate that jointing is widely spaced, and has not influenced the weathering pattern. Joints are about equally divided between strike and dip joints, with occasional oblique joints.

Four joint patterns were found, two of which appear to be most significant. The two most significant joint patterns are: strike north 55 degrees east with a dip of 61 degrees northwest, and strike north 28 degrees west with a dip of 85 degrees southwest. The other two joint patterns are: strike north 9 degrees west with a dip of 67 degrees southwest and strike northsouth with a dip of 74 degrees west. The strike and dip of the joints are shown on Figure 2-47.

#### 2.5.1.2.4 Ground Water

Subsurface water is typical of Piedmont area. The top of the zone of saturation, or water table, follows the topography, but is deeper in the uplands and more shallow in valley bottoms. It migrates through the pores of the weathered rock, where the feldspars have disintegrated and left intersticial spaces between the quartz grains. Additional water is contained in the deeper fractures and joints below the sound rock line. The water table is not stationary, but fluctuates continually as a reflection seasonal precipitation. Additional information on ground water is included in Section 2.4.13, "Groundwater" on page 2-41.

## 2.5.2 VIBRATORY GROUND MOTION

### 2.5.2.1 Seismicity

Two different methods of evaluating earthquakes are in general used. These are the Modified Mercalli (MM) Intensity (damage) Scale and the Richter Magnitude Scale. The magnitude of, and the intensities resulting from, an earthquake are only indirectly related. The Richter Magnitude is an approximate

measure of the total amount of energy released by an earthquake. The Modified Mercalli Intensity, however, is an estimate of the amount of damage caused at a particular site by an earthquake. The intensity of an earthquake at a particular site is only a general indicator of the amount of ground motion since it is a damage criteria and, therefore, dependent on structural considerations as well as ground motion amplitude. The actual amplitude of ground motion at a particular site is dependent upon the following factors:

1. the total amount of energy released by earthquake;
2. the distance of the site from the focus of the earthquake; and
3. the thickness and dynamic properties of the materials above the basement rock complex.

A considerable number of earthquakes have been felt in the region. However, most of these shocks resulted in a little or no damage. A plot of the more significant shocks, those having a recorded intensity of Modified Mercalli V or larger, is shown on Figure 2-48, Earthquake Epicenters.

Accurate locations for earthquake epicenters have only been available since the installation of modern seismographs in the region. Previous to these installations, epicentral locations, based upon known damage and reports of people who felt the earthquake, could be in considerable error. Even with instrumental locations, epicenters could be in error by 20 miles or so. It is estimated that major shocks in the region would probably have been recorded for at least 200 years. However, smaller earthquakes before about 1850 were probably either unrecorded or were unreliably located.

Several large earthquakes outside the area shown on Figure 2-48 have been felt in the region. North of the region, the closest major shocks had epicenters in the St. Lawrence Rift valley or on the folded and faulted coast of Massachusetts. The catastrophic earthquakes of 1811 and 1812 near New Madrid, Missouri, approximately 480 miles from the site, are the closest known large earthquakes to the west. These shocks were probably related to the Ozark Dome. With the exception of the earthquakes at Charleston, South Carolina, no major shocks have occurred south or east of the site within the continental United States. These distant large earthquakes are unrelated to any of the known faulting within the crystalline-metamorphic or overthrust zones in which the site is located.

The largest earthquakes close to the site occurred near Charleston in August, 1886, some 200 miles from the site. Two shocks occurring closely in time, had an intensity estimated to be about Modified Mercalli IX at the epicenter and were perceptible over an area of greater than two million square miles. However, damage was confined to a relatively small area. Aftershocks of the main earthquake had intensities ranging up to Modified Mercalli VII. These shocks may be associated with a downfaulted Triassic basin under the coastal plain.

There have been two moderate earthquakes in the immediate vicinity of the plant since construction began.

In 1971, an earthquake occurred near Seneca, South Carolina. The descriptions of this event which occurred at 07:42 (EST) on July 13, 1971 have been examined from various sources. A MM intensity VI was assigned to the event by USGS based primarily on the report of a cracked chimney near Newry, about 10 km south of the present epicentral area. A detailed examination of the buildings and chimneys by Sowers and Fogle (1978) convinced them that the chimney in question had been broken and in a state of disrepair before the shock. They assigned an intensity IV (MM) to the shaking at Newry.

The July 13, 1971 event at 07:42 AM EDT was preceded by a felt shock at about 4:15 AM EDT and followed by at least one felt aftershock at 7:45 AM (Sowers and Fogle, 1978).

On August 25, 1979 (9:31 PM EDST, Aug. 26) a magnitude 3.7 earthquake occurred in the vicinity of Lake Jocassee, South Carolina. This MM intensity VI event was felt in an area of about 15,000 sq. km and was recorded locally on the three station Lake Jocassee seismographic network, and regionally on seismic stations in South Carolina, North Carolina, Georgia, Tennessee, and Virginia. During the period (August 26, 1979 - September 15, 1979) 26 aftershocks were recorded and they ranged in magnitude from -.60 to 2.0.

A list of earthquakes in the region is provided in Table 2-94.

### 2.5.2.2 Geologic Structures and Tectonic Activity

The region (defined as North Carolina and South Carolina, and parts of Georgia, Alabama, Tennessee, and Virginia) is comprised of three large northeast-southwest trending tectonic zones: The coastal plain, the crystalline-metamorphic zone and the overthrust zone. These zones are shown on Figure 2-49 Regional Tectonics.

The site is located nearly in the center of the crystalline-metamorphic zone, which consists of six generally recognized metamorphic belts. From southeast to northwest these are: The Carolina slate belt, Charlotte belt, Kings Mountain belt, Inner Piedmont belt, Brevard belt, and Blue Ridge belt. The site location is within the Inner Piedmont belt. The rocks in the belts consist of metamorphosed sediments and volcanics that have been folded, faulted, and intruded with igneous rocks. These belts are delineated by differing degrees of metamorphism. Generally, the degree of metamorphism becomes progressively less from the northwest to the southeast.

The oldest metamorphic rocks are located in the Blue Ridge belt. The more easterly belts of younger rocks have undergone progressively less metamorphism.

To the north and west are found a series of fault systems. Since these faults are both numerous and extensive, they can be grouped together and referred to as the overthrust zone, as shown on Figure 2-49. These faults no doubt resulted from the formation of the Appalachians.

The great system of thrust faults in the overthrust zone and most of the known faulting within the crystalline-metamorphic zone apparently occurred during the last period of metamorphism (260 million years ago).

During the Triassic Period (180 to 225 million years ago), sediments were deposited over parts of the exposed metamorphic belts. These deposits and the older metamorphics were intruded by a system of northwest-trending diabase dikes and were faulted by northeast-trending normal faults in the late Triassic Time (200 million years ago). Some of the older faults within the crystalline-metamorphic zone may have been active at this time.

From the late Triassic time until the present, the coastal plain has accumulated a sedimentary cover over its crystalline-metamorphic bedrock. These sediments overlap the bedrock and thicken toward the southeast, effectively masking any ancient faulting in the basement.

It is considered possible that igneous activity has occurred in the region after the Triassic because volcanic bentonitic clays of Eocene (approximately 50 million years ago) and possible Miocene age (12 million years ago) have been mapped in the sediments of the coastal plain in South Carolina. The source of this volcanic activity is presently unknown.

Faulting: The names, distances and directions from the proposed site, and the probable age of the known faulting in the region are as follows:

| Name | Distance-Direction From Site | Probable Age Millions of Years |
|---|---|---|
| Brevard Fault | 11 Miles NW | 260 |
| Dahlonega Fault | 40 Miles W | 260 |
| Whitestone Fault | 47 Miles NW | 260 |
| Towaliga Fault | 90 Miles S | 260 |
| Cartersville Fault | 104 Miles W | 260 |
| Gold Hill Fault | 115 Miles E | 260 |
| Goat Rock Fault | 140 Miles SW | 260 |
| Triassic, Deep River Basin, N.C. and S.C. | 140 Miles E | 200 |
| Triassic, Danville Basin, N.C. | 145 Miles NE | 200 |
| Crisp and Dooly Counties, Ga. | 190 Miles SW | 12 to 70 |
| Probable Triassic Basin Charleston, S.C. | 200 Miles SE | 200 |

The locations of these faults with respect to the site are shown on Figure 2-49.

The first seven faults are all associated with the last metamorphic period. The Brevard, Whitestone, Dahlonega, and Cartersville faults apparently form an interrelated system. This system separates the eastern metamorphic belts from the Blue Ridge metamorphic belt and the overthrust zone on the west.

The Towaliga, Goat Rock, and Gold Hill Faults, and the Kings Mountain belt apparently form another interrelated alignment within the eastern metamorphic belts. The Kings Mountain belt is not considered a fault. Its association and alignment in relation to the three known faults mentioned and the location of earthquake epicenters within the area bounded by these features, lead to the conclusion that these features form an interrelated alignment.

There is no surface indication that any of these three faults have been active since the Triassic Period (200 million years).

Two fault locations in the region have been thoroughly investigated by borings. These are the Cartersville fault near the Allatoona Dam, and the Oconee-Conasauga fault in Georgia. These faults were found to be completely healed and not to have moved in many millions of years.

The Triassic basins of the Carolinas and further north may be due to the release of the compressional forces which formed the Appalachians. These basins are down-faulted grabens which are filled with Triassic sediments. Two earthquakes in the vicinity of McBee, South Carolina, may be related to an extension of a Triassic basin which has been inferred in the Chesterfield-Durham area.

Some faulting within the tertiary sediments in Dooly, Crisp, and Clay Counties, Georgia, has been mapped. The true areal extent of this faulting is unknown. This faulting apparently ranges from Cretaceous to possibly Miocene in age (70 to 12 million years).

The earthquake activity near Charleston, South Carolina, may indicate an active fault in that region. However, no evidence of surface faulting has been found.

### 2.5.2.3 Correlation of Earthquake Activity with Geologic Structures or Tectonic Provinces

The region surrounding the site can be divided into three major areas on the basis of the regional tectonics and the seismic history. These major seismic areas are:

1. the overthrust zone and Blue Ridge metamorphic belt;

2. the crystalline-metamorphic zone, exclusive of the Blue Ridge belt; and
3. the coastal plain.

The greatest number of recorded shocks have occurred within the overthrust zone and the Blue Ridge metamorphic belt northwest of the Brevard, Whitestone, Dahlonega, and Cartersville fault system. The epicenters in this area are generally widely scattered.

There have been a small number of earthquakes within the crystalline-metamorphic zone, exclusive of the Blue Ridge metamorphic belt. These earthquakes, extending from central Georgia to North Carolina, may be associated with the Towaliga, Goat Rock, Gold Hill, Kings Mountain alignment.

The coastal plain has experienced few earthquakes outside of the Charleston area. Four shocks, at Wilmington, North Carolina and Savannah, Georgia, have occurred but are unrelated to any known faulting, although the Wilmington shocks were adjacent to the Cape Fear Arch.

The only earthquake which does not closely fit this system of seismic areas is the 1924 shock in Pickens County, South Carolina (MM V Intensity). However, it is likely that this earthquake is associated with the overthrust-Blue Ridge seismic area.

### 2.5.2.4 Maximum Earthquake Potential

The assignment of probable future earthquake activity can only be based upon the previous record and the known geology of the area. Although the seismic history of the region is fairly short, a reasonable picture of the seismicity of the area becomes apparent from a study of the epicenter locations and the regional tectonics.

There are three significant zones of seismic activity in the general vicinity of the site; the Brevard and related faults zone, the overthrust zone, and the Towaliga, Goat Rock, Gold Hill, Kings Mountain alignment.

An evaluation of the earthquake activity and the regional geology can result in the selection of a series of maximum-sized shocks which are likely to occur in these various areas. Conservatively, we can assume that the previous maximum-sized shock on a particular fault zone can occur during the economic life of the proposed power station at perhaps the nearest approach of the particular fault system to the proposed site.

| Zone | Location | (MM) Intensity at Epicenter | Estimated Magnitude (Richter) |
|---|---|---|---|
| Brevard Fault Zone | 11 Miles NW | VI | Less than 4½ to 5 |
| Overthrust | 75 Miles NW | VIII | Less than 5½ to 6 |
| Towaliga, Goat Rock Gold Hill, Kings Mountain Alignment | 30 Miles SE | VII-VIII | Less than 5½ to 6 |

### 2.5.2.5 Seismic Wave Transmission Characteristics of the Site

Static and dynamic engineering properties of the soil and rock materials that underlie the site are discussed in Section 2.5.4, "Stability of Subsurface Materials and Foundations" on page 2-56. Design response spectra that include considerations of the thickness and distribution of these materials are discussed in Section 2.5.2.8, "Design Response Spectra" on page 2-55.

### 2.5.2.6 Maximum Hypothetical Earthquake (MHE)

The MHE acceleration value is 0.10 g for Class 1 structures founded on bedrock and 0.15 g for structures founded on overburden. The design response spectra are covered in Section 2.5.2.8, "Design Response Spectra."

### 2.5.2.7 Design Base Earthquake

It is considered likely that the shocks listed in Section 2.5.2.4, "Maximum Earthquake Potential" on page 2-54 could occur no closer than the indicated distances from the site during the life of the planned facilities. Since the magnitudes of these shocks are fairly small, the distance from the epicenter becomes extremely important. Ground accelerations would diminish rapidly with the distance from the epicenter. Although larger earthquakes occur within other fault zones, the highest ground accelerations at the site would be experienced from an earthquake along the Brevard fault zone. The assumption of a shock of less than Richter Magnitude five occurring along the Brevard fault zone at its closest location to the site (11 miles), would give ground motions on the order of five percent of gravity at the site. Vertical ground accelerations, as contrasted to the horizontal accelerations, would be only slightly less than five percent of the gravity in the competent rock at the site.

The DBE acceleration value is 0.05 g for both vertical and horizontal ground acceleration. The design response spectra are covered in Section 2.5.2.8, "Design Response Spectra."

### 2.5.2.8 Design Response Spectra

The Recommended Ground Motion for the 0.05 g, 0.10 g, and 0.15 g earthquakes are presented on Figure 2-50, Figure 2-52, and Figure 2-54.

The Recommended Ground Motion shows the expected maximum ground acceleration, velocity and displacement versus frequency at the site for the DBE and MHE. These plots are the expected ground motions of a particle within the rock at foundation level, and does not indicate the motions to be expected within a structure.

The Recommended Response Spectra curves for the 0.05 g, 0.10 g, and 0.15 g earthquakes are presented
2 on Figure 2-51, Figure 2-53, and Figure 2-55. The upper curve on the Recommended Response Spectra shows the expected maximum acceleration, velocity and displacement versus frequency that would be experienced by a simple inverted pendulum which has no damping if the pendulum was excited by the ground motions specified in the Recommended Ground Motion Spectrum. The other curves on the graph are plotted to show the effects of damping.

## 2.5.3 SURFACE FAULTING

This information is discussed in Section 2.5.1, "Basic Geologic and Seismic Information" on page 2-47 and Section 2.5.2, "Vibratory Ground Motion" on page 2-50.

## 2.5.4 STABILITY OF SUBSURFACE MATERIALS AND FOUNDATIONS

### 2.5.4.1 Geologic Features

This information is discussed in Section 2.5.1, "Basic Geologic and Seismic Information" on page 2-47.

### 2.5.4.2 Properties of Subsurface Materials

The materials underlying the site can be characterized by four zones. These four zones are shown on the subsurface profiles in Figure 2-56 through Figure 2-64 and are described in the following sections.

Zone 1 (Red Sandy Silty Clay or Clayey Silty Sand)

This residual soil derived from the in-place weathering of the parent rock, is the zone at the surface. This soil has been severely desiccated and partially cemented by oxidation of the iron it contains. This soil is strong, incompressible, and should not swell appreciably when saturated.

Zone 2 (Micaceous Silty Sand)

The second zone, like the first is derived from the in-place weathering of the parent rock. This zone consists of micaceous silty sand; decomposed rock that retains the relic structure of the original rock, often termed "saprolite." As is indicated by the standard penetration resistance, it is firm near the ground surface in the switchyard area (where it is thickest) but becomes denser with increasing depth. At this plant site, much of this zone has penetration resistances of 30 blows per foot or more and could be described either as a dense soil or a very soft rock. In general, this stratum is elastic and somewhat compressible because it has lost most of the intercrystalline bonds of the rock due to weathering, while much of the mica has not weathered sufficiently to lose its resiliency. The compressibility decreases and the rigidity increases with increasing density as reflected in the penetration resistances. In spite of this elastic nature, it is strong when confined and exhibits limited cohesion (both inter-particle bonding and capillary tension) as well as internal friction.

Zone 3 (Alternate Seams of the Soft Decomposed Rock and Hard Partially Decomposed Rock)

The third zone is the transition between soil and rock. This zone of alternate hard and soft weathered rock is exceedingly variable in its properties depending on the relative thicknesses of the contrasting seams. It is stronger than the saprolite zone above in shear across the seams but no stronger than the weakest seam parallel to them. The elasticity and compressibility are in proportion to the thickness of the soft seams because by comparison, the harder seams do not appreciably deflect under stress.

Zone 4 (Relatively Sound Rock)

The relatively sound rock below is both strong and rigid. The strength and elastic properties of small intact portions of the rock range from those of good concrete to several times those of concrete. The properties of the mass, however, are partially controlled by the joints and fissures. Therefore, the modulus of elasticity, the strength and the deflection of the mass are all somewhat lower than might be deduced from small scale laboratory tests of individual samples.

### 2.5.4.3 Exploration

A grid pattern of borings is established to provide the maximum amount of information for determining the foundation and soil conditions and permit flexibility in final plant layout, alignment, and elevation.

The general station area is shown on the included Location and Topographic Map, Figure 2-46 and the site and boring layout is shown on the Boring Plan, Figure 2-65.

The drilling, sampling, and rock coring are performed in accordance with methods specified by the American Society for Testing and Materials:

"Penetration Testing and Split Barrel Sampling of Soils" - D-1586-64T

"Diamond Core Drilling for Site Investigation" - D-2311-62T

"Thin Walled Tube Sampling of Soils" - D-1587-63T

NX and BX size rock cores were drilled at this site. The respective diameters of the rock cores are 2-1/8 and 1-5/8 inches. Boring logs are given in Figure 2-66 through Figure 2-115.

A limited amount of auger drilling, not required by the plant foundation exploration outline, was done in the vicinity of boring NA-9 in conjunction with seismic field testing. Also, auger boring was done for a piezometer installation to be used during percolation inflow tests made for groundwater analysis and evaluation.

Various laboratory tests were run on cores from Borings NA-4 and NA-9.

Compressional wave velocity and specific gravity measurements were performed on four cores. The results of these measurements are shown in Table 2-95.

Measurements were run on eight cores from the two borings to determine Young's modulus, Poisson's ratio, and ultimate crushing strength. The results of these measurements are shown in Table 2-96.

### 2.5.4.4 Geophysical Surveys

An uphole velocity survey was Performed on Boring NA-9. A Dynametric Interval Timer, Model 117-A, capable of measuring times of 0.0001 seconds, was used. Explosives in the boring of up to one-half pound of dynamite were used to create the shock wave.

The calculated velocities from this survey are somewhat anomalous because of the weathered and fractured character of the rock.

Two seismic refraction lines were shot across the site. A Mandrel Industries Interval Timer, ER-75, 12-trace refraction seismograph was used to record the lines. Explosives were used to provide the shock waves.

The location of the uphole boring and the seismic lines are shown on Figure 2-117.

Two cross sections through the site along the seismic refraction lines is shown on Figure 2-118. The interpretations on these cross sections are based upon the uphole velocity survey, the seismic refraction lines and velocity measurements on core samples. This interpretation of the velocities is considered generally reliable. These velocities are general averages and small areas within the site may not fit the cross section because the character and the depth and degree of weathering of the rock at the site varies greatly in short distances. The water table elevation may also vary somewhat from that shown on the cross sections.

The pattern of microtremor motion was recorded at the site. The instrument used is capable of a maximum gain of 150,000. However, this site is extremely quiet and no appreciable amplitudes were

recorded. (For example, a truck passing along the road less than 75 feet from the geophone produced double amplitudes of only 2.5 x $10^{-6}$ inches of ground motion.)

Because of the extremely low amplitudes of both the microtremor and the refraction energies, it was decided to perform an attenuation curve of the ground motion produced by explosives. Both the microtremor equipment and a Sprengnether Blast Recorder were used to measure the ground motion at 50, 100, 200, and 400 feet from 40-pound charges. This attenuation curve was compared with attenuation curves from sites with known characteristics to gain a better idea of the probable ground motion characteristics of the site. The results of this data indicated a marked attenuation of ground motion with distance.

0 **2.5.4.5 Deleted per 1990 Update**

### 2.5.4.6 Groundwater Conditions

This information is discussed in Section 2.4.13, "Groundwater" on page 2-41.

### 2.5.4.7 Response of Soil and Rock to Dynamic Loading

Under dynamic load the elastic materials may deform significantly. Experience with vibratory loading at a number of high-pressure pumping stations has demonstrated sufficient elastic response which can develop to be troublesome. The site is in a region of definite but infrequent seismic activity of moderate intensity. Under such dynamic loadings, foundations supported upon any appreciable thickness of the resilient micaceous materials could respond unfavorably, developing some magnification of the amplitude compared to the more rigid rock below.

Detailed studies of the elastic qualities of the soil-rock mass supporting the critical structures could probably develop a configuration for the structure-foundation system that would not provide amplification for the seismic frequencies anticipated. Such an analysis, however, is dependent on (1) an accurate evaluation of the rock-soil-structure elastic response and (2) an accurate knowledge of seismic frequency spectra. Available theories on soil-structure response are approximate at best and must be corrected from empirical observations made during earthquakes. Realistic frequency spectra must properly be determined from observations of ground motion during seismic activity of the same intensity as anticipated. Unfortunately, there was no instrumental observation of any of the earthquakes of the region sufficiently close to the site that either reliable frequency spectra or structural response of the soil can be evaluated. Microtremors, while of academic interest, are not of sufficient magnitude to make a reliable evaluation of earthquake response of the magnitude of those observed. In fact, there is some evidence that microseisms may arise from different mechanisms, particularly superficial, near surface strains and adjustments.

0 **2.5.4.8 Deleted per 1990 Update**

### 2.5.4.9 Earthquake Design Basis

The earthquake design basis is discussed in Section 2.5.2, "Vibratory Ground Motion" on page 2-50.

### 2.5.4.10 Static Stability

Although the individual critical station units may not tolerate substantial settlement, they are functionally inter-connected only by piping. This can absorb some differential movements if it is anticipated in the design.

Because of the relatively small thickness of the surface clayey soils and the irregular topography, the upper zone does not have an appreciable influence on the design of foundations for the major structures. This stratum does furnish excellent support for the smaller structures where there is no cut or only shallow fill.

Under static load alone, a major design consideration for heavy structures is the elastic deflection and consolidation of the micaceous soils of the saprolite zone and the micaceous, more weathered layers of the zone of alternate hard and soft seams. Experience, confirmed by laboratory tests, has shown that these materials can support power station loadings without appreciable settlement when the densities are sufficient, that is the penetration resistances consistently exceed 30 blows per foot.

0 ## 2.5.5 DELETED PER 1990 UPDATE

## 2.5.6 EMBANKMENTS AND DAMS

0 ### 2.5.6.1 Deleted per 1990 Update

### 2.5.6.2 Exploration

A thorough investigation has been made of the Keowee-Little River dam foundations (including the dam at the east end of the Oconee intake canal) by the Law Engineering Testing Company under the direction of Professor George F. Sowers.

A total of 74 soil and rock borings have been made to investigate the foundations of the Keowee and Little River dams and that of the dike at the east end of the Oconee intake canal. One hundred forty-six additional borings have been made to investigate foundations of nearby Keowee and Oconee structures and waterways.

At Keowee, 23 undisturbed samples were taken for laboratory testing to determine shear strength of the foundations.

At Little River, 19 undisturbed samples were taken for laboratory testing to determine shear strengths of the foundation.

### 2.5.6.3 Foundation and Abutment Treatment

At Keowee dam, based on test results, the extent of removal of material is specified such that shear strength of remaining material would equal or exceed shear strength of dam embankment. All alluvial material is removed. Since monitoring of any seepage in vicinity of the river itself would be extremely difficult due to backwater of Hartwell reservoir, a shallow grout curtain (10 ft-15 ft) is installed between and below the elevation 685 contours. The foundation report specifically notes that grouting is not required "to improve stability, reduce consolidation, or increase impermeability." The permeability of the intact reservoir soils varied between $1 \times 10^{-3}$ and $1 \times 10^{-4}$ feet per second as determined by laboratory tests.

Due to proximity of Keowee powerhouse (and its excavation) to left embankment, a core trench to rock is installed to provide a positive cutoff. A shallow grout curtain is placed below the bottom of core trench.

At Little River dam, as at Keowee, all material weaker in shear than the embankment materials and alluvium is excavated. A shallow (10 ft-15 ft) grout curtain is placed between and below elevation 675 contours. The permeability of the intact reservoir soils varied between $1 \times 10^{-4}$ and $1 \times 10^{-6}$ feet per second as determined by laboratory tests.

At Keowee and Little River dams and at Oconee intake canal dike, a three layer graded filter is placed under the downstream third of the dams and dike to intercept safely any seepage through the embankment and foundation. The dam abutments and upstream reservoir areas have natural blankets of residual, impervious material, and it is expected that these will prevent excessive seepage through the foundation.

0 ## 2.5.6.4 Deleted per 1990 Update

## 2.5.6.5 Slope Stability

### 2.5.6.5.1 Static Analyses

Static analyses are performed for both Keowee and Little River dams, and these studies are checked by re-analyzing the most critical circles of failure independently. The conditions studied, both upstream and downstream, included "steady state seepage," "sudden drawdown," and "construction" before the reservoir was filled, utilizing the appropriate shear strength data for each condition.

### 2.5.6.5.2 Seismic Analyses

The static analyses extend to include the effect of acceleration and the resulting "inertia forces" on stability. The method utilized is that proposed by N. Newmark (1965) in the Rankine Lecture at the Institution of Civil Engineers (London).

In this analysis a steady acceleration is assumed to be applied to the centroid of the potentially sliding segment of soil in the direction which produces the greatest increase in overturning moment.

The results show that the embankments will have safety factors of 1.0 or more when the steady state acceleration is introduced. Of course, as Dr. Newmark points out, this dynamic approach is not rigorous because earthquakes loadings are transient, not steady, but the results should be on the safe side.

For earthquake loadings, the minimum permissible safety factor considered prudent by such organizations as the Corps of Engineers is 1.0 when combined with steady state seepage.

### 2.5.6.5.3 Shear Parameters

The shear parameters utilized in Section 2.5.6.5.1, "Static Analyses" and Section 2.5.6.5.2, "Seismic Analyses" are the consolidated-undrained or R values which impose a rapid change in stress upon a soil that has consolidated under sustained load. The load change is applied so rapidly that no change in water content could occur even though the soils are saturated. The rate of loading, however, could not be termed "dynamic." In dynamic loading of such clayey soils, viscous forces would be mobilized, and therefore, the strength would be somewhat greater.

Only one loading cycle is employed. In loose cohesionless soils or sensitive clays repeated loading can cause a change in structure and progressive loss in strength. Previous experience with the undisturbed soils of the region, as well as the compacted soils, shows that the soils do not suffer progressive breakdown with repeated load. Therefore, the static shear parameters should be safe and the steady state acceleration, N, for seismic loading will be substantially the same as for static.

## 2.5.6.6 Seepage Control

Investigation and corrective action are discussed in Section 2.5.6.2, "Exploration" on page 2-59 and Section 2.5.6.3, "Foundation and Abutment Treatment" on page 2-59 respectively. Permeability is discussed in Section 2.4, "Hydrologic Engineering" on page 2-37.

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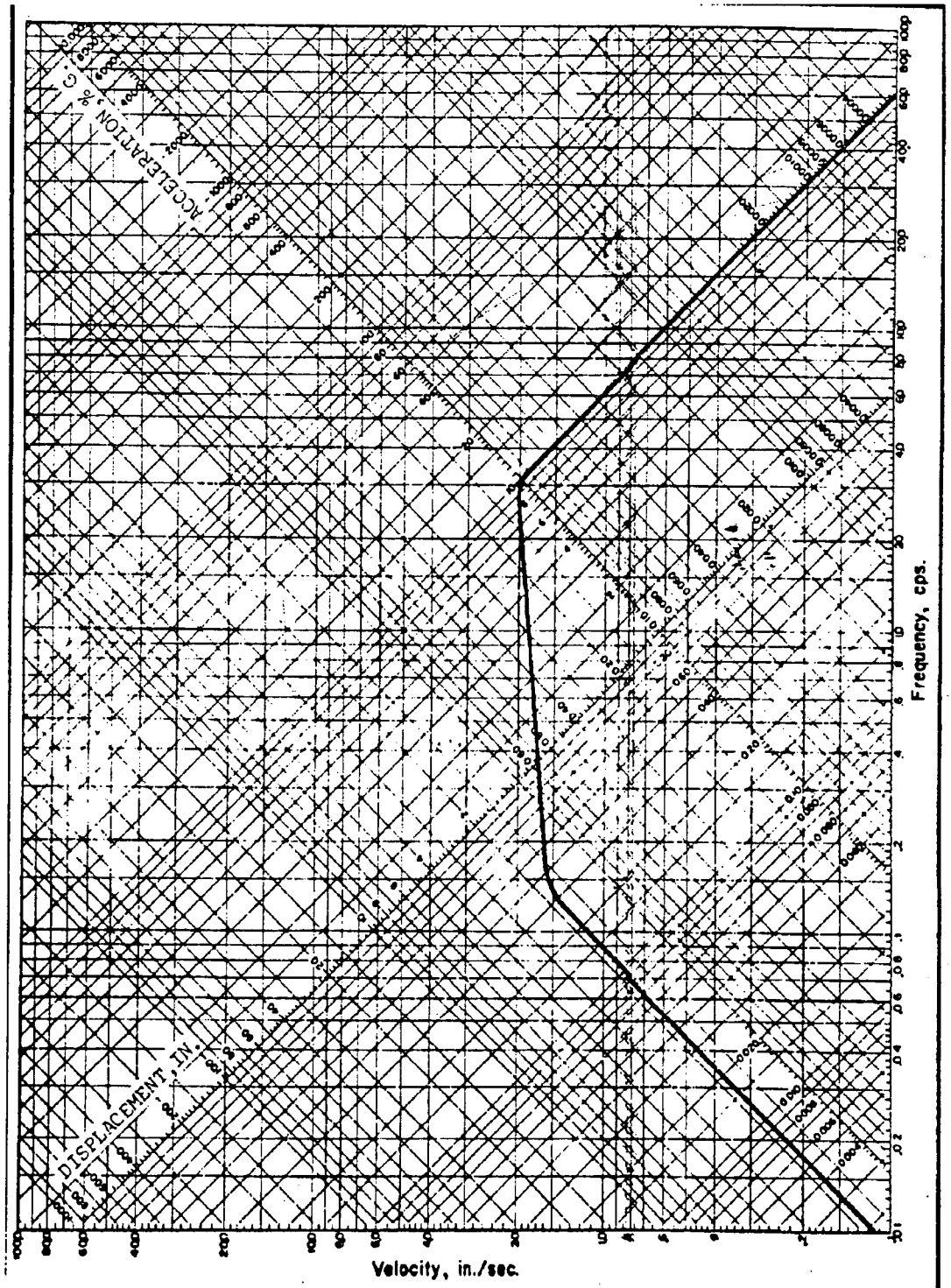



GROUND MOTION SPECTRUM

Figure 2-52.
Ground Motion Spectra

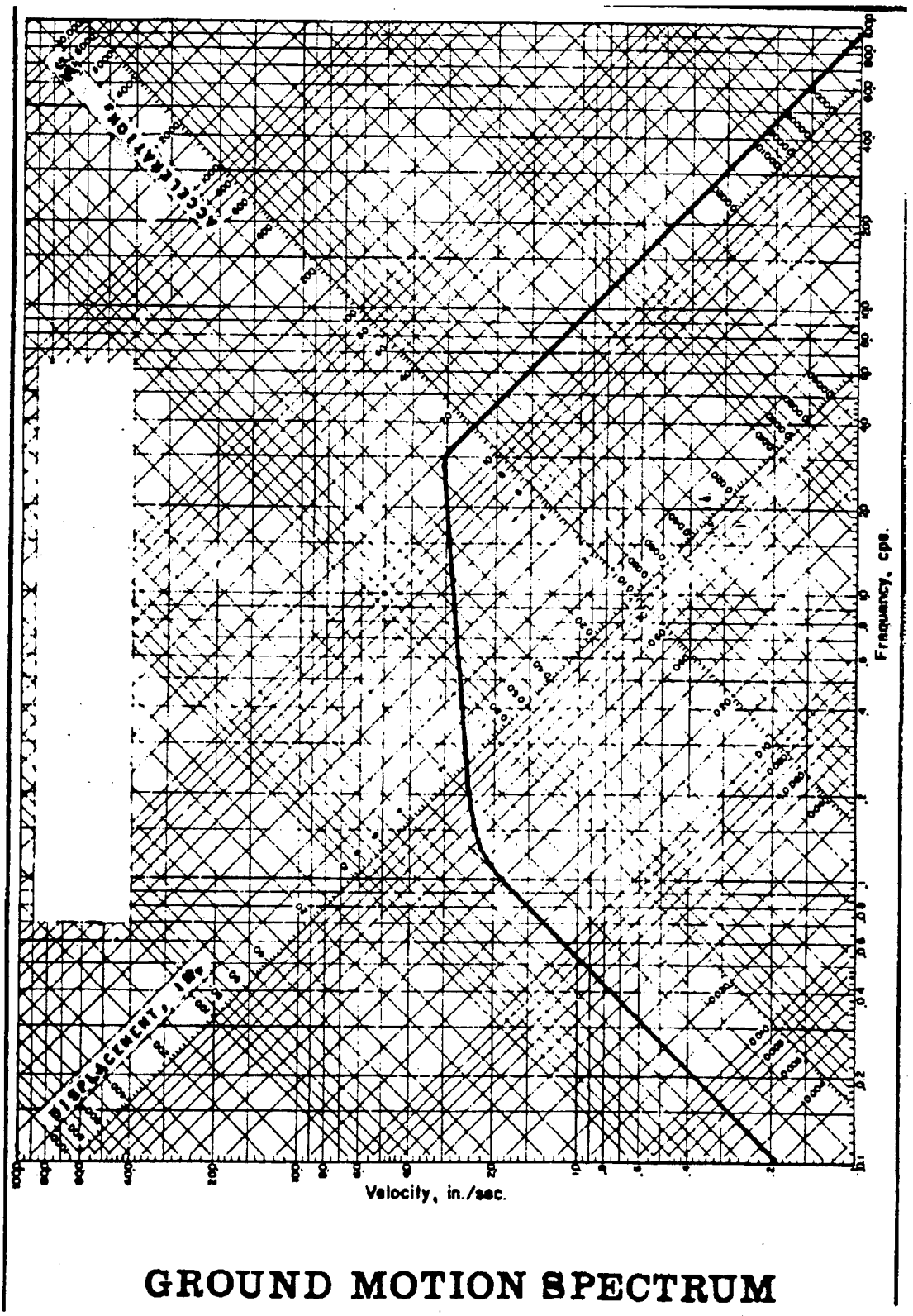



Figure 2-54.
Ground Motion Spectra

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 0 | | | | 794.1 | |
| | RED MICACEOUS SANDY CLAYEY SILT | | | | |
| 3.6 | STIFF TO VERY STIFF RED YELLOW BROWN MICACEOUS FINE TO MEDIUM SANDY SILT | | | 789.1 | N = 11 |
| | | | | | Undisturbed Sample 4.0 to 5.0 feet. |
| | | | | 784.1 | N = 20 |
| | | | | 779.1 | N = 26 |
| 18.0 | VERY STIFF TO HARD GRAY BROWN MICACEOUS FINE TO MEDIUM SANDY SILT | | | 774.1 | N = 28 |
| | | | | 769.1 | N = 38 |
| 28.0 | VERY DENSE GRAY BROWN BLACK MICACEOUS SILTY FINE TO COARSE SAND | | | 764.1 | N = 50 |
| 32.0 | VERY DENSE YELLOW BROWN MICACEOUS SILTY FINE TO COARSE SAND | | | 759.1 | N = 65 |
| 40.0 | | | | 754.1 | |

| | | | |
|---|---|---|---|
| LI | Light | HD | Hard |
| DK | Dark | BI | Biotite |
| GY | Gray | HO | Hornblende |
| BK | Black | GRAN | Granite |
| MOD | Moderately | GN | Gneiss |
| MED | Medium | | |

End of Boring

WATER TABLE

Figure 2-66.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 40.0 | | | | 754.1 | |
| | SOFT GRAY AND BROWN GRANITE GNEISS | 75% | | | |
| 44.9 | | | | 749.1 | |
| 45.6 | MOD HD LI GY GN | 93% | | | |
| 45.8 | SOFT DK GY BROWN BI HO GN | | | | |
| 46.0 | MOD LI GY GRAN GN | | | | 28° Foliation Plane at 45.1 Feet |
| 47.3 | MOD HD DK GY BROWN BI HO GN | | NX | | |
| 48.2 | SOFT LI GY BROWN GRAN GN | | | | |
| 48.5 | SOFT DK GY BROWN BI HO GN | | | | |
| 52.7 | SOFT AND MOD HD ALTERNATING LAYERS LI GY BROWN GRAN GN | | | 744.1 | |
| 53.6 | HARD LI GY QUARTZ SEAM | | | | |
| | HD LI GY GRAN GN | | | 739.1 | Vertical Joint at 50 Feet |
| | | 100% | | | |
| 58.5 | | | | | |
| 58.8 | HD DK GY BI HO GN | | | | |
| 60.9 | HD LI GY GRAN GN | | | 734.1 | |
| 61.1 | HD DK GY BI HO GN | | | | |
| | HD LI GY GRAN GN | | BX | | |
| 63.4 | | | | | |
| 64.1 | HD DK GY BI HO GN | | | | |
| | HD LI GY GRAN GN WITH INCLUSIONS OF HORNBLENDE CONCENTRATIONS | | | 729.1 | |
| 66.2 | | 100% | | | |
| 67.4 | MOD HD DK GY BK BI HO GN | | | | |
| 67.9 | HD LI GY GRAN GN | | | | |
| 68.2 | HD DK GY BI HO GN | | | | |
| 70.1 | HD LI GY GRAN GN | | | 724.1 | |
| 70.3 | HD DK GY BI HO GN | | | | |
| 73.8 | HD LI GY GRAN GN WITH THIN HD GN SEAMS | | | 720.3 | |
| | CORING TERMINATED | | | 719.1 | |

WATER TABLE

Figure 2-67.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 0 | | | | 812.5 | |
| | BROWN MICACEOUS SANDY SILT | | | | |
| 3.8 | | | | | |
| 4.9 | MOD HD LI GY BROWN GRAN GN | | | 807.5 | |
| | VERY SOFT GRAN GN AND BROWN SANDY SILT | 36% | | | |
| | | | | 802.5 | |
| 10.5 | MOD HD LI GY BROWN GRAN GN WITH SOIL SEAMS | 86% | | 797.5 | |
| 18.8 | VERY SOFT LI GY GRAN GN WITH THIN LAYERS OF MOD HD GRAN GN | | | 792.5 | |
| | | 50% | NX | 787.5 | |
| | | | | 782.5 | |
| | | 28% | | 777.5 | |
| 38.8 | | | | | |
| 40.0 | HD LI GY GRAN GN | 65% | | 772.5 | |

| | | | |
|---|---|---|---|
| LI | Light | HD | Hard |
| DK | Dark | BI | Biotite |
| GY | Gray | HO | Hornblende |
| BK | Black | GRAN | Granite |
| MOD | Moderately | GN | Gneiss |
| MED | Medium | | |

Figure 2-68.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 40.0 | | | | 772.5 | |
| | HD LI GY GRAN GN | 65% | NX | | |
| | | | | 767.5 | |
| | | 78% | | | |
| | | | | | 80° Joint at 46.0 - 47.5 Feet |
| 48.7 | | | | | |
| | SOFT DK GY BROWN BI HO GN | | | 762.5 | |
| 50.9 | | | | | |
| | HD LI GY GRAN GN | 85% | | | |
| | | | | 757.5 | |
| | | | BX | | |
| 58.3 | | | | | |
| 59.3 | MOD HD DK GY BI HO GN | | | | |
| | HD LI GY GRAN GN | 100% | | 752.5 | |
| 63.2 | | | | | |
| 63.5 | HD DK GY BK BI HO GN | | | | |
| 65.1 | HD LI GY GRAN GN | | | 747.5 | |
| | HD DK GY BK BI HO GN | | | | |
| 68.2 | | | | | |
| | HD LI GY GRAN GN | 100% | | 742.5 | |
| | | | | 739.7 | |
| 72.8 | | | | | |
| | CORING TERMINATED | | | | |
| | | | | 737.5 | |

Figure 2-69.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 0 | | | | 744.7 | |
| | BROWN MICACEOUS SANDY SILT | | | | |
| | | | | 739.7 | N = 25<br>Undisturbed Sample<br>5.5 - 6.5 Feet |
| | | | | 734.7 | N = 99 |
| | | | | 729.7 | Refusal |
| | | | | 724.7 | Refusal |
| 22.0 | MOD HD DK GY BK BI HO GN | | | | |
| 22.9 | HD LI GY GRAN GN | | | 719.7 | |
| 26.2 | | | | | |
| 26.3 | MOD HD DK GY BK BI HO GN | | | | |
| 26.7 | HD LI GY GRAN GN | 85 | % | | |
| 26.8 | MOD HD DK GY BK BI HO GN | | | | |
| | HD LI GY GRAN GN | | NX | 714.7 | |
| | | 100 | % | | |
| 33.5 | MOD HD DK GY BI HO GN | | | | |
| 34.2 | HD LI GY GRAN GN | | | 709.7 | |
| 37.7 | | 100 | % | | |
| | MOD HD DK GY BK BI HO GN | | BX | | |
| 38.0 | | | | | |
| 40.0 | HD LI GY GRAN GN | | | 704.7 | |

LI - Light
DK - Dark
GY - Gray
BK - Black
MOD - Moderately
MED - Medium
HD - Hard

BI - Biotite
HO - Hornblende
GRAN - Granite
GN - Gneiss

Figure 2-70.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 40.0 | | | | 704.7 | |
| | HD LI GY GRAN GN | 100% | BX | 702.7 | |
| 42.0 | | | | | |
| | CORING TERMINATED | | | | |
| | | | | 699.7 | |

Figure 2-71.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 0 | | | | 810.42 | |
| | VERY STIFF YELLOW BROWN MICACEOUS FINE TO MEDIUM SANDY SILT | | | | N = 32 |
| | | | | 805.42 | |
| 7.8 | | | | | |
| | MOD HD LI GY GRAN GN | 96% | | | Lost Water 9.9 Feet |
| | | | | 800.42 | 82° Joint at 11.8 Feet |
| | | | | | 82° Joint at 12.2 Feet |
| | | | | | 81° Joint at 13.2 Feet |
| | | | | | 54° Joint at 14.0 Feet |
| | | | NX | 795.42 | |
| 15.6 | | | | | |
| | SOFT LI GY GRAN GN | 60% | | | |
| 17.9 | | | | | |
| | MOD HD LI GY GRAN GN | | | | |
| | | | | 790.42 | |
| 21.1 | | | | | |
| 21.7 | SOFT DK GY BK BI HO GN | 90% | | | |
| | HD LI GY GRAN GN | | | | |
| | | | | 785.42 | |
| 26.5 | | 100 | % | | |
| 26.7 | HD DK GY BI HO GN | | | | |
| | HD LI GY GRAN GN | 100 | % | | |
| | | | | 780.42 | |
| 30.8 | | | | | |
| | HD DK GY BK BI HO GN | 100 | % | | |
| 34.1 | | | BX | | |
| 34.4 | HD LI GY GRAN GN | | | 775.42 | |
| 36.6 | HD DK GY BK BI HO GN | | | | 23° Foliation Plane at 36 Feet |
| | HD LI GY GRAN GN | | | | |
| 39.3 | | 100 | % | | |
| 40.0 | HD LI GY QUARTZ PEGMATITE | | | 770.43 | |

31.0 Feet of 2" Plastic Pipe left in hole

| | | | |
|---|---|---|---|
| LI | Light | HD | Hard |
| DK | Dark | BI | Biotite |
| GY | Gray | HO | Hornblende |
| BK | Black | GRAN | Granite |
| MOD | Moderately | GN | Gneiss |
| MED | Medium | | |

WATER TABLE

Figure 2-72.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 40.0 | | | | 770.42 | |
| | HD LI GY QUARTZ PEGMATITE | 100% | | | |
| 43.0 | | | | | |
| | HD LI GY GRAN GN | 100 % | | 765.42 | |
| 47.6 | | | | | |
| 48.1 | HD DK GY BI HO GN | | | | |
| | HD DK GY GRAN GN | 100 % | | 760.42 | |
| 50.9 | | | | | |
| 51.1 | HD DK GY BI HO GN | | BX | | |
| | HD LI GY GRAN GN | 100 % | | 755.42 | |
| 56.8 | | | | | |
| 57.3 | HD DK GY BK BI HO GN | | | | |
| 58.0 | HD LI GY GRAN GN | | | | |
| 58.2 | HD DK GY BK BI HO GN | | | | |
| 58.7 | HD LI GY GRAN GN | 100 % | | 750.42 | |
| 58.9 | HD DK GY BK BI HO GN | | | | |
| 59.3 | HD LI GY GRAN GN | | | | |
| 59.7 | HD BK BI HO GN | | | | |
| | HD LI GY GRAN GN | 100 % | | 745.42 | |
| | | 95% | | 740.02 | |
| 70.4 | CORING TERMINATED | | | 735.42 | |

Not to scale

Figure 2-73.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. 802 | REMARKS |
|---|---|---|---|---|---|
| 0 | DENSE RED BROWN MICACEOUS SILTY FINE TO MEDIUM SAND | | | 797 | N = 42 |
| 9.0 | SOFT AND SOME MOD HD LI GY GRAN GN | 33% | NX | 792 | 75° Joint at 10 Feet |
| | | | | 787 | |
| | | 39% | | 782 | |
| | | | | 777 | |
| 28.0 | | 86% | | | |
| 29.7 | MOD HD LI GY GRAN GN | | | | |
| 30.1 | MOD HD LI GY AND DK GY BK | *1 | | 772 | |
| 31.2 | MOD HD LI GY GRAN GN | | | | |
| 31.5 | MOD HD DK GY GN | | | | |
| | MOD HD LI GY GRAN GN | | | | |
| 34.2 | HD LI GY GRAN GN | | | 767 | |
| 38.8 | | 100% | | | |
| 39.0 | MOD HD DK GY BI HO GN | | | | |
| 39.5 | HD LI GY GRAN GN | | | | |
| 40.0 | MOD HD DK GY BL BI HO GN | | | 762 | |

*1 ALTERNATING THIN LAYERS OF GRAN GN AND BI HO GN

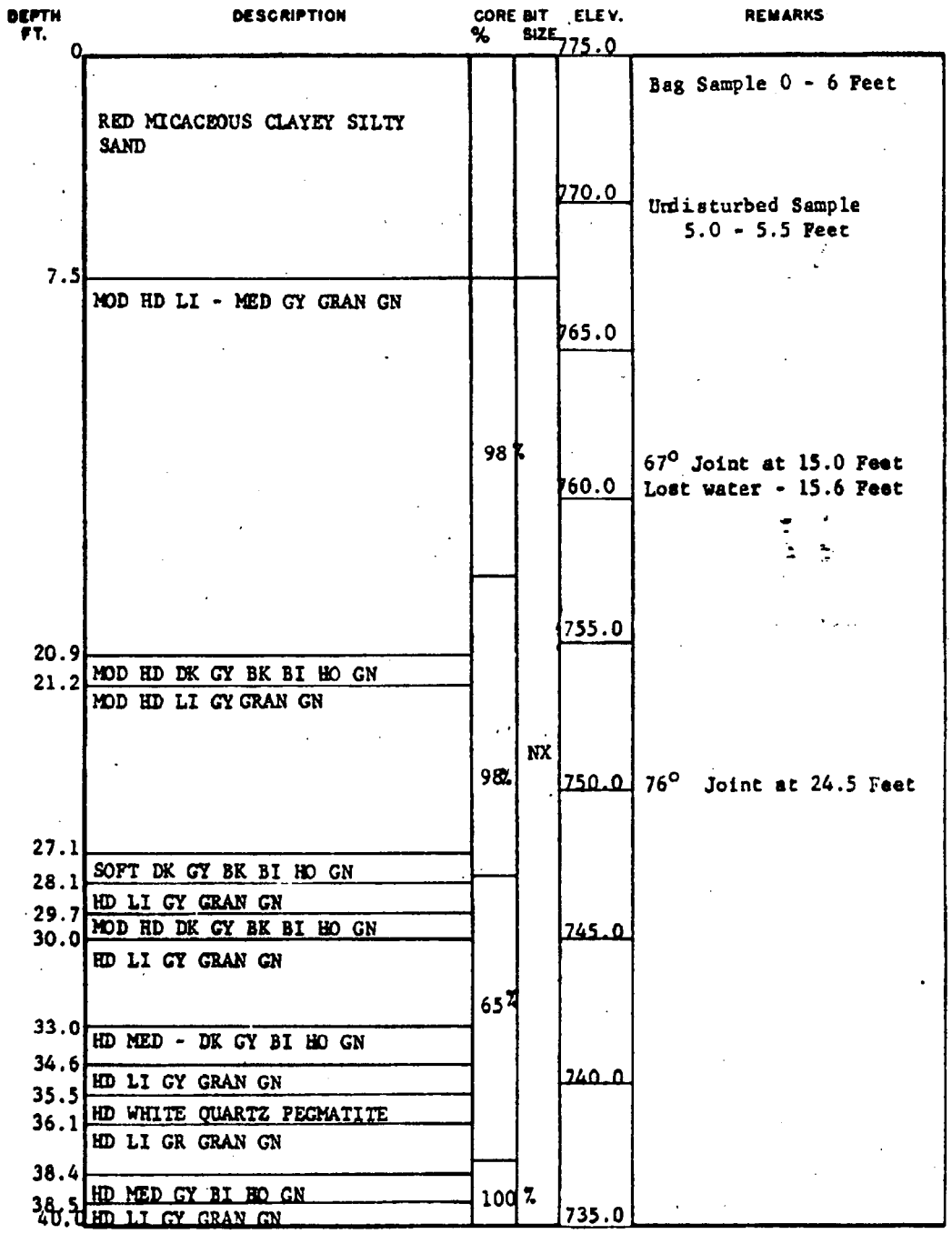
| | | | |
|---|---|---|---|
| LI | Light | HD | Hard |
| DK | Dark | BI | Biotite |
| GY | Gray | HO | Hornblende |
| BK | Black | GRAN | Granite |
| MOD | Moderately | GN | Gneiss |
| MED | Medium | | |

No Water Encountered

Figure 2-74.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 40.0 | | | | 762 | |
| 40.7 | MOD HD DK GY BK BI HO GN | | | | |
| | HD LI GY GRAN GN | 92% | | | |
| 44.5 | | | | | |
| 44.8 | MOD HD DK GY BK BI HO GN | | | 757 | |
| 45.8 | HD LI GY GRAN GN | 86% | | | |
| | MOD HD DK GY BK BI HO GN | | | | |
| 47.8 | | | | | |
| | HD LI GY GRAN GN | | | 752 | |
| 51.8 | | | | | |
| 52.5 | MOD HD DK GY BK BI HO GN | 98% | NX | | |
| | HD LI GY GRAN GN | | | 747 | |
| 57.3 | | | | | |
| 58.2 | HD DK GY BI HO GN | | | | |
| 58.8 | HD LI GY GRAN GN | | | | |
| 58.9 | HD DK GY BI HO GN | | | 742 | |
| 61.5 | HD LI GY GRAN GN | | | | |
| | HD DK GY BK BI HO GN | | | | |
| 63.2 | | 100% | | | |
| 64.2 | HD LI GY GRAN GN | | | | |
| 64.4 | HD DK GY BK BI HO GN | | | 737 | |
| | HD LI GY GRAN GN | 99% | | | |
| 67.1 | | | | | |
| 68.1 | HD DK GY BI HO GN | | | | |
| 68.9 | HD LI GY GRAN GN | | | | |
| 69.3 | HD DK GY BI HO GN | | BX | 732 | |
| | HD LI GY GRAN GN | | | 731.2 | |
| 70.8 | | | | | |
| | CORING TERMINATED | | | | |
| | | | | 727 | |

Figure 2-75.
Core Boring Record




LI - Light
DK - Dark
GY - Gray
BK - Black
MOD - Moderately
MED - Medium

HD - Hard
BI - Biotite
HO - Hornblende
GRAN - Granite
GN - Gneiss

Not to scale

Figure 2-76.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 40.0 | | | | 735.0 | |
| 40.7 | HD LI GY GRAN GN | | | | |
| 40.8 | HD MED GY BI HO GN | | | | |
| | HD LI GY GRAN GY | | | | |
| 43.7 | | | | | |
| 44.7 | HD LI-DK GY ALTERNATING LAYERS*1 | 100 % | | 730.0 | |
| 45.6 | HD LI GY GRAN GN | | | | |
| 45.9 | HD MED GY BI HO GN | | NX | | |
| 46.6 | HD LI GY GRAN GN | | | | |
| 46.7 | HD MED GY BI HO GN | | | | |
| 47.0 | HD LI GY GRAN GN | | | | |
| 47.1 | HD DK GY BK BI HO GN | | | | |
| 50.3 | HD LI GY GRAN GN | 100 % | | 725.0 | |
| 50.9 | HD MED-DK GY BI HO GN | | | | |
| 51.6 | HD LI GY GRAN GN | | | | |
| 51.7 | HD MED DK GY BI HO GN | | | 723.3 | |
| | CORING TERMINATED | | | | |

*1 OF BI HO GN AND GRAN GN

Not to scale

Figure 2-77.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 0 | | | | 784.5 | |
| | VERY DENSE RED MICACEOUS CLAYEY SILTY SAND WITH SOME QUARTZ FRAGMENTS | | | 779.5 | N=50 Undisturbed sample 4.5 to 5.5 feet |
| 5.0 | VERY DENSE GRAY TO WHITE MICACEOUS SILTY SAND | | | 774.5 | N=79 |
| | | | | 769.5 | N=80 |
| | | | | 764.5 | N=44 |
| | | | | 759.5 | |
| 25.0 | SOFT MED GY GRAN GN | | | | |
| 29.8 | MOD HD LI-MED GY GRAN GN | | | 754.5 | |
| | | 80% | NX | | |
| | | | | 749.5 | |
| | | 100% | BX | | |
| 40.0 | | | | 744.5 | |

36.0 Feet of 2" Plastic Pipe left in hole

LI - Light
DK - Dark
GY - Gray
BK - Black
MOD - Moderately
MED- Medium

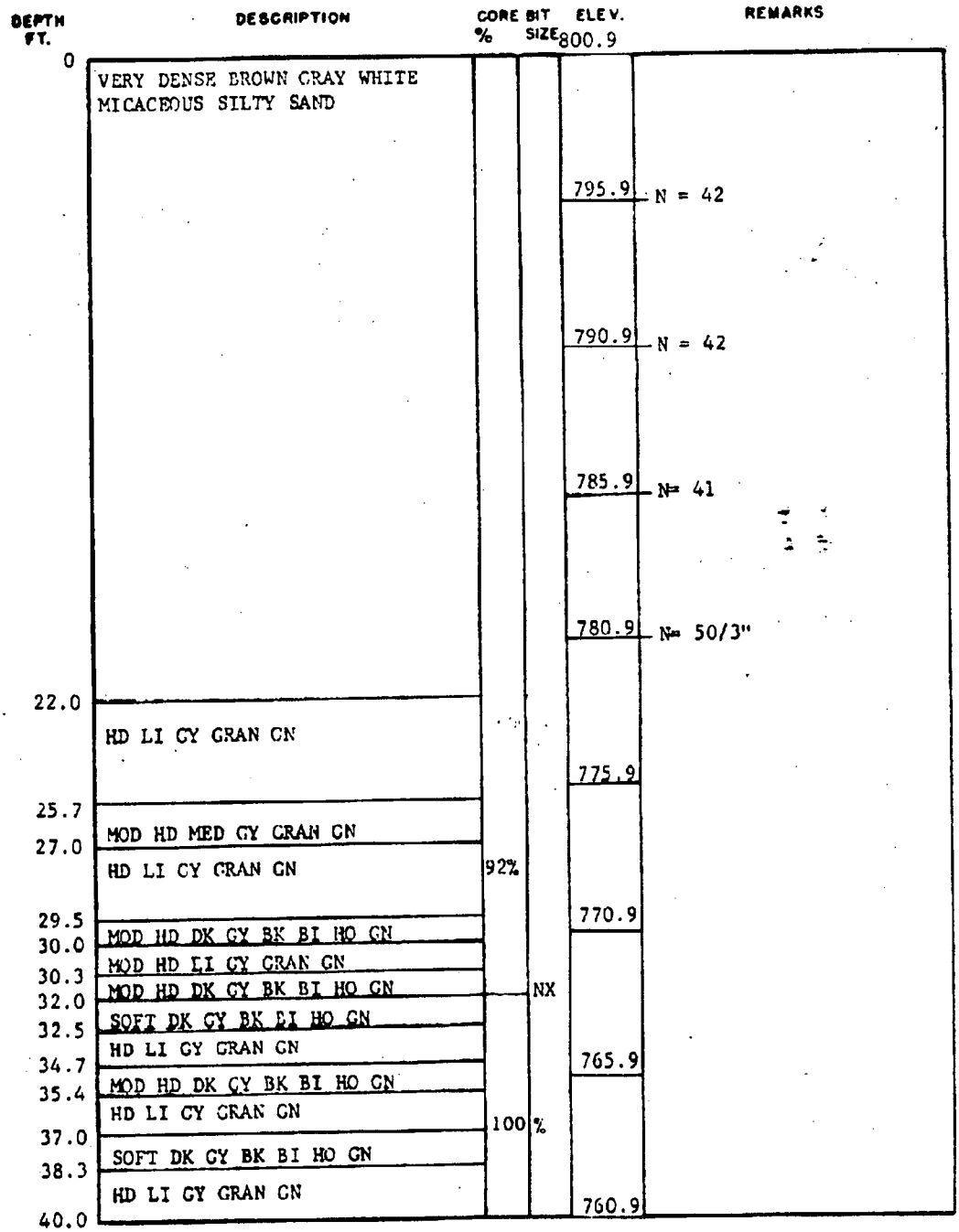
HD - Hard
BI - Boitite
HO - Hornblende
GRAN - Granite
GN - Gneiss

Figure 2-78.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 40.0 | | | | 744.5 | |
| 40.1 | MOD HD LI-MED GY GRAN GN | | | | |
| | MOD HD DK GY BK BI HO GN | 100% | | | 50° Joint at 42.7 Feet<br>50° Joint at 43.5 Feet |
| 44.8 | | | | 739.5 | |
| | MOD HD LI GY GRAN GN | | | | |
| 46.5 | | | | | |
| 46.6 | MOD HD DK GY BK BI HO GN | 100% | | | |
| 47.7 | MOD HD LI GY GRAN GN | | | | |
| 47.8 | MOD HD DK GY BK BI HO GN | | | | |
| 49.2 | MOD HD LI GY GRAN GN | | | | |
| 49.3 | MOD HD DK GY BI HO GN | | | | |
| 49.5 | MOD HD LI GY GRAN GN | | | | |
| 49.8 | MOD HD DK GY BI HO GN | | | 734.5 | |
| 50.5 | MOD HD LI GY GRAN GN | | BX | | |
| 53.0 | HD WHITE QUARTZ PEGMATITE | 98% | | | |
| 54.0 | HD LI GY GRAN GN | | | | |
| 55.2 | HD WHITE QUARTZ PEGMATITE | | | 729.5 | |
| 58.0 | HD LI GY GRAN GN | | | | |
| 58.1 | MOD HD DK CY BI HO GN | | | | |
| 59.5 | HD LI CY GRAN GN | | | | |
| 60.7 | HD WHITE QUARTZ PEGMATITE | 100% | | 724.5 | |
| 63.7 | HD LI CY GRAN GN | | | | |
| 64.7 | SOFT LI-MED CY GRAN GN | | | 719.5 | |
| 65.3 | HD LI GY QUARTZ PEGMATITE | | | | |
| 66.0 | HD LI GY GRAN GN | | | | |
| 66.7 | HD WHITE QUARTZ PEGMATITE | 100% | | | |
| 69.3 | HD LI GY GRAN GN | | | | |
| 70.4 | MOD HD LI-MED GY GRAN GN | | | 713.5 | |
| 70.9 | MOD HD DK GY GK BI HO GN | | | | |
| 71.3 | MOD HD LI-MED GY GRAN GN | | | 713.2 | |
| | CORING TERMINATED | | | | |
| | | | | 709.5 | |

Not to scale

Figure 2-79.
Core Boring Record




LI - Light
DK - Dark
GY - Gray
BK - Black
MOD - Moderately
MED - Medium

HD - Hard
BI - Biotite
HO - Hornblende
GRAN - Granite
GN - Gneiss

Figure 2-80.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 40.0 | | | | 760.9 | |
| 40.9 | HD LI GY GRAN GN | 100 % | NX | | 42° Joint at 40.8 Feet |
| 41.7 | MOD HD DK GY BK BI HO GN | | | | |
| | MOD HD LI-MED GY GRAN GN | | | 755.9 | |
| 46.7 | | 100 % | | | |
| 46.8 | MOD HD DK GY BK BI HO GN | | | | |
| 47.1 | MOD HD LI GY GRAN GN | | | | |
| 47.2 | MOD HD DK GY BK BI HO GN | | | | |
| 50.0 | HD LI GY GRAN GN | | BX | 750.9 | |
| 50.2 | HD DK GY BK BI HO GN | | | | |
| | HD LI GY GRAN GN | | | | |
| 53.4 | | | | | |
| 54.4 | HD DK GY BK BI HO GN | | | 745.9 | |
| | HD LI GY GRAN GN | 100 % | | | |
| 56.0 | HD DK GY BK BI HO GN | | | | |
| 57.7 | | | | | |
| 58.2 | HD LI GY GRAN GN | | | | |
| 58.7 | HD DK GY BK BI HO GN | | | 740.9 | |
| | HD LI GY GRAN GN | | | | |
| 61.1 | | | | | |
| 61.5 | HD DK GY BK BI HO GN | | | | |
| 61.6 | HD LI GY GRAN GN | | | | |
| 62.1 | HD DK GY BK BI HO GN | 100 % | | | |
| 64.5 | HD LI GY GRAN GN | | | 736.4 | |
| | CORING TERMINATED | | | 735.9 | |

Not to scale

Figure 2-81.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 0 | | | | 822.7 | |
| | STIFF RED BROWN MICACEOUS FINE TO COARSE SANDY CLAYEY SILT WITH SOME QUARTZ FRAGMENTS | | | | N = 27<br>N = 13 |
| | | | | 817.7 | N= 25/4"<br>N= 20/0" |
| 7.5 | MOD HD LI GY TO WHITE QUARTZ PEGMATITE | 54% | NX | 812.7 | |
| 12.5 | HAWTHORNE BIT - VERY DENSE GY SILTY SAND | | | 807.7 | - Refusal |
| | | | | 802.7 | |
| 20.0 | SOFT LI GY GRAN GN | | | | |
| 20.4 | MOD HD LI TO MED GY GRAN GN | 48 % | NX | 797.7 | |
| | | 93% | | 792.7 | 60° Joint at 30.7 Feet<br>Lost water at 30.8 Feet |
| | | 94 | | 787.7 | |
| | | 100% | BX | | |
| 38.4 | | | | | |
| 39.3 | VERY SOFT MED GY GRAN GN | | | | |
| 40.0 | MOD HD LI TO MED GY GRAN GN | 100% | | 782.7 | |

31.0 Feet of 2" Plastic Pipe left in hole

LI - Light
DK - Dark
GY - Gray
BK - Black
MOD - Moderately
MED - Medium

HD - Hard
BI - Biotite
HO - Hornblende
GRAN - Granite
GN - Gneiss

bjs

WATER TABLE

Figure 2-82.
Core Boring Record

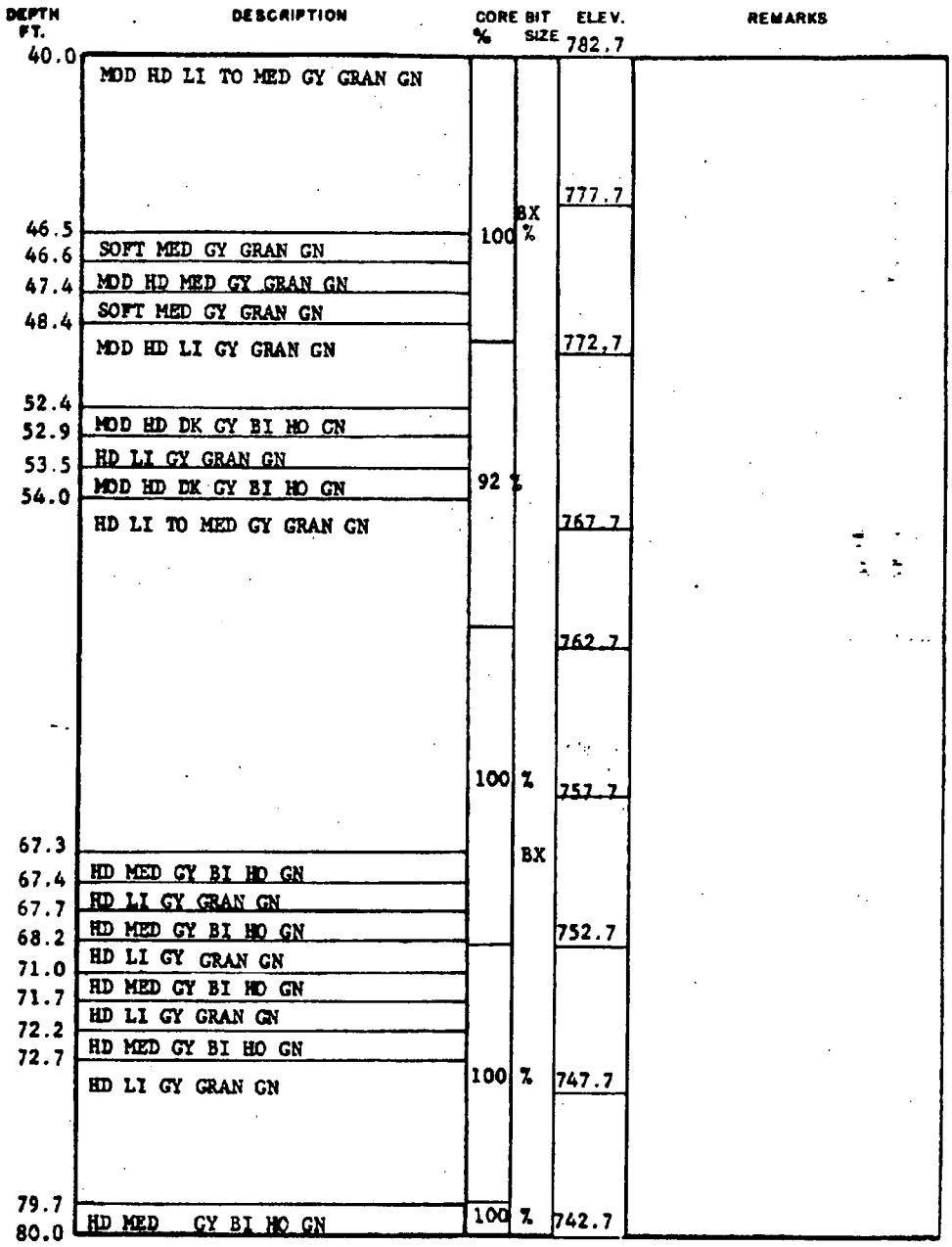



Figure 2-83.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 80.0 | | | | 742.7 | |
| 80.2 | HD MED GY BI HO GN | | | | |
| 81.0 | HD LI GY GRAN GN | | | | |
| 81.3 | MOD HD DK GY BI HO GN | | | | |
| 82.0 | HD LI GY GRAN GN | | | | |
| 83.0 | HD MED GY BI HO GN | | | | |
| 83.7 | HD LI GY GRAN GN | | | | |
| 83.9 | HD DK GY BI HO GN | 100 | | | |
| 84.3 | HD LI GY GRAN GN | | | | |
| 84.7 | HD MED TO DK GY BI HO GN | | | | |
| 87.2 | HD LI GY GRAN GN | | BX | 737.7 | |
| 87.5 | HD MED GY BI HO GN | | | | |
| 88.1 | HD WHITE QUARTZ PEGMATITE | | | | |
| 92.8 | HD LI GY GRAN GN | | | 732.7 | |
| 93.4 | HD DK GY BI HO GN | | | | |
| 94.0 | HD LI GY GRAN GN | | | | |
| 94.4 | HD MED TO DARK GY BI HO GN | | | | |
| 95.7 | HD LI GY GRAN GN | 98 | | 727.7 | |
| 96.5 | HD MED GY BI HO GN | | | | |
| 96.8 | HD LI GY GRAN GN | | | | |
| 97.3 | HD MED GY BI HO GN | | | | |
| 99.2 | HD LI GY GRAN GN | | | 723.5 | |
| | CORING TERMINATED | | | | |

Not to scale

WATER TABLE

Figure 2-84.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 0 | | | | 826.4 | |
| 0.4 | TOPSOIL AND GRASS | | | | |
| | STIFF BROWN MICACEOUS CLAYEY SANDY SILT | | | | N = 10 |
| | | | | 821.4 | |
| 6.0 | FIRM TO VERY DENSE BROWN GRAY WHITE MICACEOUS SILTY FINE TO COARSE SAND WITH ROCK FRAGMENTS | | | | N = 9 |
| | | | | 816.4 | |
| | | | | 811.4 | N = 20/0" (Refusal) |
| | | | | | N = 52 |
| | | | | 806.4 | |
| 22.0 | VERY STIFF BROWN MICACEOUS FINE SANDY SILT | | | | N = 16 |
| | | | | 801.4 | |
| 26.0 | VERY STIFF YELLOW BROWN MICACEOUS FINE SANDY SILT | | | | N = 16 |
| | | | | 796.4 | |
| 30.0 | FIRM WHITE GRAY SILTY FINE TO COARSE SAND | | | | N = 28 |
| | | | | 791.4 | |
| 37.7 | MOD HD LI TO MED GY GRAN GN | | | | |
| 40.0 | | 90% | NX | 786.4 | |

| | | | |
|---|---|---|---|
| LI | Light | HD | Hard |
| DK | Dark | BI | Biotite |
| GY | Gray | HO | Hornblende |
| BK | Black | GRAN | Granite |
| MOD | Moderately | GN | Gneiss |
| MED | Medium | | |

WATER TABLE

Figure 2-85.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 40.0 | | | | 786.4 | |
| 40.1 | MOD HD LI TO MED GY GRAN GN | 90% | | | 74° Joint at 40.0 Feet |
| 41.6 | SOFT MED GY BROWN GRAN GN | | | | |
| 42.8 | MOD HD LI TO MED GY GRAN GN | | | | |
| | HD LI GY GRAN GN | 94 % | NX | 781.4 | |
| 46.0 | | | | | |
| 46.8 | MOD HD LI TO MED GY GRAN GN | | | | |
| 48.5 | MOD HD DK GY BK BI HO GN | | | | |
| 48.9 | HD LI GY GRAN GN | | | | |
| 49.5 | MOD HD DK GY BK BI HO GN | 100% | | 776.4 | |
| | HD LI GY GRAN GN | | | | |
| | | | | 771.4 | |
| | | 100% | | | |
| 59.4 | | | | 766.4 | |
| | HD DK GY BK BI HO GN | | | | |
| 62.5 | | | | | |
| 63.2 | HD LI GY GRAN GN | | | | |
| 63.8 | HD DK GY BI HO GN | | | | |
| 64.7 | HD LI GY GRAN GN | | BX | | |
| 64.9 | HD DK GY BI HO GN | 100 | % | | |
| 65.5 | HD LI TO DK GY ALTERNATING THIN | *1 | | 761.4 | |
| 65.7 | HD DK GY BI HO GN | | | | |
| 65.9 | HD LI GY GRAN GN | | | | |
| 66.1 | HD DK GY BI HO GN | | | | |
| 66.7 | HD LI TO DK GY ALTERNATING | *1 | | | |
| 66.9 | HD DK GY BI HO GN | | | | |
| | HD LI GY GRAN GN | | | 756.4 | |
| | | | | 751.4 | |
| 78.7 | | 100% | | | |
| 78.9 | HD DK GY BI HO GN | | | | |
| 80.0 | HD LI GY GRAN GN | | | 746.4 | |

*1 LAYERS OF BI HO GN AND GRAN GN

WATER TABLE

Figure 2-86.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 80.0 | | | | 746.4 | |
| | HD LI GY GRAN GN | 100 | | | |
| | | 100 | | | |
| 85.0 | | | | 741.4 | |
| | HD WHITE QUARTZ PEGMATITE | | BX | | |
| 88.0 | | | | | |
| 88.6 | HD DK GY BI HO GN | | | | |
| | HD LI GY GRAN GN | | | 736.4 | |
| 92.4 | | 96% | | | |
| 92.5 | HD DK GY BI HO GN | | | | |
| 93.0 | HD LI GY GRAN GN | | | | |
| 94.2 | HD DK GY BI HO GN | | | | |
| | HD LI GY GRAN GN | | | 731.4 | |
| 96.7 | | | | 729.7 | |
| | CORING TERMINATED | | | | |
| | | | | 726.4 | |

Not to scale

WATER TABLE

Figure 2-87.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 0 | | | | 763.85 | |
| | LOOSE RED BROWN SILTY MICACEOUS FINE TO MEDIUM SAND | | | 758.85 | N = 9 |
| 8.0 | STIFF TO VERY STIFF GRAY BROWN MICACEOUS FINE TO MEDIUM SANDY SILT | | | 753.85 | N = 12 |
| | | | | 748.85 | N = 14 |
| | | | | 743.85 | N = 26 |
| 23.0 | DENSE BROWN MICACEOUS SILTY FINE SAND | | | 738.85 | N = 31 |
| | | | | 733.85 | N = 30 |
| 32.0 | WEATHERED ROCK FRAGMENTS | | | | |
| 35.0 | | | | 728.85 | |
| | VERY SOFT DK BROWN GRAN GN AND BI HD GN | 49% | NX | | 77° Joint at 35.5 Feet<br>47° Joint at 36.6 Feet |
| 40.0 | | | | 723.85 | |

| | | | |
|---|---|---|---|
| LI | Light | HD | Hard |
| DK | Dark | BI | Biotite |
| GY | Gray | HD | Hornblende |
| BK | Black | GRAN | Granite |
| MOD | Moderately | GN | Gneiss |
| MED | Medium | | |

End of Boring

WATER TABLE

Figure 2-88.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 40.0 | | | | 723.85 | |
| | VERY SOFT DK BROWN GY GRAN GN AND BI HO GN | | | | |
| 42.5 | | 49% | | | 46° Joint at 43.0 Feet (Recemented) |
| 43.6 | SOFT TO MEDIUM GY GRAN GN | | | | |
| | MOD HD LI GY GRAN GN | | | 718.85 | |
| | | 80% | | | |
| | | | | 713.85 | |
| | | | | | 58° Joint at 55.3 Feet |
| 55.0 | | | | 708.85 | 59° Joint at 55.9 Feet |
| 56.4 | MOD HD DK GY BK BI HO GN | | | | |
| 57.6 | HD LI GY GRAN GN | 99% | NX | | 64° Joint at 57.Feet |
| 58.8 | HD DK GY BK BI HO GN | | | | |
| | HD LI GY GRAN GN | | | 703.85 | |
| 61.4 | | | | | |
| 62.1 | HD DK GY BI HO GN | | | | |
| | HD LI GY GRAN GN | | | | |
| 64.5 | | | | 698.85 | |
| | HD DK GY BI HO GN | | | | |
| 67.3 | | | | | 60° Joint at 67.8 Feet |
| 68.4 | HD LI GY QUARTZ PEGMATITE | 92% | | | |
| | HD LI GY GRAN GN | | | 693.85 | |
| 73.0 | | | | | |
| | HD LI GY GRAN GN AND BI HO GN | | | 688.85 | |
| 75.0 | CORING TERMINATED | | | | |
| 80.0 | | | | | |

WATER TABLE

Figure 2-89.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 0 | | | | 780.0 | |
| | RED BROWN CLAYEY MICACEOUS SLIGHTLY SANDY SILT | | | | |
| 4.0 | | | | 775.4 | |
| | YELLOW BROWN MICACEOUS SANDY SILT | | | | |
| 10.0 | | | | 770.7 | |
| | GRAY BROWN MICACEOUS SILTY SAND | | | | |
| | | | | 766.1 | |
| | | | | 761.5 | |
| | | | | 756.9 | |
| 27.8 | | 100 | NX | | |
| | HD LI GY GRAN GN | | | 752.2 | |
| 32.2 | | | | | |
| 32.6 | HD DK GY BK BI HO GN | | | | |
| | HD LI GY GRAN GN | 95% | BX | 747.6 | |
| 38.1 | | | | | |
| 39.1 | HD DK GY BK BI HO GN | | | | |
| 39.2 | HD WHITE QUARTZ PEGMATITE | 94% | | 740.8 | 74° Joint at 39.4 Feet |

| | | | |
|---|---|---|---|
| LI | Light | HD | Hard |
| DK | Dark | BI | Biotite |
| GY | Gray | HO | Hornblende |
| BK | Black | GRAN | Granite |
| MDD | Moderately | GN | Gneiss |
| MED | Medium | | |

Hole inclined 22° from the vertical

Figure 2-90.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 39.2 | | | | 740.8 | |
| 39.7 | HD DK GY BK BI HO GN | | | | |
| 39.9 | SOFT DK GY BK BI HO GN | | | | |
| 40.3 | HD DK GY BK BI HO GN | | | 742.9 | |
| 42.5 | HD LI GY GRAN GN | | | | |
| 43.0 | HD DK GY BK BI HO GN | | | | |
| 46.0 | HD LI GY GRAN GN | 94% | | | |
| 46.4 | HD DK GY BK BI HO GN | | | 738.3 | |
| 46.5 | HD WHITE QUARTZ PEGMATITE | | | | |
| 46.8 | HD DK GY BK BI HO GN | | | | |
| | HD LI GY GRAN GN | | | | |
| | | | | 733.7 | |
| | | | BX | | |
| | | 100 | % | 729.1 | |
| | | | | 724.4 | |
| | | 100% | | | |
| | | | | 719.8 | |
| 71.9 | | | | | |
| 72.2 | HD DK GY BK BI HO GN | | | 715.2 | |
| 72.8 | HD LI GY GRAN GN | | | | |
| 73.2 | HD DK GY BK BI HO GN | | | | |
| | HD LI GY GRAN GN | 98% | | | |
| | | | | 710.5 | |
| | | | | 702.3 | |
| 77.7 | CORING TERMINATED | | | | |
| | | | | 705.9 | Not to scale |

Figure 2-91.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 0 | | | | 793.0 | |
| | HARD BROWN GRAY MICACEOUS SLIGHTLY CLAYEY SANDY SILT | | | 788.0 | N = 50 + (Refusal) |
| | | | | 783.0 | N = 50 + (Refusal) |
| 11.7 | SOFT MED GY GRAN GN | 75% | | | |
| 13.8 | | | | | |
| 14.4 | MOD DK LI TO MED GY GRAN GN | | | 778.0 | |
| 15.6 | SOFT DK GY BK BI HO GN | | NX | | |
| | MOD HD LI TO MED GY GRAN GN | 78% | | | |
| | | | | 773.0 | |
| 21.8 | | | | | |
| 23.0 | MOD HD DK GY BK BI HO GN | | | | |
| | MOD HD LI GY GRAN GN | 89% | | 768.0 | |
| | | | BX | | 69° Joint at 28.0 Feet |
| | | | | | Vertical Joint at 29.5 Ft |
| | | | | 763.0 | |
| 31.9 | | | | | |
| 32.4 | VERY SOFT BK BK BI HO GN | | | | |
| 34.5 | HD LI GY GRAN GN | | | | |
| 34.7 | SOFT MED GY GRAN GN | | | 758.0 | |
| 35.0 | HD LI GY GRAN GN | | | | |
| 36.2 | HD DK GY BK BI HO GN | 95% | | | |
| 37.3 | HD LI GY GRAN GN | | | | |
| 38.0 | HD MED GY BI HO GN | | | | |
| 38.8 | HD LI GY GRAN GN | | | 754.2 | |

| | | | |
|---|---|---|---|
| LI | Light | HD | Hard |
| DK | Dark | BI | Biotite |
| GY | Gray | HO | Hornblende |
| BK | Black | GRAN | Granite |
| MOD | Moderately | GN | Gneiss |
| MED | Medium | | |

11.0 feet of 2" plastic pipe left in hole

Not to scale

WATER TABLE

Figure 2-92.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 38.8 | | | | 754.2 | |
| 39.5 | HD MED TO DK GY BI HO GN | 95% | | 753.0 | |
| 43.8 | HD LI GY GRAN GN | 100 | % | | |
| 44.0 | MOD HD DK GY BK BI HO GN | | | | |
| 47.5 | HD LI GY GRAN GN | | | 748.0 | |
| 47.7 | HD DK GY BK BI HO GN | | | | |
| 48.1 | HD WHITE QUARTZ PEGMATITE | | BX | | |
| 48.6 | HD LI GY GRAN GN | 100 | % | | |
| 49.0 | HD DK GY BK BI HO GN | | | | |
| 49.3 | HD LI GY GRAN GN | | | | |
| 49.7 | HD MOD DK GY BI HO GN | | | 743.0 | |
| 52.3 | HD LI GY GRAN GN | | | | |
| 53.5 | HD WHITE QUARTZ PEGMATITE | | | | |
| 54.0 | HD LI DK GY ALTERNATING LAYERS | *1 | | 739.0 | |
| | CORING TERMINATED | | | 738.0 | |

*1 OF BI HO GN AND GRAN GN

Not to scale

WATER TABLE

Figure 2-93.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 0 | | | | 793.0 | Bag Sample 0.0 - 5' |
| | STIFF YELLOW BROWN MICACEOUS SLIGHTLY CLAYEY SANDY SILT | | | 788.0 | N = 9<br>Bag Sample 5 - 10'<br>Undistrubed Sample 5.0 - 6.0 feet<br>Undisturbed Sample 10.0-11.0 feet |
| 9.0 | | | | 783.0 | N = 19<br>Bag Sample 10-15 feet |
| | FIRM LIGHT GRAY MICACEOUS SILTY SAND | | | | |
| 13.0 | | | | 778.0 | N = 9 Bag Sample 15-20'<br>Undisturbed Sample 15.0-16.0 feet |
| | STIFF YELLOW BROWN MICACEOUS SLIGHTLY CLAYEY SANDY SILT | | | | |
| 17.5 | | | | 773.0 | N = 13 Bag Sample 20-25'<br>Undisturbed Sample 20.0-21.0 Feet |
| | FIRM VERY DENSE MICACEOUS GRAY BROWN SANDY SILT | | | 768.0 | N = 47 Bag Sample 25-30'<br>Undistrubed Sample 25.0-26.0 Feet |
| | | | | 763.0 | N = 67<br>Undisturbed Sample 30.0-26.0 Feet<br>Undisturbed Sample 33.5 - 34.5 feet |
| | | | | 758.0 | Refusal |
| 40.0 | | | | 753.0 | Refusal |

| | | | |
|---|---|---|---|
| LI | Light | HD | Hard |
| DK | Dark | BI | Biotite |
| GY | Gray | HO | Hornblende |
| BK | Black | GRAN | Granite |
| MOD | Moderately | GN | Gneiss |
| MED | Medium | | |

End of Boring

Water Table

Figure 2-94.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 40.0 | | | | 753.0 | |
| 41.0 | FIRM-VERY DENSE GRAY BROWN SANDY | *1 | | | |
| | MOD HD LI TO MED GY GRAN GN | 75 | NX | | |
| 44.7 | | 87 | | | |
| 44.8 | MOD HD DK GY BK BI HO GN | | | 748.0 | |
| | HD LI GY GRAN GN | | | | |
| | | | | 743.0 | |
| 51.5 | HD LI GY WHITE QUARTZ PEGMATITE | 100 | BX | | 72° Joint at 53.0 Feet |
| 54.5 | | | | 738.0 | |
| | HD LI GY GRAN GN | | | | |
| 56.7 | | | | | |
| 57.8 | HD DK GY BK BI HO GN | | | | |
| 58.1 | HD LI GY GRAN GN | | | | |
| 58.4 | HD DK GY BK BI HO GN | | | | |
| 59.0 | HD LI GY GRAN GN | | | | |
| 59.5 | HD DK GY BK BI HO GN | 100 | | 733.0 | |
| 61.9 | HD LI GY GRAN GN | | | | |
| 62.1 | HD DK GY BK BI HO GN | | | | |
| 63.4 | HD LI GY GRAN GN | | | | |
| 63.8 | HD DK GY BK BI HO GN | | | 729.2 | |
| | CORING TERMINATED | | | 728.0 | |

*1 SILT

Not to scale

End of Boring
WATER TABLE

Figure 2-95.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 0 | | | | 766.1 | |
| 0.3 | TOPSOIL | | | | |
| | VERY STIFF BROWN CLAYEY MICACEOUS FINE TO MEDIUM SANDY SILT | | | | N = 22 |
| | | | | 761.1 | |
| 7.0 | | | | | |
| | STIFF BROWN FINE TO MEDIUM SANDY MICACEOUS SILT | | | | N = 9" |
| | | | | 756.1 | |
| 12.0 | | | | | |
| | FIRM TO HARD BROWN GRAY MICACEOUS FINE TO MEDIUM SANDY SILT | | | | N = 6" |
| | | | | 751.1 | |
| | | | | | N = 7" |
| | | | | 746.1 | |
| | | | | | N = 40/3" |
| | | | | 741.1 | |
| 27.0 | | | | | |
| | VERY STIFF BROWN GRAY WHITE MICACEOUS FINE TO COARSE SANDY SILT | | | | N = 19 |
| | | | | 736.1 | |
| 32.0 | | | | | |
| | VERY DENSE BROWN GRAY WHITE MICACEOUS SILTY FINE TO COARSE SAND | | | | N = 20/0" |
| | | | | 731.1 | |
| | | | | | N = 40/3" |
| 40.0 | | | | 726.1 | |

| | | | |
|---|---|---|---|
| LI | Light | HD | Hard |
| DK | Dark | BI | Biotite |
| GY | Gray | HO | Hornblende |
| BK | Black | GRAN | Granite |
| MOD | Moderately | GN | Gneiss |
| MED | Medium | | |

36.0 Feet of 2" plastic pipe left in hole

WATER TABLE

Figure 2-96.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 40.0 | | | | 726.1 | |
| | VERY DENSE BROWN GRAY WHITE MICACEOUS SILTY FINE TO COARSE SAND | | | | N = Refusal |
| | | | | 721.1 | |
| 49.5 | | | | | |
| 50.8 | MOD HD LI GY GRAN GN | | | 716.1 | |
| 51.0 | MOD HD DK GY BI HO GN | | | | 50° Joint at 50.4 Feet |
| 51.6 | MOD HD LI GY GRAN GN | | | | |
| | MOD HD DK GY BI HO GN | 100 | | | |
| 55.6 | | | | 711.1 | |
| | HD LI GY GRAN GN | | | | |
| 57.2 | | | | | |
| 57.6 | MOD HD DK GY BI HO GN | 100 | NX | | |
| 58.9 | HD LI GY GRAN GN | | | | |
| 59.2 | MOD HD DK GY BI HO GN | | | | |
| 59.6 | HD LI GY GRAN GN | | | | |
| 60.0 | HD DK GY BI HO GN | | | 706.1 | |
| 61.1 | HD LI GY GRAN GN | | | | |
| 61.4 | MOD HD DK GY BK BI HO GN | 100 | | | |
| 62.2 | HD LI GY GRAN GN | | | | |
| 62.5 | HD DK GY BI HO GN | | | 701.1 | |
| 67.3 | HD LI GY GRAN GN | | | | |
| 67.5 | MOD HD DK GY BI HO GN | | | | |
| 67.6 | HD LI GY GRAN GN | 100 | | | |
| 67.8 | MOD HD DK GY BI HO GN | | | | |
| | | | | 696.6 | |
| 69.5 | HD LI GY GRAN GN | | | | |
| | | | | 696.1 | |
| | CORING TERMINATED | | | | |

Not to scale

Page 2 of 2

WATER TABLE

Figure 2-97.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 0 | | | | 767.9 | Bag Sample 0 - 25' |
| | STIFF RED BROWN MICACEOUS CLAYEY SANDY SILT | | | | Undisturbed Sample 3.5-4.5 Feet |
| | | | | 762.9 | N = 11 |
| 7.5 | STIFF YELLOW BROWN MICACEOUS SANDY SILT | | | | N = 8 |
| | | | | 757.9 | Undisturbed Sample 13.5-14.5 Feet |
| | | | | 752.9 | N = 12 |
| | | | | | N = 9 |
| | | | | 747.9 | Undisturbed Sample 23.5-24.5 Feet |
| 22.0 | FIRM GRAY BROWN SILTY MICACEOUS SAND WITH SOME QUARTZ FRAGMENTS | | | 742.9 | N = 16 |
| | | | | | N = 24 |
| | | | | 737.9 | Undisturbed Sample 33.5 - 34.5 Feet. |
| 31.5 | VERY STIFF BROWN SLIGHTLY SANDY MICACEOUS SILT | | | 732.9 | N = 17 |
| | | | | | N=23 |
| 40.0 | | | | 727.9 | |

85.0 Feet of 2" Plastic pipe left in hole

| | | | |
|---|---|---|---|
| LI | Light | HD | Hard |
| DK | Dark | BI | Biotite |
| GY | Gray | HO | Hornblende |
| BK | Black | GRAN | Granite |
| MOD | Moderately | GN | Gneiss |
| MED | Medium | | |

Figure 2-98.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 40.0 | | | | 727.9 | |
| | VERY DENSE GRAY BROWN WHITE MICACEOUS SILTY SAND WITH SOME QUARTZ FRAGMENTS | | | | N=50/4" |
| | | | | 722.9 | |
| | | | | 717.9 | |
| | | | | 712.9 | |
| | | | | 707.9 | |
| | | | | 702.9 | |
| 69.0 | | | | | |
| | MOD HD LIGHT TO MED GY GRAN GN | 100 | | 697.9 | 85° Joint at 72.7 Feet |
| 74.9 | | | | | |
| 75.4 | MOD HD DK GY BK BI HO GN | | NX | 692.9 | |
| 75.6 | MOD HD MED GY GRAN GN | | | | 71° Joint at 76.0 Feet |
| 76.5 | MOD HD DK GY BK BI HO GN | 100 | | | 77° Joint at 77.0 Feet |
| 77.7 | MOD HD MED GY GRAN GN | | | | |
| 78.0 | SOFT DK GY BK BI HO GN | | | | |
| 78.2 | MOD HD LI GY GRAN GN | | | 687.9 | |

WATER TABLE

Figure 2-99.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 78.2 | | | | 687.9 | |
| 79.1 | MOD HD DK GY BK BI HO GN | 100 | | | |
| 79.6 | MOD HD MED GY GRAN GN | | | | |
| 79.7 | SOFT DK GY BK BI HO GN | | | | |
| 80.0 | MOD HD LI GY GRAN GN | | | 682.9 | |
| 80.2 | SOFT DK BY BK BI HO GN | | | | |
| 81.3 | MOD HD LI GY GRAN GN | 100 | | | |
| 82.1 | MOD HD MED TO DK GY BI HO GN | | NX | | |
| 84.0 | MOD HD LI GY GRAN GN | | | | |
| 84.5 | HD LI GY GRAN GN | | | | |
| 84.7 | MOD HD DK GY BK BI HO GN | | | | |
| 86.0 | HD LI GY GRAN GN | | | | |
| 86.3 | SOFT DK GY BK BI HO GN | 100 | | | |
| 87.3 | HD LI GY GRAN GN | | | | |
| 89.0 | MOD HD LI TO DK GY ALTERNATE *1 | | | 678.9 | |
| | CORING TERMINATED | | | 672.9 | |

*1 LAYERS OF BI HD GN AND GRAN GN

Not to scale

WATER TABLE

Figure 2-100.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 0 | | | | 816.0 | |
| | FIRM BROWN CLAYEY SILTY MICACEOUS SAND | | | | |
| | | | | 811.0 | N = 15 Undisturbed Sample 4.0 - 5.0 Feet |
| | | | | 806.0 | N = 11 |
| 12.0 | | | | | |
| | STIFF TO VERY STIFF BROWN GRAY SANDY MICACEOUS SILT | | | 801.0 | N = 12 |
| | | | | 796.0 | N = 13 |
| | | | | 791.0 | N = 18 |
| 28.0 | | | | | |
| | | | | 786.0 | N = 19 |
| | DENSE TO VERY DENSE GRAY WHITE SILTY MICACEOUS SAND WITH SOME QUARTZ FRAGMENTS | | | | |
| | | | | 781.0 | N = 23 |
| 40.0 | | | | 776.0 | N=28 |

58.0 Feet of 2" plastic pipe left in hole

| | | | |
|---|---|---|---|
| LI | Light | HD | Hard |
| DK | Dark | BI | Biotite |
| GY | Gray | HO | Hornblende |
| BK | Black | GRAN | Granite |
| MOD | Moderately | GN | Gneiss |
| MED | Medium | | |

End of Boring

WATER TABLE

Figure 2-101.
Core Boring Record

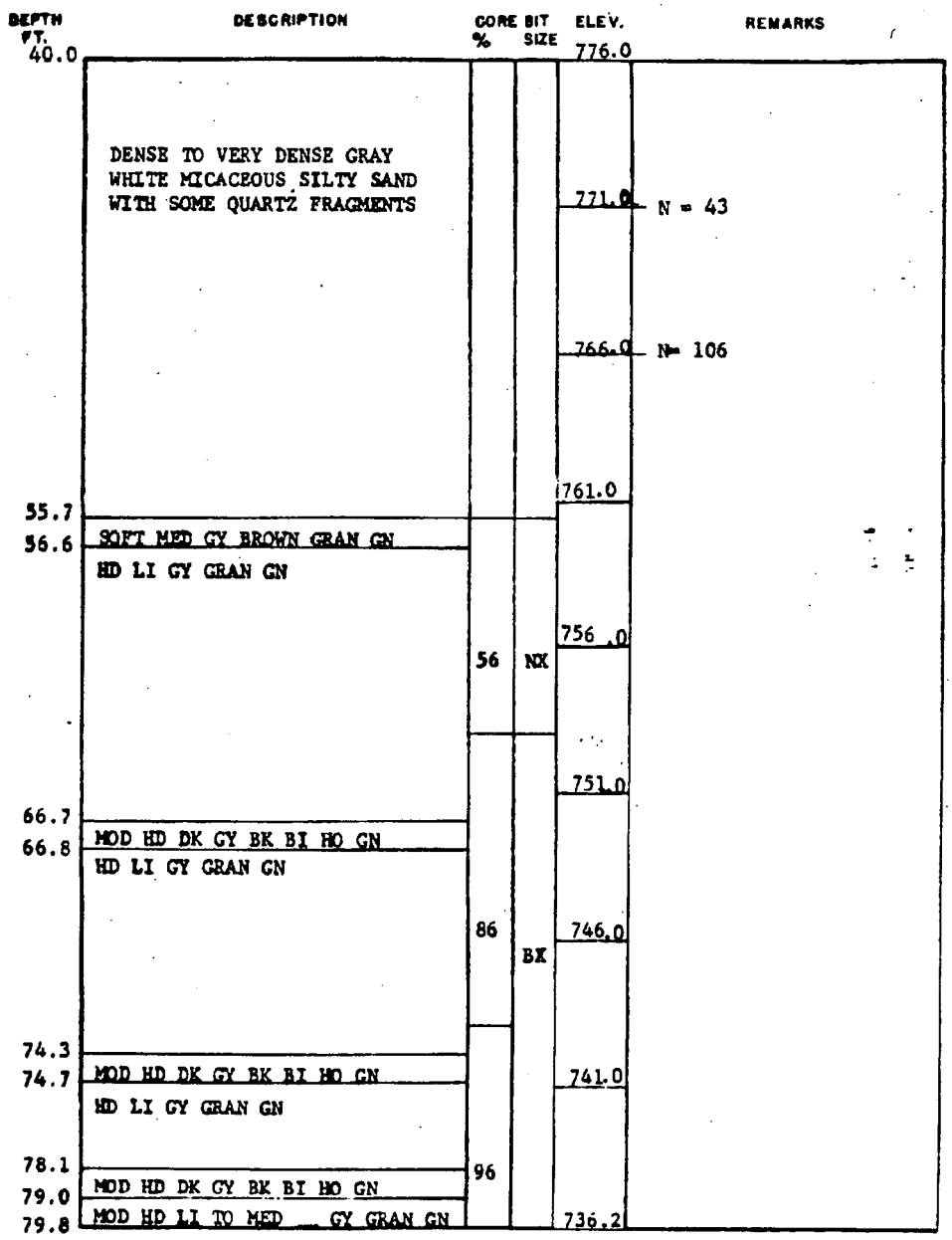



Figure 2-102.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 79.8 | | | | 736.2 | |
| 80.1 | MOD HD DK GY BK WI HO GN | | | 736.0 | |
| | HD LI TO MED GY GRAN GN | 96 | BX | 732.8 | |
| 83.2 | CORING TERMINATED | | | 731.0 | |

WATER TABLE

Figure 2-103.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 0 | | | | 745.6 | |
| 0.3 | TOPSOIL AND GRASS | | | | Bag Sample 0 - 8.0 Feet |
| | VERY STIFF RED BROWN SILTY MICACEOUS CLAY | | | | Undisturbed Sample 3.5 - 4.5 Feet |
| | | | | 740.6 | N = 18 |
| | | | | | Undisturbed Sample 8.5-9.5 Feet |
| 8.0 | STIFF TO VERY STIFF BROWN GRAY WHITE MICACEOUS SANDY SILT | | | | Bag Sample 8.0 -25.0 Feet |
| | | | | 735.6 | N = 9 |
| | | | | | Undisturbed Sample 13.5-14.5 Feet |
| | | | | 730.6 | N = 9 |
| | | | | | Undisturbed Sample 18.5 -19.5 Feet |
| | | | | 725.6 | N= 11 |
| | | | | | N = 14 |
| | | | | 720.6 | |
| | | | | | N = 21 |
| | | | | 715.6 | |
| | | | | | N = 25 |
| | | | | 710.6 | |
| 36.0 | VERY STIFF TO HARD GRAY BROWN BLACK MICACEOUS SANDY SILT | | | | |
| | | | | | N = 25 |
| 40.0 | | | | 705.6 | |

| | | | |
|---|---|---|---|
| LI | Light | HD | Hard |
| DK | Dark | BI | Biotite |
| GY | Gray | HO | Hornblende |
| BK | Black | GRAN | Granite |
| MOD | Moderately | GN | Gneiss |
| MED | Medium | | |

WATER TABLE

Figure 2-104.
Core Boring Record

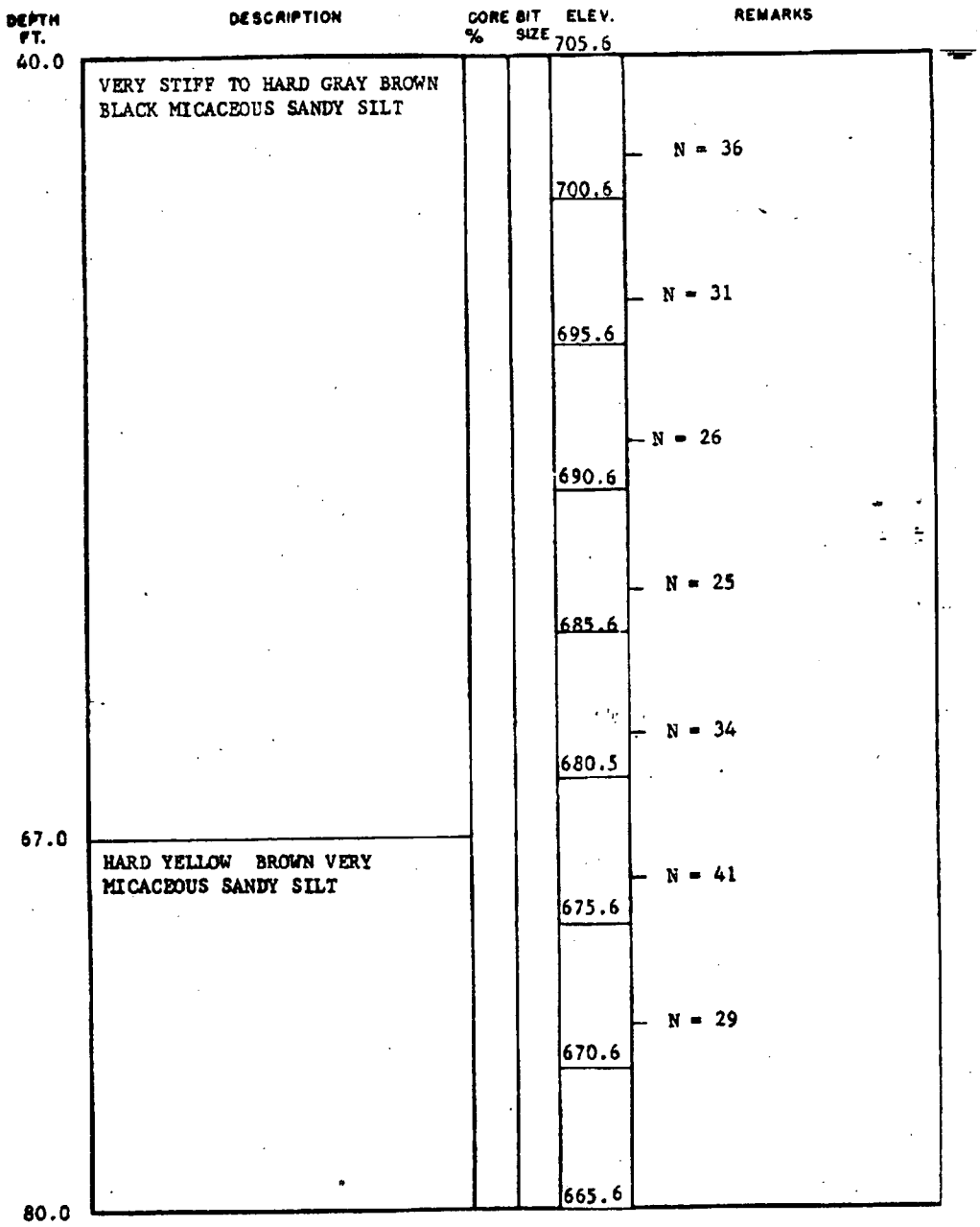



Figure 2-105.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 80.0 | | | | 665.6 | |
| | VERY SOFT ROCK (NO RECOVERY) | 0 | | | |
| 84.0 | | | | 660.6 | |
| | SOFT MED GY AND BR GRAN GN | 60 | BX | | 74° Joint at 86.0 Feet |
| 90.0 | | | | 655.6 | |
| | HAWTHORNE BIT - VERY DENSE GY SILTY SAND | | | 650.6 | |
| | | | | 645.6 | |
| 104.0 | | | | | |
| 104.5 | QUARTZ VEIN | | | 640.6 | |
| | VERY SOFT GY GRAN GN | | | | |
| 108.0 | | | | | |
| | MOD HD MED GY AND BR GRAN GN | 66 | BX | 635.6 | |
| 115.7 | | | | 630.6 | |
| 115.9 | MOD HD DK GY BK BI HO GN | | | | 75° Joint at 115.7 Feet |
| 116.1 | MOD HD LI TO MED GY GRAN GN | | | | |
| 118.0 | SOFT DK GY BK BI HO GN | | | | |
| | MOD HD LI TO MED GY GRAN GN | 80 | | | |
| 120.0 | | | | 625.6 | |

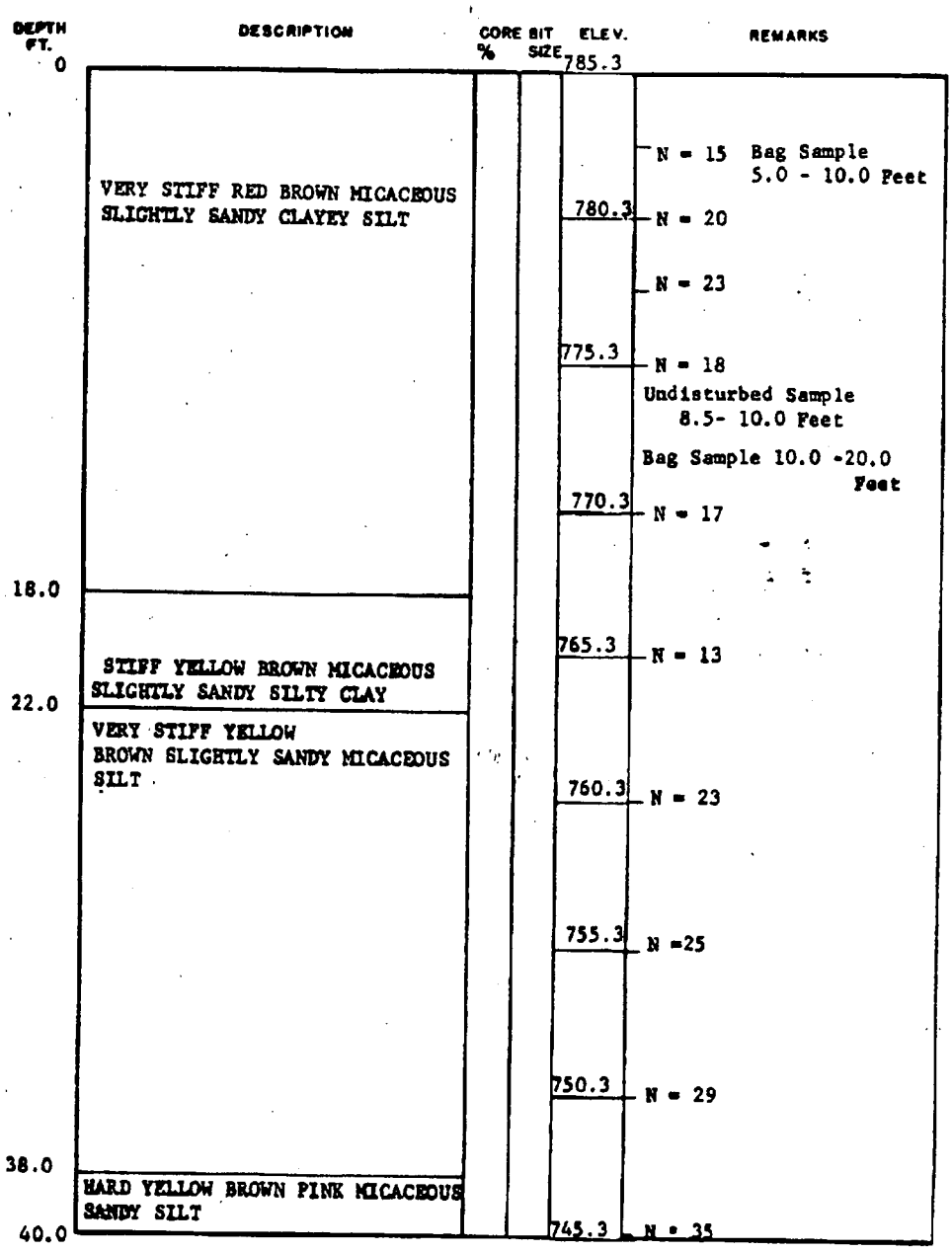
WATER TABLE

Figure 2-106.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 120.0 | | | | 625.6 | |
| | MOD HD LI TO MED GY GRAN GN | | | | |
| 122.2 | | | | | |
| 123.1 | SOFT DK GY BK BI HO GN | | | | |
| | MOD HD LI TO MED GY GRAN GN | 80 | | | |
| | | | | 620.6 | 78° Joint at 123.0 Feet |
| 125.3 | | | | | |
| | VERY SOFT MED GY GRAN GN | | BX | | 57° Joint at 124.0 Feet |
| | | 38 | | | |
| 130.0 | | | | 615.6 | |
| | CORING TERMINATED | | | | |

WATER TABLE

Figure 2-107.
Core Boring Record




| | | | |
|---|---|---|---|
| LI | Light | HD | Hard |
| DK | Dark | BI | Biotite |
| GY | Gray | HO | Hornblende |
| BK | BLACK | GRAN | Granite |
| MOD | Moderately | GN | Gneiss |
| MED | Medium | | |

WATER TABLE

Figure 2-108.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 40.0 | | | | 745.3 | |
| | HARD YELLOW BROWN PINK MICACEOUS SANDY SILT | | | | |
| 41.5 | HARD GRAY BROWN MICACEOUS SANDY SILT | | | | |
| | | | | 740.3 | N= 39 |
| | | | | 735.3 | N = 45 |
| | | | | 730.3 | N = 44 |
| | | | | 725.3 | N = 42 |
| | | | | 720.3 | N = 44 |
| 67.5 | | | | | |
| | DENSE YELLOW BROWN AND GRAY MICACEOUS SILTY SAND WITH SOME QUARTZ FRAGMENTS | | | 715.3 | N = 41 |
| | | | | 710.3 | N = 35 |
| 80.0 | | | | 705.3 | N = 36 |

WATER TABLE

Figure 2-109.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 80.0 | | | | 705.3 | |
| | DENSE YELLOW BROWN AND GRAY MICACEOUS SILTY SAND WITH SOME QUARTZ FRAGMENTS | | | 700.3 | N = 40 |
| | | | | 695.3 | |
| | | | | 690.3 | |
| | | | | 685.3 | |
| 101.0 | DENSE SILTY SAND WITH THIN LAYERS OF QUARTZ | | | 680.3 | |
| 105.3 | DENSE YELLOW BROWN AND GRAY SILTY SAND WITH SOME QUARTZ FRAGMENTS | | | 675.3 | |
| 110.0 | HD LI GY GRAN GN | | | | |
| 112.3 | MOD HD MED GY BK BI HO GN | | | | 50° Joint at 112.0 Feet |
| 112.8 | HD LI GY GRAN GN | | | 670.3 | |
| 115.5 | MOD HD DK GY BK BI HO GN | 100 | BX | | |
| 118.3 | LD LI GY GRAN GN | | | | |
| 120.0 | | | | 665.3 | |

WATER TABLE

Figure 2-110.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 120.0 | | | | 665.3 | |
| | HD LI GY GRAN GN | | | | |
| 125.0 | | | | 660.3 | |
| | MOD HD LI GY GRAN GN | 100 | BX | | |
| 130.0 | | | | 655.3 | |
| | CORING TERMINATED | | | | |

Figure 2-111.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 0 | | | | 742.6 | |
| 0.3 | TOPSOIL AND GRASS | | | | Bag Sample 0 - 4.0 Feet |
| | STIFF BROWN MICACEOUS SANDY SILT | | | | Undisturbed Sample 3.5 -4.5 feet |
| | | | | 737.6 | N = 8 |
| | | | | | Bag Sample 4.0 -25.0 Feet |
| 7.0 | FIRM TO VERY STIFF BROWN GRAY WHITE MICACEOUS SANDY SILT | | | | |
| | | | | 732.6 | N = 6 Undisturbed Sample 8.5-9.5 Feet |
| | | | | 727.6 | N = 12 Undisturbed Sample 13.5-14.5 Feet |
| | | | | | Undisturbed Sample 18.5-19.5 Feet |
| | | | | 722.6 | N = 13 |
| | | | | | Undisturbed Sample 23.5-24.5 Feet |
| | | | | 717.6 | N = 15 |
| | | | | | N = 17 |
| | | | | 717.6 | |
| | | | | | N = 16 |
| | | | | 707.6 | |
| 37.0 | DENSE BROWN GRAY WHITE MICACEOUS SILTY FINE TO MEDIUM SAND | | | | N = 31 |
| 40.0 | | | | 702.6 | |

| | | | |
|---|---|---|---|
| LI | Light | HD | Hard |
| DK | Dark | BI | Biotite |
| GY | Gray | HO | Hornblende |
| BK | Black | GRAN | Granite |
| MOD | Moderately | GN | Gneiss |
| MED | Medium | | |

36.0 Feet of 2" Plastic Pipe left in hole

WATER TABLE

Figure 2-112.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 40.0 | | | | 702.6 | |
| 41.0 | DENSE BROWN GRAY WHITE HARD BROWN GRAY VERY MICACEOUS SANDY SILT | *1 | | | |
| | | | | | N = 70 |
| | | | | 697.6 | |
| | | | | | N = 40/3" (refusal) |
| | | | | 692.6 | |
| | | | | | N = Refusal |
| | | | | 687.6 | |
| | | | | | N = Refusal |
| | | | | 682.6 | |
| | | | | | N = Refusal |
| | | | | 677.6 | |
| | | | | | N= Refusal |
| | | | | 672.6 | |
| | | | | | N = Refusal |
| | | | | 667.6 | |
| 77.0 | MOD HD LI TO MED GY GRAN GN | | | | |
| 79.6 | | 100 | NX | | ½ inch shear displacement |
| 80.0 | MOD HD DK GY BK BI HO GN | | | 662.6 | |

*1 MICACEOUS SILTY FINE TO MEDIUM SAND

WATER TABLE

Figure 2-113.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 80.0 | | | | 662.6 | |
| | MOD HD LI TO MED GY GRAN GN | 100 | | | 82° Joint at 80.0 Feet |
| | | | | | 85° Joint at 82.3 Feet |
| 83.7 | | | | | |
| 83.9 | MOD HD DK GY BK BI HO GN | | | | |
| | MOD HD LI GY AND BR GRAN GN | 100 | | 657.6 | |
| 88.6 | | | | | |
| 88.8 | MOD HD DK GY BK BI HO GN | | | | |
| 89.4 | MOD HD LI TO MED GY &BR GRAN GN | 100 | | 652.6 | |
| 90.1 | MOD HD DK GY BK BI HO GN | | | | |
| 92.0 | MOD HD LI TO MED GY GRAN GN | | | | |
| 92.2 | SOFT DK GY BR BI HO GN | | | | |
| | MOD HD LI GY GRAN GN | 100 | | 647.6 | |
| 97.0 | | | | | |
| 97.5 | MOD HD DK GY BK BI HO GN | | | | |
| 99.8 | MOD HD LI GY GRAN GN | | | | 74° Joint at 97.0 Feet |
| 100.3 | MOD HD DK GY BK BI HO GN | | | 642.6 | |
| | MOD HD LI GY GRAN GN | 100 | NX | | |
| 104.6 | | | | | |
| 105.7 | MOD HD LI TO MED GY BI HO GN | | | 637.6 | |
| 106.1 | MOD HD LI GY GRAN GN | | | | 73° Joint at 104.7 feet |
| 106.9 | MOD HD DK GY BK BI HO GN | 95 | | | |
| | MOD HD LI TO MED GY GRAN GN | | | | |
| 109.6 | | 93 | | | |
| 110.0 | SOFT DK GY BK BI HO GN | | | 632.6 | |
| 110.2 | MOD HD LI GY GRAN GN | | | | |
| 110.4 | MOD HD DK GY BK BI HO GN | | | 630.6 | |
| 112.0 | MOD HD LI GY GRAN GN | | | | |
| | CORING TERMINATED | | | 627.6 | |

Figure 2-114.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 0 | | | | 728.9 | |
| 0.4 | TOPSOIL | | | | |
| | VERY STIFF RED BROWN MICACEOUS FINE SANDY SILTY CLAY | | | 723.9 | N = 21<br>Undisturbed Sample 3.5-4.5 Feet |
| 8.0 | VERY STIFF BROWN MICACEOUS FINE SANDY SILT | | | 718.9 | N = 16<br>Undisturbed Sample 8.5-9.5 Feet |
| 12.0 | LOOSE BROWN GRAY MICACEOUS SILT FINE SAND | | | 713.9 | Undisturbed Sample 13.5-14.5 Feet<br>N = 4 |
| 17.5 | STIFF BROWN GRAY WHITE MICACEOUS FINE TO MEDIUM SANDY SILT | | | 708.9 | Undisturbed Sample 18.5-19.5 Feet<br>N = 10<br>Undisturbed Sample 23.5-24.5 Feet |
| 21.0 | VERY STIFF TO HARD MICACEOUS BROWN GRAY WHITE SILTY FINE TO COARSE SAND | | | 703.9 | N = 17 |
| | | | | 698.9 | N = 45 |
| | | | | 693.9 | N = 60 |
| 36.0 | HARD DARK GRAY BLACK AND WHITE FINE SANDY SILT | | | 688.9 | N = 30/1" (Refusal) |
| 40.0 | | | | | |

| | | | |
|---|---|---|---|
| LI | Light | HD | Hard |
| DK | Dark | BI | Biotite |
| GY | Gray | HO | Hornblende |
| BK | Black | GRAN | Granite |
| MOD | Moderately | GN | Gneiss |
| MED | Medium | | |

WATER TABLE

Figure 2-115.
Core Boring Record

| DEPTH FT. | DESCRIPTION | CORE % | BIT SIZE | ELEV. | REMARKS |
|---|---|---|---|---|---|
| 40.0 | | | | 688.9 | |
| | HARD DARK GRAY BLACK AND WHITE FINE SANDY SILT (DECOMPOSED HORNBLENDE GNEISS) | | | | N = 20/0" (Refusal) |
| | | | | 683.9 | |
| | | | | | N = Refusal |
| | | | | 678.9 | |
| 53.5 | | | | | |
| | MOD HD LI TO MED GY GRAN GN | | | | |
| 55.1 | | | | 673.9 | |
| 55.5 | SOFT MED GY GRAN GN | | | | |
| 56.0 | VERY SOFT DK GY BK BI HO GN | | | | |
| | MOD HD LI TO MED GY GRAN GN | | | | Vertical Joint at 57.5 Feet |
| 59.9 | | | | | |
| 60.2 | VERY SOFT DK GY BK BI HO GN | | | 668.9 | |
| 61.7 | MOD HD LI TO MED GY GRAN GN | 100 | | | |
| 62.1 | MOD HD DK GY BI HO GN | | | | |
| 62.2 | VERY SOFT DK GY BI HO GN | | | | |
| 62.4 | SOFT MED GY GRAN GN | | NX | | |
| 62.7 | SOFT DK GY BI HO GN | | | | |
| 63.0 | SOFT MED GY GRAN GN | | | | |
| 63.5 | MOD HD LI TO MED GY GRAN GN | | | | |
| 63.8 | SOFT DK GY BI HO GN | | | | |
| 64.0 | SOFT LI TO MED GY GRAN GN | | | | |
| 64.2 | MOD HD DR GY BI HO GN | | | 663.9 | |
| 68.8 | MOD HD LI TO MED GY GRAN GN | | | | 61° Joint at 65.0 Feet |
| 70.6 | MOD HD DK GY BK BI HO GN | | | | |
| 71.4 | MOD HD LI TO MED GY GRAN GN | 100 | | | |
| 71.5 | VERY SOFT MED GY GRAN GN | | | | |
| 73.5 | MOD HD LI TO MED GY GRAN GN | | | 655.4 | |
| | CORING TERMINATED | | | 658.9 | |

Not to scale

WATER TABLE

Figure 2-116.
Core Boring Record

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# LIST OF TABLES

# LIST OF FIGURES

# CHAPTER 3. DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

# 3.1 CONFORMANCE WITH NRC GENERAL DESIGN CRITERIA

0 **Note**

0 This section of the FSAR contains information on the design bases and design criteria of this
0 system/structure. Additional information that may assist the reader in understanding the system is
0 contained in the design basis document (DBD) for this system/structure.

The principal design criteria for Oconee 1, 2 and 3 were developed in consideration of the seventy General Design Criteria for Nuclear Power Plant Construction Permits proposed by the AEC in a proposed rule-making published for 10CFR Part 50 in the Federal Register of July 11, 1967. Listed below are the seventy criteria proposed by the AEC, together with the applicant's response indicating the applicant's
4 interpretation of an agreement with the intent of each criterion. The criteria (were) categorized as
4 Category A or Category B. Experience (had) shown that more definitive information (was) needed at the
4 construction permit stage for the items listed in Category A than for those in Category B. In the discussion of each criterion, sections of the report containing more detailed information are referenced.

## 3.1.1 CRITERION 1 - QUALITY STANDARDS (CATEGORY A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes or standards on design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance levels to be used shall be identified. A showing of sufficiency and applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance levels used is required.

Discussion

1. Essential Systems and Components

5 The integrity of systems, structures, and components (SSCs) essential to accident prevention and to
5 mitigation of accident consequences has been included in the reactor design evaluations. These
5 systems, structures, and components are:

5 a. Reactor Coolant System

5 b. Reactor vessel internals

5 c. Reactor Building

5 d. Engineered Safeguards System

5 e. Electric emergency power sources

5 2. Codes and Standards

5 The following table references applicable sections where codes, quality control, and testing are included in
5 the FSAR. The Quality Assurance program is discussed in detail in Chapter 17, "Quality Assurance" on
5 page 17-1.

| Item | Codes | Quality Control | Testing |
|---|---|---|---|
| Reactor Coolant System | Section 5.2.2 | Section 5.2.3.11 | Sections 5.2.3.11; 4.4.4 |
| Reactor Vessel Internals | Section 4.5.1 | Section 4.5.4 | Section 4.5.4 |
| Reactor Building | Sections 3.8.1.2; 3.8.3; 3.8.1.4; 3.8.1.5 | Section 3.8.1.6 | Section 3.8.1.7 |
| Engineered Safeguards System | Sections 6.2.2.2.2; Chapter 6; 6.3.2.4 | Sections Chapter 6; 6.6 | Sections 6.3.4; 6.4.3; 6.5.1.4; 6.2.2.4; 6.2.4 |
| Electric Emergency Power Sources | | | Section 8.3.1.1.6 |

### 3.1.1.1 Oconee QA-1 Program

To meet the requirements of 10CFR50 Appendix B, Oconee has defined its QA-1 program. The QA-1 program shall be applied to the "essential systems and components" listed above. The scope of these systems and components is provided in greater detail below. The QA-1 program shall also be applied to the Reactor Protective System, and shall be applied to any systems and components committed to the NRC as being classified as QA-1 per any correspondence subsequent to the orginal QA-1 licensing basis.

Therefore, the general criteria used to determine if a SSC is QA-1 is divided into two categories:

First category - provides general QA-1 criteria based on the original licensing basis of ONS, and

Second Category - provides general criteria for SSCs that were added to the QA-1 licensing basis after issuance of the original operating licenses for ONS.

**First Category, Original Oconee QA-1 Licensing Basis**

This first category includes the integrity of SSCs essential to prevention and mitigation of the Large Break LOCA coincident with loss of offsite power for the following five SSCs: 1) Reactor Coolant System, 2) Reactor Vessel Internals, 3) Reactor Building, 4) Engineered Safeguards System, and 5) Emergency Electric Power Sources. In addition, 6) Reactor Protective System, another system not addressed in FSAR Section 3.1.1, "Criterion 1 - Quality Standards (Category A)" on page 3-3, was interpreted to be included in the QA-1 scope, even though not listed.

Clarification regarding the six SSCs identified above is provided below.

1. Reactor Coolant System

From a quality assurance perspective, the Reactor Coolant System consists of all connecting piping, valve bodies, pump casings, heat exchangers, or vessels out to and including the first isolation valve. The integrity of the pressure boundary of the connecting piping, valve bodies, pump casings, heat exchangers, or vessels is the function which determines applicability of the quality assurance program.

5 2. Reactor Vessel Internals

5 The Reactor Vessel Internals consist of the plenum assembly and the core support assembly. The core
5 support assembly consists of the core support shield, vent valves, core barrel, lower grid, flow distributor,
5 incore instrument guide tubes, thermal shield, and surveillance holder tubes. The plenum assembly
5 consists of the upper grid plate, the control rod guide assemblies, and a turnaround baffle for the outlet
5 flow.

5 Reactor vessel internals do not include fuel assemblies, control rod assemblies, surveillance specimen
5 assemblies, or incore instrumentation.

5 3. Reactor Building

5 The Reactor Building consists of the following:

5 • The structure which consists of a post-tensioned reinforced concrete cylinder and dome connected to
5 and supported by a massive reinforced concreted foundation slab.

5 • The entire interior surface of the structure (a steel plate liner).

5 • Welded steel penetrations through which numerous mechanical and electrical systems pass into the
5 Reactor Building.

5 • Access openings to the Reactor Building.

5 4. Engineered Safeguards System

5 The Engineered Safeguards System consists of structure, systems, or components necessary to:

5 • Provide emergency cooling to assure structural integrity of the core:

5 High Pressure Injection System

5 Low Pressure Injection System

5 Core Flooding System

5 • Maintain the integrity of the Reactor Building

5 Reactor Building Spray System

5 Reactor Building Cooling System

5 Reactor Building Isolation System (this includes all piping penetration isolation paths)

5 • In addition, support systems necessary to ensure that the above systems can perform their intended
5 safety functions are considered QA-1. These systems are:

5 Low Pressure Service Water portions necessary to supply cooling water to:

5 Reactor Building Cooling Units

5 Decay Heat Removal Coolers

5 Keowee emergency start, load shed, and emergency power switching logic

5 Analog and Digital ES Channels and DC Power to support operability of these channels

5 5. Emergency Electric Power Sources

5 The following power sources and distribution systems are QA-1.

- Keowee Hydroelectric Units 1 and 2, including:

  Keowee Hydro-Generator and Emergency Start Circuits,

  Keowee 600/208/120 VAC Auxiliary Power System, and

  Keowee 125 VDC Power System.

  The following mechanical Keowee SSCs:

  - Governor Oil System
  - Governor Air System
  - Guide Bearing Oil System
  - Turbine Sump System
  - Cooling Water System
- Underground Emergency Power Path, including:

  Underground cable,

  Transformer CT4, and

  Standby Busses.
- Overhead Emergency Power Path, including:

  Keowee Main Step-Up Transformer,

  Associated Tranmission and 230KV Switch\yard Components (e.g., transmission lines and power circuit breakers),

  230 KV Switchyard Yellow Bus,

  230 KV Switchyard 125 VDC Power System, and

  Unit Start-up Transformers (CT1, CT2, and CT3).
- Unit Main Feeder Busses
- 4160 VAC Safety Auxiliary Power System
- 600/208 VAC Safety Auxiliary Power System
- 120 VAC Vital I&C Power System
- 125 VDC Vital I&C Power System

6. <u>Reactor Protective System</u>

The Reactor Protective System (RPS) is not covered by the equipment categories identified in FSAR Section 3.1.1, "Criterion 1 - Quality Standards (Category A)" on page 3-3. However, the RPS was listed in Section 1.41 of the PSAR and subsequently in FSAR Appendix 1B. The RPS is required for LBLOCA/LOOP mitigation and has always been QA-1. Therefore DPC believes that it warrants inclusion into the category of "original QA-1 licensing basis."

**Second Category, Oconee QA-1 SSCs Added To The Original Licensing Basis.**

In this category DPC includes any commitments to the NRC to treat other SSCs as QA-1 per correspondence subsequent to the original Oconee QA-1 licensing basis.

These commitments are as follows:

1. The following portions of the emergency feedwater (EFW) systems are QA-1.

   - the motor-driven (MD) EFW pumps
   - the piping from the MD EFW pumps to the steam generators
   - the EFW flow control valves (excluding the operators)
   - the power supply to the MD EFW pumps and controls
   - piping from the upper surge tanks (USTs) to the MD EFW pumps
   - UST level monitoring circuitry and associated solenoid valves
   - EFW flow transmitters upstream of the flow control valves
   - MD and turbine-driven EFW pump initiation signals

2. The anticipatory reactor trips on (1) loss of main feedwater and (2) turbine trip are QA-1.

3. The following instruments are QA-1 per the Duke response to Regulatory Guide 1.97:

   Two channels of wide range Reactor Coolant System (RCS) pressure
   24 core exit thermocouples (12 per train)
   Two channels of pressurizer level (one per train)
   Two channels of saturation margin (one monitoring loop A and the core, the other monitoring loop B and the core)
   Two channels of steam geneator (SG) level per SG (O-388″ range)
   Two channels of SG pressure per SG
   Two channels of borated water storage tank level
   Two channels of high pressure injection (HPI) flow
   Two channels of low pressure injection (LPI) flow
   Two channels of Reactor Building spray flow
   Two channels of Reactor Building hydrogen concentration
   Two channels of upper surge tank level (one per tank)
   Two channels of fulll range neutron flux
   Two channels of wide range RCS hot leg temperature (one per loop)
   Two channels of reactor vessel head level
   Two channels of hot leg level (one per loop)
   Two channels of wide range Reactor Building sump level
   Two channels of Reactor Building pressure
   One channel of valve position for each electrically-controlled Reactor Building isolation valve
   Two channels of high range Reactor Building radiation level
   Two channels of EFW flow per SG
   One channel of low pressure service water (LPSW) flow to the LPI coolers (per cooler)

4. The RCS hot leg and reactor vessel high point vents (piping, valves, and power supplies) are QA-1.

5. Duke has made explicit QA-1 commitments for the following portions of the Standby Shutdown Facility:

   SSF reactor coolant emergency makeup piping and components
   SSF auxiliary service water piping and components
   SSF cooling water piping for the diesel generator and HVAC

   Duke is taking the position that all portions of the SSF required for mitigation of a seismic-induced Turbine Building flood shall be QA-1.

6. The Control Rod Drive System AC breakers, DC breakers, and associated undervoltage devices are QA-1.

5 7. The power supplies and position indications for valves 2LP-3 and 3LP-3 are QA-1.

5 8. The equipment installed for the automatic Keowee auxiliary load center transfer modification is QA-1.

5 9. The 230 kV Degraded Grid Protection System (DGPS) and the CT-5 DGPS are QA-1.

5 10. The suction source for the Low Pressure Service Water (LPSW) System will be QA-1 following
5 planned modifications.

5 11. The instrument tubing on the systems that comprise the ECCS are to be reclassified as QA-1.

5 12. The pressure transmitters, logic, actuation circuitry, and solenoid valves for the MSLB Detection and
5 FDW Isolation System are QA-1.

5 13. The maintenance and test procedures for certain 6.9 kV and 4 kV switchgear breakers are QA-1.
5 Components that are used in future maintenance on these breakers that may impact the ability to
5 shed non-safety loads are also QA-1.

5 14. The two hydrogen recombiner packages and the interfacing piping systems shall be QA-1. This does
5 not include the power supplies, which are instead subject to specific docketed agreements involving
5 load-shed power.

5 15. No regulatory commitment exists for Duke to treat Oconee Class F piping as QA-1 solely on the
5 basis of its Class F designation. However, Duke has always and expects to continue to treat Oconee
5 Class F piping as QA-1 in the future. This explicit clarification is noted here, for it has been the
5 cause of some confusion both within Duke and for the NRC.

## 3.1.2 CRITERION 2 - PERFORMANCE STANDARDS (CATEGORY A)

Those systems and components of reactor facilities which are essential to the prevention of accidents which could affect the public health and safety or to mitigation of their consequences shall be designed, fabricated and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena such as earthquakes, tornadoes, flooding conditions, winds, ice, and other local site effects. The design bases so established shall reflect: a) appropriate consideration of the most severe of these natural phenomena that have been recorded for the site and the surrounding area and, b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

Discussion

1. Essential Systems and Components

   The integrity of systems, structures, and components essential to accident prevention and to mitigation of accident consequences has been included in the reactor design evaluations. These systems, structures, and components are:

   a. Reactor Coolant System

   b. Reactor vessel internals

   c. Reactor Building

   d. Engineered Safeguards Systems

   e. Electric emergency power sources.

2. Natural Phenomena

These essential systems and components have been designed, fabricated, and erected to performance standards that will enable the facility to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena. The designs are based upon the most severe of the natural phenomena recorded for the vicinity of the site, with an appropriate margin to account for uncertainties in the historical data.

These natural phenomena are listed below. Design bases are presented elsewhere in this report where specific systems, structures, and components are discussed.

a. Earthquake

b. Tornado

c. Ground Water and Flood

d. Wind and Hurricane

e. Snow and Ice

f. Other Local Site Effects

## 3.1.3 CRITERION 3 - FIRE PROTECTION (CATEGORY A)

The reactor facility shall be designed: 1) to minimize the probability of events such as fires and explosions and, 2) to minimize the potential effects of such events to safety. Noncombustible and fire-resistant materials shall be used whenever practical throughout the facility, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

Discussion

The reactor facility is designed to minimize the probability of fire and explosion. Noncombustibles and fire-resistant materials were used whenever practical throughout the facility.

The control rooms are constructed and furnished with non-flammable equipment. Adequate fire extinguishers are supplied, and combustible materials, such as records, are kept to a minimum as indicated in Section 7.7.5, "Occupancy" on page 7-91. The control rooms are equipped with emergency breathing apparatus to permit continuous occupancy in the unlikely event of a fire.

Electrical distribution equipment will be physically located to reduce vulnerability of vital circuits to physical damage as a result of accidents. Locations to achieve this result are described in Section 8.3.1.4, "Independence of Redundant Systems" on page 8-16.

## 3.1.4 CRITERION 4 - SHARING OF SYSTEMS (CATEGORY A)

Reactor facilities shall not share systems or components unless it is shown safety is not impaired by the sharing.

Discussion

Portions of the following systems are shared as indicated. Where sharing between Oconee 1 and 2 is indicated, a separate system is provided for Oconee 3. Safety is not impaired by the sharing.

| System | Shared by Units | Reference |
|---|---|---|
| Chemical Addition and Sampling | 1, 2 | 9.3.2 |
| Spent Fuel Cooling | 1, 2 | 9.1.3 |

| System | Shared by Units | Reference |
|---|---|---|
| Liquid Waste Disposal | 1, 2, 3 | 11.2.2 |
| Gaseous Waste Disposal | 1, 2 | 11.3.2 |
| Solid Waste Disposal | 1, 2, 3 | 11.4.1.2 |
| Coolant Treatment | 1, 2, 3 | 9.3.6 |
| Recirculated Cooling Water | 1, 2, 3 | 9.2.2.2.4 |
| Low Pressure Service Water | 1, 2, 3 | 9.2.2.2.3 |
| High Pressure Service Water | 1, 2, 3 | 9.2.2.2.2 |
| Control Room Ventilation | 1, 2 | 9.4.1 |
| Auxiliary Building Ventilation | 1, 2 | 9.4.3 |
| Turbine Building Ventilation | 1, 2, 3 | 9.4.4 |
| Area Radiation Monitoring | 1, 2 | 12.3.3 |
| Process Radiation Monitoring | 1, 2 | 11.5 |
| 4.16 kV Standby Power Buses | 1, 2, 3 | 8.3.1.1.3 |
| 125/250 Volt DC Power System | 1, 2, 3 | 8.3.2.1.2 |
| 120 Volt AC Vital Power System | 1, 2, 3 | 8.3.2.1.4 |
| 120 Volt Regulated Power System | 1, 2, 3 | 8.3.2.1.6 |

## 3.1.5 CRITERION 5 - RECORDS REQUIREMENTS (CATEGORY A)

Records of the design, fabrication, and construction of essential components of the plant shall be maintained by the reactor operator or under his control throughout the life of the reactor.

Discussion

Duke Power Company will have under its control or will have access to all records of major essential components for the life of the plant. Records maintained by Duke Power Company will include:

1. A complete set of as-built facility plans and systems diagrams which will include arrangement plants, system diagrams, major structural plans, and technical manuals of major installed equipment.
2. A set of completed test procedures as associated data for all plant testing outlined in Chapter 14, "Initial Tests and Operation" on page 14-1.
3. Quality assurance data generated during fabrication and erection of the essential components of the plant as defined by the quality assurance program within the scope of Section 3.1.1, "Criterion 1 - Quality Standards (Category A)" on page 3-3.

## 3.1.6 CRITERION 6 - REACTOR CORE DESIGN (CATEGORY A)

The reactor core shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core design, together with reliable process and decay heat removal systems, shall provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and for transient situations which can be anticipated, including the effects of the loss of power to recirculation pumps, tripping out of a turbine generator set, isolation of the reactor from its primary heat sink, and loss of all off-site power.

Discussion

The reactor is designed with the necessary margins to accommodate, without fuel damage, expected transients from steady-state operation including the transients given in the criterion. Fuel clad integrity is ensured under all normal and abnormal modes of anticipated operation by avoiding clad overstressing and overheating. The evaluation of clad stresses includes the effects of internal and external pressures,

temperature gradients and changes, clad-fuel interactions, vibrations, and earthquake effects. Clad fatigue due to power and pressure cycling is minimized by pre-pressurizing with helium all fuel rods except those in the low burnup region of Core 1, Oconee 1. The free-standing clad design prevents collapse at the end volume region of the fuel rod and provides sufficient radial and end void volume to accommodate clad-fuel interactions and internal gas pressures (Section 4.5.2, "Core Components" on page 4-57).

Clad overheating is prevented by satisfying the core thermal and hydraulic criteria (Section 4.4, "Thermal and Hydraulic Design" on page 4-41):

1. At the design overpower, no fuel melting will occur.
2. A 99 percent confidence exists that at least 99.5 percent of the fuel rods in the core will be in no jeopardy of experiencing a Departure from Nucleate Boiling (DNB) during continuous operation at the design overpower of 114 percent.

The design margins allow for deviations of temperature, pressure, flow, reactor power, and reactor-turbine power mismatch. Above 15 percent power, the reactor is operated at a constant average coolant temperature and has a negative power coefficient to damp the effects of power transients. The Reactor Control System will maintain the reactor operating parameters within preset limits, and the Reactor Protection System will shut down the reactor if normal operating limits are exceeded by preset amounts (Sections 7.2, "Reactor Protective System" on page 7-7 and 15.1, "Uncompensated Operating Reactivity Changes" on page 15-3).

Reactor decay heat will be removed through the stream generators until the reactor coolant system is cooled to 250°F. Steam generated by decay heat will supply the steam-driven main feedwater pump turbine and can also be vented to atmosphere and/or bypassed to the condenser. The steam generators are supplied feedwater from either the main steam-driven feedwater pumps, the motor-driven emergency feedwater pumps, or from a steam-driven emergency feed pump, sized at 7.5 percent of full feedwater flow.

The main feedwater pumps supply the steam generators with water contained in the feedwater train and the condensate storage tank. The emergency feed pumps take suction from the upper surge tank or from the condenser hotwell. These sources provide sufficient coolant to remove decay heat for about one day after reactor shutdown with the primary heat sink (condenser) isolated. The condenser is normally available so that water inventory is not depleted (Chapter 10, "Steam and Power Conversion System" on page 10-1), even in the event of loss of electrical power.

The reactor coolant pumps are provided with sufficient inertia to maintain adequate flow to prevent fuel damage if power to all pumps is lost. Natural circulation coolant flow will provide adequate core cooling after the pump energy has been dissipated (Section 15.6, "Loss of Coolant Flow Accident" on page 15-17).

## 3.1.7 CRITERION 7 - SUPPRESSION OF POWER OSCILLATIONS (CATEGORY B)

The core design, together with reliable controls, shall ensure that power oscillations which could cause damage in excess of acceptable fuel damage limits are not possible or can be readily suppressed.

Discussion

Power oscillations resulting from variations of coolant temperature are minimized by constant average coolant temperature when the reactor is operated above 15 percent power. Power oscillations from spatial

xenon effects are minimized by the large negative power coefficient and axial power shaping rod assemblies.

The ability of the reactor control and protection system to control the oscillations resulting from variation of coolant temperature within the control system dead band and from spatial xenon oscillations has been analyzed. Variations in average coolant temperature provide negative feedback and enhance reactor stability during that portion of core life in which the moderator temperature coefficient is negative. When the moderator temperature coefficient is positive, rod motion will compensate for the positive feedback. The maximum rate of power change resulting from temperature oscillations within the control system dead band has been calculated to be less than 1 percent/minute. Since the unit has been designed to follow ramp load changes of 10 percent/minute, this is well within the capability of the control system (Section 7.7.2, "Information Display and Control Functions" on page 7-89).

Control flexibility, with respect to xenon transients, is provided by the combination of control rods and nuclear instrumentation. Axial, radial, or azimuthal neutron flux changes will be detected by the nuclear instrumentation. Individual control rods or groups of control rods can be positioned to suppress and/or correct flux changes (Section 4.3.2.2, "Reactivity Control" on page 4-20). The analysis of xenon-related power effects is presented in BAW-10010, "Stability Margin for Xenon Oscillation."

## 3.1.8 CRITERION 8 - OVERALL POWER COEFFICIENT (CATEGORY B)

The reactor shall be designed so that the overall power coefficient in the power operating range shall not be positive.

Discussion

The overall power coefficient is negative in the power operating range (Section 4.3.1, "Design Bases - Nuclear Design" on page 4-19).

## 3.1.9 CRITERION 9 - REACTOR COOLANT PRESSURE BOUNDARY (CATEGORY A)

The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

Discussion

The Reactor Coolant System pressure boundary meets the criterion through the following:

1. Material selection, design, fabrication, inspection, testing, and certification in accordance with ASME codes for all components excluding piping, which is done in accordance with the USAS B31.1 and B31.7 codes.
2. Manufacture and erection in accordance with approved procedures.
3. Inspection in accordance with code requirements plus additional requirements imposed by the manufacturer.
4. System analysis to account for cyclic effects of thermal transients, mechanical shock, seismic loadings, and vibratory loadings.
5. Selection of reactor vessel material properties to give due consideration to neutron flux effects and the resultant increase of the nil ductility transition temperature.

The materials, codes, cyclic loadings, and non-destructive testing are discussed further in Chapter 5, "Reactor Coolant System and Connected Systems" on page 5-1.

## 3.1.10 CRITERION 10 - CONTAINMENT (CATEGORY A)

Containment shall be provided. The containment structure shall be designed to sustain the initial effects of gross equipment failures, such as a large coolant boundary break, without loss of required integrity, and, together with other engineering safety features as may be necessary, to retain for as long as the situation requires the functional capability to protect the public.

Discussion

Containment is provided by the Reactor Building. The Reactor Building has the capability to sustain, without loss of integrity, the effects of gross equipment failures, including the transient peak pressure associated with a hypothetical rupture of any pipe in the Reactor Coolant System including the effects of metal-water reactions described in Section 15.6, "Loss of Coolant Flow Accident" on page 15-17.

The design parameters for the Reactor Building are tabulated in Section 3.8, "Design of Class 1 Structures" on page 3-75 and Engineered Safety Systems have been evaluated for various combinations of credible energy releases as discussed in Section 15.6, "Loss of Coolant Flow Accident" on page 15-17. Sufficient redundancy is provided both in equipment and control to ensure the functional availability and capability of systems required to protect the public.

## 3.1.11 CRITERION 11 - CONTROL ROOM (CATEGORY B)

The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10CFR20 limits. It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost due to fire or other cause.

Discussion

The reactors and associated equipment are controlled from panels located in the control rooms. The control rooms are designed to permit continuous occupancy following a maximum hypothetical accident
0 (MHA) (Section 7.7.5, "Occupancy" on page 7-91).

All controls and instrumentation required to monitor and operate the reactors and electric power generating equipment are located within the control rooms. This includes indication of power level; process variables such as temperatures, pressures, and flows; valve positions; and control rod positions.

All Engineered Safety Systems equipment are controlled and monitored from the control rooms. The status of all dynamic equipment (pumps, valves, etc.)--as well as pertinent pressures, temperatures, and flows--is displayed. The Radiation Monitoring System has provisions for alarms and for display of instrumentation readouts in the control room.

The concrete Reactor Buildings and control room walls and roofs are designed to provide adequate protection against direct radiation to control room personnel at all times. Control room personnel on eight-hour shifts during a 90-day period following the MHA would not receive an integrated whole body dose in excess of 3 rem from all sources of direct radiation, including exposure during egress and ingress for shift changes.

The control rooms are provided with independent ventilation and filtration systems to minimize ingress of airborne radioactive contaminants escaping from the Reactor Building. The details of the control room ventilation system and its operation following an accident are described in Section 9.4.1, "Control Room Ventilation" on page 9-53.

The control rooms are constructed and furnished with non-flammable equipment. Adequate fire extinguishers are supplied and combustible materials, such as records, are kept to a minimum as per
0 Section 7.7.5, "Occupancy" on page 7-91. Emergency breathing apparatus is provided in the control room to permit occupancy in the unlikely event of a fire.

Adequate instrumentation and controls are provided to maintain the reactor in a safe hot shutdown condition from outside the control room if access to the control room is lost or if the room must be evacuated temporarily in the unlikely event of a fire or other causes.

## 3.1.12 CRITERION 12 - INSTRUMENTATION AND CONTROL SYSTEMS (CATEGORY B)

Instrumentation and controls shall be provided as required to monitor and maintain variables within prescribed operating ranges.

Discussion

Reactor regulation is based upon the use of movable control rods and a chemical neutron absorber (boron in the form of boric acid) dissolved in the reactor coolant. Input signals to the reactor controls include reactor coolant average temperature, megawatt demand, and reactor power. The reactor controls are designed to maintain a constant average reactor coolant temperature over the load range from 15 to 100 percent of rated power. The steam system operates at constant pressure for all loads. Adequate instrumentation and controls are provided to maintain operating variables within their prescribed ranges (Section 7.7.2, "Information Display and Control Functions" on page 7-89).

The non-nuclear instrumentation measures temperatures, pressures, flows, and levels in the Reactor Coolant System, Steam System, and Auxiliary Reactor Systems, and maintains these variables within prescribed limits (Section 7.4.2, "Non-Nuclear Process Instrumentation" on page 7-29).

## 3.1.13 CRITERION 13 - FISSION PROCESS MONITORS AND CONTROLS (CATEGORY B)

Means shall be provided for monitoring and maintaining control over the fission process throughout core life and for all conditions that can reasonably be anticipated to cause variations in reactivity of the core, such as indication of position of control rods and concentration of soluble reactivity control poisons.

Discussion

This criterion is met by reactivity control means and control room display. Reactivity control is by movable control rods and by chemical neutron absorber (in the form of boric acid) dissolved in the reactor coolant. The position of each control rod will be displayed in the control room. Changes in the reactivity status due to soluble boron will be indicated by changes in the position of the control rods. Actual boron concentration in the reactor coolant is determined periodically by sampling and analysis (Sections 7.7.1, "General Layout" on page 7-89 and 9.3.3.2, "System Description and Evaluation" on page 9-44).

## 3.1.14 CRITERION 14 - CORE PROTECTION SYSTEMS (CATEGORY B)

Core protection systems, together with associated equipment, shall be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.

Discussion

The reactor design meets this criterion by reactor trip provisions and engineered safety features. The Reactor Protection System is designed to limit reactor power which might result from unexpected reactivity changes, and provides an automatic reactor trip to prevent exceeding acceptable fuel damage limits. In a loss-of-coolant accident, the Engineered Safeguards System automatically actuates the High-Pressure and Low-Pressure Injection Systems. The core flooding tanks are self-actuating. Certain long-term operations in the emergency Core Cooling Systems which do not require immediate actuation are performed manually by the operator, such as remote switching of the low-pressure injection pumps to the recirculation mode and sampling of the recirculated coolant (Sections 7.2, "Reactor Protective System" on page 7-7 and 7.3, "Engineered Safeguards Protective System" on page 7-19).

## 3.1.15 CRITERION 15 - ENGINEERED SAFETY FEATURES PROTECTION SYSTEMS (CATEGORY B)

Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.

Discussion

The Engineered Safeguards Actuation System senses Reactor Coolant System pressure and Reactor Building pressure and initiates Emergency Core Cooling, Reactor Building isolation, and Reactor Building cooling at the appropriate levels. It also initiates starting of the Standby Emergency Power Sources (Sections 6.3.2, "System Design" on page 6-34 and 8.3.1.1.3, "4160 Volt Auxiliary System" on page 8-10).

## 3.1.16 CRITERION 16 - MONITORING REACTOR COOLANT PRESSURE BOUNDARY (CATEGORY B)

Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage.

Discussion

Reactor coolant pressure boundary integrity can be continuously monitored in the control room by surveillance of variation from normal conditions for the following:

1. Reactor Building temperature and sump level.
2. Reactor Building radioactivity levels.
3. Condenser off-gas radioactivity levels and main steam line N-16 monitors (to detect steam generator tube leakage).
4. Decreasing letdown storage tank water level (indicating system leakage).

Gross leakage from the reactor coolant boundary will also be indicated by a decrease in pressurizer water level and a rapid increase in the Reactor Building sump water level (Section 5.2.3.8, "System Incident Potential" on page 5-29).

## 3.1.17 CRITERION 17 - MONITORING RADIOACTIVITY RELEASES (CATEGORY B)

Means shall be provided for monitoring the containment atmosphere, the facility effluent discharge paths and the facility environs for radioactivity that could be released from normal operations, from anticipated transients and from accident conditions.

Discussion

Various process radiation monitoring system detectors are used to measure airborne gaseous and particulate radioactivity, including iodine, in the Reactor Buildings; in releases from Waste Gas Tanks; and in effluent activity in the vent stacks (Section 11.5, "Process and Effluent Radiological Monitoring and Sampling Systems" on page 11-17). These detectors have extended ranges to cover anticipated levels during normal operation, transient and accident conditions. They are also shielded against the background radiation levels expected to exist during an accident so that their readings will be valid under these conditions. Detectors are also located on the radioactive liquid waste discharge line which are interlocked to close the discharge valve on high activity. These instruments have been calibrated and have individual built-in secondary calibration sources of long half-life. Batch samples can also be collected for laboratory analysis and counting prior to the release of liquid and gaseous effluents. Service water, main steam lines, and turbine air ejector off-gas are also monitored to detect leakage of radioactivity in operation.

4

As part of the Environmental Radioactivity Monitoring Program, several sampling locations will be located within the Exclusion Area. One of these is located where the highest annual ground level concentrations of radioactivity from unit vent releases is expected to exist based on site meteorological studies. Another location is downstream of the liquid waste discharge point. Dosimeters are located at numerous points along the site boundary fence. Vegetation, surface water, shoreline sediment, fish, and integrated dose are monitored (Section 12.4, "Radiation Protection Program" on page 12-13).

In addition, environmental monitoring locations have been established in various populated areas and towns surrounding the site at distances up to 12 miles.

## 3.1.18 CRITERION 18 - MONITORING FUEL AND WASTE STORAGE (CATEGORY B)

Monitoring and alarm instrumentation shall be provided for fuel and waste storage and handling areas for conditions that might contribute to loss of continuity in decay heat removal and to radiation exposures.

Discussion

All refueling operations will be carried out with the fuel under borated water to provide cooling for fuel assemblies and shielding for personnel.

Level indicators are provided to alarm low water level in the spent fuel storage pool. Penetrations of the pool liner are arranged to prevent accidental drainage of the pool (Section 9.1.4.2.3, "Safety Provisions" on page 9-20)

Temperature sensors and flow monitors in the spent fuel pool cooling loop alarm on high temperature or loss of flow (Section 9.1.3, "Spent Fuel Cooling System" on page 9-13).

Radiation monitors and alarms are provided in the Reactor Building, in all refueling areas, and in the waste storage and processing areas to warn operating personnel of excessive radiation levels (Section 12.3.3, "Area Radiation Monitoring System" on page 12-11).

## 3.1.19 CRITERION 19 - PROTECTION SYSTEMS RELIABILITY (CATEGORY B)

Protection systems shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed.

Discussion

The Protection Systems design meets this criterion by specific instrument location, component redundancy, and in-service testing capability. The major design criteria stated below have been applied to the design of the instrumentation.

1. No single component failure shall prevent the protection systems from fulfilling their protective function when action is required.
2. No single component failure shall initiate unnecessary protection system action, provided implementation does not conflict with the criterion above.

Test connections and capabilities are built into the Protection Systems to provide for:

1. Pre-operational testing to give assurance that the protection systems can fulfill their required functions.
2. On-line testing to assure availability and operability (Section 7.1.2.1, "Design Bases" on page 7-3).

## 3.1.20 CRITERION 20 - PROTECTION SYSTEMS REDUNDANCY AND INDEPENDENCE (CATEGORY B)

Redundancy and independence designed into Protection Systems shall be sufficient to assure that no single failure or removal from service of any component or channel of a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served. Different principles shall be used where necessary to achieve true independence of redundant instrumentation components.

Discussion

Reactor protection is by four channels with 2/4 coincidence, and engineered safeguards features are by three channels with 2/3 coincidence. All Protection System functions are implemented by redundant sensors, instrument strings, logic, and action devices that combine to form the protection channels. Redundant protection channels and their associated elements are electrically independent and packaged to provide physical separation. The Reactor Protection System initiates a trip of the channel involved when modules or equipment are removed (Section 7.1.2.1, "Design Bases" on page 7-3).

## 3.1.21 CRITERION 21 - SINGLE FAILURE DEFINITION (CATEGORY B)

Multiple failures resulting from a single event shall be treated as a single failure.

Discussion

The Protection Systems meet this criterion in that the instrumentation is designed so that a single event cannot result in multiple failures that would prevent the required protective action (Section 7.3, "Engineered Safeguards Protective System" on page 7-19).

## 3.1.22 CRITERION 22 - SEPARATION OF PROTECTION AND CONTROL INSTRUMENTATION SYSTEMS (CATEGORY B)

Protection Systems shall be separated from control instrumentation systems to the extent that failure or removal from service of any control instrumentation system component or channel, or of those common to control instrumentation and protection circuitry, leaves intact a system satisfying all requirements for the protection channels.

Discussion

The Protection Systems' input channels are electrically and physically independent. Shared instrumentation for protection and control functions satisfies the single failure criteria by the employment of isolation techniques to the multiple outputs of various instrument strings.

## 3.1.23 CRITERION 23 - PROTECTION AGAINST MULTIPLE DISABILITY FOR PROTECTION SYSTEMS (CATEGORY B)

The effects of adverse conditions to which redundant channels or Protection Systems might be exposed in common, either under normal conditions or those of an accident, shall not result in a loss of the protection function.

Discussion

The Protection Systems are designed to extreme ambient conditions. The Protection Systems' instrumentation will operate from 40°F to 140°F and sustain the loss-of-coolant building environmental conditions, including 100 percent relative humidity, without loss of operability. Out-of-core neutron detectors, however, will withstand 90 percent relative humidity. The protection systems' instrumentation will be subject to environmental (qualification) testing as required by the proposed IEEE "Criteria for Nuclear Power Plant Protection Systems," IEEE No. 279, dated August, 1968. Protective equipment outside the Reactor Building (control room and relay room) is designed for continuous operation in an ambient temperature and relative humidity representative of loss-of-coolant accident conditions (Section 7.1.2.1, "Design Bases" on page 7-3).

## 3.1.24 CRITERION 24 - EMERGENCY POWER FOR PROTECTION SYSTEMS (CATEGORY B)

In the event of loss of all off-site power, sufficient alternate sources of power shall be provided to permit the required functioning of the Protection Systems.

Discussion

In the event of loss of all off-site power to all units at Oconee or to any unit alone, sufficient power for operation of the Protection Systems of any unit will be available from either of two on-site independent hydroelectric generators. Details of the Emergency Power Generation System are described in Section 8.3.1.1.1, "Keowee Hydro Station" on page 8-9.

Redundant battery power is provided for vital instrumentation and control.

## 3.1.25 CRITERION 25 - DEMONSTRATION OF FUNCTIONAL OPERABILITY OF PROTECTION SYSTEMS (CATEGORY B)

Means shall be included for testing Protection Systems while the reactor is in operation to demonstrate that no failure or loss of redundancy has occurred.

Discussion

Test circuits are supplied which utilize the redundant, independent, and coincidence features of the Protection Systems. This makes it possible to manually initiate on-line trip signals in any single protection channel in order to test trip capability in each channel without affecting the other channels (Section 7.3, "Engineered Safeguards Protective System" on page 7-19).

## 3.1.26 CRITERION 26 - PROTECTION SYSTEMS FAIL-SAFE DESIGN (CATEGORY B)

The Protection Systems shall be designed to fail into a safe state or into a state established as tolerable on a defined bases if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water), are experienced.

Discussion

The Reactor Protection System will trip the reactor on loss of power. The Engineered Safeguards Systems are supplied with multiple sources of electric power for control and valve action. A total loss of electrical power to the Engineered Safeguards Actuation System will cause it to assume a tripped position with the exception of the control relays. These relays require power to trip. However, since the engineered safeguards equipment also requires power to operate, this relay need not assume the tripped position upon a total loss of power.

The system is designed for continuous operation under adverse environments, as described in the discussion of Criterion 23 (Sections 7.1.2.1, "Design Bases" on page 7-3 and 7.2, "Reactor Protective System" on page 7-7).

Redundant instrument channels are Provided for the Reactor Protection and Engineered Safeguards Actuation Systems. Loss of power to each individual reactor protection channel will trip that individual channel. Loss of all instrument power will trip the Reactor Protection System and activate the Engineered Safeguards System instrumentation (with the exception of the Reactor Building spray valves).

Manual reactor trip is designed so that failure of the automatic reactor trip circuitry will not prohibit or negate the manual trip. The same is true with respect to manual operation of the engineered safeguards equipment.

## 3.1.27 CRITERION 27 - REDUNDANCY OF REACTIVITY CONTROL (CATEGORY A)

At least two independent Reactivity Control Systems, preferably of different principles, shall be provided.

Discussion

This criterion is met by movable control rods and soluble boron poison (Section 7.7.1, "General Layout" on page 7-89).

## 3.1.28 CRITERION 28 - REACTIVITY HOT SHUTDOWN CAPABILITY (CATEGORY A)

At least two of the Reactivity Control Systems provided shall independently be capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes, sufficiently fast to prevent exceeding acceptable fuel damage limits.

Discussion

A single Reactivity Control System consisting of 61 control rods is provided to rapidly make the core subcritical upon a trip signal. Trip levels are set to protect the core from damage due to the effects of any operating transient. The Soluble Absorber Reactivity Control System can add negative reactivity to make the reactor subcritical. However, its action is slow and its ability to protect the core from the damage, which might result from rapid load changes such as a full load turbine trip, is not a design criterion for this system. The high degree of redundancy in the Control Rod System is considered sufficient to meet the intent of this criterion (Section 4.3.2, "Description - Nuclear Design" on page 4-20).

## 3.1.29 CRITERION 29 - REACTIVITY SHUTDOWN CAPABILITY (CATEGORY A)

At least one of the Reactivity Control Systems provided shall be capable of making the core subcritical under any conditions (including anticipated operation transients), sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margins greater than the maximum worth of the most effective control rod when fully withdrawn shall be provided.

Discussion

The reactor design meets this criterion both under normal operating conditions and under the accident conditions set forth in Chapter 14, "Initial Tests and Operation" on page 14-1. The reactor is designed with the capability of providing a shutdown margin of at least 1 percent $\Delta k/k$ with the single most reactive control rod fully withdrawn at any point in core life with the reactor at a hot, zero power power condition. The minimum hot shutdown margin for Oconee 1 of 5.5 percent $\Delta k/k$ occurs at the end of life (Section 4.3.2.3, "Reactivity Shutdown Analysis" on page 4-21).

## 3.1.30 CRITERION 30 - REACTIVITY HOLDOWN CAPABILITY (CATEGORY B)

At least one of the Reactivity Control Systems provided shall be capable of making and holding the core subcritical under any conditions with appropriate margins for contingencies.

Discussion

The reactor meets this criterion with control rods for hot shutdown under normal operating conditions and for shutdown under the accident conditions set forth in Chapter 14, "Initial Tests and Operation" on page 14-1. Reactor subcritical margin is maintained during cooldown by changes in soluble boron concentration. The rate of reactivity compensation from boron addition is greater than the reactivity change associated with the reactor cooldown rate of 100°F/hour. Thus, subcriticality is assured during cooldown with the most reactive control rod totally unavailable (Section 4.3.2, "Description - Nuclear Design" on page 4-20).

## 3.1.31 CRITERION 31 - REACTIVITY CONTROL SYSTEMS MALFUNCTION (CATEGORY B)

The Reactivity Control Systems shall be capable of sustaining any single malfunction, such as unplanned continuous withdrawal (not ejection) of a control rod, without causing a reactivity transient which could result in exceeding acceptable fuel damage limits.

Discussion

5 The reactor design meets the intent of this criterion. A reactor trip will protect against any single
5 malfunction of the reactivity control system. This conclusion is based on the analysis for a continuous
5 rod group withdrawal accident (Section 15.3, "Rod Withdrawal Accident at Rated Power" on page 15-7).

5 **Note:** Design Criterion 31 implies by example that an unplanned continuous single rod withdrawal
5 accident analysis may be performed. ONS did not perform a single rod withdrawal accident
5 analysis in order to meet this design criterion. A single rod withdrawal accident cannot occur with
5 a single reactivity control systems malfunction under any normal conditions of plant startup,
5 shutdown, or operation. In addition, the NRC reviewed and approved the concept of using a
5 group rod withdrawal accident analysis as the basis for meeting this design criterion.

## 3.1.32 CRITERION 32 - MAXIMUM REACTIVITY WORTH OF CONTROL RODS (CATEGORY A)

Limits, which include considerable margin, shall be placed on the maximum reactivity worth of control rods or elements, and on rates at which reactivity can be increased to insure that the potential effects of a sudden or large change of reactivity cannot: a) rupture the reactor coolant pressure boundary or b) disrupt the core, its support structures, or other vessel internals sufficiently to impair the effectiveness of emergency core cooling.

Discussion

The reactor design meets this criterion by safety features which limit the maximum reactivity insertion rate. These include rod-group withdrawal interlocks, soluble boron concentration reduction interlock, maximum rate of dilution water addition, and dilution-time cutoff (Section 15.4, "Moderator Dilution Accident" on page 15-11). In addition, the rod drives and their controls have an inherent feature that limits overspeed in the event of malfunctions (Section 4.5.3, "Control Rod Drives" on page 4-59). Ejection of the maximum-worth control rod will not lead to further coolant boundary rupture or to internals damage which would interfere with emergency core cooling (Section 15.12, "Rod Ejection Accident" on page 15-41).

## 3.1.33 CRITERION 33 - REACTOR COOLANT PRESSURE BOUNDARY CAPABILITY (CATEGORY A)

The reactor coolant pressure boundary shall be capable of accommodating without rupture, and with only limited allowance for energy absorption through plastic deformation, the static and dynamic loads imposed on any boundary component as a result of any inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.

Discussion

The reactor design meets this criterion. There are no credible mechanisms whereby damaging energy releases are liberated to the reactor coolant. Ejection of the maximum worth control rod will not lead to further coolant boundary rupture (Section 15.12, "Rod Ejection Accident" on page 15-41).

## 3.1.34 CRITERION 34 - REACTOR COOLANT PRESSURE BOUNDARY RAPID PROPAGATION FAILURE PREVENTION (CATEGORY A)

The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given a) to the notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve, b) to the state of stress of materials under static and transient loadings, c) to the quality control specified for materials and component fabrication to limit flaw sizes, and d) to the provisions for control over service temperature and irradiation effects which may required operation restrictions.

Discussion

The reactor coolant pressure boundary design meets this criterion by the following:

1. Development of reactor vessel plate material properties opposite the core to a specified Charpy-V-notch test result of 30 ft/lb or greater at a nominal low NDTT.
2. Determination of the fatigue usage factor resulting from expected static and transient loading during detailed design and stress analysis.
3. Quality control procedures including permanent identification of materials and non-destructive testing.
4. Operating restrictions to prevent failure towards the end of design vessel life resulting from increase in the nil-ductility transition temperature (NDTT) due to neutron irradiation, as predicted by a material irradiation surveillance program (Section 5.2.3.13, "Reactor Vessel Material Surveillance Program" on page 5-35).

## 3.1.35 CRITERION 35 - REACTOR COOLANT PRESSURE BOUNDARY BRITTLE FRACTURE PREVENTION (CATEGORY A)

Under conditions where Reactor Coolant Pressure Boundary System components constructed of ferritic materials may be subjected to potential loadings, such as a reactivity-induced loading, service temperature shall be at least 120°F above the nil ductility transition (NDT) temperature of the component material if the resulting energy release is expected to be absorbed by plastic deformation or 60°F above the NDT temperature of the component material if the resulting energy release is expected to be absorbed within the elastic strain energy range.

Discussion

The reactor vessel is the only Reactor Coolant System component exposed to a significant level of neutron irradiation and is, therefore, the only component subject to material irradiation damage. Unit operating procedures will limit the operating pressure to 20 percent of the design pressure when the Reactor Coolant System temperature is below NDTT +60°F throughout unit life. Analysis has shown no potential reactivity-induced conditions which will result in energy release to the primary system in the range expected to be absorbed by plastic deformation (Section 5.2.3.3, "Reactor Vessel" on page 5-15).

## 3.1.36 CRITERION 36 - REACTOR COOLANT PRESSURE BOUNDARY SURVEILLANCE (CATEGORY A)

Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leak-tight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with ASTM-E-185-66 shall be provided.

Discussion

The reactor coolant pressure boundary components meet this criterion. Space is provided for non-destructive testing during plant shutdown. A reactor pressure vessel material surveillance program conforming to ASTM-E-185-66 has been established (Section 5.2.3.13, "Reactor Vessel Material Surveillance Program" on page 5-35).

## 3.1.37 CRITERION 37 - ENGINEERED SAFETY FEATURES BASIS FOR DESIGN (CATEGORY A)

Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. As a minimum, such engineered safety features shall be designed to cope with any size reactor coolant pressure boundary break up to and including the circumferential rupture of any pipe in that boundary assuming unobstructed discharge from both ends.

Discussion

The reactor design meets this criterion. The Emergency Core Cooling Systems can protect the reactor for any size leak up to and including the circumferential rupture of the largest reactor coolant pipe (Section 15.14, "Loss of Coolant Accidents" on page 15-55).

## 3.1.38 CRITERION 38 - RELIABILITY AND TESTABILITY OF ENGINEERED SAFETY FEATURES (CATEGORY A)

All engineered safety features shall be designed to provide high functional reliability and ready testability. In determining the suitability of a facility for a proposed site, the degree of reliance upon and acceptance of the inherent and engineered safety afforded by the systems, including engineered safety features, will be influenced by the known and the demonstrated performance capability and reliability of the systems, and by the extent to which the operability of such systems can be tested and inspected where appropriate during the life of the plant.

Discussion

All Engineered Safeguards Systems are designed so that a single failure of an active component in a system will not prevent operation of that system or reduce its capacity below that required to maintain a safe condition. Two independent Reactor Building Cooling Systems, each having full heat removal capacity, are provided to prevent overpressurization (Section 7.3, "Engineered Safeguards Protective System" on page 7-19).

The High-Pressure Injection, Core-Flooding, and Low-Pressure Injection Systems have separate equipment and instrumentation strings to ensure availability of capacity.

Some portions of the Engineered Safeguards Systems have both a normal and an emergency function, thereby providing nearly continuous demonstration of operability. During normal operation, the standby and operating units will be rotated into service on a scheduled basis.

Engineered Safeguards Systems equipment piping that is not fully projected against LOCA missile damage utilizes dual lines to preclude loss of the protective function as a result of the secondary failure.

Testing and inspection of the Engineered Safeguards Systems is further described in Chapter 6, "Engineered Safeguards" on page 6-1.

## 3.1.39 CRITERION 39 - EMERGENCY POWER FOR ENGINEERED SAFETY FEATURES (CATEGORY A)

Alternate power systems shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning required of the engineered safety features. As a minimum, the on-site power system and the off-site power system shall each, independently, provide this capacity assuming a failure of a single active component in each power system.

Discussion

The electrical systems meet the intent of the criterion as discussed in Chapter 8, "Electric Power" on page 8-1.

Three alternate emergency electric power supplies are provided for the station from which power to the engineered safety feature buses of each unit can be supplied. These are the 230 KV switching station with multiple off-site interconnections and two on-site independent 87,500 KVA hydroelectric generating units. Each nuclear unit can receive emergency power from the 230 KV switching station through its start-up transformer as a preferred source. Each unit can receive emergency power from one hydroelectric generating unit through a 13.8 KV underground connection to standby transformer CT4. The other hydroelectric generating unit serves as a standby emergency power source and can supply power to each unit's startup transformer when required. Both on-site hydroelectric generating units will start automatically upon loss of all normal power or upon an engineered safety feature action.

Upon completion of Oconee 2 and 3, two additional sources of alternate power will be available, as each nuclear unit is capable of supplying any other unit through the 230 KV switching station. In addition, a connection to the 100 KV transmission network is provided as an alternate source of emergency power whenever both hydroelectric generating units are unavailable.

## 3.1.40 CRITERION 40 - MISSILE PROTECTION (CATEGORY A)

Protection for engineered safety features shall be provided against dynamic effects and missiles that might result from plant equipment failures.

Discussion

Engineered safety features are redundant and either physically separated or shielded to provide protection against dynamic effects and missiles resulting from hypothesized plant equipment failure (Section 3.5, "Missile Protection" on page 3-51).

## 3.1.41 CRITERION 41 - ENGINEERED SAFETY FEATURES PERFORMANCE CAPABILITY (CATEGORY A)

Engineered safety features such as Emergency Core Cooling and Containment Heat Removal Systems shall provide sufficient performance capability to accommodate partial loss of installed capacity and still fulfill the required safety function. As a minimum, each engineered safety feature shall provide this required safety function assuming a failure of a single active component.

Discussion

All Engineered Safeguards Systems are designed so that a single failure of an active component will not prevent operation of that system or reduce the system capacity below that required to maintain a safe condition. Redundancy is provided in equipment and piping so that the failure of a single active component of any system will not impair the required safety function of that system (Section 7.3, "Engineered Safeguards Protective System" on page 7-19).

## 3.1.42 CRITERION 42 - ENGINEERED SAFETY FEATURES COMPONENTS CAPABILITY (CATEGORY A)

Engineered safety features shall be designed so that the capability of each component and system to perform its required function is not impaired by the effects of a loss-of-coolant accident.

Discussion

The Engineered Safeguards System design meets this criterion. A single-failure analysis of the Emergency Core Cooling Systems (Section 6.3.2.6, "System Reliability" on page 6-38) and the Reactor Building Heat Removal Systems (Sections 6.2, "Containment Systems" on page 6-9; 6.2.2, "Containment Heat Removal Systems" on page 6-22) demonstrates that these systems have sufficient redundancy to perform their design functions.

The core flooding tanks contain check valves which operate to permit flow of emergency coolant from the tanks to the reactor vessel. These valves are self-actuating and need no external signal or external supplied energy to make them operate. Accordingly, it is not considered credible that they would fail to operate when needed.

The engineered safeguards features are designed to function in the unlikely event of a loss of coolant accident with no impairment of function due to the effects of the accident.

## 3.1.43 CRITERION 43 - ACCIDENT AGGRAVATION PREVENTION (CATEGORY A)

Engineered safety features shall be designed so that any action of the engineered safety features which might accentuate the adverse after-effects of the loss of normal cooling is avoided.

Discussion

The Engineered Safeguards Systems are designed to meet this criterion. The water injected to ensure core
3 cooling is sufficiently borated to ensure core subcriticality. Water sources that are not required to mitigate
3 the consequences of an accident inside the Reactor Building are automatically isolated to prevent dilution
3 of the borated coolant. Sources of necessary post-accident cooling waters are monitored for boron
3 concentration to prevent additions which may lead to dilution of boron content. An analysis has been made to demonstrate that the injection of cold water on the Hot Reactor Coolant System surfaces will not lead to further failure. The design of the equipment and its actuating system ensures that water injection will occur in a sufficiently short time period to preclude significant metal-water reactions and consequent energy release to the Reactor Building (Section 15.14, "Loss of Coolant Accidents" on page 15-55).

## 3.1.44 CRITERION 44 - EMERGENCY CORE COOLING SYSTEMS CAPABILITY (CATEGORY A)

At least two Emergency Core Cooling Systems, preferable of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each Emergency Core Cooling System and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each Emergency Core Cooling System shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall nor share other features or components unless it can be demonstrated that: a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident.

Discussion

Emergency core cooling is provided by pumped injection and pressurized core flooding tanks. Pumped injection is subdivided in such a way that there are two separate and independent strings, each including both high pressure and low pressure coolant injection, and each capable of providing 100 percent of the necessary core injection with the core flooding tanks. There is no sharing of active components between the two subsystems in the post-accident operating mode. The core flooding tanks are passive components which are needed for only a short period of time after the accident, thereby assuring 100 percent availability when needed. This equipment prevents clad melting for the entire spectrum of Reactor Coolant System failures ranging from the smallest leak to the complete severance of the largest reactor coolant pipe (Section 15.14, "Loss of Coolant Accidents" on page 15-55).

## 3.1.45 CRITERION 45 - INSPECTION OF EMERGENCY CORE COOLING SYSTEMS (CATEGORY A)

Design provisions shall be made to facilitate physical inspection of all critical parts of the Emergency Core Cooling System including reactor vessel internals and water injection nozzles.

Discussion

All critical parts of the Emergency Core Cooling Systems, including the reactor vessel internals, can be inspected during plant shutdown (Section 5.2.3.12, "Tests and Inspections" on page 5-33).

## 3.1.46 CRITERION 46 - TESTING OF EMERGENCY CORE COOLING SYSTEMS COMPONENTS (CATEGORY A)

Design provisions shall be made so that active components of the Emergency Core Cooling Systems, such as pumps and valves, can be tested periodically for operability and required functional performance.

Discussion

The design of Emergency Core Cooling Systems and components has incorporated adequate test and operational features to permit periodic testing of active components to assure operability and functional capability. Core flooding tank functional performance will be demonstrated only in pre-operational testing.

## 3.1.47 CRITERION 47 - TESTING OF EMERGENCY CORE COOLING SYSTEMS (CATEGORY A)

A capability shall be provided to test periodically the delivery capability of the Emergency Core Cooling Systems at a location as close to the core as is practical.

Discussion

The High-Pressure (makeup water) and Low-Pressure (decay-heat removal) Injection Systems are included as part of Normal Service Systems. Consequently, the active components can be tested periodically for delivery capability. The Core Flooding System delivery capability will be demonstrated during startup testing. In addition, all valves will be periodically cycled to ensure operability. With these provisions, the delivery capability of the Emergency Core Cooling Systems can be periodically demonstrated (Section 6.3.4, "Tests and Inspections" on page 6-48).

## 3.1.48 CRITERION 48 - TESTING OF OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEMS (CATEGORY A)

A capability shall be provided to test, under conditions as close to design as practical, the full operational sequence that would bring the Emergency Core Cooling Systems into action, including the transfer to alternate power sources.

Discussion

The operational sequence that would bring the Emergency Core Cooling Systems into action, including transfer to alternate power sources, can be tested in parts (Sections 6.3.4, "Tests and Inspections" on page 6-48 and 7.3, "Engineered Safeguards Protective System" on page 7-19).

## 3.1.49 CRITERION 49 - CONTAINMENT DESIGN BASIS (CATEGORY A)

The containment structure, including access openings and penetrations, and any necessary containment heat removal systems shall be designed so that the containment structure can accommodate without exceeding the design leakage rate, the pressures and temperatures resulting from the largest credible energy release following a loss-of-coolant accident, including a considerable margin for effects from metal-water or other chemical reactions that could occur as a consequence of failure of Emergency Core Cooling Systems.

Discussion

The Reactor Building, access openings and penetrations, have been designed to accommodate a pressure of 59 psig at 286°F (Section 6.2.1, "Containment Functional Design" on page 6-9). As described in Section 15.14, "Loss of Coolant Accidents" on page 15-55 these conditions exceed the greatest transient peak pressure associated with a hypothetical rupture of a pipe in the Reactor Coolant System, including the margin for the effects of metal-water reactions. The capacity of each Reactor Building Cooling System (Sections 6.2, "Containment Systems" on page 6-9 and 6.2.2, "Containment Heat Removal
2 Systems" on page 6-22) is designed to remove heat from the Reactor Building to reduce pressure
2 following a loss-of-coolant accident.

2 Components of the Reactor Building Cooling System - including electric motors, valves, and damper operators, which function within the Reactor Building during accident conditions - are capable of operation as required to accomplish the safeguards function.

## 3.1.50 CRITERION 50 - NDT REQUIREMENT FOR CONTAINMENT MATERIAL (CATEGORY A)

Principle load-carrying components of ferritic materials exposed to the external environment shall be selected so that their temperatures under normal operating and testing conditions are not less than 30°F above nil-ductility transition (NDT) temperature.

Discussion

The Reactor Building liner has been designed so that it is not susceptible to a low temperature brittle fracture.

All principal load-carrying components of ferritic materials for the containment vessel exposed to the external environment have been selected and tested to confirm that their ductile-to-brittle-transition (NDT) temperature is at least 30°F below the minimum service metal temperature. The ferritic materials exposed to the external environment consist of the penetrations and large openings (equipment access hatch and personnel locks), for which materials have been selected to conform with ASME Boiler and Pressure Vessel Code, Section III, for Class "B" Vessels. Material specifications for the penetrations are more completely described in Section 3.8.1.1, "Description of the Containment" on page 3-75.

## 3.1.51 CRITERION 51 - REACTOR COOLANT PRESSURE BOUNDARY OUTSIDE CONTAINMENT (CATEGORY A)

If part of the reactor coolant pressure boundary is outside the containment, appropriate features, as necessary, shall be provided to protect the health and safety of the Public in case of an accidental rupture in that part. Determination of the appropriateness of features, such as isolation valves and additional containment, shall include consideration of the environmental and population conditions surrounding the site.

Discussion

The reactor coolant pressure boundary is defined as those piping systems or components which contain reactor coolant at high pressure and temperature. With the exception of the normal reactor coolant sampling line and the post accident reactor coolant sampling line, the reactor coolant pressure boundary, as defined above, is located entirely within the Reactor Building. These sampling lines are provided with remotely operated valves for isolation. The normal reactor coolant sampling line is used only during actual sampling operations. The post accident reactor coolant sampling line is used during performance testing of the post accident sampling system and/or actual post accident sampling operations. No significant environmental dose would result from these sources (Sections 6.2.3, "Containment Isolation System" on page 6-27, 11.2.2, "Disposal System Design" on page 11-5.

## 3.1.52 CRITERION 52 - CONTAINMENT HEAT REMOVAL SYSTEMS (CATEGORY A)

Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided.

Discussion

Two systems of different principles are provided to remove heat from each Reactor Building following an
2 accident. The systems are discussed in Sections 6.2, "Containment Systems" on page 6-9 and 6.2.2,
2 "Containment Heat Removal Systems" on page 6-22. Analysis of peak accident pressure in containment
5 following an accident is addressed in Sections 6.2.1.3, "Mass and Energy Release Analyses for Postulated
5 Loss-of-Coolant Accidents" on page 6-17 and 6.2.1.4, "Mass and Energy Release Analyses for Postulated
5 Secondary System Pipe Ruptures Inside Containment" on page 6-20 respectively. The analysis shows
2 containment to be capable of withstanding peak accident pressure without the Reactor Building Spray
2 System.

The Reactor Building Cooling System removes heat by circulating building atmosphere over cooling coils.

The Reactor Building Spray System supplies droplets of cool, borated water which absorb sensible and latent heat from the containment atmosphere.

## 3.1.53 CRITERION 53 - CONTAINMENT ISOLATION VALVES (CATEGORY A)

Penetrations that require closure for the containment function shall be protected with redundant valving and associated apparatus.

Discussion

Piping penetrations that require closure under accident conditions are provided with double valves so that no single credible failure or malfunction could result in a loss of isolation. Valves are manually,
3 electrically or pneumatically operated. Alternately, check valves are used in certain applications. All isolation valves inside the Reactor Building requiring remote operation are electrically operated.

## 3.1.54 CRITERION 54 - CONTAINMENT LEAKAGE RATE TESTING (CATEGORY A)

Containment shall be designed so that an integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations and the leakage rate measured over a sufficient period to verify its conformance with required performance.

Discussion

The Reactor Buildings are designed so that leakage rate can be determined at design pressure after completion and installation of all penetrations. The leak-rate test will verify that the maximum integrated leak rate does not exceed the design leakage rate (Section 3.8.1.7.3, "Initial Leakage Tests" on page 3-121).

## 3.1.55 CRITERION 55 - CONTAINMENT PERIODIC LEAKAGE RATE TESTING (CATEGORY A)

The containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during plant lifetime.

Discussion

The Reactor Building has been structurally designed to permit integrated leakage rate testing at design pressure (Section 3.8.1.7.4, "Leakage Monitoring" on page 3-122) but retesting is at lesser pressures.

## 3.1.56 CRITERION 56 - PROVISIONS FOR TESTING OF PENETRATIONS (CATEGORY A)

Provisions shall be made for testing penetrations which have resilient seals or expansion bellows to permit leak tightness to be demonstrated at design pressure at any time.

Discussion

All Reactor Building penetrations with resilient seals or expansion bellows are constructed so that they may be pressurized to design pressure for leak tests at any time (Section 3.8.1.7.4, "Leakage Monitoring" on page 3-122 and Section 3.8.1.5.4, "Penetrations" on page 3-102).

## 3.1.57 CRITERION 57 - PROVISIONS FOR TESTING OF ISOLATION VALVES (CATEGORY A)

Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valves leakage does not exceed acceptable limits.

Discussion

All remotely operated valves serving an Engineered Safeguards function have the capability for testing their functional operability. These tests can be conducted from the control rooms.

Isolation valves that are required to be closed from an Engineered Safeguards signal have test provisions for leak testing (Table 6-7).

## 3.1.58 CRITERION 58 - INSPECTION OF CONTAINMENT PRESSURE - REDUCING SYSTEMS (CATEGORY A)

Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as pumps, valves, spray nozzles, torus, and sumps.

Discussion

Provision is made to permit periodic physical inspection of components of the two containment pressure-reducing systems, the Reactor Building Spray System and the Reactor Building Cooling System. The Reactor Building spray pumps and the valves and operators associated with piping in each of these systems are located outside the Reactor Building, permitting the inspection of these components. The fan units of the Reactor Building cooling units are located so that physical inspection is possible during normal operation.

The cooling coils of the Reactor Building cooling units can be inspected during shutdown. The spray header and nozzles of the Reactor Building Spray System, located in the dome of the Reactor Building, can be inspected visually during shutdown. The sumps can be inspected and the screens cleaned during shutdown.

## 3.1.59 CRITERION 59 - TESTING OF CONTAINMENT PRESSURE-REDUCING SYSTEM COMPONENTS (CATEGORY A)

The containment pressure-reducing systems shall be designed so that active components, such as pumps and valves can be tested periodically for operability and required functional performance.

Discussion

The containment pressure-reducing systems have the capability of being periodically tested as follows:

1. Reactor Building Cooling Units
   a. The air fans can be individually tested for low speed operation.
   b. The cooling coil low pressure service water valves can be operated through their full travel with resulting flow alarm indication.
   c. The stand-by low pressure service water pumps can be tested for automatic starting.
2. Reactor Building Spray System
   a. The operation of the spray pumps can be tested by recirculating to the borated water storage tank through a test line.
   b. The building spray isolation valves can be operated through their full travel.

## 3.1.60 CRITERION 60 - TESTING OF CONTAINMENT SPRAY SYSTEMS (CATEGORY A)

A capability shall be provided to periodically test the delivery capability of the Containment Spray System at a position as close to the spray nozzles as is practical.

Discussion

The delivery capability of the spray nozzles will be tested by blowing low pressure air through the system and verifying flow through the nozzles.

The delivery capability of the pumps will be tested by recirculating to the borated water storage tank and monitoring the resultant flow.

## 3.1.61 CRITERION 61 - TESTING OF OPERATIONAL SEQUENCE OF CONTAINMENT PRESSURE REDUCING SYSTEMS

A capability shall be provided to test, under conditions as close to the design as practical, the full operational sequence that would bring the containment pressure-reducing systems into action including the transfer to alternate power sources.

Discussion

Each of the three redundant 4 kV switchgear buses supplying power to essential loads receives its power from two 4 kV main feeder buses. These main feeder buses are supplied by: 1) the main unit auxiliary transformers, 2) the startup transformer, and 3) the underground feeder from Keowee Hydro plant. Each main feeder bus is fed from each of the three sources above. In normal operation the two main feeders will be supplied through breakers from the unit auxiliary transformer and the breakers from the start-up transformer and the underground feeder will be open.

To test the transfer to alternate power source, the three breakers associated with one of the main feeders will be placed in test position with the normal breaker closed and the two alternate power sources breakers open. A low voltage simulation will be used to trip the normal breaker and close the start-up breaker. A low voltage and an Engineered Safeguards (ESG) simulation will be used to trip the start-up breaker and close the underground feeder breaker. In making these tests, the automatic dropping of load will not take place.

Testing the two independent channels for the Reactor Building Cooling System and the Building Spray System by inserting an analog signal can be accomplished without placing the systems in operation.

## 3.1.62 CRITERION 62 - INSPECTION OF AIR CLEANUP SYSTEMS

Design provisions shall be made to facilitate physical inspection of all critical parts of containment air cleanup systems such as ducts, filters, fans, and dampers.

Discussion

The Penetration Room Ventilation System is design to collect and process the leakage from penetrations as noted in Section 6.5.1, "Engineered Safeguards (ES) Filter Systems" on page 6-55. All components of this system are located in the Auxiliary Building, permitting periodic physical inspection.

## 3.1.63 CRITERION 63 - TESTING OF AIR CLEANUP SYSTEM COMPONENTS

Design provisions shall be made so that active components of the Air Cleanup Systems, such as fans and dampers, can be tested periodically for operability and required functional performance.

Discussion

The Penetration Room Ventilation System (Section 6.5.1.2, "System Design" on page 6-55) is designed so that active components can be tested periodically. Each penetration room fan can be manually started periodically to demonstrate operability. Provision is made for pressure testing the valves for leak tightness.

## 3.1.64 CRITERION 64 - TESTING OF AIR CLEANUP SYSTEMS

A capability shall be provided for in situ periodic testing and surveillance of the Air Cleanup Systems to ensure; a) filter bypass paths have not developed and, b) filter and trapping materials have not deteriorated beyond acceptable limits.

Discussion

The design of the penetration room ventilation system (Section 6.5.1.2, "System Design" on page 6-55) incorporates provisions for testing and surveillance of the filters. Connections and instrumentation for each filter bank allow in situ testing to ensure that filter performance has not deteriorated beyond acceptable limits.

## 3.1.65 CRITERION 65 - TESTING OF OPERATIONAL SEQUENCE OF AIR CLEANUP SYSTEMS (CATEGORY A)

A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the Air Cleanup Systems into action including the transfer to alternate power sources and the design air flow delivery capability.

Discussion

The Penetration Room Ventilation System is designed to allow testing the operation sequences required to bring the system into operation including the transfer to alternate power sources and the design air flow delivery capability.

Actuation of the Penetration Room Ventilation System and the transfer to alternate power sources are identical to the description provided in Criterion 3.1.61 for the Containment Pressure Reducing Systems.

## 3.1.66 CRITERION 66 - PREVENTION OF FUEL STORAGE CRITICALITY (CATEGORY B)

Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

Discussion

Criticality in new and spent fuel storage is prevented by designing storage facilities to maintain an eversafe geometric spacing between assemblies. Fuel assemblies cannot be placed in other than the prescribed locations (Section 9.1.2, "Spent Fuel Storage" on page 9-3).

## 3.1.67 CRITERION 67 - FUEL AND WASTE STORAGE DECAY HEAT (CATEGORY B)

Reliable Decay Heat Removal Systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to plant operating areas or the public environs.

Discussion

This criterion is met by the Spent Fuel Cooling System which incorporates provisions to maintain water cleanliness, temperature, and water level. Three pumps and three coolers will be adequate to maintain the spent fuel pool temperature within acceptable limits. The pumps in the system can be operated from the standby bus in case of loss of outside power to provide continuous cooling capability in the fuel storage facility (Section 9.1.3, "Spent Fuel Cooling System" on page 9-13).

## 3.1.68 CRITERION 68 - FUEL AND WASTE STORAGE RADIATION SHIELDING (CATEGORY B)

Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities to meet the requirements of 1OCFR20.

Discussion

Shielding meeting the requirements of 10CFR 20 is provided for protection of operating personnel:

1. During all phases of spent fuel removal and storage (Section 12.3.2, "Shielding" on page 12-9).
2. From radioactive waste holdup tanks and other containers containing potentially radioactive solutions, resins, or gases (Section 12.3.2, "Shielding" on page 12-9).

## 3.1.69 CRITERION 69 - PROTECTION AGAINST RADIOACTIVITY RELEASE FROM SPENT FUEL AND WASTE STORAGE (CATEGORY B)

Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

Discussion

Analyses in Chapter 15, "Accident Analyses" on page 15-1 have demonstrated that accidental release of the maximum activity content of a tank containing waste gases or liquids will not cause excessive off-site doses. The fuel handling accident, analyzed in Chapter 15, "Accident Analyses" on page 15-1 does not result in excessive off-site doses.

## 3.1.70 CRITERION 70 - CONTROL OF RELEASES OF RADIOACTIVITY TO THE ENVIRONMENT (CATEGORY B)

The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified: a) on the basis of 10CFR 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and b) on the basis of 10CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

Discussion

The Waste Disposal System is designed to insure that station personnel and the general public are protected against excessive exposure to radioactive material in accordance with the regulations of 10CFR 20.

The gaseous, liquid, and solid waste storage facilities are discussed in Chapter 11, "Radioactive Waste Management" on page 11-1 where it is demonstrated that adequate holdup capacity is provided. Gaseous and liquid wastes will be sampled before release and will be monitored for activity level at all times during release, or independent sampling and analysis will be performed prior to release when the appropriate monitor is out of service.

Control of leakage following a reactor accident is accomplished by the Containment System. This system consists of the Reactor Building and the Reactor Building Penetration Room Ventilation System. Experience has shown that Reactor Building leakage is more likely at penetrations than in liner plates or weld joints. Any potential penetration leakage will be into the Penetration Room which has a ventilation system designed to collect and filter leakage from penetrations. The ventilation system maintains a slightly negative pressure within the Penetration Room. Discharge from the Penetration Room Ventilating System is to the unit vent through a series filter system consisting of a prefilter, an absolute filter and a charcoal filter.

The release of radioactive materials produced by a reactor accident or waste gas tank failure are within the guidelines set by 10CFR 100.

# 3.2 CLASSIFICATION OF STRUCTURES, COMPONENTS, AND SYSTEMS

## 3.2.1 SEISMIC CLASSIFICATION

### 3.2.1.1 Structures

The design bases for normal operating conditions are governed by the applicable building design codes. The basic design criterion for the worst loss-of-coolant accident and seismic conditions is that there shall be no loss of function if that function is related to public safety.

AEC publication TID 7024, "Nuclear Reactors and Earthquake," as amplified in Chapter 3, "Design of Structures, Components, Equipment, and Systems" on page 3-1 is used as the basic design guide for seismic analysis.

The design basis earthquake ground acceleration at the site is 0.05g. The maximum hypothetical earthquake ground acceleration is 0.10g. and 0.15g for Class 1 structures founded on bedrock and overburden respectively.

The plant structures are classified as one of three classes according to their function and the degree of integrity required to protect the public.

#### 3.2.1.1.1 Class 1

Class 1 structures are those which prevent uncontrolled release of radioactivity and are designed to withstand all loadings without loss of function. Class 1 structures include the following:

Portions of the Auxiliary Building that house engineered safeguards systems, control room, fuel storage facilities and radioactive materials.

Reactor Building and its penetrations.

2

3 CT4 Transformer and 4KV Switchgear Enclosures (Blockhouses) (Reference Section 8.3.1.4.1,
3 "Auxiliary Transformers" on page 8-16.)

Unit Vent.

#### 3.2.1.1.2 Class 2

Class 2 structures are those whose limited damage would not result in a release of radioactivity and would
2 permit a controlled plant shutdown but could interrupt power generation. Class 2 structures include the
2 following:

2 Oconee Intake Structure
2 Oconee Turbine and Auxiliary Buildings, except as included in Class 1
2 Oconee Intake Canal Dike
2 Oconee Intake Underwater Weir
2 Keowee Powerhouse
2 Keowee Spillway
2 Keowee Service Bay Substructure

2 Keowee Breaker Vault
2 Keowee Intake Structure
2 Keowee Power and Penstock Tunnels
2 Keowee Dam
2 CCW Intake Piping
2 CCW Discharge Piping
2 ECCW Piping (Structural Portion outside of Turbine Building)
2 Little River Dam and Dikes

#### 3.2.1.1.3 Class 3

Class 3 structures are those whose failure could inconvenience operation, but which are not essential to power generation, orderly shutdown or maintenance of the reactor in a safe condition. They include all structures not included in Classes 1 and 2.

### 3.2.1.2 Components and Systems

Capability is provided to shutdown safely all three units in the event of a maximum hypothetical earthquake. Equipment and portions of systems that can withstand the maximum hypothetical earthquake are identified in Section 3.2.2, "System Quality Group Classification."

## 3.2.2 SYSTEM QUALITY GROUP CLASSIFICATION

This section defines the design criteria used with respect to the loss-of-coolant accident (LOCA), and natural phenomena and also explains the division of components and piping into classifications related to design and function. These criteria are as follows:

A maximum hypothetical earthquake will not result in a loss-of-coolant accident (LOCA), but the simultaneous occurrence of these events will not result in loss of function to vital safety related
5 components or systems. The simultaneous occurrence of the maximum hypothetical earthquake and a
5 LOCA is only a design criteria. A LOCA is not postulated to occur simultaneously with a maximum
5 hypothetical earthquake during accident analysis. In addition, pipe failures during a maximum
5 hypothetical earthquake are not postulated as part of the accident analysis.

A tornado will not be allowed to cause a LOCA.

A tornado does not occur simultaneously with or following a LOCA.

A tornado and earthquake do not occur simultaneously.

5 An earthquake can occur simultaneously with a loss of offsite power.

A turbine missile will not be allowed to cause a LOCA.

A turbine missile does not occur simultaneously with a LOCA.

The following design objectives result from consideration of the design criteria:

1. Loss-of-Coolant Accident

   Capability is provided to assure necessary protective actions, including reactor trip and operation of the Emergency Core Cooling System, to protect the public during a LOCA, even in the event of a simultaneously occurring maximum hypothetical earthquake.

2. Turbine Missile Accident

   The Reactor Coolant System will not be damaged by a turbine missile. Capability is provided to safely shutdown the affected units.

3. Earthquake

   The following equipment and portions of systems can withstand the maximum hypothetical earthquake:

   a. Reactor Coolant System.

   b. Borated water storage tank and piping to high pressure and low pressure injection pumps and Reactor Building spray pumps.

   c. HP injection pumps and piping to Reactor Coolant System.

   d. LP injection pumps, LP injection coolers and piping to both Reactor Coolant System and Reactor Building spray pumps.

   e. Core flood tanks and piping to Reactor Coolant System.

   f. Reactor Building spray pumps, piping to spray headers, and the spray headers.

   g. Reactor Building coolers.

   h. Low pressure service water (LPSW) pumps, LPSW piping to LP injection coolers and Reactor Building coolers and LPSW piping from these coolers to the condenser circulating water (CCW) discharge.

   2 i. CCW intake structure, CCW pumps, pump motors, CCW intake piping to the LPSW pumps,
   2 also through the condenser and emergency CCW discharge piping and CCW discharge piping.

   j. Upper surge tanks, and piping to the emergency feedwater pump.

   k. Emergency feedwater pump and turbine and auxiliary feedwater piping to the steam generators.

   l. Main steam lines to and including turbine stop valves. Turbine bypass system up thru Main Steam System isolation valves, and steam supply lines to the emergency feedwater pump turbine.

   m. Penetration Room Ventilation System.

   n. Reactor Building penetrations and piping through isolation valves.

   o. Electric power for above.

4. Tornado

   The Reactor Coolant System will not be damaged by a tornado. Capability is provided to shutdown safely all three units.

   The Reactor Coolant System, by virtue of its location within the Reactor Building, is protected from tornado damage. A sufficient supply of secondary side cooling water for safe shutdown is assured by an auxiliary service water pump located in the Auxiliary Building and taking suction from Oconee 2 CCW intake piping.

   Protected or physically separated lines are used to supply cooling water to each steam generator. One of the six sources of electric power for the pump is supplied from Keowee Hydro Station.

   An external source of cooling water is not immediately required due to the large quantities of water stored underground in the intake and discharge CCW piping. The stored volumes of water would provide sufficient cooling water for all three units for these approximate times after trip of the three reactors.

| | |
|---|---|
| Intake and discharge lines below elevation 791 ft | 37 days |
| Intake lines only below elevation 791 ft | 17 days |
| Intake and discharge lines below elevation 775 ft | 78 hours |
| Intake lines only below elevation 775 ft | 51 hours |

Furthermore, a sufficient supply of primary side makeup water is assured during a tornado initiated loss of offsite power by several backup systems.

a. The SSF Reactor Coolant Makeup Pump can take suction from the Spent Fuel Pool. The pump can be supplied power from the SSF Diesel.

b. A High Pressure Injection Pump can take suction from either the Borated Water Storage Tank or the Spent Fuel Pool. Either the "A" or "B" High Pressure Injection Pump can be powered from Keowee via the Auxiliary Service Water Pump Switchgear.

### 3.2.2.1 System Classifications

Plant piping systems, or portions of systems, are classified according to their function in meeting design objectives. The systems are further segregated depending on the nature of the contained fluid. For those systems which normally contain radioactive fluids or gases, the Nuclear Power Piping Code, USAS B31.7 and Power Piping Code USAS, B31.1.0 are used to define material, fabrication, and inspection requirements.

Diagrams for each system are included in the FSAR sections where each system is described.

Fabrication and erection of piping, fittings, and valves are in accordance with the rules for their respective classes. Welds between classes of systems (Class I to II, I to III, or II to III) are performed and inspected in accordance with the rules for the higher class. This preceding sentence does not apply to valves where the class break has been determined to occur at the valve seat, and to pipe with 1″ nominal diameter and less.

In-line instrument components such as turbine meters, flow nozzle assemblies, and control valves, etc. are classified with their associated piping unless their penetration area is equal to or less than that of a 1 inch i.d. pipe of appropriate schedule for the system design temperature and pressure, in which case they are placed in Class III. Definitions of the three classes are listed below:

Class I

This class is limited to the Reactor Coolant System. The connecting piping out to and including the first isolation valve is Class I in material, fabrication, erection, and supports and restraints. A fatigue analysis is planned to address portions of this connecting piping. Isolation valves can be either stop, relief, or check valves. Piping 1 inch and less is excluded from Class I.

Class II

Class II systems, or portions of systems, are those whose loss or failure could cause a hazard to plant personnel but would represent no hazard to the public. Class II systems normally contain radioactive fluid whose temperature is above 212°F, and in addition, those portions of Engineered Safeguards Systems outside the Reactor Building which may see recirculated reactor building sump water following a LOCA. Piping 1 inch and less is excluded.

Class III

Class III systems, or portions of systems, are those which would normally be Class II except that the contained fluid is less than 212°F. Valves, piping, instrument fittings and thermowells with a penetration area equal to or less than a 1 inch i.d. pipe or less (all schedules) are placed in Class III regardless of system temperature or pressure, when such equipment is connected to Class I, II, or III systems.

### 3.2.2.2 System Piping Classifications

System piping is divided into eight classes, depending on the required function of the system or portion of a system. These eight piping classes result from the combination of the preceding system classifications with and without design for seismic loading, as indicated in Table 3-1. Piping classes A through C meet the intent of USAS B31.7 Nuclear Power Piping Code (February 1968) and Addenda (June 1968) with the exception of those portions of the code which lack adequate definition for complete application.

Code Applicability: Due to the numerous code references located throughout this FSAR, no attempt is made to revise these references as Codes are amended, superseded or substituted. The existing Code references are the basis for design and materials; however, it is Duke Power Company's intent to comply with portions of, or all of, the latest versions of existing Codes unless material and/or design commitments have progressed to a stage of completion such that it is not practical to make a change. When only portions of Code Addenda are utilized, the appropriate engineering review of the entire addenda will be made to assure that the overall intent of the Code is still maintained. Detailed information for each station unit and code applicability with respect to design, material procurement, fabrication techniques, Nondestructive Testing (NDT) requirements and material traceability for each piping system class is described in the station piping specifications.

Table 3-1 applies uniformly to all piping except auxiliary systems in the Reactor Building. Due to schedule commitments, and concern over lack of definitive design guidance in B31.7, it was decided to use B31.1 and applicable nuclear cases in the Reactor Building, but the materials were bought, erected, and inspected to the standards set down in B31.7. The Reactor Coolant System was designed to B31.7, Class I.

### 3.2.2.3 System Valve Classifications

In the absence of definitive codes, the non-destructive testing criteria applied to system valves are consistent with the intent of Par. 1-724 of USAS B31.7 Nuclear Power Piping Code (Feb. 1968) and the piping classification applicable to that portion of the system which includes the valve. On this basis, valves are grouped into the same eight classes as shown for piping in Table 3-1, and a valve is in the same class as the portion of system piping which includes the valve.

### 3.2.2.4 System Component Classification

In the absence of definitive codes, the design criteria applied to pressure retaining system components are generally consistent with the intent of Sections III and VIII of the ASME Boiler and Pressure Vessel Code, the piping system classification applicable to that portion of the system which includes the component, and the required function of the component. Atmospheric water storage tanks important to safety conform to American Waterworks Association Standard for Steel Tanks, Standpipes, Reservoirs and Elevated Tanks for Water Storage, D100, or equivalent.

Components are listed by system in Table 3-2. This tabulation shows the code to which the component was designed, whether the component was designed to withstand the seismic load imposed by the maximum hypothetical earthquake, and the analytical technique employed in seismic analysis.

# 3.3 WIND AND TORNADO LOADINGS

All Class 1 structures, except those structures not exposed to wind, are designed to withstand the effects of wind and tornado loadings, without loss of capability of the systems to perform their safety functions.

## 3.3.1 WIND LOADINGS

### 3.3.1.1 Design Wind Velocity

The design wind velocity for all Class 1 structures is 95 mph. This is the largest wind velocity for a 100-year occurrence as shown in Figure 1(b) of Reference 1 on page 3-45.

### 3.3.1.2 Determination of Applied Forces

The applied wind pressures are computed by the means outlined in ASCE Paper 3269 which states that the equivalent static force on a building is equal to the dynamic pressure (q) times the drag coefficient ($C_D$) multiplied by the elevation area. The dynamic pressure is the product of one-half the air density and the square of the velocity (the kinetic energy per unit volume of moving air). For air at 15° C at 760 mm Hg: $q = 0.002558\ V^2$ with q in psf and V in mph. The drag coefficient is based on test data and tabulated in Reference 1 on page 3-45. For these high wind velocities, this equation may be excessively conservative, but no credit is taken for this possible pressure reduction.

## 3.3.2 TORNADO LOADINGS

All Class 1 structures, except those structures not exposed to wind, are designed for tornado loads.

### 3.3.2.1 Applicable Design Parameters

Simultaneous external loadings used in the tornado design are:

a. Differential pressure of 3 psi developed over 5 seconds.

b. External wind forces resulting from a tornado having a velocity of 300 mph.

The spectrum and characteristics of tornado-generated missiles is covered in Section 3.5.1.3, "Missiles Generated by Natural Phenomena" on page 3-56.

### 3.3.2.2 Determination of Forces on Structures

Tornado wind loadings are calculated in accordance with Section 3.3.1.2, "Determination of Applied Forces," using the tornado wind velocities given in Section 3.3.2.1, "Applicable Design Parameters." The tornado loading combination used for design of Class 1 structures is:

$$Y = \frac{1}{\phi}(1.0D + 1.0W_t + 1.0P_i)$$

Where Y, $\phi$, and D are as defined in Table 3-14.

| | | |
|---|---|---|
| $W_t$ | = | Stress induced by design tornado wind velocity (drag, lift and torsion) |
| $P_i$ | = | Stress due to differential pressure |

Shape factors will be applied in accordance with ASCE Paper 3269. No height or gust factors will be used with tornado loadings.

### 3.3.2.3 Effect of Failure of Structures or Components Not Designed for Tornado Loads

The Reactor Coolant System will not be damaged by a tornado. Capability is provided to shutdown safely all three units.

The Reactor Coolant System, by virtue of its location within the Reactor Building, is protected from tornado damage. A sufficient supply of cooling water for safe shutdown is assured by an auxiliary service water pump located in the Auxiliary Building and taking suction from Unit 2 CCW intake piping.

### 3.3.2.4 Wind Loading for Class 2 and 3 Structures

The wind loads are determined from the largest wind velocity for a 100-year occurrence as shown in Figure 1(b) of Reference 1 on page 3-45. This is 95 mph at the site.

## 3.3.3 REFERENCES

1. *Wind Forces on Structures*, Task Committee on Wind Forces, ASCE Paper No. 3269.

# 3.4 WATER LEVEL (FLOOD) DESIGN

## 3.4.1 FLOOD PROTECTION

### 3.4.1.1 Flood Protection Measures for Seismic Class 1 Structures

The plant yard elevation is 796.0 ft. msl. All of the man-made dikes and dams forming the Keowee Reservoir rise to an elevation of 815.0 ft. msl with a full pond elevation of 800.0 ft. msl. However, Class 1 structures and components are not subject to flooding since the Probable Maximum Flood (PMF) would be contained by the Keowee Reservoir. The minimum external access elevation for the Auxiliary, Turbine, and Service Buildings is 796.5 ft. msl which provides a 6 inch water sill. Also, the plant site is provided with a surface water drainage system that protects the plants facilities from local precipitation.

#### 3.4.1.1.1 Current Flood Protection Measures for the Turbine and Auxiliary Buildings

In the current Turbine Building flood handling analysis, it was found that there exists a remote possibility of flooding in the Turbine Building at the basement level due to failure of expansion joints in the Condenser Circulating Water System near the condenser water box inlet or outlet nozzles.

Condenser circulating water intake and discharge pipes are embedded in the Turbine Building substructure mat at points immediately below the inlet or outlet connections on the condenser inlet and outlet water boxes. At each waterbox connection, a 78 inch steel pipe is turned up and projected above the basement level and connected to a butterfly valve. A rubber expansion joint is located between each valve and waterbox connection. The rubber joint spans across a 4¼ inch physical gap in the 78 inch intake pipe and across a 2 inch physical gap in the 78 inch discharge. At maximum flow conditions through any condenser, a complete rupture of the 4¼ inch intake pipe point (all rubber removed) would result in a 235 cfs leak into the Turbine Building basement area. This is the worst case leak condition due to the higher head and wider possible gap situation that exists on the intake side of the condenser.

Each foot of depth in this 202 feet wide by 790 feet long structure contains a volume of 160,000 cubic feet. Therefore, a joint rupture would fill the Turbine Building at the rate of 0.088 feet per minute until the water surface reaches the height of the rupture and a reduced rate thereafter due to reduced differential head conditions, provided all flood water could be contained in the Turbine Building.

Curbs 1.75 feet high have been provided around doorway entrances to the Auxiliary Building from the Turbine Building to contain flood water in the Turbine Building until action is taken to control flooding. (This will provide 20 minutes storage in the Turbine Building basement.) Turbine Building sump, moisture separator drain pits, and Hotwell Pump Pit level alarms will alert the control room operators.

The Turbine/Auxiliary Building wall along column line "N" is capable of withstanding a flood to a depth of 20 ft. above elevation 775 + 0. Six doors originally located on this wall have been made flood barriers. Three of the doors are permanently sealed while the remaining three have been replaced with "submarine type" flood doors. All other penetrations through the wall to elevation 795 + 0 have been sealed.

A Turbine Building Flood Statalarm is provided in each control room to indicate flood conditions in the Turbine Building basement. This alarm has a 2 out of 3 logic, a range of 0 to 7 feet, and sensitivity of ±1.5 inches. The operator is provided adequate time to confirm the flood conditions by visual inspection and initiate the appropriate valve operation to control the flooding or initiate a reactor trip. Each half of

each condenser shell of each unit can be isolated from the remainder of the cooling water system without unit shutdown in the event of joint failure.

5 A push button in each control room provides capability to close the Condenser Circulating Water (CCW) pump discharge valves to protect against CCW siphoning into the turbine building basement. This flood mitigation station modification has been installed pursuant to the recommendations made in the Oconee Probabilistic Risk Assessment Study.

#### 3.4.1.1.2 Flood Protection Measures Inside Containment

The primary means for detecting leakage in the Reactor Building is the level indication for the normal sump. This indication has a range of 0-to-30 inches, with a statalarm occurring at 15 inches increasing level and a computer alarm at approximately 22 inches. These alarms would alert the operators in the control room such that appropriate actions could be taken. In addition to the alarms, sump level is input to the plant computer and is typed out on the utility typer every five minutes. Level is also recorded on a trend recorder in each control room. Safety related redundant level transmitters with a range of 3 inches to 24 inches are also provided in the normal sump. Both transmitter levels are indicated in the control room on receiver gauges and one train is recorded. Thus, the operators have several methods for monitoring changes in sump level.

The sump fill rate is routinely measured to determine leakage rate. The sump capacity is 15 gallons per inch of height and each graduation on the indicator level indicates 7.5 gallons of leakage into the sump. A 1 gal/min leak would therefore be detectable within less than 10 minutes.

In addition to the normal sump level, indication of the emergency sump level is also provided by redundant safety related systems with a range of 0 to 3 feet. Both trains of instrumentation are indicated on receiver gauges in the control room and one train is recorded. This indication can be used in conjunction with the normal sump level indication to detect abnormal leakage in the Reactor Building. Two additional trains of containment level transmitters are installed in each Reactor Building to provide wide range level indication and recording with a range of 0 to 15 feet.

The normal sump is routinely pumped to the miscellaneous waste holdup tanks whenever the alarm point (15 inches) is reached. Pumping of the sump water is started manually, but terminates automatically when the sump level has dropped to 6 inches (which clears the statalarm). Each time the sump is pumped, it is recorded in the Unit Reactor Operator's Log Book. During pumping, a decreasing sump level indication and/or increasing miscellaneous waste holdup tank level indication can be used to verify flow from the normal sump. The flow rate from the sump can be determined using the rate of change in sump level.

In order to provide periodic monitoring of sump levels, the recording of normal and emergency sump levels is done daily. Daily monitoring of level indications is useful in confirming that level instrumentation are operable, while verifying the sump pumps are operable and maintaining the sump level at or below the alarm point. Calibration of the normal and emergency sump indications is performed during refueling.

In the event of increased leakage to the Reactor Building, sampling may be performed to determine the origin of the leakage (e.g., LPSW, feedwater, component cooling, or RC system).

Leakage from the LPSW system in containment can also be detected by the monitoring of other parameters. For example, the inlet and outlet LPSW flows for each Reactor Building Cooling Unit (RBCU) are monitored for any differences which could be indicative of a cooler leak. If a flow difference is detected, an alarm is provided to the control room. The operator can then promptly isolate the affected cooler by closing remote operated valves.

The Reactor Coolant Pump (RCP) motor parameters are also continuously monitored. A leak in the motor stator winding cooler would be alarmed in the control room. A leak in either of the motor bearing oil coolers could be detected by changing motor temperature in conjunction with increasing sump level. The pump could then be stopped and the cooling water isolated from the control room.

In-leakage of reactor coolant is detected by radiation monitor and an increase in surge tank level which will be annunciated. Out-leakage from the system will result in a decreasing surge tank level which is annunciated. Volume of the surge tank is 50 $ft^3$ and allows relatively small volumes of in-leakage or out-leakage to be observed.

## 3.4.2 REFERENCES

1. Elevations taken from Figure 2-2 of FSAR and Oconee FSAR 2.2.6.
2. Response to Question of Effects of Failure of Non-Category I Equipment, Oconee FSAR, Supplement 13 of January 29, 1973, Item No. 7347. Information received from Steam Department.
3. Response to Bulletin 80-24 on Cooling Systems Inside Containment, Attachment to Mr. W. O. Parker, Jr.'s letter of January 6, 1981, Item No. 760. Information received from Steam Department.

# 3.5 MISSILE PROTECTION

## 3.5.1 MISSILE SELECTION AND DESCRIPTION

### 3.5.1.1 Internally Generated Missiles (Inside Containment)

The major components including reactor vessel, reactor coolant piping, reactor coolant pumps, steam generators, and the pressurizer are located within three shielded cubicles. Each of two cubicles contain one steam generator, two coolant pumps, and associated piping. One of the cubicles also contains the pressurizer. The reactor vessel is located within the third cubicle or primary shield. The reactor vessel head and control rod drives extend into the fuel transfer canal.

Penetrations in the generators, piping, and the pressurizer are located such that missiles which may be generated, such as valves, valve bonnets, valve stems, or reactor coolant temperature sensors will not escape the cubicles or possess sufficient energy to damage the Reactor Building liner plate.

Openings are provided in the lower shield walls to provide vent area. To assure that no missile will impact on the Reactor Building liner plate, concrete shielding is provided for the liner plate area opposite the openings. The shielding extends beyond the openings so that any missile will impact on the shields.

Pipe lines carrying high pressure injection water are routed outside the shield walls entering only when connecting to the loop. Missiles which may be generated in one cubicle cannot rupture high pressure injection lines for the other loop. Low pressure injection lines and core flooding lines are routed outside of the shield walls, behind missile shield walls, and through the primary shield where they enter the reactor vessel. They are, thus, protected from missiles which might be generated in either cubicle.

A concrete missile shield is located above the control rod drives to stop a control rod drive should it become a missile. The shield is removed during refueling.

The reactor cavity annulus seal ring is analyzed as a potential missile following a postulated pipe rupture inside the reactor vessel cavity. The analysis indicates that missile shielding located over the reactor vessel is adequate to prevent this postulated missile from reaching the containment.

Items that could become missiles are oriented so they impinge on concrete surfaces.

Analysis of the missile penetration is based on the methods described in Nav. Docks P-51, Design of Protective Structures by Amirikan (Bureau of Yards and Docks, August 1950).

The penetration formulae are:

$$D = k\,ApV'$$

where:

$$V' = \log_{10} 1 + \frac{V^2}{215000}$$

$$K = \frac{D^1}{D} = 1 + e^{-4(a^1 - 2)}$$

where:

$$a^1 = \frac{T}{D}$$

where:

D = Penetration in a slab of infinite thickness (ft.)

$D^1$ = Penetration in a slab of thickness "T" (ft.)

T = Thickness of slab (ft.)

Ap = Sectional pressure, obtained by dividing the weight of missile by its cross sectional area (psf)

V = Velocity of missile (fps)

k = Material's coefficient, in our case, $k = 2.30 \times 10^{-3}$ for reinforced concrete

Formulae for determining energy loss due to drag:

$$\frac{Ti}{Tc} = \frac{1}{1 + \frac{2Tc}{WL}};$$

$$L = \frac{2W}{SAC_d}$$

5 **Note:** The above equation was revised in 1995 update.

where:

A = Average area

$C_d$ = Drag coefficient (Cd = 1.0 in our case)

Ti = Kinetic energy on impact

Tc = Kinetic energy after leaving casing

W = Weight in lbs.

S = Air density = 0.074 #/ft$^3$

In addition to the penetration calculation, the overall structural strength of the removable concrete slabs, its supports and anchors are analyzed based on the research paper "Impact Effect of Fragments Striking Structural Elements" by R. A. Williamson and R. R. Alvy.

The following three missiles are used to design the removable concrete slabs:

| Description | Wt. Lbs. | Imp. Area $In^2$ | Velocity FPS | Kin. Energy Ft-lbs. |
|---|---|---|---|---|
| C. R. Drive Assembly | 1500 | 64.0 | 254 | $1.49 \times 10^6$ |
| CRD Vent Cap w/Valve | 55 | 13.4 | 546 | $0.12 \times 10^6$ |
| CRD Motor and Clutch Assem. | 750 | 47.0 | 483 | $1.35 \times 10^6$ |

The properties of other missiles postulated by the Nuclear Steam System Supply (NSSS) vendor are given in Table 3-3 to Table 3-9.

Missile protection is provided to comply with the following criteria:

1. The Reactor Building and liner are protected from loss of function due to damage by such missiles as might be generated in a loss-of-coolant accident for break sizes up to and including the double-ended severance of a main coolant pipe.
2. The engineered safeguards system and components required to maintain Reactor Building integrity are protected against loss of function due to damage by the missiles defined below.

During the detailed plant design, the missile protection necessary to meet the above criteria was developed and implemented using the following methods:

1. Components of the Reactor Coolant System are examined to identify and to classify missiles according to size, shape and kinetic energy for purposes of analyzing their effects.
2. Missile velocities are calculated considering both fluid and mechanical driving forces which can act during missile generation.
3. The Reactor Coolant System is surrounded by reinforced concrete and steel structures designed to withstand the forces associated with double-ended rupture of a main coolant pipe and designed to stop missiles.
4. The structural design of the missile shielding takes into account both static and impact loads and is based upon the state of the art of missile penetration data.

The types of missiles for which missile protection is provided are:

1. Valve stems.
2. Valve bonnets.
3. Instrument thimbles.
4. Various types and sizes of nuts and bolts.

Protection is not provided for certain types of missiles for which postulated accidents are considered incredible because of the material characteristics, inspections, quality control during fabrication, and conservative design as applied to the particular component. Included in this category are missiles caused by massive, rapid failure of the reactor vessel, steam generator, pressurizer, main coolant pump casings and drives.

### 3.5.1.2 Turbine Missiles

The turbine-generator supplier has made a study of failure of rotating elements of steam turbines and generators. The postulated types of failures are: (1) failure of rotating components operating at or near normal operating speed and, (2) failure of components that control admission of steam to the turbine resulting in destructive shaft rotational speed.

#### 3.5.1.2.1 Failure at or Near Operating Speed

All of the known turbine and generator rotor failures at near rated speed resulted from the combination of severe strain concentrations in relatively brittle materials. New alloys and processes have been developed and adopted to minimize the probability of brittle fracture in rotors, wheels, and shafts. Careful control of chemistry and detailed heat treating cycles have greatly improved the mechanical properties of all of these components. Transition temperatures (the temperature at which the character of the fracture in the steel changes from brittle to ductile, often identified as FATT) have been reduced on the low temperature wheel and rotor applications for nuclear units to well below startup temperatures. Improved steel mill practices in vacuum pouring and alloy addition have resulted in forgings which are much more uniform and defect free than ever before. More comprehensive vendor and manufacturer tests involving improved ultrasonic and magnetic particle testing techniques are better able to discover surface and internal defects than in the past. Laboratory investigation has revealed some of the basic relationships between structure strength, material strength, FATT and defect size, and location so that the reliability of the rotor as a structure has been significantly improved over the past few years.

New starting and loading instructions have been developed to reduce the severity of surface and bore thermal stress cycles incurred during service. The new practices include:

1. Better temperature sensors.
2. Better control devices for acceleration and loading.
3. Better guidance for station operators in the control speed, acceleration, and loading rates to minimize rotor stresses.

Progress in design, better materials and quality control, more rigorous acceptance criteria, and improved machine operation have substantially reduced the likelihood of burst failures of turbine-generator rotors operating at or near rated speed.

#### 3.5.1.2.2 Failure at Destructive Shaft Rotational Speeds

Improvements of rotor quality discussed above, while reducing the chance of failures at operating speed, tend to increase the hazard level associated with unlimited overspeed because of higher bursting speed. Therefore, turbine overspeed protection systems have been evaluated as follows:

1. Main and secondary steam inlets have the following valves in series:
   a. Control valves - controlled by the speed governor and tripped closed by emergency governor and backup overspeed trip, thus providing three levels of control redundancy.
   b. Stop valves or trip throttle valve - actuated by the emergency governor and backup overspeed trip, thus providing two levels of control redundancy.

      Since 1948 there have been over 650 turbines, of over 10,000 kw each, placed in service by the Oconee turbine supplier with no report of main stop valves failing to close when required to protect the turbine. Impending sticking has been disclosed by means of the fully closed test feature so that a planned shutdown could be made to make the necessary correction. This almost always involves the removal of the oxide layer which builds up on the stem and bushing and which would not occur on a low temperature nuclear application.
   c. Combined stop and intercept valves in cross around systems - these are actuated by the speed governor, emergency, and backup overspeed trips. These valves also include the testing features described above.

   The speed sensing devices for the governor and emergency governor are separate from each other, thus providing two independent lines of defense.

2. Uncontrolled Extraction Lines to Feedwater Heaters

   If the energy stored in an uncontrolled extraction line is sufficient to cause a dangerous overspeed, two positive closing nonreturn valves are provided, to be actuated by the emergency governor and backup overspeed trip. These are designed for remote manual periodic tests to assure proper operation. The station piping, heater, and check valve system are reviewed during the design stages to make sure the entrained steam cannot overspeed the unit beyond safe limits.

Special field tests are made of new components to obtain design information and to confirm proper operation. These include the capability of controls to prevent excessive overspeed on loss of load.

Careful analysis of all past failures has led to design, inspection, and testing procedures to substantially eliminate destructive overspeed as a possible cause of failure in modern design units.

The study of postulated ruptures made by the turbine-generator supplier concludes that the missile having the highest combination of weight, size, and energy is the last stage wheel. The properties of this missile are summarized in Table 3-10. Initial velocities and energies shown in the table are based on 180 percent overspeed. As the missile penetrates the casing, 50 percent of the initial energy is considered absorbed in the casing.

Analysis of the above missile is based on calculations using methods presented in Reference 1 on page 3-57 to determine the depth to which this missile would penetrate the concrete Reactor Building. Conservatively, no reduction of missile energy is made for penetration of the Turbine Building and/or impact with intervening equipment and structural components after leaving turbine shell. The energy loss from 23.25 x $10^6$ ft-lbs to 18.0 x $10^6$ ft-lbs is caused by air friction. This effect has been calculated by using a drag coefficient of 1.0. Since the offset between the Turbine and Reactor Buildings is relatively short, about 170 feet, no account has been taken for air friction losses for the case in which the missile is ejected nearly horizontally to strike the cylinder wall. Following are results of analysis:

Case I:

"Side on" impact. Missile could penetrate the concrete cylinder wall to a depth of approximately 6 inches and the dome to a depth of approximately 5½ inches. The tendons will not be damaged since they are protected to a depth of 7 3/4 inches in the cylinder wall and 8 inches in the dome.

Case II:

"End on" impact. In this case the missile could penetrate the concrete cylinder wall to a depth of approximately 13 3/4 inches and the dome to a depth of approximately 12¼ inches. The tendon arrangement is such that the missile could strike two adjacent tendons in the dome or a maximum of three horizontal and one vertical tendons in the cylinder wall. The local effect on the tendons could be one of either partial deflection or possible severance. However, analysis of the structure indicates that the structure can withstand the loss of three horizontal and three vertical tendons in the cylinder wall or five adjacent tendons in the dome without loss of function and a greater number of tendons without building failure.

Case III:

As a final analysis, an extreme case is considered in which none of the initial kinetic energy of the missile is absorbed by its penetration through the turbine casing. The total initial energy of 46.5 x $10^6$ ft-lbs is available for penetration of the cylinder wall and 29.3 x $10^6$ ft-lbs for penetration of dome where the reduction is due to air friction only. The maximum depth of penetration of cylinder wall is 35½ inches and the dome is 25 inches. The missile can strike five tendons in the dome or three horizontal and one

vertical tendons in the cylinder wall. The local effect in the impact area would be as described in Case II above even though the depth of penetration is greater.

Depths of penetration of Reactor Building wall are summarized in Table 3-11.

Since the thicknesses of the cylinder wall and dome are 45 inches and 39 inches respectively, it can be seen that the turbine missile, even under extreme assumptions, does not penetrate the Reactor Building.

### 3.5.1.3 Missiles Generated by Natural Phenomena

For an analysis of missiles created by a tornado having maximum wind speeds of 300 mph, two missiles are considered. One is a missile equivalent to a 12 foot long piece of wood 8 inches in diameter traveling end on at a speed of 250 mph. The second is a 2000 pound automobile with a minimum impact area of 20 square feet traveling at a speed of 100 mph.

For the wood missile, calculations based on energy principle indicate that because the impact pressure exceeds the ultimate compressive strength of wood by a factor of about four, the wood would crush due to impact. However, this could cause a secondary source of missiles if the impact force is sufficiently large to cause spalling of the free (inside) face. The compressive shock wave which propagates inward from the impact area generates a tensile pulse, if it is large enough, will cause spalling of concrete as it moves back from the free (inside) surface. This spalled piece moves off with some velocity due to energy trapped in the material. Successive pieces will spall until a plane is reached where the tensile pulse becomes smaller than the tensile strength of concrete. From the effects of impact of the 8 inch diameter by 12 foot long wood missile, this plane in a conventionally reinforced concrete section would be located approximately 3 inches from the free (inside) surface. However, since the Reactor Building is prestressed, there will be residual compression in the free face, as the tensile pulse moves out and spalling will not occur. Calculations indicate that in the impact area a 2 inch or 3 inch deep crushing of concrete should be expected due to excessive bearing stress due to impact.

For the automobile missile, using the same methods as in the turbine failure analysis, the calculated depth of penetration is ¼ inch and for all practical purposes the effect of impact on the Reactor Building is negligible.

From the above, it can be seen that the tornado generated missiles neither penetrate the Reactor Building wall nor endanger the structural integrity of the Reactor Building or any components of the Reactor Coolant System.

## 3.5.2 BARRIER DESIGN PROCEDURES

The Reactor Building and Engineered Safeguards Systems components are protected by barriers from all credible missiles which might be generated from the primary system. Local yielding or erosion of barriers is permissible due to jet or missile impact provided there is no general failure.

The final design of missile barrier and equipment support structures inside the Reactor Building is reviewed to assure that they can withstand applicable pressure loads, jet forces, pipe reactions and earthquake loads without loss of function. The deflections or deformations of structures and supports are checked to assure that the functions of the Reactor Building and engineered safeguards equipment are not impaired. Missile barriers are designed on the basis of absorbing energy by plastic yielding.

## 3.5.3 REFERENCES

1. Amirikian, A., Design of Protective Structures, Bureau of Yards and Docks, Department of the Navy, *NAVDOCKS P-51*, 1950.
2. Alvy, R. R., and Willimson, R. A., "Impact Effect of Fragments Striking Structural Elements."

# 3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

## 3.6.1 POSTULATED PIPING FAILURES IN FLUID SYSTEMS INSIDE AND OUT SIDE CONTAINMENT

### 3.6.1.1 Design Bases

The basic design criteria for pipe whip protection is as follows:

1. All penetrations are designed to maintain containment integrity for any loss of coolant accident combination of containment pressures and temperatures.
2. All penetrations are designed to withstand line rupture forces and moments generated by their own rupture as based on their respective design pressures and temperatures.
3. All primary penetrations, and all secondary penetrations that would be damaged by a primary break, are designed to maintain containment integrity.
4. All secondary lines whose break could damage a primary line and also breach containment are designed to maintain containment integrity.

### 3.6.1.2 Description

The major components including reactor vessel, reactor coolant piping, reactor coolant pumps, steam generators, and the pressurizer are located within three shielded cubicles. Each of two cubicles contain one steam generator, two coolant pumps, and associated piping. One of the cubicles also contains the pressurizer. The reactor vessel is located within the third cubicle or primary shield. The reactor vessel head and control rod drives extend into the fuel transfer canal.

Openings are provided in the lower shield walls to provide vent area. Pipe lines carrying high pressure injection water are routed outside the shield walls entering only when connecting to the loop.

### 3.6.1.3 Safety Evaluation

The analysis of effects resulting from postulated piping breaks outside containment is contained in Duke Power MDS Report No. OS-73.2, dated April 25, 1973 including Supplement 1, dated June 22, 1973.

An evaluation of potential non-safety grade control system interactions during design basis high energy line break accidents is contained in the Duke Power/Babcock and Wilcox Report dated October 5, 1979.

## 3.6.2 REFERENCES

1. Duke Power Mechanical Design Section Report OS-73-2, April 25, 1973, including Supplement 1, June 22, 1973.
2. Duke Power/B&W Report, Oconee Nuclear Station, "Evaluation of Potentially Adverse Environmental Effects on Non-Safety Grade Control Systems", October 5, 1979.

# 3.7 SEISMIC DESIGN

## 3.7.1 SEISMIC INPUT

### 3.7.1.1 Design Response Spectra

The design response spectra curves for the 0.05g Design Base Earthquake (DBE), the 0.10g Maximum Hypothetical Earthquake (MHE) for Class 1 Structures founded on rock, and the 0.15g Maximum Hypothetical Earthquake (MHE) for structures founded on overburden are given in Figure 2-51, Figure 2-53, Figure 2-55, respectively.

### 3.7.1.2 Design Time History

The Time History record of the N-S, May 1940 El Centro earthquake is used (vertical and N-S horizontal components).

### 3.7.1.3 Critical Damping Values

The following damping values are used for the seismic design of Class 1 structures:

| Item | Percent of Critical Damping |
|---|---|
| Welded carbon and stainless steel assemblies (This includes reactor internals, supports and similar weldments.) | 1 |
| Steel frame structures (Both welded and high strength bolted) | 2 |
| Reinforced concrete equipment supports | 2 |
| Reinforced concrete frames and buildings | 5 |
| Prestressed concrete structures under design earthquake forces | 2 |
| Prestressed concrete structures under maximum hypothetical earthquake | 5 |
| Vital piping | 0.5 |

### 3.7.1.4 Supporting Media for Seismic Class 1 Structures

The supporting media for each seismic Class 1 structure are defined in Section 2.5, "Geology, Seismology, and Geotechnical Engineering" on page 2-47.

## 3.7.2 SEISMIC SYSTEM ANALYSIS

### 3.7.2.1 Seismic Analysis Methods

#### 3.7.2.1.1 Reactor Building

Seismic loading of the structure controls in all cases over that of tornado or wind loading. The seismic analysis is conducted in the following manner: The loads on the Reactor Building caused by earthquake are determined by a dynamic analysis of the structure. The dynamic analysis is made on an idealized structure of lumped masses and weightless elastic columns acting as spring restraints. The analysis is

performed in two stages: the determination of the natural frequencies of the structure and its mode shapes, and the response of these modes to the earthquake by the spectrum response method.

#### 3.7.2.1.2 Auxiliary Building

In determining the response of the building to the earthquake the spectrum response technique is utilized. For this technique the earthquake is described by spectrum response curves presented in Figure 2-51 and Figure 2-53. From the curves, acceleration levels are determined as associated with the natural frequency and damping value of each mode. The standard spectrum response technique uses these values to determine inertial forces, shears, moments, and displacements per mode. These results are then combined on the basis of the absolute sum to obtain the structural response. The process is accomplished by the Bechtel computer program, CE641.

#### 3.7.2.1.3 Turbine Building

Seismic analysis of Turbine Building is discussed in Section 3.8.5.4, "Design and Analysis Procedures" on page 3-135.

### 3.7.2.2 Natural Frequencies and Response Loads

#### 3.7.2.2.1 Reactor Building

The natural frequencies and mode shapes are computed using the matrix equation of motion shown below for a lumped mass system. The form of the equation is:

$$(K)(\Delta) = \omega^2(M)(\Delta)$$

| | | |
|---|---|---|
| K | = | matrix of stiffness coefficients including the combined effects of shear, flexure, rotation, and horizontal translation. |
| M | = | matrix of concentrated masses. |
| $\Delta$ | = | matrix of mode shape |
| $\omega$ | = | angular frequency of vibration. |

The results of this computation are the several values of $\omega_n$ and mode shapes $\Delta_n$ for n = 1, 2, 3, . . . m, where m is the number or degrees of freedom (i.e., lumped masses) assumed in an idealized structure.

#### 3.7.2.2.2 Auxiliary Building

The natural frequencies and mode shapes of the structure are obtained by the Bechtel computer program, CE617. This program utilizes the flexibility coefficients and lumped weights of the model. The flexibility coefficients are formulated into a matrix and inverted to form a stiffness matrix. The program then uses the technique of diagonalization by successive rotations to obtain the natural frequencies and mode shapes. The results are shown in Figure 3-1.

### 3.7.2.3 Procedure Used for Modeling

#### 3.7.2.3.1 Reactor Building

The modeling of the Reactor Building is discussed in Section 3.7.2.4, "Development of Floor Response Spectra" on page 3-63.

#### 3.7.2.3.2 Auxiliary Building

The mathematical model of the structure is constructed in terms of lumped masses and stiffness coefficients. At appropriate locations within the building, points are chosen to lump the weights of the structure. Between these locations properties are calculated for moments of inertia, cross sectional areas, effective shear areas, and lengths. A sketch of the model is shown on Figure 3-3. The properties of the model are utilized in the IBM computer program, STRESS, along with unit loads to obtain the flexibility coefficients of the building at the mass locations. In Figure 3-4 are presented the moments, shears, displacements, and accelerations for the model subjected to 0.05 g ground motion and 5 percent damping.

#### 3.7.2.3.3 Turbine Building

This information is outlined in Section 3.8.5.4, "Design and Analysis Procedures" on page 3-135.

### 3.7.2.4 Development of Floor Response Spectra

#### 3.7.2.4.1 Reactor Building

The actual structural system is idealized as a mathematical model in form of a lumped mass system interconnected by elastic members. Lumped masses, which are a summation of structure and equipment masses, are located at pertinent floor levels and at other levels where response spectra are desired. These other levels would include equipment support elevations, pipe support elevations, etc.

In the case of the Reactor Building, two mathematical models are generated to describe the complete Reactor Building. The first model represents the Reactor Building shell, and the second model represents the internal structure. Modifications to each model are required to determine response from ground motions. After these models are developed, the following procedure is employed for all the models:

The flexibility matrix of the structural system is determined by using Bechtel program CE309. The procedure for each loading condition is to apply a unit load at each mass point and determine the deflection at all mass points.

The mode shapes and frequencies for the lumped mass systems are obtained by means of either of two Bechtel computer programs; CE548, "Symbolic Matric Interpretive System" or CE617, "Diagonalization Method for Eigenvalues and Eigenvectors."

The Time-History record of the N-S, May 1940 El Centro earthquake is used (vertical and N-S horizontal components). Using the mode shapes and frequencies and the time history (time vs. acceleration record), properly scaled, the time history of the accelerations, velocities and displacements of the lumped masses are obtained. Bechtel program CE611 is utilized for this computation.

The acceleration time-history is applied at the base of a single degree of freedom system. Initially the system is set with specific values for its natural frequency and damping. The time-history response of the mass is determined and examined for the value of maximum acceleration. The same process is repeated over a range of natural frequencies. The resulting maximum G levels and frequencies are tabulated and plotted into the spectrum curve for a single structure elevation. The resulting curve is labeled with the damping value. The process is repeated for required structure elevations and damping values. Bechtel computer program CE591, "Spectral Analysis," is used to obtain the acceleration and velocity response spectra at each floor for each percentage of damping required.

A sample of the acceleration spectrum curves at different floor levels of a building is shown in Figure 3-6. For these curves, the horizontal axis is logarithmic in cycles per second and the vertical axis is linear in G's. The curves are for 1½ percent of critical damping. The building has natural frequencies of 4.8 cps

at the first mode and 10 cps at the second mode. Thus maximum accelerations occur between 1.0 cps and 10.0 cps. At the far right end, the curve converges on the peak value of the input earthquake as the single degree of freedom system becomes rigid, relative to the seismic excitation. At progressively higher locations, the building amplifies the input earthquake, especially in the vicinity of its natural frequencies. Note the sharp peak in each curve at the natural frequency of the building.

When using response curves for piping systems which are located at different elevations, it is necessary to superimpose several curves and plot the envelope curve for the system inputs. At the maximum acceleration peak of each specific curve used for the envelope curve, the envelope has a plateau of approximately $\pm 10$ percent to avoid the condition where a small change in frequency could result in a significant change in acceleration. Through the ME 601 program, the natural frequency and mode shapes of the pipe are found and combined with the spectrum curves to find the seismic forces on the pipe.

#### 3.7.2.4.2 Auxiliary Building

The spectrum response curves for equipment inside the building are generated by the time history technique of seismic analysis. The sample earthquake utilized is that recorded at El Centro, California, N-S, May 18, 1940. Essentially the curves are generated by applying the recorded earthquake to the structure and obtaining the time history at selected mass points. Each of these time histories is then applied to a single degree of freedom system of which the values for damping and natural frequency are varied. The curves for Units 1 and 2 Auxiliary Buildings are accomplished by the Bechtel program, CE611. The curves for Unit 3 Auxiliary Building are generated by Duke. The spectrum curves were generated for both directions (East-West and North-South). At the high frequency end of the curve, the acceleration levels converge to the value of the location inside the building.

Digital computer program, CE 617, CE 641, CE 611, and CE 591 are proprietary programs of the Bechtel Corporation.

### 3.7.2.5 Components of Earthquake Motion

Seismic forces are applied in the vertical and in any horizontal direction. The horizontal and vertical components of ground motion are applied simultaneously.

### 3.7.2.6 Combination of Modal Responses

#### 3.7.2.6.1 Reactor Building

The response of each mode of vibration to the design earthquake computed by the response spectrum technique, as follows:

1. The base shear contribution of the $n^{th}$ mode $V_n = W_n S_{an}(\omega_n \Upsilon)$ where:

   $W_n$ = effective weight of the structure in the $n^{th}$ mode.

   $$W_n = \frac{(\Sigma_x \Delta_{xn} w_x)^2}{\Sigma_x (\Delta_{xn})^2 w_x},$$

   where the subscript x refers to levels throughout the height of the structure, and $w_x$ is the weight of the lumped mass at level x.

   $\omega_n$ = angular frequency of the $n^{th}$ mode.

   $S_{an}(\omega_n \Upsilon)$ = spectral acceleration of a single degree of freedom system with a damping coefficient of $\Upsilon$, obtained from the response system.

2. The horizontal load distribution for the $n^{th}$ mode is computed as:

$$F_x = V_n \frac{(\Delta_{xn} w_x)}{\Sigma_x \Delta_{xn} w_x}$$

5 **Note:** The above equation was revised in 1995 update.

The several mode contributions are then combined to give the final response of the structure to the design earthquake.

3. The number of modes to be considered in the analysis is determined to adequately represent the structure being analyzed. The analytical model and results for the 0.05 g earthquake and 2 percent damping and for the 0.1 g earthquake and 5 percent damping are shown in Figure 3-7.

#### 3.7.2.6.2 Auxiliary Building

For description of combining of modal responses, see Section 3.7.2.1, "Seismic Analysis Methods" on page 3-61.

### 3.7.2.7 Method Used to Account for Torsional Effects

0 Torsional modes are not considered in the seismic analysis. Insignificant torsional shear stresses exist, assuming a minimum of 10 percent eccentricity, based on "Torsion in Symmetrical Buildings," N. M. Newark.

### 3.7.2.8 Methods for Seismic Analysis of Dams

The methods for the seismic analysis of dams are defined in Section 2.5.6.5.2, "Seismic Analyses" on page 2-60.

### 3.7.2.9 Determination of Seismic Class 1 Structure Overturning Moments

The safety factor against overturning due to maximum hypothetical earthquake moment is 3.6.

### 3.7.2.10 Analysis Procedure for Damping

Damping values for the structural system are selected based upon evaluation of the materials and mode shapes. Appropriate damping values of individual materials are presented in Section 3.7.1.3, "Critical Damping Values" on page 3-61. Evaluation of the mode shapes makes possible the selection of damping values to be associated with each mode.

## 3.7.3 SEISMIC SUBSYSTEM ANALYSIS

### 3.7.3.1 Seismic Analysis Methods

The criteria for determining whether systems or portions of systems require a seismic analysis is defined in Section 3.2.1, "Seismic Classification" on page 3-37. Piping is further classified according to the required function of the system or portion of a system as shown in Table 3-1.

Two analytical techniques are employed in the seismic analyses: dynamic and static methods. The results obtained by the Section 3.7.3.3, "Use of Equivalent Static Load Method of Analysis of Piping Systems" on page 3-69 static method are more conservative than the results calculated by the dynamic analysis. The use of the static analysis procedure is limited to piping systems which are not considered complex and where the anticipated seismic effects are minimal.

All seismically designed systems penetrating the Reactor Building wall are designed as follows: Within the Reactor Building, a dynamic analysis is performed except where noted below. As each penetration serves as an anchor to the system passing through the Reactor Building wall, a separate analysis is run on the piping outside the Reactor Building.

The design of the B, C, and F Systems outside the Reactor Building is based on a static analysis using a 0.5 g design acceleration. However, subsequent floor response spectra presented in Bechtel "Seismic Analysis Auxiliary Building" report dated January, 1970 and subsequent floor response spectra for Turbine Building developed by Duke Power Company show that there are peak accelerations greater than 0.5 g. Consequently, additional analysis is done to ensure that either (1) span lengths are reduced to avoid fundamental frequencies corresponding to accelerations above 0.5 g or (2) piping stresses and restraint load capabilities are reviewed for adequacy for the appropriate accelerations. Conservative manual methods will be used to determine span frequencies. Also, piping spans will be kept simple to avoid the necessity for modal analysis. Where this technique cannot be applied with confidence, a dynamic analysis will be performed.

Seismically designed systems which penetrate the Reactor Building with a very minor portion of the system inside the Reactor Building (i.e., from the penetration point to the inside isolation valve) are statically analyzed. These systems are as follows:

Reactor Building Penetration Room Ventilation System

Coolant Storage System

Liquid Waste Disposal System

Miscellaneous Non-Nuclear Service Systems; i.e., Service Air, Nitrogen, Demineralized Water, Filtered Water, etc.

Although there is not a seismic classification type interface, the Reactor Coolant System is a B&W Duke system interface.

The scope of NRC Inspection and Enforcement Bulletin 79-14 was defined as all piping that was computer analyzed for seismic loadings and all piping greater than or equal to 2½″ diameter that was
0 seismically analyzed using criteria methods. The design inputs for the IEB 79-14 seismic analysis have
0 been reconciled with the as-built. A rigorous computer analysis has been performed for all pipe reanalyzed for IEB 79-14.

Each pipe is idealized as a mathematical model consisting of lumped masses connected by elastic members. Lumped masses are located at carefully selected points in order to adequately represent the dynamic and elastic characteristics of the pipe system. Using the elastic properties of the pipe, the flexibility matrix for the pipe is determined. The flexibility calculations include the effects of torsional, bending, shear, and axial deformations. In addition, for curved members, the stiffness is decreased in accordance with USAS B31.1-1967, Code for Power Piping.

Once the flexibility and mass matrices of the mathematical model are calculated, the frequencies and mode shapes for all significant modes of vibration are determined. All modes having a period greater than 0.05 seconds are used in the analysis. The mode shapes and frequencies are solved in accordance with the following equation:

$$(K - w_n^2 M)\phi_n = 0$$

in which:

K = square stiffness matrix of the pipe

M = mass matrix for the pipe

$w_n$ = frequency for the $n^{th}$ mode

$\phi_n$ = mode shape matrix of the $n^{th}$ mode

After the frequency is determined for each mode, the corresponding spectral acceleration is read from the appropriate response spectrum for the pipe. Using these spectral accelerations, the response for each mode is found by solving the following equation:

$$Y_n \max = \frac{R_n Sa_n D}{M_n w_n^2}$$

in which:

$Y_n$max = response of the $n^{th}$ mode

$R_n$ = participation factor for the $n^{th}$ mode = $\Sigma M_i \phi_{in}$

$Sa_n$ = spectral acceleration for the $n^{th}$ mode

D = earthquake direction matrix

M = generalized mass matrix for the $n^{th}$ mode = $\Sigma M_i \phi_{in}^2$

Using these results, the maximum displacements for each mode are calculated for each mass point in accordance with the following equation:

$$V_{in} = \phi_{in} Y_n \max$$

in which:

$V_{in}$ = maximum displacement of mass i for mode n

The total displacement for each mass is determined by taking the square root of the sum of the squares of the maximum deflection for each mode:

$$V_i = \sqrt{\Sigma V_{in}^2}$$

in which:

$V_i$ = maximum displacement of mass i due to all modes calculated

The inertia forces for each direction of earthquake for each mode are then determined from:

$$Q_n = K\ V$$

in which:

$Q_n$ = inertia force matrix for mode n

V = displacement matrix corresponding to $Q_n$

Each mode's contribution to the total displacements, internal forces, moments, and stresses are determined from standard structural analysis methods using the inertia forces for each mode as an external loading condition. The total combined results are obtained by taking the square root of the sum of the squares of each parameter under consideration, in a manner similar to that done for displacements.

The computer program PISOL, used for original Oconee piping analysis performed by Duke, was provided and maintained by EDS Nuclear of San Francisco, California. Subsequent revision piping analyses on Oconee have been performed by Duke using updated versions of PISOL and SUPERPIPE, by the NUS Corporation, and by Nuclear Power Services (NPS) of Secaucus, New Jersey using their proprietary program.

Both EDS and NPS have reviewed their programs and have verified that the algebraic summation methods were not used in either the earthquake co-directional responses or in the inter-modal responses per IE Bulletin 79-07.

Certain piping analyses on Oconee were performed by Bechtel Corporation, Gaithersburg, Maryland. Bechtel has verified that algebraic summation methods, as noted above, were not used in the piping analysis performed for Duke on Oconee. Bechtel's analysis was performed by EDS on EDS programs. Certain piping analyses on Oconee were supplied by Babcock and Wilcox Co., Lynchburg, Virginia. B & W has verified that algebraic summation methods, as noted above, were not used in this piping analysis. B & W contracted this analysis to Dynatech who used the Southern Services program entitled "General Thermal Pipe Stress and Deflection Program." B & W contracted other analyses to Bechtel, San Francisco and performed some analysis in-house on the computer codes ST3DS/LUMS.

The verification of computer programs was done in a combination of ways. Due to the non-existence of the ASME benchmark problems during the time of the original analyses, original versions of programs were verified with hand calculated results. As more and more programs became commercially available, comparisons were made with these programs and with the ASME problems.

Specifically, EDS has used a combination of any or all of the following methods:

1. Comparison to ASME Benchmark Problem #1
2. Benchmark Problems Utilizing EDS Programs and Other Industry Programs (PIPESD, NUPIPE, ADLPIPE, ME-101).
3. Comparison to Hand Calculations.
4. Comparison Between EDS Programs and Updated Versions.

NPS has verified its program against PIPESD and ANSYS.

B & W has verified through hand calculations that the methodology of inter-modal and earthquake co-directional responses combinations are in agreement with their intended methodology for the analysis performed by Dynatech and for the in-house ST3DS/LUMS programs. The analysis performed by Bechtel, San Francisco, for B & W was done on PISOL (verified above).

### 3.7.3.2 Procedure Used for Modeling

A general description of the modeling for the specific programs used for seismic analysis is included in Section 3.7.3.1, "Seismic Analysis Methods" on page 3-65. The following figures are isometric drawings of typical piping models:

| | | | |
|---|---|---|---|
| 1. | System 01A | Figure 3-9 | Main Steam System - West Generator |
| | Problem #1-01-08 | | |
| 2. | Systems 53A and 59 | Figure 3-10 | Core Flooding Tank 1A |
| | Problem #1-53-9 | Figure 3-11 | Low Pressure Injection System - West |
| 3. | Systems 51A and 59 | Figure 3-12 | Reactor Coolant Pump Piping to High |
| | Problem #1-55-3 | Figure 3-13 | Pressure Injection Letdown Coolers |
| | | Figure 3-14 | |
| | | Figure 3-15 | |

The practice of overlapping analysis problems was used in the original analytical work performed for the Oconee Nuclear Station piping systems. This approach was utilized to avoid erecting in-line pipe anchors for the sole purpose of defining piping analysis problems. In the reanalysis work required for IE Bulletin 79-14, every effort was made to reduce the number of problems with overlap regions. This was done by combining individual analysis problems into one larger problem. However, this could not be accomplished for all problems due to computer capacity limitations.

When necessary to separate analysis problems the models will be "overlapped" to obtain adequate boundary conditions. The overlap region (pipe modeled in both problems) shall be selected based on engineering judgement, considering the specific geometry to be modeled, to give acceptably accurate results at the problem boundary. As a minimum, the overlap region must include five effective restraints in each of three orthogonal directions. One axial restraint is effective for the entire run between changes of direction. The overlap region should be located in the most rigid portion of the pipe to obtain maximum isolation between problems.

In the overlap region, S/R loads from both problems will be enveloped to obtain a conservative design load.

### 3.7.3.3 Use of Equivalent Static Load Method of Analysis of Piping Systems

#### 3.7.3.3.1 Piping

Duke Engineering Design Report, Static Method of Seismic Analysis of Piping Systems for Oconee 1, 2, 3, File #OS-27-B, dated June 5, 1970, describes the approach and a sample problem for seismic piping outside the Reactor Building.

The method for determining seismic response based on static analysis for Reactor Building piping is as follows:

The envelope of response curve(s) developed for the dynamic analysis are used for the static analysis which is based on the assumption that the natural frequency of the piping system is at the critical frequency.

Static loads at points of support are determined by utilizing the computer program ME553-Piping Flexibility Analysis - to perform a modified weight analysis which is based on applying the maximum horizontal forces in the positive X or Z directions simultaneously with the maximum, vertical force.

The horizontal forces are obtained by using the maximum acceleration peak from the appropriate envelope curves as the multiplier to convert uniform pipe weight into forces. The vertical force is obtained from the pipe weight density multiplied by the vertical peak acceleration.

The valves and special fittings on the system are mathematically expressed in the analysis as equivalent pipe of the same weight as the valve or fitting.

The combination of all maximum forces in the positive directions produces resulting static loads of greater magnitude than the dynamic analysis.

#### 3.7.3.3.2 Components

The seismic analysis of the component coolers shown in Figure 3-16 (3 sheets) is an example of the static analysis applied to components in Class B, C, and F Systems outside the Reactor Building.

### 3.7.3.4 Components of Earthquake Motion

Seismic forces are applied in the vertical and in any horizontal direction. The horizontal and vertical components of ground motion are applied simultaneously.

### 3.7.3.5 Combination of Modal Response

This information is addressed in Section 3.7.3.1, "Seismic Analysis Methods" on page 3-65.

### 3.7.3.6 Analytical Procedures for Piping

General Analytical Procedures are discussed in Section 3.7.3.1, "Seismic Analysis Methods" on page 3-65.

### 3.7.3.7 Multiple Supported Equipment and Components with Distinct Inputs

Floor response spectra developed as discussed in Section 3.7.2.5, "Components of Earthquake Motion" on page 3-64 are used as input for the piping analysis. When the pipe is supported from more than one elevation or structure, the response spectra for all support levels are enveloped and the envelope spectra are used in the analysis, except when the Independent Support Motion technique is used. In certain instances where one group of supports attach to a structure and another group of supports attach to a structure with a definite distinction in structural seismic response, ISM methods have been used in the qualification of existing pipe and supports during reanalysis for IEB 79-14 to mitigate the consequences and excessive conservatism of the total enveloped spectra method. Such consequences may include undue radiation exposure to personnel or undue hardship in implementing field modifications.

For piping passing from one building into another building, the maximum movements of the two buildings (deflections produced by earthquake) are summed absolutely and the piping system is subjected to these movements through the piping system restraints. The stresses produced in the piping by the building movements are considered additive to the stresses resulting from accelerations or thermal expansion.

Rocking of the turbine support structure has been considered with respect to the Main Steam System analysis and movements of the turbine support are negligible as compared to other design movements of the main turbine piping leads attached to the main steam stop valve and control valve assembly.

2 #### 3.7.3.8 Buried Piping Tunnels Designed for Seismic Conditions

2 The CCW intake piping furnishing water to the LPSW pumps is placed on concrete bedding which rests on bed rock at the point of entry into the station. There will not be any differential movement at the piping-structural interface with the rock base, thereby precluding any stress problems. Except for the CCW piping described above, other seismically designed safety-related buried lines for the Oconee Project are the 48″ emergency discharge CCW pipe and the SSF Auxiliary Service Water pump discharge line.

2 #### 3.7.3.9 Interaction of other Piping with Piping Designed for Seismic Conditions

The interaction between seismic/non-seismic lines are considered and safety system integrity is assured by the following methods:

Seismic/non-seismic lines are physically separated insofar as possible such that failure of a non-seismic line has no effect on safety-related piping.

Seismic/non-seismic boundaries are established by valves which are designed to meet the seismic design criteria. Failure in the non-seismic portion of the system cannot cause loss of function to the safety system in that automatic or remote manual-operated valves are used for valves normally open during Reactor Operation.

The seismic/non-seismic boundary valve is protected from seismic effects by restraining or anchoring the non-seismic portion of the system downstream of the valve.

#### 3.7.3.10 Seismic Analysis of Reactor Internals

The core support structure is designed as a Class 1 structure, as defined in Section 3.8, "Design of Class 1 Structures" on page 3-75 to resist the effects of seismic disturbances. The basic design guide for the seismic analysis is AEC publication TID-7024, "Nuclear Reactors and Earthquakes."

Lateral deflection and torsional rotation of the lower end of the core support assembly is limited in order to prevent excessive deformation resulting from seismic disturbance thereby assuring insertion of control rod assemblies (CRA). Core drop in the event of failure of the normal supports is limited by guide lugs so that the CRA do not disengage from the fuel assembly guide tubes. Additional information on design of the Reactor Internals is included in Section 3.9.2, "Dynamic Testing and Analysis" on page 3-141.

#### 3.7.3.11 Analysis Procedures for Damping

A 0.5 percent critical damping value is used for vital piping analysis (see Section 3.7.1.3, "Critical Damping Values" on page 3-61).

### 3.7.4 SEISMIC INSTRUMENTATION PROGRAM

#### 3.7.4.1 Location and Description of Instrumentation

Earthquake instrumentation being provided is a strong motion accelerograph designated SMA-3 and manufactured by Kinemetrics, Inc., of Pasadena, CA. This system consists of a central recording system, control panel, one TS-3 triaxial seismic trigger package, and two force-balance triaxial accelerometer packages.

The operations sequence is as follows:

The seismic trigger senses the initial earthquake ground motion with a normal setting of 0.01g and actuates the SMA-3 to full operation in less than 0.1 second.

The SMA-3 operates for as long as the trigger detects the earthquake, plus an additional 10 seconds.

The accelerograph can thus record a single earthquake or a sequence of earthquakes and aftershocks lasting as long as 30 minutes.

The output of each triaxial sensor is recorded using frequency modulation on a single four track cassette tape. Three of the tracks on the tape are the data tracks; the fourth is a reference track used for tape speed and amplitude compensation.

The Seismic Trigger and one Force Balance accelerometer of the SMA-3 system are located in the Unit 1 Tendon Gallery. Also, a second Force Balance accelerometer is located directly above at elevation 797′ + 6″ in the Oconee 1 Reactor Building. The recorder for the system is located in the Unit 1 Cable Room.

Also, a seismic trigger/switch is located in the Unit 1 tendon gallery. The Kinemetrics Model TS-3A has a preset acceleration threshold of 0.05g which activates the statalarm in Units 1 and 3 control rooms, when design conditions occur.

Six 2g peak recording accelorometers, manufactured by Engdahl-Model PAR 400, are also installed at various locations within the Oconee 1 Reactor Building as follows:

1. Adjacent to the strong motion accelerograph located in Tendon Access Gallery.
2. Support of the pressurizer vessel.
3. Support of Core Flood Tank 1A.
4. On the main steam line pipe hanger.
5. On the feedwater line pipe hanger.
6. On the core flood injection line pipe hanger.

The major Class 1 structures, Reactor Building and Auxiliary Buildings, will be founded on a common rock foundation and will have similar base motions. The dynamic structural properties and responses of these structures are generated using similar assumptions and analytical techniques. Therefore, the response of these structures can be determined based upon the instrumentation in one structure.

Top of soil (free field) responses will not provide useful analytical data for the evaluation of major Class 1 structures founded on rock. Therefore, it is felt that free field instrumentation will not contribute to the evaluation of these structures.

### 3.7.4.2 Comparison of Measured and Predicted Responses

In the event of an earthquake, the data will be analyzed to determine the magnitude of the earthquake. If the design earthquake is exceeded, the units would be shut down and structures, systems, and equipment thoroughly investigated. Responses from instruments located on selected structures, systems and components will be compared to calculated responses for those structures, systems and components at the respective location when subjected to the same base response.

The recorded seismic data will be used for comparison and verification of seismic analysis assumptions, damping characteristics, and the analytical model used for the plant seismic design.

## 3.7.5 REFERENCES

1. Bechtel Report, "Seismic Analysis Auxiliary Building", January, 1970.
2. Duke Power Engineering Design Report, "Static Method of Seismic Analysis of Piping Systems for Oconee 1, 2 and 3", File OS-27-B, June 6, 1970.
3. AEC Report TID-7024, "Nuclear Reactors and Earthquakes".
4. Newmark, N. M., "Torsion in Symmetrical Buildings".

# 3.8 DESIGN OF CLASS 1 STRUCTURES

Class 1 structures are those which prevent uncontrolled release of radioactivity and are designed to withstand all loadings without loss of function.

## 3.8.1 CONCRETE CONTAINMENT

The concrete/steel containment is a free standing structure and is referred to as the Reactor Building. It is constructed of reinforced concrete and structural liner plate steel with no separation between the two.

### 3.8.1.1 Description of the Containment

The structure consists of a post-tensioned reinforced concrete cylinder and dome connected to and supported by a massive reinforced concrete foundation slab as shown in Figure 3-19. The entire interior surface of the structure is lined with a ¼ inch thick welded ASTM A36 steel plate to assure a high degree of leak tightness. Numerous mechanical and electrical systems penetrate the Reactor Building wall through welded steel penetrations as shown in Figure 3-20 and Figure 3-21. The mechanical penetrations and access openings are design, fabricated, inspected, and installed in accordance with Subsection B, Section III, of the ASME Pressure Vessel Code.

Principal dimensions are as follows:

| | |
|---|---|
| Inside Diameter | 116 ft |
| Inside Height (Including Dome) | 208½ ft |
| Vertical Wall Thickness | 3-3/4 ft |
| Dome Thickness | 3-¼ ft |
| Foundation Slab Thickness | 8-½ ft |
| Liner Plate Thickness | ¼ in. |
| Internal Free Volume | 1,910,000 cu ft<br>(As-built volume 1,836,000 cu ft) |

The Reactor Building is shown in Figure 1-2 through Figure 1-9.

In the concept of a post-tensioned Reactor Building, the internal pressure load is balanced by the application of an opposing external pressure type load on the structure. Sufficient post-tensioning is used on the cylinder and dome to more than balance the internal pressure so that a margin of external pressure exists beyond that required to resist the design accident pressure. Nominal, bonded reinforcing steel is also provided to distribute strains due to shrinkage and temperature. Additional bonded reinforcing steel is used at penetrations and discontinuities to resist local moments and shears.

The internal pressure loads on the foundation slab are resisted by both the external bearing pressure due to dead load and the strength of the reinforced concrete slab. Thus, post-tensioning is not required to exert an external pressure for this portion of the structure.

The post-tensioning system consists of:

1. Three groups of 54 dome tendons oriented at 120° to each other for a total of 162 tendons anchored at the vertical face of the dome ring girder.
2. 176 vertical tendons anchored at the top surface of the ring girder and at the bottom of the base slab.

3. Six groups of 105 hoop tendons plus two additional tendons enclosing 120° of arc for a total of 632 tendons anchored at the six vertical buttresses.

Each tendon consists of ninety ¼ inch diameter wires with buttonheaded BBRV type anchorages, furnished by The Prescon Corporation. The tendons are housed in spiral wrapped corrugated thin wall sheathing. After fabrication, the tendon is shop dipped in a petroleum corrosion protection material, bagged and shipped. After installation, the tendon sheathing is filled with a corrosion preventive grease.

Ends of all tendons are covered with pressure tight grease filled caps for corrosion protection.

ASTM A615, Grade 60 reinforcing steel, mechanically spliced with T-series CADWELDS, is used throughout the foundation slab and around the large penetrations. A615, Grade 40 steel is used for the bonded reinforcing throughout the cylinder and dome as crack control reinforcing. At areas of discontinuities where additional steel is used, such steel is generally A615, Grade 60 to provide an additional margin of elastic strain capability.

The ¼ inch thick liner plate is attached to the concrete by means of an angle grid system stitch welded to the liner plate and embedded in the concrete. The details of the anchoring system are provided in Figure 3-19. The frequent anchoring is designed to prevent significant distortion of the liner plate during accident conditions and to insure that the liner maintains its leak tight integrity. The design of the liner anchoring system also considers the various erection tolerances and their effect on its performance. The liner plate is coated on the inside with 3 mils of inorganic zinc primer and 4 mils of Phenoline 305 for corrosion protection. See Table 3-12 for Reactor Building coatings. There is no paint on the side in contact with concrete.

The concrete used in the structure is made with crushed marble aggregate obtained from Blacksburg, South Carolina. Such aggregate produces an excellent high strength, dense, sound concrete. The design strengths are 5000 psi at 28 days for the shell and foundation slab.

Personnel and equipment access to the structure is provided by a double door personnel hatch with double seals on the outer door and by a 19 ft. - 0 in. clear diameter double gasketed single door equipment hatch as shown in Figure 3-21. A double door emergency personnel escape hatch is also provided. These hatches are designed and fabricated of A516, Grade 70 firebox quality steel made to A3000 specification, Charpy V-notch impact tested to 0°F in accordance with Section III of the ASME Pressure Vessel Code. All piping penetrations are furnished to the same requirements.

Structural brackets provided for the Reactor Building polar crane runway are fabricated of A36 steel shapes and A516, Grade 70 insert plates (Figure 3-19). Structural brackets and thickened plates are shop fabricated, stress relieved and shipped to the jobsite for welding into the ¼ inch liner plate similar to the penetration assemblies.

### 3.8.1.2 Applicable Codes, Standards, and Specifications

The following codes, standards, and specifications were used during the design, construction, testing and inservice inspection of Class 1 Structures:

ACI 301 - Specification for Structural Concrete for Buildings

ACI 318-63 - Building Code Requirements for Reinforced Concrete

ACI 347 - Recommended Practice for Concrete Framework

ACI 605 - Recommended Practice for Hot Weather Concreting

| | | |
|---|---|---|
| ACI 613 | - | Recommended Practice for Selecting Proportions for Concrete |
| ACI 614 | - | Recommended Practice for Measuring, Mixing and Placing Concrete |
| ACI 315 | - | Manual of Standard Practice for Detailing Reinforced Concrete Structures |
| ASME-1965 | - | Boiler and Pressure Vessel Code, Sections III, VIII, and IX |
| AISC | - | Steel Construction Manual, 6th ed. |
| PCI | - | Inspection Manual |
| ACI 505 | - | Specification for Design and Construction of Reinforced Concrete Chimneys |
| ACI | - | American Concrete Institute |
| ASME | - | American Society of Mechanical Engineers |
| AISC | - | American Institute of Steel Construction |
| PCI | - | Prestressed Concrete Institute |

## 3.8.1.3 Loads and Load Combinations

### 3.8.1.3.1 Loads Prior to Prestressing

Under this condition the structure is designed as a conventionally reinforced concrete structure. It is designed for dead load, live loads (including construction loads), and a reduced wind load. Allowable stresses are according to ACI 318-63.

### 3.8.1.3.2 Loads at Transfer of Prestress

The Reactor Building is checked for prestress loads and the stresses compared with those allowed by ACI 318-63 with the following exceptions: ACI 318-63, Chapter 26, allows concrete stress of $0.60f'_{ci}$ at initial transfer. In order to limit creep deformations, the membrane compression stress is limited to $0.30f'_{ci}$; whereas, in combination with flexural compression, the maximum allowable stress will be limited to $0.60f'_{ci}$ per ACI 318-63.

For local stress concentrations with nonlinear stress distribution as predicted by the finite element analysis, $0.75f'_{ci}$ is permitted when local reinforcing is included to distribute and control these localized strains. These high local stresses are present in every structure but they are seldom identified because of simplifications made in design analysis. These high stresses are allowed because they occur in a very small percentage of the cross section, are confined by material at lower stress and would have to be considerably greater than the values allowed before significant local plastic yielding would result. Bonded reinforcing is added to distribute and control these local strains.

Membrane tension and flexural tension are permitted provided they do not jeopardize the integrity of liner plate. Membrane tension is permitted to occur during the post-tensioning sequence but will be limited to $1.0\sqrt{f'_{ci}}$. When there is flexural tension but no membrane tension, the section is designed in accordance with Section 2605(a) of the ACI Code. The stress in the liner plate due to combined membrane tension and flexural tension is limited to 0.5 fy.

Shear criteria are in accordance with the ACI 318-63 Code, Chapter 26, as modified by the equations in Section 3.8.1.3.6, "Loads Necessary to Cause Structural Yielding" on page 3-79 using a load factor of 1.5 for shear loads.

#### 3.8.1.3.3 Loads Under Sustained Prestress

The conditions for design and the allowable stresses for this case are the same as above except that the allowable tensile stress in nonprestressed reinforcing is limited to 0.5 fy. ACI 318-63 limits the concrete compression to $0.45f'_c$ for sustained prestress load. Values of $0.30f'_c$ and $0.60f'_c$ are used as described above which bracket the ACI allowable value. However, with these same limits for concrete stress at transfer of prestress, the stresses under sustained load are reduced due to creep.

#### 3.8.1.3.4 Service Loads

This loading case is the basic "working stress" design. The Reactor Building is designed for the loading cases shown in Table 3-13.

Sufficient prestressing is provided in the cylindrical and dome portions of the vessel to eliminate membrane tensile stress (tensile stress across the entire wall thickness) under design loads. Flexural tensile cracking is permitted but is controlled by bonded reinforcing steel.

Under the design loads the same performance limits stated in Section 3.8.1.3.2, "Loads at Transfer of Prestress" on page 3-77 apply with the following exceptions:

1. If the net membrane compression is below 100 psi, it is neglected and a cracked section is assumed in the computation of flexural bonded reinforcing steel. The allowable tensile stresses in bonded reinforcing are 0.5 fy.
2. When the maximum flexural stress does not exceed $6\sqrt{f'_c}$ and the extent of the tensions zone is not more than 1/3 the depth of the section, bonded reinforcing steel is provided to carry the entire tension in the tension block. Otherwise, the bonded reinforcing steel is designed assuming a cracked section. When the bending moment tension is additive to the thermal tension, the allowable tensile stress in the bonded reinforcing steel is 0.5 fy minus the stress in reinforcing due to the thermal gradient as determined in accordance with the method of ACI-505.
3. The problem of shear and diagonal tension in a prestressed concrete structure should be considered in two parts: membrane principal tension and flexural principal tension. Since sufficient prestressing is used to eliminate membrane tensile stress, membrane principal tension is not critical at design loads. Membrane principal tension due to combined membrane tension and membrane shear is considered under Section 3.8.1.3.6, "Loads Necessary to Cause Structural Yielding" on page 3-79.

   Flexural principal tension is the tension associated with bending in planes perpendicular to the surface of the shell and shear stress normal to the shell (radial shear stress). The present ACI 318-63 provisions of Chapter 26 for shear are adequate for design purposes with proper modifications as discussed under Section 3.8.1.3.6, "Loads Necessary to Cause Structural Yielding" on page 3-79 using a load factor 1.5 for shear loads.

Crack control in the concrete is accomplished by adhering to the ACI and American Society of Civil Engineers Code Committee standards for the use of reinforcing steel. These criteria are based upon a recommendation of the Prestressed Concrete Institute and are as follows:

0.25 percent reinforcing shall be provided at the tension face for small members

0.20 percent for medium size members

0.15 percent for large members

A minimum of 0.15 percent bonded steel reinforcing is provided in two perpendicular directions on the exterior faces of the wall and dome for proper crack control.

The liner plate is attached on the inside faces of the wall and dome. Since, in general, there is no tensile stress due to temperature on the inside faces, bonded reinforcing steel is not necessary at the inside faces.

The Reactor Building shell is also designed for the following loads:

1. Dead load
2. Prestress forces
3. Live load including allowances for piping, ductwork and cable trays
4. Wind, including tornado
5. Earthquake
6. Thermal expansion of pipes attached to the Reactor Building wall

The external design pressure of the Reactor Building shell is 3 psig. This value is approximately 0.5 psig beyond the maximum external pressure that could be developed if the Reactor Building were sealed during a period of low barometric pressure and high temperature and, subsequently, the Reactor Building atmosphere were cooled with a concurrent rise in barometric pressure. Vacuum breakers are not provided.

#### 3.8.1.3.5 Loadings Common to all Structures

Ice or Snow Loading

A uniform distributed live load of 20 pounds per square foot is considered for roofs as stated in Section 1203.2 of the Southern Standard Building Code.

#### 3.8.1.3.6 Loads Necessary to Cause Structural Yielding

The structure is checked for the factored loads and load combinations that will cause structural yielding.

The load factors are the ratio by which loads will be multiplied for design purposes to assure that the load/deformation behavior of the structure is one of elastic, low-strain behavior. The load factor approach is being used in this design as a means of making a rational evaluation of the isolated factors which must be considered in assuring an adequate safety margin for the structure. This approach permits the designer to place the greatest conservatism on those loads most subject to variation and which most directly control the overall safety of the structure. It also places minimum emphasis on the fixed gravity loads and maximum emphasis on accident and earthquake or wind loads. The final design of the Reactor Building satisfies the loading combinations and factors tabulated in Table 3-14.

The load combinations, considering load factors referenced above, are less than the yield strength of the structure. The yield strength of the structure is defined as the upper limit of elastic behavior of the effective load carrying structural materials. For steels (both prestress and nonprestress), this limit is taken to be the guaranteed minimum yield given in the appropriate ASTM specification. For concrete, it is the ultimate values of shear (as a measure of diagonal tension) and bond per ACI 318-63 and the 28-day ultimate compressive strength for concrete in flexure ($f'_c$). The ultimate strength assumptions of the CI Code for concrete beams in flexure are not allowed; that is, the concrete stress is not allowed to go beyond yield and redistribute at a strain of three or four times that which causes yielding.

The maximum strain due to secondary moments, membrane loads and local loads exclusive of thermal loads is limited to that corresponding to the ultimate stress divided by the modulus of elasticity ($f'_c/E_c$) and a straight-line distribution from there to the neutral axis assumed.

For the loads combined with thermal loads, the peak strain is limited to 0.003 inch/inch. For concrete membrane compression, the yield strength is assumed to be $0.85f'_c$ to allow for local irregularities in accordance with the ACI approach. The reinforcing steel forming part of the load carrying system is allowed to go to, but not to exceed, yield as is allowed for ACI ultimate strength design.

A further definition of yielding is the deformation of the structure which causes strains in the steel liner plate to exceed 0.005 inch/inch. The yielding of nonprestress reinforcing steel is allowed, either in tension or compression, if the above restrictions are not violated. Yielding of the prestress tendons is not allowed under any circumstances.

Principal concrete tension due to combined membrane tension and membrane shear, excluding flexural tension due to bending moments or thermal gradients, is limited to $3\sqrt{f'_c}$. Principal concrete tension due to combined membrane tension, membrane shear, and flexural tension due to bending moments or thermal gradients is limited to $6\sqrt{f'_c}$. When the principal concrete tension exceeds the limit of $6\sqrt{f'_c}$, bonded reinforcing steel is provided in the following manner:

1. Thermal Flexural Tension - Bonded reinforcing steel is provided in accordance with the methods of ACI-505. The minimum area of steel provided is 0.15 percent in each direction.
2. Bending Moment Tension - Sufficient bonded reinforcing steel is provided to resist the moment on the basis of cracked section theory using the yield stresses stated above with the following exception: When the bending moment tension is additive to the thermal tension, the allowable tensile stress in the reinforcing steel is fy minus the stress in reinforcing due to the thermal gradient as determined in accordance with the methods of ACI-505.

Shear stress limits and shear reinforcing for radial shear are in accordance with Chapter 26 of ACI 318-63 with the following exceptions: Formula 26-12 of the code shall be replaced by

$$V_{ci} = Kb'd\sqrt{f'_c} + M_{cr}\left(\frac{V}{M'}\right) + V_i$$

Where:

$$K = \left[1.75 - \frac{0.036}{np'} + 4.0\,np'\right]$$

but not less than 0.6 for $p' \geq 0.003$.

For $p' < 0.003$, the value of K shall be zero.

$$M_{cr} = \frac{I}{Y}\left[6\sqrt{f'_c} + f_{pe} + f_n + f_i\right]$$

$f_{pe}$ = Compressive stress in concrete due to prestress applied normal to the cross section after all losses (including the stress due to any secondary moment) at the extreme fiber of the section at which tension stresses are caused by live loads.

$f_n$ = Stress due to axial applied loads ($f_n$ shall be negative for tension stress and positive for compression stress).

$f_i$ = Stress due to initial loads at the extreme fiber of a section at which tension stresses are caused by applied loads (including the stress due to any secondary moment). $f_i$ shall be negative for tension stress and positive for compression stress.

$n = \frac{505}{\sqrt{f'_c}}$

$p' = \frac{A'_s}{bd}$

V = Shear at the section under consideration due to the applied loads.

M′ = Moment at a distance d/2 from the section under consideration, measured in the direction of decreasing moment, due to applied loads.

$V_i$ = Shear due to initial loads (positive when initial shear is in the same direction as the shear due to applied loads).

Lower limit placed by ACI 318-63 on $V_{ci}$ as $1.7b'd\sqrt{f'_c}$ is not applied.

Formula 26-13 of the Code shall be replaced by

$$V_{cw} = 3.5b'd\sqrt{f'_c} \quad \sqrt{1 + \frac{f_{pc} + f_n}{3.5\sqrt{f'_c}}}$$

The term $f_n$ is as defined on the previous page. All other notations are in accordance with Chapter 26, ACI 318-63.

1. This formula is based on the tests and work done by Dr. A. H. Mattock of the University of Washington.
2. This formula is based on the commentary for proposal redraft of Section 2610, ACI-318, by Dr. A. H. Mattock, dated December 1962.

When the above-mentioned equations show that allowable shear in concrete is zero, radial horizontal shear ties are provided to resist all the calculated shear.

### 3.8.1.4 Design and Analysis Procedures

The strength of the Reactor Building at working stress and overall yielding is compared to various loading combinations to assure safety. The Reactor Building is examined with respect to strength, the nature and the amount of cracking, the magnitude of deformation, and the extent of corrosion to assure proper performance. The structure is designed and constructed in accordance with design criteria based upon ACI 318-63, ACI 301, and ASME Pressure Vessel Code, Sections III, VIII, and IX to meet the performance and strength requirements prior to prestressing, at transfer of prestress, under sustained prestress, at design loads and at yield loads.

It is the intent of the criteria to provide a structure of unquestionable integrity that will meet the postulated design conditions with a low strain elastic response. The Oconee Reactor Building meets these criteria because:

1. The design criteria are, in general, based on the proven stress, strain, and minimum proportioning requirements of the ACI or ASME Codes. Where departures or additions from these codes have been made, they have been done in the following manner:
   a. The environmental conditions of severity of load cycling, weather, corrosion conditions, maintenance, and inspection for this structure have been compared and evaluated with those for code structures to determine the appropriateness of the modifications.

b. The consultant firm of T. Y. Lin, Kulka, Yang and Associates was retained to assist in the development of the criteria. In addition to assisting with the criteria submitted in the PSAR, they have been involved in the continuing updating of the criteria and the review of design methods to assure that the criteria were being implemented as intended.

c. Dr. Alan H. Mattock of the University of Washington was retained to assist in developing the proper design criteria from combined shear, bending, and axial load.

d. All criteria, specifications, and details relating to the liner plate and penetrations, and corrosion protection have been referred to Bechtel's Metallurgy and Quality Control Department. This department maintains a staff to advise the corporation on problems of welding, quality control, metallurgy, and corrosion protection.

e. The design of the Oconee Reactor Building was continually reviewed as the criteria were improved for successive license applications to assure that this structure does meet the latest criteria.

2. The primary membrane integrity of the structure is provided by the unbonded post-tensioning tendons, each one of which is stressed to 80 percent of ultimate strength during installation and performs at approximately 50 percent - 60 percent during the life of the structure. Thus, the main strength elements are individually proof-tested prior to operation of the plant.

3. 970 such post-tensioning elements have been provided, 162 in the dome, and 176 vertical and 632 hoop tendons in the cylinder. Any three adjacent tendons in any of these groups can be lost without significantly affecting the strength of the structure due to the load redistribution capabilities of the shell structure. The bonded reinforcing steel provided for crack control assures that this redistribution capability exists.

4. The unbonded tendons are continuous from anchorage to anchorage, being deflected around penetrations and isolated from secondary strains of the shell. Thus, the membrane integrity of the shell can be assured regardless of conditions of high local strains.

5. The unbonded tendons exist in the structure at a slightly ever-decreasing stress due to relaxation of the tendon and creep of the concrete and, even during pressurization, are subject to a stress change of very small magnitude (2 percent to 3 percent of ultimate strength). Thus, the main structural system is never subject to large changes in load, even during accident conditions.

6. The concrete portion of the structure, similar to the tendons, is subject to the highest state of stress during the initial post-tensioning. During pressurization, it is subject to a large change in load (or state of stress) but the change is, in general, a decrease in load. The large membrane compressive forces are diminished, and replaced, by relatively small radial pressures and stresses.

7. The deformations of the structure during plant operation, or due to accident conditions, are relatively minor due to the low strain behavior of the concrete. The largest deformations occur at the time of initial post-tensioning and shortly, thereafter, prior to operation. This low strain behavior, and the inherent strength of the structure, permit the anchoring of all piping penetrating the structure directly to the shell. Such details (see Figure 3-21) eliminate the use of expansion bellow seals and significantly reduce the likelihood of leaks developing at the penetrations.

The analysis for the Reactor Building falls into two parts, axisymmetric and nonaxisymmetric. The axisymmetric analysis is performed through the use of a finite element computer program for the individual loading cases of dead load, live load, temperature, prestress, and pressure, as described in Section 3.8.1.4.1, "Axisymmetric Techniques" on page 3-83. The axisymmetric finite element approximation of the Reactor Building shell does not consider the buttresses, penetrations, brackets, and anchors. These items of configuration, the lateral loads due to seismic or wind, and concentrated loads are considered in the nonaxisymmetric analysis described in Section 3.8.1.4.2, "Nonaxisymmetric Analysis" on page 3-89.

This section discusses analytical techniques, references and design philosophy. The results of these analyses are discussed in Section 3.8.1.5, "Structural Acceptance Criteria" on page 3-97. The design criteria and analysis have been reviewed by Bechtel's consultants, T. Y. Lin, Kulka, Yang and Associates.

#### 3.8.1.4.1 Axisymmetric Techniques

The finite element technique is a general method of structural analysis in which the continuous structure is replaced by a system of elements (members) connected at a finite number of nodal points (joints). Conventional analysis of frames and trusses can be considered to be examples of the finite element method. In the application of the method to an axisymmetric solid (e.g., a concrete Reactor Building), the continuous structure is replaced by a system or rings of quadrilateral cross section which are interconnected along circumferential joints. Based on energy principles, work equilibrium equations are formed in which the radial and axial displacements at the circumferential joints are unknowns of the system. The results of the solution of this set of equations are the deformation of the structure under the given loading conditions. For the output, the stresses are computed knowing the strain and stiffness of each element.

The finite element mesh used to describe the structure is shown in Figure 3-22. The upper portion and lower portion of the structure were analyzed independently to permit a greater number of elements to be used for those areas of the structure of major interest such as the ring girder area and the base of the cylinder. The finite element mesh of the structure base slab was extended down into the foundation material to take into consideration the elastic nature of the foundation material and its effect upon the behavior of the base slab. The tendon access gallery is separated from the Reactor Building base slab by 3 in. compressible material. No moments or forces are transmitted from the base slab to the tendon access gallery. The maximum vertical elastic displacement of the base slab is one inch due to the maximum loading combinations. The tendon access gallery was designed as a separate structure with no reactions being generated from the bedrock to the ring shaped gallery structure.

The finite element mesh for the Reactor Building does not include the interior structure. The interior structure was included in the finite element input as a lump weight. The finite elements provide stresses for axisymmetric loads. The stresses from the eccentric interior structure loads and earthquake loads are superimposed analytically to the finite element stresses. The final algebraic summation of all stresses was used to design the base slab.

| Stresses for Axisymmetric Loads | Stresses with Non-Axisymmetric Loads |
|---|---|
| 11.0 kips/sq.ft. | 26.0 kips/sq.ft. |

The use of the finite element computer program permitted an accurate estimate of the stress pattern at various locations of the structure. The following material properties were used in the program for the various loading conditions:

| | Load Conditions | |
|---|---|---|
| | $D, F, T_O, T_A$ | P |
| $E_{concrete}$, Foundation (psi) | $3.0 \times 10^6$ | $3.0 \times 10^6$ |
| $E_{concrete}$, Shell (psi) | $3.0 \times 10^6$ | $3.0 \times 10^6$ |
| $\mu_{concrete}$ (Poisson's Ratio) | 0.17 | 0.17 |
| $\alpha_{concrete}$ (Coefficient of Expansion) | $0.55 \times 10^{-5}$ | – |
| $E_{subgrade}$ (psi) | $4.5 \times 10^6$ | $4.5 \times 10^6$ |
| $E_{liner}$ (psi) | $29 \times 10^6$ | $29 \times 10^6$ |
| $f_{y\ liner}$ (psi) | 36,000 | 36,000 |

The major benefit of the program is the capability to predict shears and moments due to internal restraint and the interaction of the foundation slab relative to the subgrade. The structure is analyzed assuming an uncracked homogeneous material. This is conservative because the decreased relative stiffness of a cracked section would result in smaller secondary shears and moments.

In arriving at the tabulated values of E, the effect of creep is included by using the following equation for long-term loads such as thermal load, dead load and prestress:

$$E_{cs} = E_{ci}\ (\varepsilon_i/(\varepsilon_s + \varepsilon_i))$$

Where:

$E_{cs}$ = sustained modulus of elasticity of concrete.

$E_{ci}$ = instantaneous modulus of elasticity of concrete.

$\varepsilon_i$ = instantaneous strain, inch/inch per psi.

$\varepsilon_s$ = creep strain, inch/inch per psi.

The thermal gradients used for design are shown in Figure 3-24. The gradients for both the design accident condition and the factored load condition are based on the temperature associated with the factored pressure (factored loads are described in Section 3.8.1.3.6, "Loads Necessary to Cause Structural Yielding" on page 3-79). The design pressure and temperature of 59 psig and 286°F became 88.5 pisg and 286°F at factored conditions.

The upper stress limit for a linear stress-strain relationship was assumed to be 3000 psi (0.6 $f'_c$) for use with analyses made by the use of the axisymmetric finite element analytical method. (The analyses referred to considered the concrete as uncracked and the analytical model is the entire containment.) However, the maximum predicted compressive stress was about 2559 psi. The load combination considered was $0.95D + F + P + E' + T_A$ and the location for the predicted stress was for Section EF in ring girder (see Table 3-16). Therefore, only the linear portion of the stress strain curve was used in the analyses that used the entire containment structure as a model.

The compressive stress and strain level is the highest (after the LOCA when temperature is still relatively high, 200°F, and pressure is dropping rapidly) at the inside face of the concrete at the edge of openings and also under the liner plate anchors. Neither concentration is a result of what may be considered a real load. In the case of an opening, the real stress is a result of prestress, reduced pressure and dead load. Applying stress concentration factors to these loads still keeps the concrete in essentially the elastic range.

When the strain and resulting stress from the thermal gradient are also multiplied by a stress concentration factor, the total strain and resulting stress will be above the linear stress range determined as by a uniaxial compression test. The relatively high stress level is not of real concern due to the following:

1. The concrete affected is completely surrounded by either other concrete or the penetration nozzle and liner reinforcing plate. This confinement puts the concrete in triaxial compression and gives it the ability to resist forces far in excess of that indicated by a uniaxial compression test.
2. The high state of stress and strain exist at a very local area and really have no effect on the overall containment integrity.

However, to be conservative, reinforcing steel was placed in these areas, and also, the penetration nozzle will function as compressive reinforcement.

The concrete under the liner plate anchors has some limited yielding in order to get the necessary stress distribution required to resist the liner plate self-relieving loads.

The thermal loads are a result of the temperature differential within the structure. The design temperature stresses for this finite element analysis were prepared so that when temperatures are given at every nodal point, stresses are calculated at the center of each element.

Thus, the liner plate was handled as an integral part of the structure and was included in the finite element mesh of the Reactor Building, but having different material properties, and not as a mechanism which would act as an outside source to produce loading only on the concrete portion of the structure.

Figure 3-22, Sheet 1 of 2, shows the inclusion of the liner plate in the finite element mesh.

Under the design accident condition or factored load condition, cracking of the concrete at the outside face would be expected. The value of the sustained modulus of elasticity of concrete, $E_{cs}$, was used in ACI Code 505-54 to find the stresses in concrete, reinforcing steel and liner plate from the predicted design accident thermal loads and factored accident loads.

The isostress plots shown in Figure 3-25 and Figure 3-26 do not consider the concrete cracked. The thermal stresses are combined from the individual isostress output for the cases of D + F + T and D + F + 1.5P + T. The first case is critical for concrete stresses and occurs after depressurization of the Reactor Building; the second case is critical for the reinforcing stresses and it occurs when pressure and thermal loads are combined and cause cracking at the outside face. The loading cases for isostress plots shown in Figure 3-25 are D + F + 1.15P on Sheet 1, 0.95D + F + 1.5P + T on Sheet 2, D + F on Sheet 3, and T on Sheet 4. The loading cases for isostress Plots shown in Figure 3-26 are D on Sheet 1, F on Sheet 2, T on Sheet 3, 0.95D + T on Sheet 4, F + 1.15P on Sheet 5, and F + 1.5P on Sheet 6.

The general approach of determining stresses in the concrete and reinforcement required the evaluation of the stress blocks of the cross section being analyzed.

The value of stresses was taken from the computer output in case of axisymmetric loading and from analytical solutions in case of nonaxisymmetric loading. Both computations were based on homogeneous materials; therefore, some adjustment was necessary to evaluate the true stress-strain conditions when cracks develop in the tensile zone of the concrete.

An equilibrium equation can be written considering the tension force in the reinforcement, the compressive force in the concrete and the axial force acting on the section. In this manner, the neutral axis is shifted from the position defined by the computer analyses into a position which is the function of the amount of reinforcement, the modulus ratio, and the acting axial forces.

Large axial compressive force might prevent the existence of any tension stresses, as in the loading condition D + F + T; therefore, no self-relieving action exists; the stresses are taken directly from the computer output.

In the case of D + F + 1.5P + T, the development of cracks in the concrete decreases the thermal moment and this effect was considered; but the self-relieving properties of other loadings were not taken into account, even in places where they do exist, such as at discontinuities, e.g., the cylinder-base slab connection. This means that in analyzing the section, a reduced thermal moment was added to the unreduced moment caused by other loadings.

The thermal stresses in the containment are comparable to those developed in a reinforced concrete slab, which is restrained from rotation. The temperature varies linearly across the slab. The concrete will crack in tension and the neutral axis will be shifted toward the compressive extreme fiber. The cracking will reduce the compression at the extreme fiber and increase the tensile stress in reinforcing steel.

The following analysis is based on the equilibrium of normal forces; therefore, any normal force acting on the section must be added to the normal forces resulting from the stress diagram. The effects of Poisson's ratio are considered while the reinforcement is considered to be identical in both directions.

Stress - Strain relationship in compressed region of concrete:

$$E_c\Sigma_x = \sigma_x - \nu_c\sigma_y \tag{1}$$

$$E_c\Sigma_y = -\nu_c\sigma_x + \sigma_y \tag{2}$$

From the above equations (1) and (2):

$$\sigma_x = E_c \frac{\Sigma_x + \Sigma_y\nu}{1 - \nu_c^{\,2}} \tag{3}$$

$$\sigma_y = E_c \frac{\Sigma_y + \Sigma_x\nu}{1 - \nu_c^{\,2}} \tag{4}$$

Substituting,

$$\sigma_x = \sigma_y = \sigma_c \text{ and } \Sigma_x = \Sigma_y = \Sigma_c \text{ into equations (3) and (4)}$$

$$\sigma_c = E_c\Sigma_c \frac{1}{1 - \nu_c} = 1.205\, E_c\Sigma_c \text{ (if } \nu_c = .17)$$

The reinforcement is acting in one direction, independently from the reinforcement in the perpendicular direction.

Example: If $E_c = 3 \times 10^6$ and $E_s = 29 \times 10^6$

$$n_R = \frac{29}{1.205 \times 3} = 8.02$$

The liner plate is acting in two directions, similar to the concrete except for the difference caused by the Poisson's ratios:

$$\sigma_L = E_s\Sigma_s \frac{1}{1 - \nu_L} = 1.35\, E_s\Sigma_s \qquad \nu_L = .25$$

$$n_L = \frac{1.35 \times 29}{1.205 \times 3} = 10.83 \qquad \nu_c = .17$$

The following is an example of the use of the analytical method derived for D + F + P + $T_A$ + E (See Table 3-16).

The concrete and reinforcement stresses are calculated by conventional methods, from the moment caused by loading other than thermal. The analyses assume homogeneous concrete sections. Those concrete and reinforcing steel stresses are then added to the thermal stresses as obtained by the method described.

Data:

$E_c = 3 \times 10^6$ psi $\quad \nu_L = 0.25$

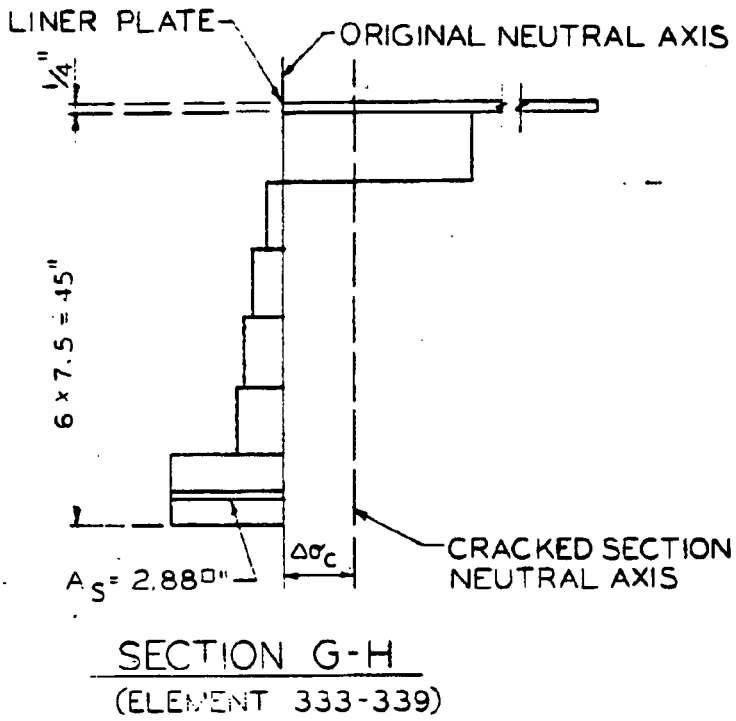
$E_s = 29 \times 10^6$ psi $\quad n_R = 8.02$

$\nu_c = 0.17 \quad n_L = 10.83$

Notation:

| | |
|---|---|
| $E_c$ | Modulus of elasticity of concrete. |
| $E_s$ | Modulus of elasticity of steel. |
| $n_L$ | Modular ratio of liner plate-concrete. |
| $n_R$ | Modular ratio of reinforcement-concrete. |
| $\Delta\sigma c$ | Reduction of concrete compressive stress, considering cracking. |
| $\Sigma_c$ | Concrete strain. |
| $\Sigma_s$ | Steel strain. |
| $\Sigma_x$ | Concrete strain in X direction. |
| $\Sigma_y$ | Concrete strain in Y direction. |
| $\nu_c$ | Poisson's ratio of concrete. |
| $\nu_L$ | Poisson's ratio of liner plate. |
| $\sigma_c$ | Stress in concrete. |
| $\sigma_L$ | Stress in liner plate. |
| $\sigma_R$ | Stress in reinforcement. |
| $\sigma_x$ | Stress in concrete in direction X. |

STRESS BLOCK FROM THE COMPUTER OUTPUT




| (Thermal) Stresses (Psi) | (Thermal) Forces (K/Ft.) Resultant |
|---|---|
| -36000 | - 108.0 |
| - 1997 | - 179.7 |
| + 277 | + 24.9 |
| + 529 | + 47.6 |
| + 595 | + 53.5 |
| + 666 | + 60.0 |
| + 1467 | + 132.0 |

EQUILIBRIUM AFTER CRACKING

$2.88\ (1467+\Delta\sigma c)\ 8.02 - (179.7+108)\ 1000 + \Delta\sigma c\ (12x7.5+3x10.83) = N = -102{,}000$

$33884 + 23.1\ \Delta\sigma c - 287700 + 124.5\ \Delta\sigma c + 102{,}000 = 0$

$$147.6\ \Delta\sigma c = 151{,}816$$

$$\Delta\sigma c = 1028.6$$

ASSUMED POSITION OF N. A. is O.K.

$$\Delta\sigma c = 1029 \text{ Psi}$$

$\sigma_s$(After Cracking) $= (1467 + 1029)\ 8.02 = 20018$ Psi

$\sigma_c = -1997 + 1029 = -968$ Psi

$\sigma_R = \sigma_{D+F+P} + \sigma_T \sigma_E$
$= -503 + 20018 \pm 96 = 19611$ (Tensile)

$\sigma_c = \sigma_{D+F+P} + \sigma_T + \sigma_E$
$= 61 - 968 \pm 11 = -918$ (Compression)

#### 3.8.1.4.2 Nonaxisymmetric Analysis

The nonaxisymmetric aspects of configuration or loading required various methods of analysis. The description of the methods used as applied to different parts of the containment is given below.

1. Buttresses

   The buttresses and tendon anchorage zones are defined as Class 1 elements and were designed in accordance with the general design criteria for the Reactor Building structure and with the applicable provisions of ACI 318-63, Chapter 26.

   The buttresses were analyzed for two effects, nonaxisymmetric and anchorage zone stresses. Both effects are shown in the results of a two-dimensional plane strain finite element analysis with loads acting in the plane of the coordinate system (Figure 3-27).

   At each buttress, the hoop tendons are alternately either continuous or spliced by being mutually anchored on the opposite faces of the buttress. Between the opposite anchorages, the compressive force exerted by the spliced tendon is twice as much as elsewhere. This value combined with the effect of the tendon which is not spliced will be 1.5 times the prestressing force acting outside of the buttresses. The cross-sectional area at the buttress is about 1.5 times that of the wall, so the hoop stresses as well as the hoop strains and radial displacements can be considered as being nearly constant all around the structure. Isostress plots of the plane strain analysis, Figure 3-28, confirm this.

   The vertical stresses and strains, caused by the vertical post-tensioning, become constant at a short distance away from the anchorages because of the stiffness of the cylindrical shell. Since the stresses and strains remain nearly axisymmetric despite the presence of the buttresses, their effect on the overall analysis is negligible when the structure is under dead load or prestressing loads.

   When an increasing internal pressure acts upon the structure, combined with a thermal gradient (Figure 3-29) such as at the design accident condition, the resultant forces being axisymmetric, the stiffness variation caused by the buttresses will decrease as the concrete develops cracks. The structure will then tend to shape itself to follow the direction of the acting axisymmetric resultant forces even more closely. Thus, the buttress effect is more axisymmetric at yield loads, which include factored pressure, than at design loads including pressure. This fact, combined with the design provision that alternate horizontal tendons terminate in a single buttress, indicates that the buttresses will not reduce the margins of safety available in the structure.

   The analysis of the anchorage zone stresses at the buttresses has been determined to be the most critical of all the various types of anchorage areas of the shell. The local stress distribution in the immediate vicinity of the bearing plates has been derived by the following three analysis procedures:

   a. The Guyon equivalent prism method: This method is based on experimental photoelastic results as well as on equilibrium considerations of homogeneous and continuous media. It should be noted that the relative bearing plate dimensions are considered.

   b. In order to include biaxial stress effects, use has been made of the experimental test results presented by S. J. Taylor at the March 1967 London Conference of the Institution of Civil Engineers (Group H, Paper 49). This paper compares test results with most of the currently used approaches (such as Guyon's equivalent prism method). It also investigates the effect of the rigid trumpet welded to the bearing plate.

   c. The finite element method, assuming homogeneous and elastic material, was used in a plane strain analysis. The mesh and results are shown in Figure 3-27 and Figure 3-28.

The Guyon method yields the following results for a loading ratio $(a'/a)^*$ = 0.9 Maximum compressive stress under the bearing plate:

$\sigma_c$ = -2400 psi

Maximum tensile stress in spalling zone:

$\sigma$ spalling = +2400 psi = $-\sigma_c$

Maximum tensile stress in bursting zones:

$\sigma$ maximum bursting = 0.04 P = +96 psi

S. J. Taylor's experimental results indicate that the anchor plate will give rise to a similar stress distribution pattern as Guyon's method; the main difference lies in the fact that the central bursting zone has a tensile stress peak of twice Guyon's value:

$\sigma$ maximum bursting = +192 psi

By finite element analysis, the symmetric buttress loading yields a tensile peak stress in the bursting zone very close to S. J. Taylor's value:

s maximum bursting = +114 psi

A state of biaxial tension in the concrete will appear on the outside face under the loading case 1.05D + 1.5P + $1.0T_A$ + 1.0F. The superposition of the corresponding state of stress with the local anchor stresses reduces the load carrying capacity of the anchorage unit and caused a reduction in the maximum tensile strain to cracking.

On the other hand, the uniform compressive state of stress (vertical prestress) applied to the anchorage zone increases the load carrying capacity of the anchorage unit, with the maximum tensile strain to cracking being increased.

The design of the buttress anchor zones considered such additional vertical stresses, leading to a state of pseudo biaxial stress, the second direction being radial through the thickness.

For the above-mentioned case, 1.05D + 1.5P + $1.0T_A$ + 1.0F, the averaged vertical (meridional) stress component is:

$$f_a \simeq +400 \text{ psi}$$

The compressive bearing plate stress at 10 inches depth below the bearing plate is:

$$f_c \simeq -1500 \text{ psi}$$

(Note: The steel trumpet carries 7.2 percent of the prestress force.)

Thus, the two values introduced in the biaxial stress envelopes proposed in S. J. Taylor's article:

$$f_c/f'_c = 1500/1500 = 0.3$$

$$f_c/f'_c = 400/5000 = 0.08$$

show that failure could occur if vertical reinforcing were not provided. In fact, the maximum allowable vertical averaged tensile stress according to Taylor's interaction curve is $f_a/f'_c = 0.03$; therefore, $f_a$ = +150 psi.

The three dimensional stress distribution in the anchor zones was analyzed in sufficient detail to permit the rational evaluation of stress concentrations. A conical wedge segment was used as the

* Ratio of width of bearing plate to width of concrete under bearing plate.

basic design element and the radial splitting tension was determined as a tangential distribution function. The summation of splitting stresses through the entire volume of the lead-in zone established the value of the splitting force. This force is a function of the a/b ratio and the cone angle and/or, a/b and h. Several different combinations of the values were analyzed and the most critical values selected. A system analysis for the vertical splitting force was carried out based on statics and the magnitude of vertical and spalling forces were also determined.

The most unfavorable loads and load combinations were considered in the analysis of the anchorage zone and stresses based on transient thermal gradients were used in all cases where the use of a steady state gradient under-estimated the stresses and strains and were superimposed on the bursting stresses determined from the triaxial stress calculations. The computed stresses are less than the ACI allowable values. The design of the concrete reinforcement is based on this conservative analysis to provide a margin of safety similar to the other components of the Reactor Building structure and to control cracking in the anchorage zone. As a result, there is no danger of delayed rupture of the concrete under sustained load, due to local overstress and microcracking.

The reinforcing details, including the method for anchoring and splicing the reinforcing, are shown on Figure 3-30.

The reinforcement required has been designed primarily to resist tensile forces and has been located such that it will efficiently resist the tensile forces. The reinforcement was provided for load cases which create the maximum tensile forces and for other load cases the relevant shear forces or stresses were superimposed.

The possibility of the concrete breaking along shear planes was considered at the intersection of (1) the buttress with the cylinder and (2) the cylinder with the base slab.

a. Buttress - Cylinder Intersection

   An increase in the compression force at the buttress corresponds to an increase in the concrete area of the same magnitude.

b. Cylinder - Base Slab Intersection

   An analysis for the most critical radial shear conditions was performed. The difference in shear stiffness between the shell and the buttress and the remainder of the shell was included as a shear amplification factor. The reinforcing required was less than the reinforcing provided.

The possibility of concrete breaking along a shear plane is excluded by providing ample reinforcing. In other locations, breakage along the shear plane has been excluded by the opposition of prestressing and anchor forces.

The following three sources of information were also considered in the design of the anchorage zone reinforcing:

a. Full-scale load tests of the anchorage on the same concrete mix used in the structure and review of prior uses of the anchorage.

b. The post-tensioning supplier's recommendations of anchorage reinforcing requirements.

c. Review of the final details of the combined reinforcing by the consulting firm of T. Y. Lin, Kulka, Yang, and Associates.

2. Large Opening (Equipment Hatch and Personnel Lock Opening)

The primary loads considered in the design of the equipment hatch and personnel lock opening, as for any part of the structure, were dead load, prestress, pressure, earthquake, and thermal loads. The secondary loads considered were the following effects caused by the above primary loads:

a. The deflection of tendons around the opening.

b. The curvature of the shell at the opening.

c. The thickening around the opening.

The primary loads listed are mainly membrane loads with exception of the thermal loads. In addition to membrane loads, accident pressure also produces punching shear around the edge of the opening. The values of these loads for design purposes were the magnitudes of these loads at the center of the opening. These are fairly simple to establish knowing the values of hoop and vertical prestressing, accident pressure, and the geometry and location of the opening.

Secondary loads were predicted by the following methods:

a. The membrane stress concentration factors and effect of the deflection of the tendons around the equipment hatch were analyzed for a flat plate by the finite element method. The stresses predicted by conventional stress concentration factors, compared with those values found from above-mentioned finite element computer program, demonstrated that the deflection of the tendons does not significantly affect the stress concentrations. This is a plane stress analysis and does not include the effect of the curvature of the shell. However, it gives an assurance of the correctness of the assumed membrane stress pattern caused by the prestressing around the opening. Results of this analysis are shown in Figure 3-31.

b. With the help of Reference 1 on page 3-138, stress resultants around the large opening were found for various loading cases. Comparison of the results found from this reference, with the results of a flat plate of uniform thickness with a circular hole, showed the effect of the cylindrical curvature on stress concentrations around the opening.

   Normal shear forces (relative to opening) were modified to account for the effect of twisting moments as shown in Reference 1 on page 3-138. These modified shear forces are called Kirschoff's shear forces. Horizontal wall ties were provided to resist a portion of these shear forces.

c. The effect of the thickening on the outside face around the large opening was considered using several methods. Reference 2 on page 3-138 was used to evaluate the effect of thickening on the stress concentration factors for membrane stress. A separate axisymmetric finite element computer analysis for a flat plate with anticipated thickening on the outside face was prepared to handle both axisymmetric and nonaxisymmetric loads to predict the effect of the concentration of hoop tendons, with respect to the Reactor Building at the top and bottom of the opening.

   For the analysis of the thermal stresses around the opening, the same method was used as for the other loadings. At the edge of the opening, a uniformly distributed moment, equal but opposite to the thermal moment existing on the rest of the shell, was applied and evaluated using the methods of the preceding Reference 1 on page 3-138. The effects were then superimposed on the stresses calculated for the other loads and effects.

   In the case of accident temperature, after the accident pressure has already been decreased, very little or no tension develops on the outside, so thermal strains will exist without the relieving effect of the cracks. However, the liner plate will reach a high strain level and so will the concrete at the inside corner of the penetration, thereby relieving the very high stresses, but still carrying a high moment in the state of redistribution stresses.

   In the case of 1.5P (prestress fully neutralized) + $1.0T_A$ (accident temperature), the cracked concrete with highly strained tension reinforcement constitutes a shell with stiffness decreased but still essentially constant in all directions. In order to control the increased hoop moment around the opening, the hoop reinforcement is about twice that of the radial reinforcement. See Figure 3-21.

   The equipment hatch opening was thickened for the following reasons:

1) To reduce the larger than acceptable predicted membrane stresses around the opening.

2) To accommodate tendon placement.

3) To accommodate bonded steel reinforcing placement.

4) To compensate for the reduction in the overall shell stiffness due to the opening.

The working stress method (elastic analysis) was applied to both the load combinations for design loads, as well as for yield loads, for the analytical procedures described above. The only difference is the higher allowable stresses under yield conditions. The various factored load combinations and capacity reduction factors are specified in Section 3.8.1.3.6, "Loads Necessary to Cause Structural Yielding" on page 3-79 and were used for the yield load combinations using the working stress design method. The design assumption of straight line variation of stresses was maintained under yield conditions.

The governing design condition for the sides of the equipment hatch opening at the outside edge of the opening is the accident condition. Under this condition, approximately 60 percent of the total bonded reinforcing steel needed at the edge of the opening at the outside face is required for the thermal load.

Excluding thermal load, the remaining stress (equivalent to approximately 40 percent of the total load including thermal) at the edge of the outside face is the sum of the following stress resultants:

1) Normal stresses resulting from membrane forces, including the effect of thickening, contribute approximately minus 35 percent (minus 14 percent of total).

2) Flexural stresses resulting from the moments caused by thickening on the outside face contribute approximately 150 percent (60 percent of total).

3) Normal and flexural stresses resulting from membrane forces and moments caused by the effect of cylindrical curvature contribute approximately minus 15 percent (minus 6 percent of total).

3. Penetrations

Analysis of the Reactor Building penetrations falls into three parts: (1) the concrete shell, (2) the liner plate reinforcement and closure to the pipe, and (3) the thermal gradients and protection requirements at the high-temperature penetrations. The three categories will be discussed separately.

a. Concrete Shell

In general, special design consideration is given to all openings in the Reactor Building. Analysis of the various openings has indicated that the degree of attention required depends upon the penetration size. Small penetrations are considered to be those with a diameter smaller than 2-½ times the shell thickness: i.e., approximately 8 feet in diameter or less. Reference 1 on page 3-138 indicates that, for openings of 8-foot diameter or less, the curvature effect of the shell is negligible. In general, the typical concrete wall thickness has been found to be capable of taking the imposed stresses using bonded reinforcement, and the thickness is increased only as required to provide space requirements for radially deflected tendons. The induced stresses, due to normal thermal gradients and postulated rupture conditions, distribute rapidly and are of a minor nature compared to the numerous loading conditions for which the shell must be designed. The small penetrations are analyzed as holes in a plane sheet. Applied piping restraint loads due to thermal expansion or accident forces are assumed to distribute in the cylinder as stated in Reference 3 on page 3-138. Typical details associated with these openings are indicated in Figure 3-20.

b. Liner Plate Closure

The stress concentrations around openings in the liner plate were calculated using the theory of elasticity. The stress concentrations were then reduced by the use of a thickened plate around the

opening. In the case of a penetration with no appreciable external load, stud bolts are used to maintain strain compatibility between the liner plate and the concrete. Inward displacement of the liner plate at the penetration is also controlled by the stud bolts.

In the case of a pipe penetration in which significant external operating loads are imposed upon the penetration, the stress level from the external loads is limited to the design stress intensity values, $S_m$, given in the ASME Boiler and Pressure Vessel Code, Section III, Article 4. The stress level in the stud bolts from external loads is in accordance with the AISC Code.

The combining of stresses from all effects is performed using the methods outlined in the ASME Boiler and Pressure Vessel Code, Section III, Article 4, Figure N-414. The maximum stress intensity is the value from Figure N-415 (A) of the previously referenced code. Figure 3-32 shows a typical penetration and the applied loads.

Design stresses for the effects of pipe loads, pressure loads, dead load, and earthquake were calculated and the stress intensity kept below $S_m$.

The stresses from the remaining effects were combined with the above-calculated stresses and the stress intensity kept below $S_a$.

c. Thermal Gradient

The only high temperature lines penetrating the Reactor Building shell are the main steam and feedwater. Cooling fans and stacks designed to maintain the temperature in the penetration below 150°F are provided.

4. Liner Plate

There are no design conditions under which the liner plate is relied upon to assist the concrete in maintaining the integrity of the structure even though the liner will, at times, provide assistance in order to maintain deformation compatibility.

Loads are transmitted to the liner plate through the anchorage system and direct contact with the concrete and vice versa. Loads may be, at times, also transmitted by bond and/or friction with the concrete. These loads cause, or are caused by, liner strain. The liner is designed to withstand the predicted strains.

Possible cracking of concrete has been considered and reinforcing steel is provided to control the width and spacing of the cracks. In addition, the design is made such that total structural deformation remains small during the loading conditions, and that any cracking will be orders of magnitude less than that sustained in the repeated attempts to fail the prestressed concrete reactor vessel "Model 1," and even smaller than the concrete strains of overpressure tests of "Model 2" (both at General Atomic). See Reference 4 on page 3-138 and Reference 5 on page 3-138.

As described, the structural integrity consequences of concrete cracking are limited by the bonded reinforcing and unbonded tendons provided in accordance with the design criteria. The effect of concrete cracking on the liner plate has also been considered. The anchor spacing and other design criteria are such that the liner will sustain orders of magnitude of strain, for example, less than did the liner of Model 1 at General Atomic (Reference 4 on page 3-138) without tensile failure.

5. Liner Plate Anchors

The liner plate anchors were designed to preclude failure when subjected to the worst possible loading combinations. The anchors were also designed such that, in the event of a missing or failed anchor, the total integrity of the anchorage system would not be jeopardized by the failure of adjacent anchors.

The following loading conditions were considered in the design of the anchorage system:

a. Prestress

b. Internal Pressure

c. Shrinkage and Creep of Concrete

d. Thermal Gradients

e. Dead Load

f. Earthquake

g. Wind or Tornado

h. Vacuum

The following factors were considered in the design of the anchorage system:

a. Initial inward curvature of the liner plate between anchors due to fabrication and erection inaccuracies.

b. Variation of anchor spacing.

c. Misalignment of liner plate seams.

d. Variation of plate thickness.

e. Variation of liner plate material yield stress.

f. Variation of Poisson's ratio for liner plate material.

g. Cracking of concrete in anchor zone.

h. Variation of the anchor stiffness.

The anchorage system satisfies the following conditions:

a. The anchor has sufficient strength and ductility so that its energy absorbing capability is sufficient to restrain the maximum force and displacement resulting from the condition where a panel with initial outward curvature is adjacent to a panel with initial inward curvature.

b. The anchor has sufficient flexural strength to resist the bending moment which would result from Condition 5a.

c. The anchor has sufficient strength to resist radial pull-out force.

When the liner plate moves inward radially as shown in Figure 3-33, the sections will develop membrane stress due to the fact that the anchors have moved closer together. Due to initial inward curvature, the section between 1 and 4 will deflect inward giving a longer length than adjacent sections and some relaxation of membrane stress will occur. It should be noted here that section 1-4 cannot reach an unstable condition due to the manner in which it is loaded.

The first part of the solution for the liner plate and anchorage system is to calculate the amount of relaxation that occurs in section 1-4, since this value is also the force across anchor 1 if it is infinitely stiff. This solution was obtained by solving the general differential equation for beams and the use of calculus to simulate relaxation or the lengthening of section 1-4. Figure 3-33 shows the symbols for the forces that result from the first step in the solution.

Using the model shown in Figure 3-34 and evaluating the necessary spring constants, the anchor was allowed to displace.

The solution yielded a force and displacement at anchor 1, but the force in section 1-2 was (N) - $K_{R(Plate)}S_1$ and anchor 2 was no longer in force equilibrium.

The model shown in Figure 3-34 was used to allow anchor 2 to displace and then to evaluate the effects on anchor 1.

The displacement of anchor 1 was $S_1 + S'_1$ and the force on anchor 1 was $K_c(S_1 + S'_1)$. Then anchor 3 is not in force equilibrium and the solution continued to the next anchor.

After the solution was found for displacing anchor 2 and anchor 3, the pattern was established with respect to the effect on anchor 1 and by inspection, the solution considering an infinite amount of anchors was obtained in the form of a series solution.

The preceding solution yielded all necessary results. The most important results were the displacement and force on anchor 1.

Various patterns of welds attaching the angle anchors to the liner plate have been tested for ductility and strength when subjected to a transverse shear load such as N and are shown in Figure 3-35.

Using the results from these tests together with data from tests made for the Fort St. Vrain PSAR, Amendment No. 2 and Oldbury vessels, Reference 6 on page 3-138, a range of possible spring constants was evaluated for the Oconee liner. By using the solution previously obtained together with a chosen spring constant, the amount of energy required to be absorbed by the anchor was evaluated.

By dividing the amount of energy that the system will absorb by the most probable maximum energy, the result then yielded the factor of safety.

By considering the worst possible loading condition which resulted from the listed loading conditions and conditions stated below, the results in Table 3-15 were obtained.

Case I – Simulates a plate with a yield stress of 36 Ksi and no variation in other parameters.

Case II – Simulates a 1.25 increase in yield stress and no variation in any other parameters.

Case III – Simulates a 1.25 increase in yield stress, a 1.16 increase in plate thickness and a 1.08 increase for all other parameters.

Case IV – Simulates a 1.88 increase in yield stress with no variation of any other parameters.

Case V – Is the same as Case III except the anchor spacing has been doubled to simulate what happens if an anchor is missing or has failed.

6. Supports

In designing for structural bracket loads applied perpendicular to the plane of the liner plate, or loads transferred through the thickness of the liner plate, the following criteria and methods have been used:

a. The liner plate was thickened to reduce the predicted stress level in the plane of the liner plate. The thickened plate with the corresponding thicker weld attaching the bracket to the plate will also reduce the probability of the occurrence of a leak at this location.

b. Under the application of a real tensile load applied perpendicular to the plane of the liner plate, no yielding is to occur in the perpendicular direction. By limiting the predicted strain to 90 percent of the minimum guaranteed yield value, this criterion was satisfied.

c. The allowable stress in the perpendicular direction was calculated using the allowable predicted strain in the perpendicular direction together with the predicted stresses in the plane of the liner plate.

d. In setting the above criteria, the reduced strength and strain ability of the material perpendicular to the direction of rolling (in plane of plate) was also considered in the bracket did not penetrate the liner thickened plate. In this case, the major stress is normal to the plane of the liner plate. The allowable stresses were reduced to 75 percent of the stress permitted in Item (3) above.

e. The necessary plate characteristics were assured by ultrasonic examination of the thickened plates for lamination defects.

### 3.8.1.5 Structural Acceptance Criteria

This section documents the manner in which the structural acceptance criteria were met by the designer.

Section 3.8.1.5.1, "Results of Analysis" consists of isostress plots and tabulations of predicted stresses for the various materials. The isostress plots of the homogeneous uncracked concrete structure indicate the general stress pattern for the structure as a whole, under various loading conditions. More specific documentation is made of the predicted stresses for all materials in the structure. In these tabulations, the predicted stress is compared with the allowable to permit an easy comparison and evaluation of the adequacy of the design.

Sections 3.8.1.5.3, "Liner Plate" on page 3-101 and 3.8.1.5.4, "Penetrations" on page 3-102 illustrate the actual details used in the design to implement the criteria.

#### 3.8.1.5.1 Results of Analysis

The isostress plots, Figure 3-25 and Figure 3-26, show the three principal stresses and the direction of the principal stresses normal to the hoop direction. The principal stresses are the most significant information about the behavior of the structure under the various conditions and were a valuable aid for the final design.

The plots were prepared by a cathode-ray tube plotter. The data for plotting were taken from the stress output of the finite element computer program for the following design load cases:

$D + F$

$D + F + 1.15P$

$D + F + 1.5P + T_A$

$D + F + T_A$

The above axisymmetric loading conditions have been found to be governing in the design since they result in highest stresses at various locations in the structure.

The containment stress analysis results for structural concrete and liner plate, including shear stresses, are shown in Table 3-16.

#### 3.8.1.5.2 Prestress Losses

In accordance with the ACI Code 318-63, the design provides for prestress losses caused by the following effects:

1. Seating of anchorage.
2. Elastic shortening of concrete.
3. Creep of concrete.
4. Shrinkage of concrete.
5. Relaxation of prestressing steel stress.
6. Frictional loss due to intended or unintended curvature in the tendons.

All of the above losses can be predicted with sufficient accuracy.

The environment of the prestress system and concrete is not appreciably different, in this case, from that found in numerous bridge and building applications. Considerable research has been done to evaluate the above items and is available to designers in assigning the allowances. Building code authorities consider it acceptable practice to develop permanent designs based on these allowances.

The following categories and values of prestress losses have been considered in the design:

| Type of Loss | Assumed Value |
|---|---|
| Seating of Anchorage | None |
| Elastic Shortening | $\frac{f_{cpi}}{3.0 \times 10^6}$ Inch/Inch |
| Creep of Concrete | $0.280 \times 10^{-6}$ Inch/Inch/psi |
| Shrinkage of Concrete | $100 \times 10^{-6}$ Inch/Inch |
| Relaxation of Prestressing Steel | 8% of $0.65f_s$ = 12.5 Ksi |
| Frictional Loss | K = 0.0003, $\mu$ = 0.156 |

There is no allowance for the seating of the BBRV anchor since no slippage occurs in the anchor during transfer of the tendon load into the structure. Sample lift-off readings will be taken to confirm that any seating loss is negligible.

The loss of tendon stress due to elastic shortening was based on the change in the initial tendon relative to the last tendon stressed.

The concrete properties study conducted at Clemson University indicated an acutal creep value of $0.280 \times 10^{-6}$ inch/inch/psi. Conversion of the unit creep data to hoop, vertical and dome stress gives these values of stress loss in the tendons:

| | |
|---|---|
| Hoop | -16.1 Ksi |
| Vertical | -8.05 Ksi |
| Dome | -16.1 Ksi |

A single creep loss figure of $420 \times 10^{-6}$ inch/inch at 1500 psi ($f_{cpi}$) was used throughout the structure. This results in a prestress loss of 12.6 Ksi.

The value used for shrinkage loss represents only that shrinkage that could occur after stressing. Since the concrete is, in general, well aged at the time of stress, little shrinkage is left to occur and add to prestress loss.

The value of relaxation loss is based on the information furnished by the tendon system vendor, The Prescon Corporation.

Frictional loss parameters for unintentional curvature (K) and intentional curvature ($\mu$) are based on full-scale friction test data. This data indicates actual values of K = 0.0003 and $\mu$ = 0.125 versus the design values of K = 0.0003 and $\mu$ = 0.156.

Column lines B, D, J, and M were braced with diagonal members. For lines D, J, and M, this bracing took the form of two members for each brace with batten plates and angle lacing tying them together.

Dynamic Seismic Analysis

4 A dynamic seismic analysis of the building was performed consisting of a three mass system. Section
4 3.8.5.3.1, "Turbine Building" on page 3-131 describes loading conditions for the dynamic seismic analysis.
4 Maximum accelerations in the transverse direction were taken as the absolute sum of the accelerations
4 associated with the first three mode shapes and were 0.47 g at the roof, 0.20 g at the Operating floor, and
4 0.16 g at the Mezzanine floor. Maximum accelerations in the longitudinal direction were taken as the
4 absolute sum of the accelerations associated with the first three mode shapes and were 0.57 g at the roof,
4 0.24 g at the Operating floor, and 0.18 g at the Mezzanine floor. It is considered that the absolute sum is a conservative value. The structure was analyzed using these accelerations and stresses were found to be within design criteria. Typical stress values, shown as percentage of allowable, are as follows:

| Location | Normal Load | Seismic Load #2 (Static Analysis) | Seismic Load Dynamic Analysis) |
|---|---|---|---|
| Col. D at basement | 83% | 68% | 44% |
| Col. D at roof | 94% | 36% | 28% |
| Col. J below oper. floor | 81% | 51% | 67% |
| Col. J above oper. floor | 78% | 37% | 48% |
| Col. J at roof | 88% | 54% | 59% |
| Col. M below oper. floor | 89% | 72% | 75% |
| Col. M at oper. floor | 84% | 36% | 47% |
| Col. M at roof | 90% | 56% | 48% |

#### 3.8.5.4.2 Keowee Structures

The Keowee structures are designed using conventional structural analytical techniques.

#### 3.8.5.4.3 Class 3 Structures

Class 3 structures are designed in accordance with design methods of accepted standards and codes insofar as they are applicable.

### 3.8.5.5 Structural Acceptance Criteria

The load combinations used in the design of the Turbine Building and Keowee structures and section strengths required to resist those load combinations are given in Section 3.8.5.3, "Loads and Load Combinations" on page 3-131.

### 3.8.5.6 Materials, Quality Control, and Special Construction Techniques

Keowee Structures

All structures utilize 3000 psi concrete, 40,000 psi reinforcing steel and A36 structural steel.

## 3.8.6 FOUNDATIONS

The foundation for the Reactor Building is described in Section 3.8.1.1, "Description of the Containment" on page 3-75.

a. Dead load plus live load with allowable stresses in accordance with ACI Code. The maximum calculated stresses were $f_s$ = 19,700 psi and $f_c$ = 1160 psi.

b. Dead load plus live load plus seismic load equal to 0.15 times the combined dead-live load. The allowable stresses were $f_s$ = 0.9 $f_y$ = 36,000 psi and $f_c$ = 0.85 f'c = 2550 psi. The maximum calculated stresses were $f_s$ = 24,000 psi and $f_c$ = 1410 psi. It is apparent that the seismic loads could be substantially increased with resulting stresses being well below those allowable.

4. Breaker Vault

The Breaker Vault is located on the Operating Floor level of the Keowee Powerhouse and was designed primarily to afford tornado protection for electrical equipment. The controlling case was dead load plus equipment loads plus tornado wind and missile. Resulting stresses for this case were $f_s$ = 38,000 psi and $f_c$ = 2190 psi.

These compare to the allowable $f_s$ = 0.9 $f_y$ = 36,000 psi and $f_c$ = 0.85 f'c = 2550 psi. The actual steel stresses were about 5-½ percent over the allowable stresses but 5-½ percent below the guaranteed minimum yield stress and are considered satisfactory for this severe loading combination.

A second case considered dead load plus seismic loads equal to 0.15 times the combined dead-live loads plus normal wind load. By inspection, it was found that this would result in substantially lower stresses than the loading combination above. Therefore, a detailed design check was not made.

5. Intake Structure

Three design cases were considered:

a. Construction condition (dead load plus wind load) with no water and allowable stresses being within the ACI and AISC Code. The resulting stresses were extremely low.

0 b. Structure unwatered and stop logs in place. Allowable stresses were based on ACI and AISC Code. Calculated stresses were found to be well within the code limits.

c. The third case considered the cylinder gate open, dead loads and seismic loads equal to 0.15 times the dead load. Maximum calculated stresses were $f_s$ = 39,700 psi and $f_c$ = 2050 psi.

The resulting steel stresses are marginally below the guaranteed minimum yield stress and are considered satisfactory for the severe loading combination.

## 3.8.5.4 Design and Analysis Procedures

### 3.8.5.4.1 Turbine Building

Based on the basic criteria and general arrangement drawings of the Turbine Building, design studies were made to determine building dimensions, type of steel, member sizes, and shapes. A computer program, "Stress," was used in the analysis of the bents.

Transverse Analysis

Each bent consisted of the three main crane columns, on lines D, J, and M, the roof girders, the columns of lines K and L and the operating and mezzanine floor framing. Where continuity of framing was not interrupted by the turbine-generator support, the short columns and operating and mezzanine floor framing were included as a part of the rigid frame. See Figure 3-38 for typical Turbine Building cross-section.

Longitudinal Analysis

S = Allowable stress due to normal loading - from AISC specifications

D = Dead Loads (Equipment loads included in the case of seismic loadings)

L = Live Loads

W = Wind Loads

E = Loads from Seismic Loading No. 1

E′ = Loads from Seismic Loading No. 2

#### 3.8.5.3.2 Keowee Structures

1. Powerhouse

A typical reinforced concrete frame was investigated for the following loading conditions using a static type analysis:

a. Dead load plus live load (1000 lbs per square foot) using allowable stresses in accordance with ACI Code. The maximum calculated stresses were $f_s$ = 18,590 psi and $f_c$ = 1122 psi.

b. Dead load plus live load (1000 lbs per square foot) plus seismic load equal to 0.10g times the dead load. The maximum calculated stresses were $f_s$ = 19,120 psi and $f_c$ = 1189 psi. Allowable stresses were $f_s = 0.9\, f_y$ = 36,000 psi and $f_c = 0.85\, f'c$ = 2550 psi.

c. Dead load plus live load (1000 lbs per square foot) plus seismic load equal to 0.20g times the dead load. The maximum calculated stresses were $f_s$ = 19,700 psi and $f_c$ = 1229 psi.

The large live loading of 1000 lbs per square foot was included to allow for heavy equipment loads expected during construction and maintenance. Therefore, to be conservative, the 1000 lbs per square foot was included to b and c above but with seismic loadings added as a function of dead load only.

2. Spillway

A typical spillway pier was investigated for the following loading conditions:

a. Dead load plus hydrostatic load with allowable stresses in accordance with ACI Code. The maximum calculated stresses were $f_s$ = 0 and $f_c$ = 61.7 psi.

b. Dead load plus hydrostatic load plus seismic load equal to 0.10 times dead load. The maximum calculated stresses were $f_s$ = 7760 psi and $f_c$ = 173 psi. The allowable stresses were $f_s = 0.9\, f_y$ = 36,000 psi and $f_c = 0.85\, f'c^*$ = 3400 psi.

c. Same as b except seismic load equal to 0.20 times dead load. The maximum calculated stresses were $f_s$ = 16,350 psi and $f_c$ = 227 psi.

* f'c = 4000 psi in piers.

In addition, the taintor gate thrust girder was investigated for the following loading conditions:

a. Dead load plus hydrostatic load with allowable stresses in accordance with AISC Code. The maximum calculated stress was $f_s$ = 23,300 psi.

b. Dead load plus hydrostatic load plus seismic load equal to 0.10 times dead load with allowable stress = $0.9\, f_y$ = 32,500 psi. The maximum calculated stress was $f_s$ = 25,000 psi.

c. Same as b except seismic load equal to 0.20 times dead load. The maximum calculated stress was $f_s$ = 28,800 psi.

3. Service Bay Substructure

The Service Bay substructure contains the Control Room, Cable Room, Equipment Room, and Battery Room areas. The substructure was investigated for the following loading conditions:

Cranes - 180 Ton Crane and 80 Ton Crane at rest and in unloaded condition.

Crane Columns and Girders - Calculated weights.

4 Loading for Dynamic Seismic Analysis - (Load Combinations)

4 Critical Damping = 2%

4 Maximum Ground Motion Acceleration - 10% of gravity.

4 Reference subsection "Dynamic Seismic Analysis" in Section 3.8.5.4.1, "Turbine Building" on
4 page 3-135 for design accelerations.

4 Loadings:

4 Roof - 25 psf

4 Operating Floor - dead load of floor plus equipment load. (Equipment load estimated at 150 psf)

4 Mezzanine Floor - dead load of floor plus equipment load. (Equipment load estimated at 150
4 psf)

4 Upper Surge Tank and Floor - 65 psf plus tank at normal operating condition.

4 Cranes - 180 Ton and 80 Ton Capacity Cranes at rest and in unloaded condition.

4 Crane Columns and Girders - calculated weights.

Seismic Loading No. 2 was introduced approximately six months after the building was analyzed for Seismic Loading No. 1. With more complete information, it was apparent that the equipment loads assumed for the Operating and Mezzanine Floors were too conservative. Therefore, the equipment loads were reduced for the analysis for Seismic Loading No. 2.

2. Longitudinal Loading

The loadings were applied as follows:

Wind Load - 30 psf.

Crane Load - 10% of Maximum wheel load.

Seismic Loading No. 1 - Same as Seismic Loading No. 1 for Transverse Analysis with the following exceptions:

Loadings - Operating Floor - Equipment load estimated at 130 psf.

Mezzanine Floor - Equipment load estimated at 110 psf.

Seismic Loading No. 2 - Same as Seismic Loading No. 2 for Transverse Analysis with the following exceptions:

Loadings - Operating Floor - Equipment load estimated at 130 psf.

Mezzanine Floor - Equipment load estimated at 110 psf.

4 Loading for Dynamic Seismic Analysis - Same as loading for dynamic seismic analysis for transverse
4 direction.

3. Loading Combinations and Factors

$S = 1.0\,D + 1.0\,L$

$1.33S = 1.0\,D + 1.0\,L + 1.0\,W$

$1.33S = 1.0\,D + 1.0\,E$

$1.64S = 1.0\,D + 1.0\,E'$

Operating Floor

- Turbine Bay - 600 psf.
- Heater Bay - 400 psf.

4 Mezzanine Floor

4 - General Area - 250 psf
4 - Moisture Separator Tube Pull Area - 400 psf
4 - Moisture Separator Lay Down Area - 30 kip concentrated load @ c/l of
4 collector beams

4 Upper Surge Tank Floor - 100 psf (all areas except those between column lines 28 &
4 29 and 44 & 45 - 250 psf)

Cranes - 180 Ton and 80 Ton Cranes fully loaded, lifted load and lateral force arranged to produce maximum stresses. The lateral forces were reduced to 15 percent of the sum of the weights of the lifted load and the crane trolleys.

Wind load - 30 psf.

Seismic Loading No. 1 - (Load Combinations)

Critical Damping - 2%

Maximum Ground Motion Acceleration - 5% of gravity

Maximum Acceleration for Design - 12% of gravity (This is the maximum value of the acceleration response curve for 2% damping.)

Loadings - Roof - 50 psf, reduced to 25 psf when the type of roof construction was finalized.

Operating Floor - dead load of floor plus equipment load. (Equipment load estimated at 250 psf.)

Mezzanine Floor - dead load of floor plus equipment load. (Equipment load estimated at 150 psf.)

Upper Surge Tank and Floor - 65 psf plus tank at normal operating condition.

Crane - 180 Ton Crane, fully loaded, at center of bay.

Crane Columns and Girders - Calculated weights.

Seismic Loading No. 2 - (Load Combinations)

Critical Damping = 2%

Maximum Ground Motion Acceleration - 10% of gravity

Maximum Acceleration for Design - 22% of gravity (This is the maximum value of the acceleration response curve for 2% damping.)

Loadings - Roof - 25 psf.

Operating Floor - dead load of floor plus equipment load. (Equipment load estimated at 200 psf.)

Mezzanine Floor - dead load of floor plus equipment load. (Equipment load estimated at 125 psf.)

Upper Surge Tank and Floor - 65 psf plus tank at normal operating condition.

2 The underwater weir retains an emergency water supply in the event that the waters of Lake Keowee
2 are released by the failure of a dam or dike.

2 6. CCW Intake Piping

2 The CCW Intake Piping conveys water from the CCW pumps on the intake structure to the
2 condenser, supplies water to the LPSW Pumps, and serves as the reservior for the Auxiliary Service
2 Water System.

2 7. CCW Discharge Piping

2 The CCW Discharge Piping conveys water from the condenser to the discharge structure and
2 supplements the CCW intake piping as a reservior for the Auxiliary Service Water System.

2 8. ECCW Piping

2 The ECCW Piping serves two different functions. 1) It can siphon the Condenser Circulating Water
2 through the Condenser to be discharged at the treatment pond. 2) It can be used for recirculation of
2 the Condenser Circulating Water back to the Intake Canal.

### 3.8.5.2 Applicable Codes, Standards, and Specifications

Class 2 structures are designed in accordance with the following codes:

ACI 318-63 - Building Code Requirements for Reinforced Concrete

AISC - Steel Construction Manual, 6th ed.

2 The working stress design method will be used for normal and seismic conditions and stress will be in
2 accordance with above codes. Class 2 structures are qualified for the Design Base Earthquake (DBE). All
2 Keowee Structures necessary for Emergency Power Generation, the Oconee Turbine and Auxiliary
2 Buildings (except as included in Class 1), the Oconee Intake Structure, the CCW Intake Piping, the CCW
2 Discharge Piping, ECCW Piping (structural portion), the Oconee Intake Canal Dike, and the Oconee
2 Intake Underwater Weir are designed for Maximum Hypothetical Earthquake (MHE).

### 3.8.5.3 Loads and Load Combinations

#### 3.8.5.3.1 Turbine Building

1. Transverse Loading

The loadings were applied as follows:

Dead Loads - Roof - 50 psf, reduced to 25 psf when the type of roof construction was finalized.

Floors - Grating Areas - 20 psf.

Concrete Areas

- Operating Floor - 11-1/2 in. masonry - 170 psf.
- Mezzanine Floor - 8 in. masonry - 115 psf.
- Upper Surge Tank Floor - 4 in. masonry - 65 psf plus tank at normal operating condition.

Crane Columns and Girders - Calculated weights.

Live Loads - Roof - 50 psf.

Grating Areas - 100 psf.

#### 3.8.4.7.1 Applicable Codes and Standards

The criteria for the re-evaluation of masonry walls is contained in Attachment 4 of Reference 14 on page 3-138. This criteria uses the American Concrete Institute "Building Code Requirements for Concrete Masonry Structures," ACI 531-79, as the governing code with supplemental allowables specified for cases not directly addressed in the code.

#### 3.8.4.7.2 Loads and Load Combinations

The design loadings for the masonry walls at Oconee are those specified in Section 3.8.4, "Other Seismic Class I Structures" on page 3-127. The only thermal effects which a masonry wall experiences are those pertinent to normal operation, and these are not considered a significant design consideration.

#### 3.8.4.7.3 Upgrade and Modification of Masonry Walls

A program of repairs was performed on selected masonry walls. The walls included in this program were not found to be unsafe in their original configuration; however, an added margin of safety was desired for these walls. The repairs provide increased factors of safety by either upgrading the walls to meet the allowable stresses set forth in the re-evaluation criteria or by shielding the safety related equipment in proximity of the walls from damage, assuming the masonry walls were to collapse. References 12 on page 3-138 through 24 on page 3-139 pertain to I.E. Bulletin 80-11.

## 3.8.5 NONCLASS 1 STRUCTURES

Class 2 structures are those whose limited damage would not result in a release of radioactivity and would
2 permit a controlled plant shutdown but could interrupt power generation. The Turbine Building, the
2 condenser circulating water structures, and the Keowee structures as listed in Section 3.2.1.1.2, "Class 2"
2 on page 3-37 are Class 2 structures.

Class 3 structures are those whose failure could inconvenience operation, but which are not essential to power generation, orderly shutdown or maintenance of the reactor in a safe condition. They include all structures not included in Class 1 and 2.

### 3.8.5.1 Description of the Structures

1. Turbine Building

   The building was constructed of reinforced concrete below grade consisting of substructure walls and a mat foundation. Above grade, the building consists of structural steel with metal siding.

2. Keowee Structures

2 The Keowee Structures considered are Powerhouse, Power and Penstock Tunnels, Spillway, Service Bay Substructure, Breaker Vault, and Intake Structure.

2 3. Dams and Dikes

2 The Keowee Dam, the Little River Dam and Dikes, and the Oconee Intake Canal Dike impound the
2 waters of Lake Keowee to provide the source of flowing water for the Keowee hydroelectric power
2 plant.

2 4. Oconee Intake Structure

2 The intake structure supports the CCW pumps, intake screens, and inlets of the CCW pipes.

2 5. Oconee Intake Underwater Weir

The Spent Fuel Pool Slab was designed for the postulated cask drop accident. Fill concrete was placed from sound rock to the bottom of the fuel pool slab in the area covered by the cask crane to prevent the shearing of a large plug from the pool slab in the event the cask was accidentally dropped.

The height of the cask drop is the maximum vertical travel of the crane and is 45 feet of which 40 feet is through water. The penetration of the cask into the slab is calculated to be 1.75 inches. No credit was taken for the water resistance nor the resistance of the liner plate.

The geometry and strength characteristics (edge radius, base material, etc.) of the cask will be specified to assure that the calculated penetration can occur without rupturing the liner plate. The analysis considers local concrete crushing and liner yielding; however, the strains in the liner plate will not exceed ultimate.

### 3.8.4.5 Structural Acceptance Criteria

The areas of the Auxiliary Building housing the facilities listed in Section 3.8.4.1, "Description of the Structure" on page 3-128 have been designed for the loads and conditions as shown in Table 3-23 with maximum allowable stresses as follows:

| Loading Condition | Maximum Allowable Stress |
|---|---|
| A | Stresses in accordance with ACI and AISC Codes |
| B, D | $fc = 0.85\, f'_c$ for Flexure |
| | $fc = 0.70\, f'_c$ for tied compression members |
| | Shear $= 1.1\sqrt{f'_c} \times 1.33$ for beams with no web reinforcing |
| | $fs = 0.90\, fy$ for Flexure |
| | $fs = 0.90\, fy$ for web reinforcing |
| | $fs = 0.85\, fy$ for reinforcing steel with lap or mechanical splices |
| | Bond $= \dfrac{3.4\sqrt{f'_c}}{D} \times 1.33$ for top bars |
| | $= \dfrac{4.8\sqrt{f'_c}}{D} \times 1.33$ other than top bars |
| C, E | Analyzed on basis of Reference 7 on page 3-138 |

### 3.8.4.6 Materials, Quality Control, and Special Construction Techniques

This information is outlined in applicable portions of Section 3.8.1.6, "Materials, Quality Control, and Special Construction Techniques" on page 3-104.

### 3.8.4.7 Concrete Masonry Walls

2 The masonry walls are in-fill panels serving as partitions with some walls having pressure, fire and radiation barrier applications. The walls are single or multiwythe, constructed of hollow or grouted concrete blocks or solid concrete blocks or bricks. All masonry walls are non-structural and constructed on a structural support system. Pursuant to I.E. Bulletin 80-11, a safety re-evaluation of all masonry walls was undertaken by Duke Power Company. As a result of this reevaluation effort certain masonry walls were modified to meet minimum factors of safety.

### 3.8.4.1 Description of the Structure

Auxiliary Building

The building was constructed on a 5.00 foot thick reinforced concrete mat foundation. Below grade, the building consists of reinforced concrete walls and slabs. Above grade, the building consists principally of reinforced concrete columns, beams, and slabs, with the slabs acting as diaphragms.

The following facilities related to the Nuclear Steam Supply System are located in the Auxiliary Building:

1. New and Spent Fuel Handling, Storage, and Shipment
2. Control Rooms
3. Waste Disposal System
4. Chemical Addition and Sampling System
5. Component Cooling System
6. Reactor Building Spray Systems
7. High and Low Pressure injection System
8. Spent Fuel Cooling System
9. Electrical Distribution System

### 3.8.4.2 Applicable Codes, Standards, and Specifications

The Class 1 Structure is designed according to the applicable codes and specifications listed in Section 3.8.1.2, "Applicable Codes, Standards, and Specifications" on page 3-76.

### 3.8.4.3 Loads and Load Combinations

The loads and load combinations considered for the design of the Auxiliary Building are listed in Table 3-23. The final design of the Auxiliary Building satisfies the loading combinations and factors tabulated in Table 3-14.

### 3.8.4.4 Design and Analysis Procedures

The design of the Auxiliary Building is performed using conventional structural analytical techniques. The provisions of the design for the Spent Fuel Pool are described below:

The Spent Fuel Pool Walls were analyzed for thermal loads in accordance with methods presented in ACI 505. The exterior wall temperature was assumed to be 60°F for areas enclosed by the Auxiliary Building and 0°F for exposed areas.

Under normal conditions, the interior wall temperature was 150°F and the maximum calculated thermal stress was 996 psi for concrete and 11,410 psi for reinforcing steel.

After prolonged outage of the cooling system, the interior wall temperature could reach 212°F and the maximum calculated thermal stress was 1681 psi for concrete and 25,600 psi for reinforcing steel. Reinforcing steel conforming with ASTM A516, Grade 60, was used.

A minimum of 0.30 percent reinforcing was used in the spent fuel pool walls to control concrete cracking. Also, a ¼ inch thick steel liner was used on the inside face of the pool for leak tightness.

West Steam Generator Compartment – 15 psi

In addition to the peak differentials, the steam generator compartment walls are designed for simultaneous action of a single jet impingement load and the safe shutdown earthquake.

Pipe whipping restraints are provided for the main steam, feedwater, and other high-pressure piping.

### 3.8.3.5 Structural Acceptance Criteria

The Reactor Building interior structure (comprising all elements inside the Reactor Building shell) is a Seismic Class 1 structure and is designed on the following bases:

1. The stresses in any portion of the structure under the action of dead load, live load, and design seismic load will be below the allowable stresses given by either the ACI Building Code, ACI 318-1963 except as noted in Section 3.8.1.3.6, "Loads Necessary to Cause Structural Yielding" on page 3-79, AISC Manual of Steel Construction, 6th Edition.
2. The stresses in any portion of the structure under the action of dead load, and thermal load will be below 133 percent of the allowable stresses given in (1).
3. The capability to safely shut down the plant will be maintained under the combined action of dead load, maximum seismic load, pressure and jet impingement load. The latter two loads are based on the rupture of one pipe in the primary loop. The deflections of structures and supports under these combined loads would be such that the functioning of engineered safeguards equipment would not be impaired. The yield load equations in Section 3.8.1.3.6, "Loads Necessary to Cause Structural Yielding" on page 3-79 are adhered to except that local yielding is permitted for pipe, jet or missile barriers provided there is no general failure.

### 3.8.3.6 Materials, Quality Control, and Special Construction Techniques

The materials used for the structural elements are as follows:

| | | |
|---|---|---|
| Structural Steel | – | ASTM A36 |
| Concrete | – | $f'_c$ = 4000 psi at 28 days |
| | – | $f'_c$ = 5000 psi at 28 days (for steam generator bases, reactor foundation, and primary shield wall). |
| Reinforcing | – | ASTM A615, Grade 40 for Bars #11 and under ASTM A615, Grade 60 for Bars larger than #11. |

Additional materials, quality control, and construction techniques and described in Section 3.8.1.6, "Materials, Quality Control, and Special Construction Techniques" on page 3-104.

### 3.8.3.7 Testing and Inservice Surveillance Requirements

Testing and inservice surveillance requirements are outlined in Section 3.8.1.7, "Testing and Inservice Inspection Requirements" on page 3-118.

## 3.8.4 OTHER SEISMIC CLASS I STRUCTURES

is pressurized to 67.8 psig during the surveillance test to check the inward displacement of the liner plate. This program was completed for Oconee 1 on January 2, 1974.

## 3.8.2 STEEL CONTAINMENT

The Reactor Building does not have a steel containment vessel separate from the concrete shell. The description of the steel liner plate and all applicable supporting data is found in Section 3.8.1, "Concrete Containment" on page 3-75.

## 3.8.3 CONCRETE AND STRUCTURAL STEEL INTERNAL STRUCTURES OF THE STEEL CONTAINMENT

### 3.8.3.1 Description of the Internal Structures

The Reactor Building interior structure consists of (1) the reactor cavity, (2) two steam generator compartments, and (3) a refueling pool which is located between the steam generator compartments and above the reactor cavity.

The reactor cavity houses the reactor vessels and serves as a biological shield wall. The reactor cavity is also designed to contain core flooding water up to the level of the reactor nozzle.

Additional descriptive information can be found in Section 3.8.1, "Concrete Containment" on page 3-75.

### 3.8.3.2 Applicable Codes, Standards, and Specifications

The interior structures are designed in accordance with the applicable codes and specifications listed in Section 3.8.1.2, "Applicable Codes, Standards, and Specifications" on page 3-76.

### 3.8.3.3 Loads and Load Combinations

The loads and load combinations considered for the design of the interior structures are described in Section 3.8.1.3, "Loads and Load Combinations" on page 3-77.

### 3.8.3.4 Design and Analysis Procedures

The Reactor Building interior structures are designed using conventional structural analytical techniques. Some of the provisions of the design are described below:

The primary functions of the steam generator compartment walls are to serve as secondary shield walls and to resist the pressure and jet loads.

The foundations for all NSSS equipment including the reactor vessel, the steam generators, and the pressurizer are designed to remain within the elastic range during rupture of any pipe combined with the "maximum earthquake."

The design pressure differential across walls and slabs of enclosed compartments in the internal structure are as follows:

| | |
|---|---|
| Reactor Cavity | − 208 psi |
| East Steam Generator Compartment | − 15 psi |

The steel-lined Reactor Building is self-sufficient, and other than valves and hatch doors, there are no operating parts.

#### 3.8.1.7.5 Engineered Safeguards Tests

The Reactor Building Spray, Penetration Room Ventilation, Reactor Building Cooling Systems, and the Reactor Building Isolation Valves will be tested during refueling or extended maintenance outages to provide approximately annual tests. These tests will include:

1. Reactor Building Cooling and Reactor Building Penetration Room Ventilation Systems.

   Each of these systems is operated periodically during normal operating periods to maintain satisfactory temperatures within the Reactor Buildings and penetration rooms respectively. This normal operator initiated operation of these systems provides verification of the operability. In addition to this normal operation, an annual test of these systems in the engineered safeguards mode will also be performed. This test will be initiated by inserting a simulated engineered safeguards signal as would occur during an accident situation. Verification of the proper operation of the components of these systems will be determined and a record of the test results made a part of the permanent plant records.

2. Reactor Building Spray System

   The Reactor Building Spray System will be tested in the same manner as the systems above with the exception that the Reactor Building spray headers will be isolated to prevent spray water from entering the spray headers. A special test connection is provided ahead of the Reactor Building isolation valves so that the portion of the system outside the Reactor Building will be in normal operation. When the test of that portion of the system outside the Reactor Building has been completed, compressed air will be blown through each of the spray headers in the Reactor Building through special test connections to verily that spray water would be directed into the Reactor Building under accident conditions. Proper operation of the various components of this system will be verified and a record of the test results made a part of the plant records.

3. Reactor Building Isolation Valves

   Proper operability of the Reactor Building isolation valves not covered in the other tests will be verified by inserting a simulated engineered safeguards signal to initiate operation of these valves.

#### 3.8.1.7.6 Post-Tensioning System

A surveillance program for the Reactor Building post-tensioning system, is executed in order to assure the continued quality of the system. The program consists of periodic inspections of nine pre-selected tendons - three horizontal tendons, three vertical tendons and three dome tendons - for symptoms of material deterioration or pre-stress force reduction. The program assesses the condition and functional capability of the system and, therefore, verifies the adequacy of the system and provides an opportunity to take proper corrective action should adverse conditions be detected.

An end anchorage concrete surveillance program for the post-tensioning system is implemented to assure the continued structural integrity of the Reactor Buildings. The program consists of periodic inspections of selected end anchorages and adjacent concrete surfaces.

#### 3.8.1.7.7 Liner Plate

A surveillance program for the Reactor Building liner plate is implemented to assure continued integrity of the liner plate. The initial surveillance is conducted in conjunction with the initial Reactor Building Structural Integrity Test. The surveillance is conducted semiannually and if no corrective action is needed due to deformations, the surveillance program is discontinued after the one year inspection. The building

Penetrations such as the personnel access and emergency hatches cannot be opened except by deliberate action and are interlocked and alarmed by failsafe devices such that the Reactor Building will not be breached unintentionally. The liner plate over the foundation slab is protected by cover concrete. Wherever access to the liner plate is blocked by interior concrete, means are provided so that weld seams can be tested for leakage. The liner plate is protected against corrosion by suitable coatings. Walls and floors for biological and missile shielding, and for access and operating purposes, also provide compartmentation which constitutes protection for the liner during operating as well as accident conditions.

Once the adequacy of the liner has been established initially, there is no reason to anticipate progressive deterioration during the life of the station which would reduce the effectiveness of the liner as a vapor barrier. Inside the Reactor Building, the atmosphere is subject to a high degree of temperature control. The outside of the liner is protected by 3-3/4 feet of prestressed concrete which is exceptionally resistant to all weather conditions.

Inspection on a periodic basis, as necessary, will be conducted in all spaces accessible under full power operation. Biological shielding is provided to reduce radiation to limits which make occupancy of spaces adjacent to the liner permissible.

All penetrations except the following are grouped within or vented to the penetration room. Any leakage that might occur from these penetrations will be collected and discharged through high efficiency particulare air (HEPA) filters and charcoal filters to the unit vent. In this manner, leakage which might occur from these penetrations will be isolated from leakage which might occur through the Reactor Building itself.

1. Main Steam Lines
2. Sump Drain Lines
3. Decay Heat Removal Lines
4. Reactor Building Equipment Drain Lines
3 5. Post-Accident Liquid Sample Lines
3 6. Fuel Transfer Tubes

The above lines are not considered a source of significant leakage because they are welded to the liner plate.

Individual major penetrations or groups of penetrations will be tested by means of permanently installed pressure connections or temporarily installed pressure or vacuum boxes. If necessary, liner plate weld seams will be tested by the vacuum box soap bubble method, where accessible, or by means of the permanently installed backup channels and angles where inaccessible.

In any event, sources of excessive leakage will be located and such corrective action as necessary will be taken. This will consist of repair or replacement. Appropriate action will also be taken to minimize the possibility of recurrence of excessive leakage, including such redesign as might prove to be necessary to protect public health and safety. Leak testing will be continued until a satisfactory leak rate has again been demonstrated.

A considerable background of operation experience is being accumulated on containments and penetrations. Full advantage of this knowledge has been taken in all phases of design, fabrication, installation, inspection, and testing. Practical improvements in design and details have been incorporated as they are developed, where applicable.

3. The engineered safeguards tests will also be performed at more frequent intervals than the integrated leak rate tests to verify the functional capability of these systems which are relied upon to limit or reduce leakage from the containment buildings in the case their service is required. These tests will be performed during outages for refueling and/or major maintenance outages.

The schedule of testing, type of test, and components to be tested are as follows:

Integrated Leak Rate Tests

Integrated leak rate tests shall be performed as follows:

1. Each Reactor Building shall be tested at the calculated peak accident pressure of 59 psig and at one-half this pressure prior to the initial fuel loading.
2. After the initial preoperational leakage rate test, two integrated leakage rate tests shall be performed on each Reactor Building at approximately equal intervals between each major shutdown for inservice inspection to be performed at 10 year intervals. In addition, an integrated test shall be performed at each 10 year interval, coinciding with the inservice inspection shutdown. The test shall coincide with a shutdown for major fuel reloading. These tests shall be conducted at or above one-half peak accident pressure ($P_t$).

Local Leak Detection and Operability Tests (Resilient Seals)

Local leak detection and operability tests shall be performed as required by the Technical Specifications.

The barrier to leakage in the Reactor Building is the one-quarter inch steel liner plate. All penetrations are continuously welded to the liner plate before the concrete in which they are embedded is placed. The penetrations, shown on Figure 3-20 and Figure 3-21, become an integral part of the liner and are so designed, installed, and tested.

The steel liner plate is securely attached to the prestressed concrete Reactor Building and is an integral part of this structure. This Reactor Building is conservatively designed and rigorously analyzed for the extreme loading conditions of a highly improbable hypothetical accident, as well as for all other types of loading conditions which could be experienced. Thorough control is maintained over the quality of all materials and workmanship during all stages of fabrication and erection of the liner plate and penetrations and during construction of the entire Reactor Building.

During construction, the entire length of every seam weld in the liner plate is leak tested. Individual penetration assemblies are shop tested. Welded connections between penetration assemblies and the liner plate are individually leak tested after installation. Following completion of construction, the entire Reactor Building, the liner, and all its penetrations are tested at 115 percent of the design pressure to establish structural integrity. The initial leak rate tests of the entire Reactor Building are conducted at the maximum calculated peak accident pressure and one-half this pressure to demonstrate vapor tightness and to establish a reference for periodic leak testing for the life of the station. Multiple and redundant systems based on different engineering principles are provided as described in Section 3.8.1.7.5, "Engineered Safeguards Tests" on page 3-125, to provide a very high degree of assurance that the accident conditions will never be exceeded and that the vapor barrier of the containment will never be jeopardized.

Under all normal operating conditions and under accidental conditions short of the worst loss-of-coolant accident, virtually no possibility exists that any leakage could occur or that the integrity of the vapor barrier could be violated in any way that would be significant to the public health and safety or to that of the station personnel. Adequate administrative controls will be enforced to minimize the possibility of human error.

rate. It will be demonstrated that the total Reactor Building leakage rate to the environment will maintain public exposure below 10CFR100 limits in the event of an accident.

#### 3.8.1.7.4 Leakage Monitoring

A program of testing and surveillance of each of the three duplicate Reactor Buildings has been developed to provide assurance, during service, of the capability of each containment system to perform its intended safety function. This program consists of tests defined as follows:

Overall integrated leak rate tests of the Reactor Buildings and systems which under post accident conditions become an extension of the containment boundary.

Local leak detection tests of components having resilient seals, gaskets, or sealant compounds that penetrate or seal the boundary of the containment system. Components included in this category are:

1. Personnel Hatches
2. Emergency Hatches
3. Equipment Hatches
3 4. Fuel Transfer Tube Covers
3 5. Electrical Penetrations
3 6. Leak Rate Test Pressurization/Exhaust Penetration

Local leak detection and operability tests of containment isolation valves in systems that vent directly to the Reactor Building atmosphere or the Reactor Coolant system that must close upon receiving an isolation signal and seal the containment under accident conditions. Valves and containment penetrations which during operation are normally valved closed and which if open could be immediately closed, will not require testing.

Operability tests of engineered safeguards systems which under post accident conditions are relied upon to limit or reduce leakage from the containment. Included in these tests are:

1. Reactor Building Spray Systems
2. Reactor Building Penetration Room Ventilation Systems
3. Reactor Building Cooling Systems
4. Reactor Building Isolation Valves not covered above

Following the integrated leakage rate tests, performed as a part of the preoperational testing, subsequent tests will be performed at a pressure of 50 percent of the maximum calculated peak accident pressure or greater. The tests will be performed on schedule based on the following considerations:

1. There are three Reactor Buildings each having the same design. Information pertaining to deterioration in performance obtained in the testing of one Reactor Building is therefore applicable to the other Reactor Buildings.
2. Local leak detection tests will be performed on a more frequent basis than the integrated tests to detect and correct excessive leakage at containment penetrations. Where feasible, these tests will be performed during operation; otherwise, they will be performed during refueling outages and/or major maintenance outages. These tests will be performed at or above the maximum calculated peak accident pressure.

3. Six inoperative gauges mark SGR-4 and SFT-5 are accessible and will be replaced to obtain data for comparison with Palisades and predicted strains for Oconee.
4. Load cells that are inoperative will be repaired or supplemented with prestress rams that have been modified with 20 psi division gauges to measure tendon forces. Prestress rams were used at Palisades and performed satisfactory. Results of measured forces can than be compared with those predicted.

The taut wire system consists of linear potentiometers (infinite resolution type) as the transducer element. Movement of the linear pontentiometers will be actuated by invar wires attached at one end to the point of measurement and at the other end to a reference point. Approximately 35 linear potentiometers will be used to measure building deformations during the structural test.

Oconee 2 and 3 Reactor Buildings are instrumented with the taut wire system for measuring building deformations as described above for Oconee 1. Displacement measurements are made at the following locations:

| | | |
|---|---|---|
| Dome | – | Four points |
| Cylinder Wall | – | Seven elevations at approximately 20 foot intervals at a buttress section and a wall section |
| Equipment Hatch | – | Nine points with six of the points on the horizontal centerline and three of the points on the vertical centerline above the hatch |
| Vertical | – | Two points |

The above locations were selected so that deformation measurements could be compared with Oconee 1 measurements.

Concrete crack patterns are recorded at the base-wall intersection, cylinder wall mid-height, springline, equipment hatch opening, buttress-cylinder wall intersection, cylinder wall-ring girder intersection, and top of ring girder. Each inspection area consisted of approximately 40 square feet. Cracks that exceed 0.01 inch in width are mapped.

#### 3.8.1.7.3 Initial Leakage Tests

Following completion of the Reactor Buildings and prior to the hot functional tests and fueling of the reactors, integrated leakage rate tests will be performed on the containment systems. One test will be performed at or above the maximum calculated peak accident pressure. A second test will be performed at a pressure of not less than 50 percent of maximum calculated peak accident pressure.

The absolute pressure-temperature and/or the reference vessel method will be used for these tests. The objectives of these tests are:

1. To determine the initial integrated leakage rate for comparison with the design leakage rate.
2. To establish representative leakage characteristics of the containment system to permit retesting at reduced pressures.
3. To establish a performance history summary of the integrated leakage rate tests.
4. To establish a test method and the equipment to be used for subsequent retesting.

The leakage rate will be measured by integrating the leakage rate for a period of not less than 24 hours. This integrated leakage will be verified by the "pump-back" method and/or introduction of a known leak rate. The necessary instrumentation will be installed to provide accurate data for calculating the leakage

Bonding and waterproofing materials such as BLH EPY150 Cement, Epoxylite 222 and Microcrystalline Wax are used to install the gauges.

Gauges were calibrated in accordance with the manufacturer's instructions and set at zero reading during installation.

The final procedures in sequence of structural proof testing are as follows:

1. Test strain gauges immediately after installation.
2. Test strain gauges immediately after pouring concrete.
3. Record strains and deflections and observe cracking at three intervals suitably spaced during prestressing and immediately after all prestressing is completed.
4. After prestressing and before testing, a certain number of readings will be taken to determine the effects of creep and shrinkage.
5. Record measurements at increments of 10 psi up to 40 psi and then at increments of 5 psi up to proof-test pressure.
6. Record measurements at 15 psi increments during depressurization.
7. Observe the development of cracks during load application. Measurement of cracks with mechanical dial gauges will be made when deemed pertinent by the test engineer.

The Reactor Building air temperature is monitored by resistance thermometers and the dewpoint temperature is monitored by a dewpoint sensor. Using the Reactor Building coolers and electric heaters, the temperature is maintained between 60° and 100°F and above the dewpoint temperature.

The status of gauges on November 28, 1970 was as follows:

| Gauge Mark | Number Inoperative | Number Operative | Number Being Replaced |
|---|---|---|---|
| SGA-1 | 114 | 4 | (See 2 below) |
| SGE-2 | 7 | 2 | (See 2 below) |
| SGC-3 | 0 | 6 | – |
| SGR-4 | 7 | 11 | 6 |
| SFT-5 | 7 | 19 | 6 |
| LC (Load Cell) | 1 | 6 | (See 4 on page 3-121 below) |
| Taut Wire System | 0 | – | – |

Since a significant number of embedded gauges are inoperative, we believe it prudent to verify the design by (a) utilizing test results from Palisades and, (b) continuing with the Oconee Structural Test, as noted below:

1. The design and construction of Palisades and Oconee Reactor Buildings are very similar. The Palisades' structural instrumentation program was successful and permitted a detailed comparison between design calculations and observed response.
2. At Oconee, the taut wire system (building deformation) will permit verification that the structural response is consistent with the predicted behavior. In addition, twenty-six Carlson SAIOS strain gauges will be surface mounted on the Reactor Building to obtain concrete strains for comparison with Palisades and those predicted for Oconee as shown on Figure 3-37, Sheet 4.

2. To provide direct verification that the in-place tendons (the major strength elements) have a strength of at least 80 percent of guaranteed ultimate tensile strength and that the concrete has the strength needed to sustain a strain range from high initial average concrete compression when unpressurized to low average concrete compression when pressurized.
3. To acquire detailed strain data which will be compared with the analytical predictions.

To achieve objectives, data will be acquired and evaluated to determine the response of the structure during and immediately after post-tensioning to determine any indication of unanticipated and continued deformation under load. A quality assurance program was instituted. In addition, each individual tendon is tensioned in place to 80 percent of the guaranteed ultimate tensile strength and then anchored at a lower load that is still in excess of those predicted to exist at test pressure levels. During pressurization of the structure, the structure's response will be measured at selected pressure levels with the highest being 1.15 times the design pressure. An indication that the structure is capable of withstanding internal pressure will result from these tests. The strain measuring program is described in Section 3.8.1.7.2, "Instrumentation."

Individual test values which fall outside the predicted range will not be considered as necessarily indicative of a lack of adequate structural integrity. Structural integrity cannot be judged on the data acquired from only one sensor since such precise devices may malfunction.

#### 3.8.1.7.2 Instrumentation

The structural response of the building will be assessed by comparing the theoretical analysis to test results of strains and deformations at boundaries, points of stress concentration, openings, areas of maximum creep, and at sections representing typical stress conditions.

The following instruments were installed in the first Reactor Building:

| | |
|---|---|
| 118 | Two element strain rosette, waterproofed BLH Company designation FAET-12-12-56, to be attached to the reinforcing bars. |
| 9 | Linear element, electric resistance strain gauges, BLH designation AS9-1 (Valore Type) to be attached to the surface of the concrete. |
| | Taut wire system for measuring building deformation. |
| 6 | Electric resistance strain gauge, Budd Company designation CP-1101 EX to be attached to the surface of the concrete for measuring crack propagation. |
| 1 | Cement Paint (Figure 3-37) to observe cracks in concrete. |
| 7 | Load cells each containing strain gauges to be attached to the tendons. |
| 18 | Three element rosette, electric resistance strain gauges BLH Company designation FAER-25-12-(60)56, to be attached to the inside and outside face of the liner and penetration nozzles. |
| 26 | Two element strain rosette, BLH Company designation FAET-25-12-5 to be attached to the inside face and outside face of the liner and penetration nozzles. |

The instrument layout is shown on Figure 3-37, sheets 1, 2, and 3. The types and locations of the gauges are described in the legend on the figure. Because of the well-known vulnerability of the bonded resistance gauges to moisture, special care is taken in bonding and waterproofing of the gauges.

In order to reduce the possibilities of faulty preparation of the gauges in the field, the gauges are encapsulated and the wires soldered to the gauge leads and then waterproofed in the shop.

*3.8.1.6.5.2 Instructions for Field Welding Inspectors*

Quality Control procedures are in accordance with the quality assurance requirements outlined in Chapter 17, "Quality Assurance" on page 17-1. Visual inspection is performed afer welding in accordance with NCIG-01, Visual Weld Acceptance Criteria.

*3.8.1.6.5.3 Qualifications for Nondestructive Examination Inspectors.*

Duke Power NDE inspectors are trained, qualified, and certified in accordance with the quality assurance requirements outlined in Chapter 17, "Quality Assurance" on page 17-1.

1. A technician will have a thorough knowledge of the type of testing he is to conduct. He will also be familiar with the welding procedure specification for the field welds he is inspecting.
2. When required by the various codes, the technician is properly certified in accordance with the applicable section of the Society for Nondestructive Testing Recommended Practice No. SNT-TC.1A.

*3.8.1.6.5.4 Instructions for Nondestructive Examination Inspectors*

NDE procedures are in accordance with the quality assurance requirements outlined in Chapter 17, "Quality Assurance" on page 17-1.

*3.8.1.6.5.5 Welding Procedures*

All welding is in strict accordance with approved welding procedure specifications.

Welder Qualification

All welders and welding operators who are to make welds under a code or standard which requires qualification of welders are tested and qualified accordingly before beginning production welding. Duke Power Company is responsible for testing and qualifying its own welders. The welding inspector is responsible in all cases for determining that the welders have passed the necessary qualification tests.

## 3.8.1.7 Testing and Inservice Inspection Requirements

### 3.8.1.7.1 Structural Test

Each of the three Reactor Buildings will be pressurized to 115 percent of design pressure for one hour following completion of construction to establish the structural integrity of the building. The structural integrity test of each building will be conducted in accordance with a written procedure. Operating units will remain in operation during the structural test of another unit. Personnel access limitations included in the written procedures will designate areas of limited access during specific periods of the test. Except for personnel access restrictions, the operation of one unit will not be affected by a building being tested.

The structural integrity test of each building will verify the workmanship involved; in addition, the test of the Oconee 1 Reactor Building will verify the design and workmanship. The response of the Oconee 1 building will be compared with the calculated behavior to confirm the design by means of instrumentation.

*3.8.1.7.1.1 Test Objectives*

1. To provide direct verification that the structural integrity as a whole is equal to or greater than necessary to sustain the forces imposed by two different and large loading conditions.

Tests are made in accordance with the following schedule for each position, bar size and grade of bar:

1 out of first 10 splices

3 out of next 100 splices

2 out of next 100 and each subsequent 100 splices

Test splices are made by having test bars of 3 feet length spliced in sequence with the production bars. In addition, two production splices are cut out and tested for each 100 test splices.

The inspections and tests are performed by individuals thoroughly trained by the CADWELD manufacturer.

For reinforcing steel of size 11 and under, lap splices are permitted in accordance with ACI 318-63, Chapter 8, "Electric Power" on page 8-1.

#### 3.8.1.6.4 Liner Plate

Construction of the liner plate conformed to the applicable portions of Part UW of Section VIII of the ASME Code. Specifically, Paragraphs UW-26 through UW-38, inclusive, applied in their entirety. In addition, the qualification of all welding procedures and welders was performed in accordance with Part A of Section IX of the ASME Code. All liner angle welding was visually inspected prior to, during, and after welding to insure that quality and general workmanship met the requirements of the applicable welding procedure specification.

The erection of the liner plate was as follows:

After the floor plate embedments in the foundation slab had been placed and welded, and concrete was poured flush, the wall liner plates were erected in 60 degree segments and 10 feet high courses. This pattern was followed to the dome spring line and then the steel dome erection trusses were placed. During the period of erection o: wall liner plates, the floor liner plate was placed and welded.

The tolerances for liner plate erection were as follows:

1. The location of any point on the liner plate shall not vary from the design diameter by more than $\pm$ 3 inches.
2. Maximum inward deflection (toward the center of the structure) of the ¼ inch liner plate between the angle stiffeners of 1/8 inch, when measured with a 15 inch straightedge placed horizontally.

#### 3.8.1.6.5 Field Welding

This section outlines the general requirements for welding quality control to assure that all field welding is performed in full compliance with the applicable job specification. These requirements include the use of qualified welding inspectors and nondestructive testing technicians and the assurance that field welding is performed only by qualified welders using qualified procedures.

##### *3.8.1.6.5.1 Qualifications for Welding Inspectors*

Duke Power welding inspectors are qualified in accordance with the quality assurance requirements outlined in Chapter 17, "Quality Assurance" on page 17-1.

Final acceptance for warranty purposes is the successful completion of the pressure testing of the Reactor Building.

#### 3.8.1.6.3 Reinforcing Steel

The concrete inspector visually inspected the shop fabricated reinforcing steel for compliance with drawings and specifications. Intermediate grade reinforcing steel conformed with ASTM A615, Grade 40 and high strength reinforcing steel conformed with ASTM A615, Grade 60. Mill test reports are submitted for engineering review and approval. Metallurgical inspection and testing of the reinforcing steel is done in accordance with the ACI Code 318-63, Chapter 8, "Electric Power" on page 8-1.

Reinforcing steel is inspected at delivery as well as at erection. The condition of the material must meet all of the requirements of ACI 318-63, as well as any additional requirements made by the inspector.

Number 14S and 18S reinforcing steel for which the ACI Code required welded or mechanical splices is spliced by the CADWELD process using full tensile strength "T" series connections. Quality control is maintained by qualification testing of the individual splicing crews, visual inspection of each completed connection, and random sampling and tensile testing of splices.

Prior to splicing operations, bar ends were inspected for damaged deformations and were power brushed to remove all loose mill scale, rust, and other foreign material. Immediately before the splice sleeve positioning, bar ends were preheated to assure complete absence of moisture.

Prior to making any production splices, each individual splicing crew prepares sample splices for tensile testing covering each bar size and position used in production to qualify. The sample splices must be properly filled, free of porous metal and meet the minimum requirement for tensile strength as stated below.

All splices are subjected to visual inspection and must meet the following standards:

1. Sound, nonporous filler metal must be visible at both ends of the splice sleeve and at the tap hole in the center of the splice sleeve. Filler metal is usually recessed ¼ inch from the end of the sleeve due to the packing material, and is not considered a poor fill.
2. Splices which contain slag or porous metal in the riser, tap hole, or at the ends of the sleeves (general porosity) are rejected. A single shrinkage bubble present below the riser is not detrimental and should be distinguished from general porosity as described above.

In addition to the above, random splices are subjected to mechanical tests and must meet the following standards:

1. The strength of 95 percent of the CADWELD splices tested will be greater than 125 percent of the specified minimum yield strength for the particular bar size and ASTM specification.
2. The strength of the average of all the splices tested will be equal to or greater than the minimum ultimate strength for the particular bar size and ASTM specification.
3. No failures of CADWELD splices below the required minimum yield strength are expected. In the unlikely event that one should occur, it would be sent to a testing laboratory for analysis of failure. Based on the testing laboratory's report, additional samples would be taken to insure that there are no other defective welds.

0 to 2500 psi - Accuracy limit of the gauge, plus or minus 50 psi.

2500 to 7030 psi - Plus or minus 2 percent of gauge reading.

Pressure gauges are recalibrated after each stressing cycle on Oconee 3 and, as requested by Duke Power, during and at the end of the tensioning operations on Oconee 1 and 2.

Strain Gauge Installation and Protection

Strain or force gauging devices are installed on certain tendon areas prior to and/or during installation. These strain devices are monitored during the tensioning operation and used during subsequent pressure testing. Approximately 4 tendon sets are instrumented with load cells.

Tests, Samples, Inspections

Sampling and testing conforms to ASTM Standard A421 and as specified herein.

Each size of wire from each mill heat shipped to the site is assigned an individual lot number and tagged in such a manner that each such lot can be accurately identified at the job site. Anchorage assemblies are likewise identified. All unidentified prestressing steel or anchorage assemblies received at the job site are rejected.

Random samples as specified in the ASTM Standard stated above are taken from each lot of prestressing steel used in the work. With each sample of prestressing steel wire that is tested, there is submitted a certificate stating the manufacturer's minimum guaranteed ultimate tensile strength of the sample tested.

For the prefabricated tendons, one completely fabricated prestressing test specimen tendon 5 feet in length, including anchorage assemblies, is tested for each size of tendon contained in individual shipping release.

No prefabricated tendon is shipped to the site without first having been released by Duke, and each tendon is tagged before shipment for identification purposes. The release of any material by Duke does not preclude subsequent rejection if the material is damaged in transit or later damaged or found to be defective.

Duke shop inspects the prefabricated tendons prior to being shipped to the job site.

The anchorages and tendons are inspected at the job site for corrosion and mechanical damage during shipment, storage, installation, and tensioning. Damaged or corroded tendons and anchorages are rejected.

Acceptance

The Reactor Building has been analyzed based on missing tendons for the various loading conditions including missiles. The stresses for the various loading conditions were within the allowable design stresses. The missing tendons will not have any affect on the structure to withstand turbine and tornado generated missiles without loss of function. The missing tendons are located on the northwest face and shielded by location from a direct turbine missile strike. However, as stated in Section 3.5.1.3, "Missiles Generated by Natural Phenomena" on page 3-56, the structure can withstand the loss of three horizontal and three vertical tendons in the cylinder wall without loss of function. The depth of penetration from tornado generated missiles as stated in Section 3.5.1.3, "Missiles Generated by Natural Phenomena" is less than the tendon concrete cover and will not endanger the structural integrity of the Reactor Building.

Phase 7
One hundred and fifty-three hoop tendons from elevation 775 feet + 0 inches to elevation 865 feet + 0 inches on buttresses at 30 degrees, 150 degrees, and 270 degrees.

Phase 8
Forty-two hoop tendons from elevation 776 feet + 0 inches to elevation 801 feet + 6 inches on buttresses at 90 degrees, 210 degrees, and 330 degrees.

Phase 9
One hundred and seventy-six vertical tendons.

Phase 10
Two hundred and fifty-two hoop tendons from elevation 801 feet + 6 inches to elevation 943 feet + 6 inches on buttresses at 90 degrees, 210 degrees, and 330 degrees.

Phase 11
Ten hoop tendons above elevation 949 feet + 10-2/3 inches on buttresses at 90 degrees, 210 degrees, and 330 degrees.

Phase 12
Ten hoop tendons above elevation 949 feet + 10-2/3 inches on buttresses at 30 degrees, 150 degrees, and 270 degrees.

Force and Stress Measurements

Force and stress measurements are made by measurement of elongation of the prestressing steel after taking up initial slack and comparing it with the force indicated by the jack-dynamometer or pressure gauge. Force jack pressure gauge or dynamometer combinations are calibrated against known precise standards before application of prestressing force. All gauges are calibrated on a dead weight calibration apparatus. The presence of two gauges, one gauge on the pump and one gauge on the jack, provides a means to maintain a constant check of the calibration of the gauges. Based on the actual calibration tests of the stressing equipment, it was concluded that the pump efficiency does not influence the equipment accuracy and that the stressing accuracy depends only on the ram efficiency. Therefore, any combination of ram, gauge, and pump may be used interchangeably. During stressing, records are made of elongations as well as pressures obtained. Jack dynamometer or gauge combinations are checked against elongation of the tendon and any discrepancy exceeding plus or minus 5 percent will be evaluated by Design Engineering. The measured elongation will differ from the calculated elongation because of the following:

1. The statistical modulus of elasticity of 29.3 million psi for straight, untwisted wire.
2. The actual length and location of the tendon sheath will vary from the theoretical position due to approved placing tolerances.
3. All wires in a tendon are equal in length and the tendon is twisted to compensate for the difference in actual arc lengths. The twisting forms a wire cable configuration which does not follow the sheath centerline and which has a modified modulus of elasticity value.
4. The friction factor used in calculations is an average value based on experience. The true influence of friction on each tendon can be significantly different from the average value used in calculations.
5. The permissible tolerance in pressure gauge accuracy combined with the possible variables in stressing techniques such as reading the gauges and scales can constitute a significant difference.

Calibration of the pressure gauges are maintained accurate within the following limits:

The amount of nitrate found in the 90,000 gallons of Nuclear Grade material made for Palisades, Point Beach, and Turkey Point plants, so far, was "0" and practically, in order to keep the trace amounts allowed, be it 2 or 4, the amounts must be kept at zero. However, the refinery requires the use of 4 parts per million figure as a maximum.

Infra-red spectographic analysis shows Visconorust 2090P and NO-OX-ID CM to be quite similar with approximately the same amounts of wetting agents and rust preventives in the petroleum carriers.

PERFORMANCE DATA

| Item | NO-OX-ID | Visconorust 2090P | ASTM Method |
|---|---|---|---|
| Weight Per Gal. | 7.2 - 7.5 lbs. | 7.3 - 7.6 lbs. | -- |
| Pour Point | 110° - 120°F | -- | D-97 |
| Flash Point (coc) | 400°F | 385°F | D-92 |
| Viscosity @ 150°F | 125 - 150 SSU | 116 SSU | D-88 |
| Viscosity @ 210°F | 55 - 75 SSU | 59 SSU | D-88 |
| Spec. Gray @ 60°F | 0.88 - 0.90 | 0.88 - 0.91 | D-287 |
| Pene. (cone) @ 77°F | 325 - 370 | 370 | D-937 |
| Water Sol Chlorides | 1 PPM | 1 PPM | D-512 |
| Water Sol Nitrates | 2 PPM | 4 PPM | D-1255 |
| Water Sol Sulfides | 1 PPM | 1 PPM | D-992 |
| Phenoloc Bodies (As Phenol) | 1 PPM | 1 PPM | -- |
| Shrinkage Factor (150°F to 70°F) | 3.5 - 4.5% | 3.5 - 4.5% | -- |

*3.8.1.6.2.7 Tensioning Schedule*

Prestressing begins after the concrete in the walls and the dome has reached the specified f'c. The dome and the hoop tendons are tensioned from both ends, and the vertical tendons are tensioned from either the top end or from both ends. Six jacks are used throughout the post-tensioning operations.

Phase 1
Twelve hoop tendons above elevation 943 feet + 6 inches on buttresses at 90 degrees, 210 degrees, and 330 degrees.

Phase 2
Thirty-six dome tendons in the periphery of the dome.

Phase 3
Twelve hoop tendons above elevation 943 feet + 6 inches on buttresses at 30 degrees, 150 degrees, and 270 degrees.

Phase 4
Remaining 126 dome tendons.

Phase 5
One hundred and forty-one hoop tendons from elevation 865 feet + 0 inches to elevation 943 feet + 6 inches on buttresses at 30 degrees, 150 degrees, and 270 degrees.

Phase 6
Close the construction opening if not closed prior to Phase 6.

The grease is sampled and laboratory tested for chemical analysis to establish conformance with specifications and for deleterious substances such as water soluble chlorides, nitrates, and sulfides.

Visconorust 2090P Casing Filler is a petroleum base corrosion preventive designed for bulk application and extended protection.

It has:

1. A three phase protective system starting with a polar agent preferentially wetting the wires and displacing any moisture, rust preventive additives molecularly attached to the wetting agent and a petroleum barrier completing the resistant coating.
2. The property to emulsify any moisture picked up in the system while being pumped through the casing and either carrying it out the other end or nullifying its rusting ability if the moisture is trapped in the casing.
3. Reserve Alkalinity - Base Number of 3. The basic formulation of Visconorust 2090P is very stable and resistant to exterior moisture encroachment as well as mild acids and alkali. However, because of the probability of picking up moisture as the rust preventive is pumped through the tendons, an additional safety factor, besides the barrier action, is available to neutralize any acids that might form between the interface of the moisture and rust preventive.

   As a comparison for a more definitive value of Base Number 3 (equivalent neutralization value of 3 mg of Potassium Hydroxide per gram of product) Crankcase motor oils which undergo constant formation of acids require only a Base Number of 6. Hence, the Base Number of 3 will provide enough additional protection, since the tendons are subject only to a static environmental condition.

   Tests have been run using volatile acids, such as Hydro Bromic Acid, in an attempt to penetrate the Visconorust 2090P film and cause corrosion, without success thus far.
4. Only a trace amount of water soluble chlorides, sulfides, or nitrates.
5. A plugging agent designed to supplement the natural tendency of the microwax crystals and amorphus solid components to form a filter cake bridging any hair line cracks in the concrete, with which the casing filler might come in contact.
6. Self-healing qualities at the ambient temperature expected during operation, to take care of any voids created by wire movement.
7. Thixotropic properties that provide pumpability below 50°F.
8. Radiation Resistance:

   Visconorust 2090P has been subjected to $1 \times 10^6$ rads by the Gamma Process Company of New York. Results show that the Gamma rays did not have any material effect on either the physical or chemical structure (as noted by a negligible change in base number).

   Corroboration of the test results is readily noted in extensive literature on this subject, a few of which are listed below:

   Bibliography:

   a. The Lubrication of Nuclear Power Plants by R. S. Barnett - NLGI - October 1960.
   b. How Radiation Affects Petroleum Lubricants - Power, Vol. 100 December 1956, Page 164.
   c. Conventional Lubricants Are Sufficiently Radiation Resistant for Most Nuclear Power Reactor Applications by E. D. Reeves SAE Journal Vol. 66, May, 1956, Page 56-57.
   d. Organic Lubricants and Polymers for Nuclear Power Plants by Bolt and Carroll.

Bearing Plates

Bearing plates are capable of developing the ultimate strength of the tendon and distributing the bearing load over the bearing surface of the concrete. Bearing plates conform to the following requirements:

1. The transfer unit compressive stress on the concrete directly underneath the plate or assembly is in conformance with the ACI Code 318-63, latest edition.
2. Bending stresses in the plates induced by the pull of the prestressing steel shall not exceed 22,000 psi for structural steel and 15,000 psi for cast steel, except as experimental data may indicate that higher stresses are satisfactory.
3. Materials shall meet requirements of ASTM A36 for structural shapes or ASTM A148, Grade 80-40 for cast steel, or higher quality materials approved by Duke to meet strain requirements.
4. Design, fabrication, and erection shall meet the requirements of the latest AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings."

*3.8.1.6.2.5 Sheaths*

Materials

Sheaths for post-tensioning tendons are ungalvanized corrugated articulated tubing and meet the following requirements:

1. The internal diameter is adequate to allow insertion of prestressing steel after concrete placement.
2. The sheaths will withstand the placing of concrete at a pour rate of two feet per hour (with mechanical vibration) without ovalling or changing alignment.
3. Sheaths are protected from corrosion during storage.

Sheath Fabrication

The sheaths are cut to length and bent to shape. The bending is accomplished without wrinkling the metal. Dented or wrinkled sheaths are replaced. Finished bent or straight dimensions are in accordance with approved drawings.

Installation (by Duke)

Sheaths are accurately installed in the forms at the location shown on the drawings to a tolerance of ± one-half (½) inch, except as otherwise indicated on the drawings. The sheaths are supported in such a manner as to prevent displacement during concrete placement. The sheath is supported at the ends and at such intervals as are in accordance with the drawings. Damaged or improperly bent sheaths are rejected.

Cleaning and Venting

Just prior to insertion of the tendon, the sheath is cleaned by the use of compressed air or other suitable means.

*3.8.1.6.2.6 Corrosion Protective Grease*

Corrosion protection is provided by grease injected into the sheaths under pressure. Grease will be Visconorust 2090P manufactured by Viscosity Oil Company.

The BBR Bureau Standard for button head splits is a maximum number of two splits with a width of 0.06 inch. The Prescon Corporation has run tests on button heads with splits; and based on an evaluation of the test results, the BBR Bureau Standard is acceptable.

Protection

Prestressing steel is protected from mechanical damage and corrosion during shipment, storage, installation, and tensioning. A thin film of No-Ox-Id (R) 500, as manufactured by Dearborn Chemical Company or Visconorust 1601, manufactured by Viscosity Oil Company, is applied to the prestressing steel after fabrication in accordance with the manufacturer's instructions. The steel is then wrapped before shipment to the site. The steel is not handled, shipped, or stored in a manner that will cause a permanent set or notch, change its material properties, or expose it to inclement weather or injurious agents such as chloride containing solutions. Damaged or corroded tendons are rejected.

Installation

The tendon installation prestressing procedure was carried out as follows:

1. To assure a clear passage for the tendons, a "sheathing Rabbit" was run through the sheathing both prior to and following placement of the concrete.
2. Tendons were uncoiled and pulled through the sheathing unfinished end first.
3. The unfinished end of the tendons was pulled out with enough length exposed so that field attachment of the anchor head and buttonheading could be performed. To allow this operation, trumplates on the opposite end had an enlarged diameter to permit pulling the shop finished ends with their anchor heads.
4. The anchor heads were attached and the tendon wires buttonheaded.
5. The shop finished end of the tendon was pulled back and the stressing jack attached.
6. The post-tensioning was done by jacking to the permissible overstressing force to compensate for friction and placing the shims precut to lengths corresponding to the calculated elongation. Proper tendon stress was achieved by comparing both jack pressure and tendon elongation against previously calculated values. The vertical tendons were prestressed from either one or both ends, while the horizontal and dome tendons were prestressed from both ends.
7. The grease caps were bolted onto anchorages at both ends and made ready for pumping the tendon sheathing filler material.
8. The tendon sheaths and grease caps were filled with sheathing filler and sealed. The sheathing filler material had limitations specified for deleterious water soluble salts.

   Corrosion protection of the tendons and interior surface of sheathing was applied prior to shipment.

   Tendon sheaths mark 24H34, 13H34 and 34V14 on Oconee 1 and 13H21 on Oconee 2 were plugged. The location of the plugged sheaths are shown in Figure 3-36.

*3.8.1.6.2.4 Anchorages and Bearing Plates*

Anchorages

Anchorages will develop the minimum guaranteed ultimate strength of the tendon and the minimum elongation of the tendon material as required by the applicable ASTM specification.

provided with a system of vertical and horizontal (hoop) tendons. Hoop tendons are placed in a 120 degree system in which three tendons form a complete ring. Six buttresses are used as anchorages.

*3.8.1.6.2.2 Detail Shop Drawings*

Subcontractor

Upon award of the contract, Duke furnished engineering design drawings which were issued for construction of the prestressing work providing information required for the preparation of shop detail drawings by the subcontractor. The subcontractor furnished the following detail drawings and erection drawings to Duke:

1. Outside dimensions of sheathing proposed for the tendon.
2. Complete details of the post-tensioned wall and dome including dimensional locations of the tendons and necessary equipment and materials to place the tendons.
3. Tendon characteristics indicating the $A_s$, $F'_s$, $f_{sy}$, and a typical stress-strain curve for the tendon used, as well as tendon force capability.
4. Details of anchorages, bearing plates, and other accessories pertinent to the post-tensioning system.
5. Erection drawings showing clearly the marking and positioning of tendons, anchorages, and sheaths, and details showing alignment and setting tolerances required.
6. Stressing sequence drawings.

*3.8.1.6.2.3 Prestressing Steel*

Materials and Fabrication

High strength steel wires are in accordance with ASTM A416 or A421 as a minimum requirement.

Wires are to be straightened if necessary to produce equal stress in all wires or wire groups that are to be stressed simultaneously or when necessary to insure proper positioning in sheaths. However, wires showing a permanent set are not to be straightened or installed if the bend exceeds 60 degrees and the radius is less than 1.25 inches.

Tests were made on wire bent to 30, 60, and 90 degrees with a bend radius of 1.25 inches (5 times wire diameter) and wire bent to 30 and 60 degrees with a zero radius. The test specimens were from two different heats of ¼ inch diameter wire. All specimens within one test series were from the same heat and coil. In the sequence of cutting, every sixth specimen fell into the same group. The first group consisted of straight specimens for comparison.

Specimens were cut to a length of 15-½ inches, bent to the prescribed angle and radius in a bend-tester, and straightened. The specimens were button headed on each end and tensile tested to failure. The test results presented in Table 3-22 show that the strength of prestressing wire is not affected by bending the wire 60 degrees around a 1.25 inch radius pin.

The button head is cold formed to a nominal diameter of 3/8 inch symmetrically about the axis of the wires. If splitting is consistent and appears in all heads or if there are more than two splits in which the opening exceeds 0.06 inch per head, the wire is rejected. No forming process is used that caused indentation in the wire. Wires showing indentations are rejected. Wires showing fabrication defects, wires having welds or joints made during manufacture, or broken wires are removed and replaced.

Vertical joints are also blasted with compressed air, cleaned, and wetted before placing concrete.

Vertical joints are placed at the center of each buttress to take advantage of the 50 percent additional horizontal prestress due to the overlapping of the anchored hoop tendons.

Horizontal joints between buttresses are at the same elevation. These joints are prepared as stated above to provide maximum possible bond. Principal tension in the membrane is limited to $3\sqrt{f'_c}$.

#### 3.8.1.6.2 Prestressing

These instructions and methods describe the quality control standards and measures applied in the control, manufacture, and field installation of the prestressing phase of construction of the Reactor Building.

The BBRV post-tensioning system furnished by The Prescon Corporation was used. Each tendon consists of ninety ¼ inch diameter wires conforming with ASTM A-421-65T, two anchor heads and two sets of shims conforming with American Iron and Steel Institute (AISI) C-1045 HR. The tendon sheathing system consists of spirally wound carbon steel tubing connecting to a trumplate (bearing plate and trumpet) at each end. The bearing plates were fabricated from steel plate conforming with AISI C-1045 HR and the trumpets from AISI C-1010 HREW material.

The C-1045 HR material used for the stressing washers, dead-end washers, shims, and bearing plates was modified by the addition of silicon to obtain a finer grain structure and cleaner steel than unmodified C-1045. The average depth of the heat affected zone resulting from flame cutting is approximately 1/16 inch and the improved general ductility of modified C-1045 material should increase resistance to cracks starting in heat affected zones and decrease the probability of crack propagation. However, a cracked plate could continue to perform its function without loss of structural integrity and should be evaluated in terms of actual functional ability.

Flame cutting is limited to sizing the bearing plate and making the center hole. All other holes in the bearing plate are drilled. The dead-end washer is flame cut to size and drilled for the tendon wires. No flame cutting is performed on the stressing washer.

##### *3.8.1.6.2.1 Control*

Supervision

The subcontractor furnishes competent, experienced supervision of the tendon installation and tensioning operation until completion of post-tensioning. The above individual exercises a close check and rigid control of all post-tensioning operations, as necessary, for full compliance with specifications.

Inspection of Duke's Work

The subcontractor is responsible for the inspection of Duke's handling and installation of tendon sheaths and bearing plates. To this end, he provides a competent technical representative to check the installation of these items by Duke. If any of Duke's work or actions jeopardize the subcontractor's work, he notifies Duke's Resident Engineer in writing. Failure to do this constitutes acceptance of Duke's work as it affects subcontractor's responsibilities.

Arrangement of Prestressing Tendons

The configuration of the tendons in the dome is based on a three-way tendon system consisting of three groups of tendons oriented at 120 degrees with respect to each other. The vertical cylinder wall is

Test cylinders are cast from the mix proportions selected for construction and the following concrete properties determined:

Uniaxial creep
Modulus of elasticity and Poisson's Ratio
Autogenous shrinkage
Thermal diffusivity
Thermal coefficient of expansion
Compressive strength

*3.8.1.6.1.6 Aggregates*

Aggregate testing is performed as follows:

1. Sand sample for gradation (ASTM C33 Fine Aggregate)
2. Organic test on sand (ASTM C40)
3. 3/4″ sample for gradation (ASTM C33, Size No. 67)
4. 1-½ inch sample for gradation (ASTM C33, Size No. 4)
5. Check for proportion of flat and elongated particles.

*3.8.1.6.1.7 Concrete Construction*

Cast-in-place concrete was used to construct the Reactor Building shell. The base slab construction was performed in seven pours utilizing large block pours. After the completion of the base slab steel liner erection and testing, an additional concrete slab was placed to provide protection for the floor liner.

The concrete placement in the walls was done in 10 ft high lifts with vertical joints at the radial center line of each of six buttresses. Cantilevered jump forms on the exterior face and interior steel wall liner served as the forms for the wall concrete.

The dome liner plate, temporarily supported by 18 radial steel trusses and purlins, served as an inner form for the initial 8 inch thick pour in the dome. The weight of the subsequent pour was supported in turn by the initial 8 inch pour. The trusses were lowered away from the liner plate after the initial 8 inches of concrete had reached design strength, but prior to the placing of the balance of the dome concrete.

The standards or specifications on quality control and tests of concrete during construction are equal to or better than requirements of ACI 301. Some of the areas where quality control exceeds the requirements of ACI 301 are as follows:

1. Requirements for water quality.
2. Placing temperature of concrete.
3. Requirements for aggregate acceptability.
4. Requirements for test cylinders.

Horizontal construction joints are prepared for receiving the next lift by blasting with compressed air. Surface set retardant compounds are not used.

Horizontal surfaces are wetted and covered with a coating of mortar of the same cement-sand ratio as used in the concrete immediately before the concrete is placed.

*3.8.1.6.1.2 Water*

Water is potable and does not contain impurities in amounts that will cause a change of more than 25 percent in setting time for the Portland Cement, nor a reduction in the compressive strength of mortar of more than 5 percent as compared with results obtained using distilled water.

*3.8.1.6.1.3 Admixtures*

Admixtures, as to be determined by detailed mix design, conform to applicable ASTM Specification covering such materials and their testing.

*3.8.1.6.1.4 Concrete Test Cylinders*

Concrete cylinders for compression testing are made and stripped within 24 hours after casting, and marked and stored in the curing room. These cylinders are made in accordance with ASTM C21, "Tentative Method of Making and Curing Concrete Compression and Flexure Test Specimens in the Field."

Slump, air content, and temperature are taken when cylinders are cast and for each 35 yards of concrete placed. Slump tests are performed in accordance with ASTM C143, "Standard Method of Test for Slump of Portland Cement Concrete." Air tests are performed in accordance with ASTM C231, "Standard Method of Test of Air Content of Freshly Mixed Concrete by the Pressure Method." Compressive strength tests are made in accordance with ASTM C39, "Method of Test for Compressive Strength of Molded Concrete Cylinders."

Six standard test cylinders are obtained and molded for concrete placed in excess of 10 cubic yards in any one day, with 6 additional cylinders for each successive 100 cubic yards placed. Two cylinders are tested at the age of 7, 28, and 90 days.

Concrete mixes are designed in accordance with "Recommended Practice for Selecting Proportions for Concrete" (ACI 613), using materials qualified and accepted for the work; and the strength, workability, and other characteristics of the mixes are ascertained before placement. Duke Power's concrete control laboratory is set up on the Oconee site. A batch-plant inspector is provided, and testing as shown below is performed. Field control is in accordance with the "Manual of Concrete Inspection" as reported by ACI Committee 611.

*3.8.1.6.1.5 Mix Design*

Only those mixes meeting the design requirements specified for Reactor Building concrete are used. Trial mixes are tested in accordance with the applicable ASTM Codes as follows:

| Test | ASTM |
|---|---|
| Air Content | C231 |
| Slump | C143 |
| Bleeding | C232 |
| Making and Curing Cylinders in Laboratory | C192 |
| Compressive Strength Tests | C39 |

Six cylinders are cast from each design mix for two tests on each of the following days: 7, 28, and 90.

#### 3.8.1.6.1 Concrete

An experienced full-time concrete inspector continuously checked concrete batching and placing operations.

Concrete mixes were designed and the associated tests run by the concrete testing laboratory at Clemson University in accordance with ACI 613. During construction, the field inspection personnel made minor modifications that were necessitated by variations in aggregate gradation or moisture content.

In determining the design mixes; air content, slump, and bleeding tests were run in accordance with the appropriate ASTM Specifications.

The concrete ingredients consist of Type II Cement (ASTM C-150), Solar 25 air entraining agent (ASTM C-260), Plastiment water reducing agent (ASTM C-494), Aggregate (ASTM C-33), and water that was free from injurious amounts of chlorides, sulphates, oil, acid, alkali, organic matter, or other deleterious substances.

Fine aggregate consists of clean, sharp, washed sand of uniform gradation from Becker County Hagood Quarry. Coarse aggregate consists of washed crushed rock having hard, strong, durable pieces of Gaffney marble from Campbell Limestone Company. The acceptability of the aggregate was based on Los Angeles Abrasion, Clay Lumps Natural Aggregates, Material Finer Number 200 Sieve, Organic impurities effect on Mortar, Organic impurities - Sands, Potential Reactivity, Seive Analysis, Soundness, Specific Gravity and Absorption, and Petrographic tests based on the appropriate ASTM Specifications.

Acceptability of aggregates is based on the following ASTM tests. These are performed by a qualified testing laboratory.

| Test | ASTM |
|---|---|
| L. A. Rattler | C131 |
| Clay Lumps Natural Aggregate | C142 |
| Material Finer No. 200 Sieve | C117 |
| Mortar making properties | C87 |
| Organic impurities | C40 |
| Potential Reactivity (chemical) | C289 |
| Potential Reactivity (mortar bar) | C227 |
| Sieve Analysis | C136 |
| Soundness | C88 |
| Specific Gravity and Absorption | C127 |
| Specific Gravity and Absorption | C128 |

##### *3.8.1.6.1.1 Cement*

Cement conforms to ASTM C150 and tested to ASTM C114.

The manufacturer submits certified copies of mill test reports showing the chemical composition and certifying that the cement complies with the specification on each shipment delivered to the site. In addition to the manufacturer's tests, cement is sampled periodically at the site and tested to ascertain conformance with ASTM Specification C150.

For typical details of piping penetrations, see Figure 3-20.

2. Electrical Penetrations

Medium voltage penetrations for reactor coolant pump power shown on Figure 3-20 are canister type using glass sealed bushings for conductor seals. The canisters are filled to a positive pressure with an inert gas. The assemblies are bolted to mating flanges which incorporate double "0" ring seals with a test port between as a means of verifying seal integrity.

Low voltage power, control and instrumentation assemblies are shown on Figure 3-20. These assemblies are designed to bolt to mating flanges mounted inside the Reactor Building. Each assembly includes two header plates to which are welded glass to metal sealed conductors. The space between the seal headers is piped to a pressure gauge and a charging valve located outside of the Reactor Building. This test volume is pressurized with an inert gas. Dual "0" rings with a test port between are used to complete the seal to the mating flange, which is welded to the penetration nozzle.

#### 3.8.1.5.5 Miscellaneous Considerations

In various cases, it has been the designer's decision to provide structural adequacy beyond that required by the design criteria. Those cases are as follows:

1. Section 3.8.1.3.4, "Service Loads" on page 3-78 requires a minimum of 0.15 percent bonded reinforcing steel in two perpendicular directions on the exterior faces of the wall and dome for proper crack control. Due to the weather exposure, a minimum of approximately 0.5 percent was provided.
2. Section 3.8.1.3.4, "Service Loads" on page 3-78 requires a minimum of 0.15 percent bonded steel reinforcing (as stated above) for any location. At the base of the cylinder, the controlling design case requires 0.25 percent vertical reinforcing. As a result of pursuing the recommendation of the AEC Staff to further investigate current research on shear in concrete, several steps were taken:

   a. The work of Dr. Alan H. Mattock was reviewed and he was retained as a consultant on the implementation of the current research being conducted under his direction. The criteria has been updated in accordance with his recommendations.

   b. Concurrently with reviewing Dr. Mattock's work, the firm of T. Y. Lin, Kulka, Yang and Associates was consulted to review the detailed design of the cylinder to slab connection. It was their recommendation to use approximately 0.5 percent reinforcing rather than the 0.25 percent reinforcing indicated by the detailed design analysis for the vertical wall dowels. This increase would assure that there was sufficient flexural steel to place the section within the lower limits of Mattock's test data (approximately 0.3 percent) to prevent flexural cracking from adversely affecting the shear capability of the section.

   Additional information concerning structural acceptance criteria for liner plate, penetrations, supports, and buttresses can be found in Section 3.8.1.4.2, "Nonaxisymmetric Analysis" on page 3-89.

### 3.8.1.6 Materials, Quality Control, and Special Construction Techniques

Test, code, and cleanliness requirements accompanied each specification or purchase order for materials and equipment. Hydrostatic, leak, metallurgical, electrical, and other tests to be performed by the supplying manufacturers are enumerated in the specifications together with the requirements, if any, for test witnessing by an inspector. Fabrication and cleanliness standards, including final cleaning and sealing, are described together with shipping procedures. Standards and tests are specified in accordance with applicable regulations, recognized technical society codes and current industrial practices. Inspection is performed in the shops of vendors and subcontractors as necessary to verify compliance with specifications.

1. The surface reinforcements either have a very large radius such as hoop bars concentric with the penetration or are practically straight, having only standard hooks as anchorages where necessary.
2. The tendons are bent around penetrations at a minimum radius of approximately 20 feet. Maximum tendon force at initial prestress is 850 kips, which results in a bearing stress of about 880 psi on the concrete.

It is also important to note that the deflected tendons are continuous past the openings and are isolated from the local effects of stress concentrations by virtue of being unbonded.

In accordance with ASME Section II, piping penetration reinforcing plates and the weldment of the pipe closure to it are stress relieved. This code requirement and the grouping of penetrations into large shop assemblies permit a minimum of field welding at penetrations.

The personnel hatch consists of a steel cylinder with 3 ft-6 in. x 6 ft-8 in. doors at each end interlocked so that only one door can be open at any time. The hatch is designed to withstand all Reactor Building design conditions with either or both doors closed and locked. Doors open toward the center of the Reactor Building and are thus sealed under Reactor Building pressure. Design live load on the hatch floor is 200 psf.

Operation of the hatch is normally manual, that is, without power assist. Interlocks will prevent opening both doors at once.

Double gaskets are provided on the outer door to permit periodic pressurizing of the space between the gaskets from outside the Reactor Building. The hatch barrel may be pressurized to demonstrate its leak tightness without pressurizing the Reactor Building. Auxiliary restraint beams are attached to the inner door in this case to help the locking bars to resist internal lock pressure, which is greatly in excess of the Reactor Building design external pressure of 3 psig. The personnel hatch was pneumatically shop tested for pressure and leakage.

Figure 3-21 shows the principal features of the personnel hatch.

An emergency hatch is provided with 30 inch diameter doors. Its features are identical to the personnel hatch.

A 19-foot diameter equipment hatch opening to the outside provides the movement of large items into and out of the Reactor Building. The door is secured by bolts on the inside of the Reactor Building wall and can be opened only from inside the Reactor Building. It is opened only when the reactor is subcritical. Double gaskets on the door permit the seals to be pressurized from outside the Reactor Building to check the integrity of the seals. During operation, the space between the double gaskets is vented to the penetration room.

Figure 3-21 shows the principal features of the equipment hatch.

1. <u>Piping and Ventilation Penetrations</u>

   All piping and ventilation penetrations are of the rigid welded type and are solidly anchored to the Reactor Building wall or foundation slab, thus precluding any requirements for expansion bellows. All penetrations and anchorages are designed for the forces and moments resulting from operating conditions. External guides and stops are provided as required to limit motions, bending and torsional moments to prevent rupture of the penetrations and the adjacent liner plate for postulated pipe rupture. Piping and ventilation penetrations have no provision for individual testing since they are of all-welded construction.

In accordance with ASME Code, Paragraph 412 (m) (2), the liner plate is restrained against significant distortion by continuous angle anchors and never exceeds the temperature limitation of 700°F and also satisfies the criteria for limiting strains on the basis of fatigue consideration.

Paragraph 412 (n), Figure N-415 (A) of the ASME Code has been developed as a result of research, industry experience, and the proven performance of code vessels, and it is a part of a recognized design code. Figure N-415 (A) and its appropriate limitations have been used as a basis for establishing allowable liner plate strains. Since the graph in Figure N-415 (A) does not extend below ten cycles, ten cycles are being used for a loss-of-coolant accident instead of one cycle.

The maximum compressive strains are caused by accident pressure, thermal loading prestress, shrinkage and creep. The maximum strains do not exceed 0.0025 inch/inch and the liner plate always remains in a stable condition.

At all penetrations the liner plate is thickened to reduce stress concentrations in accordance with the ASME Boiler and Pressure Vessel Code 1965, Section III, Nuclear Vessels.

The liner plate is anchored as shown in Figure 3-19 with anchorage in both the longitudinal and hoop direction. The anchor spacing and welds were designed to preclude failure of an individual anchor. The load deformation tests referred to in Section 3.8.1.4.2, "Nonaxisymmetric Analysis" on page 3-89 indicate that the alternate stitch fillet weld used to secure the anchor to the liner plate would first fail in the weld and not jeopardize the liner plate leak tight integrity.

Offsets at liner plate seams are controlled in accordance with ASME Section III Code, which allows 1/16 inch misalignment for ¼ inch plate. The flexural strains due to the moment resulting from the misalignment were added to calculate the total strain in the liner plate.

The liner plate plus structural shapes to support the liner are ASTM A36 or ASTM A516 steel. The selection of this material complies with "Safety Standard for Design, Fabrication and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors" prepared by Subcommittee N6.2, Containment, of ASA Sectional Committee N6, Reactor Safety Standards.

#### 3.8.1.5.4 Penetrations

Penetrations conform to the applicable sections of ASA N6.2-1965, "Safety Standard for the Design, Fabrication and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors." All personnel locks and any portion of the equipment access door extending beyond the concrete shell conform in all respects to the requirements of ASME Section III, Nuclear Vessels Code.

The basis for limiting strains in the penetration steel is the ASME Boiler and Pressure Vessel Code for Nuclear Vessels, Section III, Article 4, 1965, and therefore, the penetration structural and leak tightness integrity are maintained. Local heating of the concrete immediately around the penetration will develop compressive stress in the concrete adjacent to the penetration and a negligible amount of tensile stress over a large area. The mild steel reinforcing added around penetrations distributes local compressive stresses for overall structural integrity.

Horizontal and vertical bonded reinforcement is provided to help resist membrane and flexural loads at the penetrations. This reinforcement was located on both the inside and outside face of the concrete. Stirrups were also used to assist in resisting shear loads.

Local crushing of the concrete due to deflection of the reinforcing or tendons is precluded by the following details:

the anchorage force for the tendons was kept at or below 0.7 $f'_s$ in accordance with the interpretation described.

#### 3.8.1.5.3 Liner Plate

The design criteria which are applied to the Reactor Building liner to assure that the specified leak rate is not exceeded under accident conditions are as follows:

1. That the liner be protected against damage by missiles (see Section 3.5.1.2, "Turbine Missiles" on page 3-53).
2. That the liner plate strains be limited to allowable values that have been shown to result in leak tight vessels or pressure piping.
3. That the liner plate be prevented from developing significant distortion.
4. That all discontinuities and openings be well anchored to accommodate the forces exerted by the restrained liner plate, and that careful attention be paid to details of corners and connections to minimize the effects of discontinuities.

The most appropriate basis for establishing allowable liner plate strains is considered to be the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Article 4. Specifically, the following sections have been adopted as guides in establishing allowable strain limits:

| | |
|---|---|
| Paragraph N-412 (m) | Thermal Stress (2) |
| Paragraph N-414.5 | Peak Stress Intensity<br>Table N-413<br>Figure N-414, N-415 (A) |
| Paragraph N-412 (n) | |
| Paragraph N-415.1 | |

Implementation of the ASME design criteria requires that the liner material be prevented from experiencing significant distortion due to thermal load and that the stresses be considered from a fatigue standpoint (Paragraph N-412 (m) (2)).

The following fatigue loads are considered in the design on the liner plate:

1. Thermal cycling due to annual outdoor temperature variations. The number of cycles for this loading is 40 cycles for the plant life of 40 years.
2. Thermal cycling due to Reactor Building interior temperature varying during the startup and shutdown of the reactor system. The number of cycles for this loading is assumed to be 500 cycles.
3. Thermal cycling due to the loss-of-coolant accident will be assumed to be one cycle. Thermal load cycles in the piping systems are somewhat isolated from the liner plate penetrations by the concentric sleeves between the pipe and the liner plate. The attachment sleeve is designed in accordance with ASME Section III fatigue considerations. All penetrations are reviewed for a conservative number of cycles to be expected during the plant life.

The thermal stresses in the liner plate fall into the categories considered in Article 4, Section III, Nuclear Vessels of the ASME Boiler and Pressure Vessel Code. The allowable stresses in Figure N-415 (A) are for alternating stress intensity for carbon steel and temperatures not exceeding 700°F.

Paragraph 2606 (a) 2 of ACI 318-63 refers to "tendons" rather than to an individual tendon. Further, the paragraph does not refer to the location to be considered for the determination of $f_s$ in the manner, for example, of the "temporary jacking force" referred to in 2606 (a) 1.

Two interpretations were therefore required. Both interpretations had to consider the effect of the resultant actions on both the prestressing system and structure.

The first interpretation was that the location for measurement of the seating force, used in calculating $f'_s$ was at the end anchor and just subsequent to the measurement of the "temporary jacking force" referred to in ACI 318-63 2606 (a) 1. The advantages of this location are several. One is that it is a practical one and thus the possibility for achieving valid measurements is greater. The second is that it is the same location used for measuring the "temporary jacking force" and measurements could be made without the added complexity of additional measuring devices. The third advantage is that measurements at this location provide assurance that the calculated $f'_s$ does not anywhere exceed the maximum $f'_s$ to which that tendon has been subjected.

Several possible cases were considered for the second interpretation so as to allow anchoring of an individual tendon without exceeding the requirement stated for "tendons" collectively in ACI 318-63 2606 (a) 2. One such case assumed that the anchoring force for the typical tendon was that for a tendon anchored midway through the prestressing sequence. It further assumed that the losses to be assumed were one-half of the sum of elastic losses, and of the creep, shrinkage, and relaxation predicted to occur during the entire prestressing sequence. This interpretation, however, was not considered to be practical nor enforceable since it resulted in changing the seating forces as the actual (as compared to the schedule) time length of the prestressing period was dictated by weather and manpower availability.

Another case considered was that of anchoring each tendon at a measured force of 850 kips ($0.8f'_s$). Although there was no apparent detrimental effect to the prestressing system or structure, insertion of shims would be almost impossible. Further, it was concluded that this case would not establish compliance with ACI 318-63.

The case adopted was to seat each tendon with a measured "pressure" reading for the jack, at "lift-off" of the end anchor, of 775 kips (between 0.72 and 0.73 $f'_s$). This procedure has several advantages.

One advantage was that the force on the containment and the tendon was within the bounds of those for which it had been tested and resulted in no known detrimental effects. The second advantage was that the stressing procedure was simplified since the stressing crews did not have to accommodate a large number of different anchoring force requirements. The third advantage was that, at the completion of stressing the last tendon, the expected losses were such that the average $f'_s$ at the end anchors of the tendons would be less than 0.7 $f'_s$, thus establishing compliance with ACI 318-63 2606 (a) 1 and 2. The fourth advantage was that the percentage loss of prestressing force was less than would be the case if the tendons were anchored in such a manner the calculated value of $f'_s$ nowhere exceeded 0.7 $f'_s$.

The latter advantage deserves special mention since it plays a strong role in assuring that the Final Effective Prestress equalled or exceeded the desired value. For example, if the $f'_s$ at anchorage of the tendons were 0.1 $f'_s$, creep and shrinkage of concrete could result in the loss of almost all of the prestressing force. Assuming that the total losses due to creep, shrinkage, and elastic shortening equals 0.1 $f'_s$, then the Final Effective Prestress would be 20 percent percent of an initial prestress equivalent to 0.5 $f'_s$. If the initial prestress were equivalent to 0.7 $f'_s$, the Final Effective Prestress, neglecting relaxation for the moment, would be about 86 percent of the initial prestress. Clearly, the assurance (that the concrete creep and shrinkage losses have been properly accounted for) increases as the $f'_s$ for the anchored tendons and tendon increases. However, this design was committed to meeting the ACI 318-63 requirement and

Assuming that the jacking stress for tendons is 0.80 $f'_s$ of 192,000 psi and using the above prestress loss parameters, the following tabulation shows the magnitude of the design losses and the final effective prestress at end of 40 years for a typical dome, hoop and vertical tendon.

| | Dome (Ksi) | Hoop (Ksi) | Vertical (Ksi) |
|---|---|---|---|
| Jacking Stress | 192 | 192 | 192 |
| Friction Loss | 19 | 21.3(1) | 21 |
| Seating Loss | 0 | 0 | 0 |
| Elastic Loss | 14.5 | 14.3 | 7.2 |
| Creep Loss | 12.6 | 12.6 | 12.6 |
| Shrinkage Loss | 2.9 | 2.9 | 2.9 |
| Relaxation Loss | 12.5 | 12.5 | 12.5 |
| Final Effective Stress(2) | 130.5 | 128.4 | 135.8 |

**Note:**

(1) Average of crossing tendons.

(2) This force does not include the effect of pressurization which increases the prestress force.

To provide assurance of achievement of the desired level of Final Effective Prestress and that ACI 318-63 requirements are met, a written procedure was prepared for guidance of post-tensioning work. The procedures provided nominal values for end anchor forces in terms of pressure gauge readings for calibrated jack-gauge combinations. Force measurements were made at the end anchor, of course, since that is the only practical location for such measurements.

The procedure required the measured temporary jacking force, for a single tendon, to approach but not exceed 850 kips (0.8$f'_s$). Thus, the limits set by ACI 318-63 2606 (a) 1, and of the prestressing system supplier, were observed. Additionally, benefits were obtained by in place testing of the tendon to provide final assurance that the force capability exceeded that required by design. During the increase in force, measurements were required of elongation changes and force changes in order to allow documentation of compliance with ACI 318-63 2621 (a). The procedures required that the prestressing steel be installed in the sheath before stressing for a sufficient time period that the temperatures of the prestressing steel and concrete reach essential equilibrium, to establish conformance with ACI 318-63 2621 (e). The jacking force of 0.8$f'_s$ further provided for a means of equalizing the force in individual wires of a tendon to establish compliance with ACI 318-63 2621 (b). The procedures required compliance with ACI 318-63 such that, if broken wires resulted from the post-tensioning sequence, compliance with section 2621 (d) was documented. Each of the above procedures contributed to assurance that the desired level of Final Effective Prestress would be achieved.

The requirements of ACI 318-63 2606 (a) 2 state that $f_s$ should not exceed 0.7$f'_s$ for "post-tensioning" tendons immediately after anchoring.

Industry has been considering rewriting that requirement such that it has only one interpretation rather than the several now possible. Consideration is also being given to raising the value of 0.7$f'_s$ or eliminating the requirement entirely and, instead, retaining the 0.8$f'_s$ or some other limitation on temporary jacking force.

Foundation descriptions for Auxiliary and Turbine Buildings are given in Section 3.8.4.1, "Description of the Structure" on page 3-128 and Section 3.8.5.1, "Description of the Structures" on page 3-130, respectively.

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# 3.9 MECHANICAL SYSTEMS AND COMPONENTS

## 3.9.1 SPECIAL TOPICS FOR MECHANICAL COMPONENTS

### 3.9.1.1 Design Transients

All Reactor Coolant System components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. Design transient cycles are shown in Table 5-2.

### 3.9.1.2 Computer Programs Used in Analysis

Computer programs used to perform the code calculations on the casing for the reactor coolant pump are described in Section 5.4.1.2, "Reactor Coolant Pumps (Oconee 2 & 3)" on page 5-49.

Additional computer programs used in analysis are given in Section 3.7.3.1, "Seismic Analysis Methods" on page 3-65.

### 3.9.1.3 Experimental Stress Analysis

In August of 1964, B&W began design and construction of facilities to test full scale sections of the Once Through Steam Generator. Mechanical design testing includes vibration and structural tests. For details, see Section 5.2.3.4, "Steam Generators" on page 5-23.

### 3.9.1.4 Considerations for the Evaluation of the Faulted Condition

The analytical method used for the evaluation of faulted conditions is elastic analysis. Stress limits for the faulted conditions are established in Section 3.9.3.1, "Load Combinations, Design Transients and Stress Limits" on page 3-153.

Faulted operating conditions were not applied to any components that were not a part of the reactor coolant pressure boundary.

The design stress limits for components comparable to the ASME Code Class 2 and 3 did not allow inelastic deformation.

## 3.9.2 DYNAMIC TESTING AND ANALYSIS

### 3.9.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

It is Duke's normal practice and a startup procedure consideration to put essential and safety related systems through all of their normal and emergency modes of operation, visually observing the system for excessive movement and/or vibration. Based on operational reports indicating possible excessive movement and/or vibration, the Steam Production Department requests Design Engineering review of each case. Design Engineering observes the system making necessary measurements, readings, etc., as required to analyze the problem against existing design stress analysis and design criteria. Based on this analysis, Design Engineering either approves the system as satisfactory or requiring additional design consideration. Additional supports or suppressors are designed to accommodate the effects of valve closures, pump trips, safety valve operations, and operational vibrations as required. Any problems defined for any unit are reviewed and corrected for all three units as required.

Although not required for the Oconee project, Duke will conduct prior to station startup the following monitoring programs which are typical of Design Engineering reviews as discussed above for the purpose of comparing results with design analysis.

1. Thermal Movement Monitoring Program for the Reactor Coolant System Piping (Data will be taken on Oconee 1 only; however, the report will be qualified for all three units).
2. Thrust Movement Monitoring Program for the Main Steam Bypass to Condenser Piping (Data will be taken on Oconee 1 only; however, the report will be qualified for all three units).
3. Hanger and Restraint Setting Monitoring Program for the LP Injection System (Data will be taken on Oconee 1 only; however, the report will be qualified for all three units).

   Dynamic analysis is further described in Section 3.9.3.1, "Load Combinations, Design Transients and Stress Limits" on page 3-153.

### 3.9.2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment

When the response spectra at each elevation in the building has been determined, the G-loadings imposed on a component may then be determined. These loads are evaluated by the equipment supplier and in the case of complex components such as a heat exchanger, the design calculations performed by the supplier are reviewed by B&W Engineering or Duke, as applicable. The supplier has the freedom to use either of two alternate analytical methods to evaluate the equipment or he may choose to test it. Components may be tested by either shaker or impact tests and a certification of the test results are required. In a few cases, a manufacturer's certification that the equipment would withstand seismic conditions is acceptable based on tests of similar equipment, an example of this would be similar type pumps. Analytically the evaluation can be made by calculating the natural frequency of the component, entering the appropriate damping curve and determining the amplification factor from the response spectrum curve. The equipment is then evaluated using these G-loadings. As an alternate, the component may be evaluated without calculating the natural frequency by using the peak amplification factor from the appropriate damping curve to determine the equipment loads. This latter approach is conservative.

Special attention is given to foundation and nozzle loadings for equipment such as tanks, pumps, heat exchangers, demineralizers and filters. Loads imposed by connecting piping on a given component are included and in some cases, component nozzles have had to be reinforced to accommodate these loads. Components which are most likely to require special reinforcement due to seismic loads are long horizontal, saddle mounted tanks, vertical tanks mounted on legs, and stacked heat exchangers. These have all been evaluated and appropriately designed for the seismic conditions.

### 3.9.2.3 Pre-operational Flow-induced Vibration Testing of Reactor Internals

The test program developed to measure vibration of the reactor internals at Oconee 1 during hot functional testing is described in Topical Report *BAW-10038* (Reference 18 on page 3-175).

The main objective of the testing program is to obtain measurements of flow-induced vibration to confirm the structural adequacy of the internals. The components to be instrumented were selected on the basis of an evaluation of the distressed areas in the Oconee 1 reactor internals during previous hot functional testing and of the regions with high flow velocities. Another objective of the program is to confirm current analytical methods.

*BAW-10038* presents the documentation required by Safety Guide 20 to qualify the Oconee 1 internals as the prototype design for B&W's 177-fuel-assembly plant. Described therein are the instrumentation used

on the internals and reactor vessel, the data acquisition system, the test conditions, the on-line analysis of data, the predicted component responses, the test acceptance criteria, and the inspection program.

## 3.9.2.4 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions

This subdivision summarizes an analysis of the fuel assembly for loads caused by the depressurization transient following an instantaneous reactor coolant pipe rupture and/or seismic excitation. Four separate loading conditions are investigated: loads due to (1) the operating basis earthquake (OBE) (2) the safe shutdown earthquake (SSE), (3) a loss-of-coolant accident (LOCA), and (4) the simultaneous occurrence of a SSE and a LOCA timed so that the combined deflections are maximum. It is found that the LOCA loads are most severe for an outlet pipe rupture. Loads for an inlet pipe rupture are less severe because of the higher flow resistance of the smaller pipe and better equalization of pressures permitted by the internals vent valves.

The maximum loads or deflections occurring in the fuel assembly are discussed and tabulated. The deflections caused by these loads are tabulated for cases and locations where deformations have a potential safety implication. The analysis is based on a conservative application of loads and end conditions, etc., resulting in calculated internal loads (or deflections) that exceed the actual internal loads (or deflections) for the given applied load.

Investigation of the effects of the foregoing loadings identified two areas to be investigated:

1. Horizontal - contact between fuel assemblies due to motions in a horizontal plane, where the contact occurs primarily at the mid-span grid spacers.
2. Vertical - contact of fuel assemblies with the internals due to upward pressure, where the contact occurs between the end fittings and the grid plates.

The seismic accelerations used in these analyses are those specified for the Oconee site. The time-history method is used to evaluate the seismic effects. The calculation of LOCA forces is described in Topical Report BAW-10008, Part 1, Rev. 1 (Reference 16 on page 3-175).

The discussion and tabulated loads and deflections are submitted as a basis for the conclusion that the fuel assemblies can withstand a LOCA, the combined effects of a safe shutdown earthquake and loss-of-coolant accident, a safe shutdown earthquake, and an operating basis earthquake without exceeding the respective allowable limits.

### 3.9.2.4.1 Component Description

#### *3.9.2.4.1.1 Reactor Vessel*

The general arrangement of the reactor pressure vessel is shown on Figure 3-39. The reactor vessel consists of a cylindrical shell, a spherically dished bottom head, and a ring flange to which a removable reactor closure head is bolted. A cylinder welded to the vessel's shell supports the vessel and extends downward to a flanged base ring which is bolted to the building's foundation. The reactor vessel's ring flange includes an internal ledge to support the core and the internal structural components. The vessel has two outlet nozzles through which reactor coolant is transported to the steam generators, and four inlet nozzles through which reactor coolant re-enters the reactor vessel.

#### *3.9.2.4.1.2 Reactor Internals*

The internal components of the reactor include the plenum assembly and the core support assembly. The reactor internals assembly is shown on Figure 3-39. The reactor internals are supported by a ledge on the

inside of the reactor vessel closure flange and are designed to support the core, maintain fuel assembly alignment, and limit fuel assembly movement.

Plenum Assembly

The plenum assembly, located directly above the reactor core, consists of a plenum cover, upper grid, control rod guide tube assemblies, and a flanged plenum cylinder with openings for reactor coolant outlet flow. The upper grid assembly is attached to a flange which is bolted to the lower flange of the plenum cylinder.

Core Support Assembly

The core support assembly consists of the core support shield, core barrel, lower grid, flow distributor, and thermal shield.

*3.9.2.4.1.3 Fuel Assembly*

The canless fuel assembly (Figure 3-40) consists of six intermediate and two end spacer grids, 16 guide tubes, and two end fittings. These components form the basic structural cage. There is an instrumentation tube in the central position of the array. Seven spacer sleeve segments are positioned along the length of the fuel assembly on the instrumentation tube - one segment between each spacer grid.

The spacer grids are constructed from slotted strips assembled in egg-crate fashion and welded at each intersection. Each grid has 32 strips, 16 perpendicular to 16, to form a 15 by 15 array. Contact points are formed in each strip which extends into each square opening to contact and support fuel rods and control rod guide tubes in two mutually perpendicular planes. The end spacer grids differ from the intermediate spacer grids in that the peripheral strip is extended to form a square box which is screwed to the end fitting. The spacer grids locate fuel rods, maintain coolant channel geometry, and contribute to the lateral stiffness of fuel assemblies.

The lower end fitting is a weldment of two castings. The base casting, to which the grid casting is welded, fits into and rests on the lower grid assembly of the reactor core support assembly. The upper end fitting assembly is similar to the lower end fitting assembly. Penetrations in the upper end fitting grid are provided for the guide tubes. Attached to the end fitting are a coiled compression spring and a cast holddown spider. The spider consists of a ring of the same diameter as the spring with four radial bars that contact the upper grid assembly of the reactor plenum assembly.

The end fittings serve as rigid connectors for the 16 guide tubes, provide lateral and vertical location of fuel assemblies, and act to restrain the fuel rods vertically, thereby positively determining the vertical location of the fuel. In addition, the upper end fitting must be capable of housing the holddown spring to prevent fuel assembly lift off, and it must be capable of absorbing energy during LOCA contact of the fuel assembly with the upper grid. The guide tubes provide guidance for control rods. The spacer sleeves provide positive axial positioning of the spacer grids.

Table 3-24 provides a list of the materials used in the fuel assembly.

**3.9.2.4.2 Fuel Assembly Structural Design Criteria**

The fuel assembly is a very complex structure composed of many components; such as spacer grids, fuel cladding, etc. Due to this complexity, it is quite difficult to accurately calculate the stresses in these individual components. B&W has therefore established load or deflection limits for the individual components of the fuel assembly, rather than stress limits, as described below. Actual numerical limits are

shown in Table 3-25 and the tests conducted to determine these limits are described in Section 3.9.2.5, "Correlations of Reactor Internals Vibration - Tests with the Analytical Results" on page 3-151.

Loads and permanent deflection for the operating basis earthquake (OBE) are limited as follows:

1. Loads on the fuel assembly spacer grid shall not exceed the elastic limit of the spacer grid determined from tests performed on production grids.
2. There shall be no permanent deformation of the fuel assembly spacer grids.

Loads and permanent deflection for the safe shutdown earthquake (SSE), LOCA, and simultaneous LOCA and SSE are limited as follows:

1. Loads on the fuel assembly spacer grid are allowed to exceed the elastic limit, but the permanent deformation of the spacer grid shall not exceed that which would distort the guide tubes and prevent the insertion of the control rods. This value of permanent deformation is determined by tests on production grids.
2. To provide stability, loads on the control rod guide tubes and end spacer grid assembly are limited to 85 percent of the critical Euler buckling load. The value of 85 percent is chosen as a value, based on engineering judgment, so as not to design to failure.
3. Loads on the spacer grid welds shall be limited to 85 percent of the load that would cause failure. The value of this load is determined by tests on production grids.
4. Loads on the bolts connecting the end spacer grid skirt to the end fitting shall be limited to 85 percent of the load that would cause the bolts to fail in shear. The value of this load is determined by tests on production end grid assemblies.

The preceding criteria provide sufficient safety margin against failure. All margins in this subdivision are calculated as follows:

$$\text{Margin} = \frac{[\text{Allowable (load)} - \text{Applied (load)}] \times 100\%}{\text{Applied (load)}}$$

Since the allowable loads are based on the foregoing criteria, the margins quoted are in excess of those required by the criteria. Thus, any positive margin, including zero, is acceptable. A zero margin, for example, indicates that the criterion has just been met.

#### 3.9.2.4.3 Loads

##### *3.9.2.4.3.1 Vertical Loads On Core During LOCA*

The total force acting on a single fuel assembly for the outlet rupture is given in Figure 3-41. Figure 3-41 is a combination of Figure 6 ($\Delta P$ across core for a 36-inch-i.d. outlet break) and Figure 10 (shear force on core for a 36-inch-i.d. outlet break) from Topical Report BAW-10008, Part 1, Rev. 1, (Reference 16 on page 3-175). It is found in the following manner:

$$\text{Figure 3-41} = \frac{\text{(Fig. 6) (blocked area of core)} + \text{(Fig. 10)} - \text{(weight of core)}}{\text{177 fuel assemblies}}$$

This combined pressure and fluid friction force is sufficient to cause the fuel assemblies to lift off of the lower grid and contact the upper grid. They deflect the upper grid, causing axial loads in the control rod guide assemblies and subsequent deflection of the plenum cover beams. The resisting force from the plenum cover stops the fuel assemblies and causes them to return to the lower grid.

*3.9.2.4.3.2 Horizontal Thrust Force During LOCA*

The LOCA thrust force acting at the vessel's outlet nozzle is analyzed using the FLASH computer code and the relationship

Thrust = pressure x area.

Testing associated with the LOFT program tends to confirm that the horizontal thrust can be calculated by this relationship. The FLASH program has been used to correlate the vessel pressure and, therefore, the thrust for some of the semi-scale blowdown tests.

The results for a 36-inch outlet pipe rupture are shown in Figure 3-42.

*3.9.2.4.3.3 Seismic Excitation*

The specific seismic time history used in this analysis was determined for the Oconee site. The record used is the EL Centro 1940 NS Earthquake normalized to the Oconee site level.

Appendix A to BAW-10035 (Reference 17 on page 3-175) gives the following data:

1. A picture of the complete digitalized record used in the analysis.
2. A general description of the manner of digital-to-analog conversions of data; an estimate of the accuracy of the process and a description of the input techniques.
3. Complete acceleration response spectra comparisons at 1 and 10 percent critical damping.

**3.9.2.4.4 Models Used in Analysis**

*3.9.2.4.4.1 Horizontal Contact Analysis*

Structurally, the fuel assemblies are long slender beams which are responsive to horizontal excitations. Because of the proximity of the assemblies, these motions could result in midspan contact. The concern is that such contacts could produce unacceptable damage to the spacer grids and thus reduce coolant flow or restrict control rod motion. Two possible forms of horizontal excitation are seismic and LOCA. Seismic excitation occurs at the vessel's foundation, and the LOCA produces a thrust force (as described in Section 3.9.2.4.3.2, "Horizontal Thrust Force During LOCA") at the nozzle. The vertical component of the earthquake was considered with the horizontal analysis. However, because of the vertical stiffness of the reactor internals, the seismic contribution to the displacement of the core is negligible (about 0.002 to 0.003 inch) with respect to the horizontal motion.

Both of these excitations can produce horizontal motion of the fuel assemblies, so that a dynamic model including all the components involved - the reactor vessel, the control rod drives, the internals, and the fuel assemblies - is needed.

This overall model is divided into two segments. The first segment includes all the components named above except individual fuel assemblies, and involves recording the motions of the upper and lower grid plates, the core support shield, and earth velocity versus time. These motions are input excitations for the second segment of the overall model.

The first step in the solution is to determine a model that accurately represents the structure being investigated. The more masses used, the more accurate (but also the more complex) the solution. The investigation of different models shows that for these horizontal contacts a nine-mass model (Figure 3-43) was sufficient to describe the motions of the components. Engineering sketches of the structural features

of importance and a precise description of the location of and basis for the computation of masses and section properties/boundary conditions is given in Appendix A to BAW-10035 (Reference 17 on page 3-175).

The method used in this dynamic model is the far coupled, "lumped mass" approach, which may be considered as the vibration equivalent of the finite element technique in static stress analysis problems. The distributed mass of components of the structure is considered to be concentrated at discrete points. These mass points are connected by massless flexible elements. The behavior of the total structure is then determined from the response of these mass points. No damping is included.

Once the model is fixed, a set of simultaneous equations is written for the structure. For a system of N masses, the equations in matrix form are:

$$[M](\ddot{X}) + [K]\ (X) = (F)$$

$$NxN\ Nx1\quad NxN\ Nx1\quad Nx1$$

where:

[M] = mass matrix,

$(\ddot{X})$ = acceleration matrix,

[K] = stiffness matrix,

(X) = displacement matrix,

(F) = force matrix.

This model is then programmed for a digital computer. A modification is made to invert the final flexibility matrix generated and thus obtain a stiffness matrix. Each row of the stiffness matrix is then divided by the mass corresponding to that row to obtain a "K/M" matrix. These values are then substituted into scaled equations and solved on the analog computer.

To validate the analog representation of the model, initial displacement test are performed. The digital program generates frequencies and mode shapes. The nine-mass model on the analog is displaced into a particular mode shape and then allowed to vibrate. The frequency and mode shape of vibration from the analog compares well with the digital results.

As shown in Figure 3-43, the seismic excitation is applied at the base of the reactor vessel and the results recorded. These time-history records have been compared with the published spectrum.

The simultaneous occurrence of the SSE and LOCA is also recorded. The seismic excitation is applied at the vessel's skirt, as described above, and the LOCA thrust force is applied at the nozzle. Owing to the relative timing of these two events, maximum fuel assembly displacement is obtained. The investigation indicates that maximum displacement gives maximum contacts and hence maximum loads.

In the second part of the program the model comprises five fuel assemblies, two core baffles, and associated circuitry. Each fuel assembly is modeled by far-coupling techniques with three lumped masses. Each mass has an individual damper and an elastic-plastic spring that represents the transverse structural properties of the grid. The core baffle is represented by an elastic spring. The clearances between assemblies and between the core baffle and outside assemblies are also established during the program. The elastic-plastic properties of the spacer grid are determined by test and used as program input. The frequency and damping properties of the fuel assembly are established by test, as described in Section

3.9.2.5, "Correlations of Reactor Internals Vibration - Tests with the Analytical Results" on page 3-151, and are used as program input.

Sketches of the second segment model are presented in Appendix A to BAW-10035 (Reference 17 on page 3-175). The second segment model consists solely of fuel assemblies. To determine their response, the motion of the upper and lower grid plates must be known. The response of the grid plates is obtained from the first segment model, which also contains the mass of the entire core to provide its influence on these motions. This is also discussed in Appendix A, BAW-10035 (Reference 17 on page 3-175).

Because of the input excitation, contacts occur between adjacent assemblies or between assemblies and the core baffle. The elastic-plastic grid spring allows a maximum value of force to be exerted, and any remaining motion of the assembly creates permanent deformation of the grid. The program considers the energy loss involved as well as the change in cross-sectional size of the impacting grid. These effects influence the event as it occurs and also influence the succeeding contacts. The program sums the total grid deformation from all contacts during the seismic history and presents this value as program output. These results are compared with the general design criteria as well as the specific criterion.

*3.9.2.4.4.2 Vertical Contact Analysis*

This analysis is conducted to determine the loads acting on the various parts of the fuel assembly as a result of vertical contact with the upper grid assembly during a LOCA. The fuel assembly, when subjected to the upward pressure force caused by the instantaneous rupture of a primary outlet pipe, will suddenly accelerate vertically toward the upper grid assembly. When the fuel assembly does contact the upper grid assembly, the fuel rods will tend to slip in the upper end spacer grid. Since the stiffness of the end spacer grid assembly is substantially greater than the axial stiffness of the guide tubes, once slippage occurs, a compressive load is applied to the guide tubes by way of the lower end grid-end fitting assembly. Since any dynamic buckling of the guide tubes during a LOCA could prevent control rod insertion, investigation of the loads applied to the guide tubes is the primary concern in this analysis.

The following conservative assumptions are made:

1. Based on the flexibility of the other members, the fuel rods are considered to be rigid.
2. There is no slip of fuel pellets relative to the fuel rod cladding.

The mathematical model of the fuel assembly is shown in Figure 3-44. By appropriately combining springs in series and parallel, this model is simplified to a single-degree-of-freedom system, i.e., a single mass and spring combination. However, the single spring reflects the nonlinear characteristics of the overall structural system. A typical load-deflection curve for this spring, based on data obtained from production fuel assemblies, is shown in Figure 3-45. The spring constant variation for the fuel assembly in relation to its location within the core for that part of the load deflection curve which occurs after the gap is closed is presented in this figure. It should be noted that this spring curve is for the beginning of life (BOL). Its shape will change considerably as a function of full-power operational time because of irradiation growth, other irradiation effects, and material yield strength variation. These effects are considered in the analysis. The forcing function acting on the model is based on the LOCA pressure curves given in Topical Report BAW-10008, Part 1, Rev. 1 (Reference 16 on page 3-175) and is shown in Figure 3-41. Details of its determination are in Section 3.9.2.4.3.1, "Vertical Loads On Core During LOCA" on page 3-145.

To account for the nonlinear spring characteristics of the fuel assembly model and also for the rapidly changing forcing function, a digital program is developed. Utilizing a numerical integration routine based on the linear acceleration assumption, the program calculates the three dynamic parameters of interest-displacement, velocity, and acceleration at every quarter-millisecond. A logic system monitors the displacement and adjusts the spring rate to reflect the nonlinearities of the spring. Calculation is stopped

when a negative velocity, indicating that the fuel assembly was moving in a downward direction, is encountered.

From the digital output, the maximum displacement of the fuel assembly is used to enter the spring load-deflection curve and read directly the total fuel assembly load. This load is used to determine the maximum guide tube load.

#### 3.9.2.4.5 Results

LOCA and seismic loads rather than stresses are combined. For the horizontal analysis, the LOCA time-history force is applied to the reactor system in conjunction with the earthquake time history. The start of the LOCA relative to the start of the earthquake is such that the maximum load on the fuel assembly spacer grid is obtained.

##### *3.9.2.4.5.1 Horizontal Contact Analysis*

The design criterion for horizontal contact is that the level of permanent distortion suffered during the safe shutdown earthquake, LOCA, or SSE plus LOCA must not prevent control rod insertion.

The results of the analysis described in Section 3.9.2.4.4.1, "Horizontal Contact Analysis" on page 3-146 show that the canless fuel assembly meets the general design criteria as well as the specific design criterion stated above; the margins of safety for the three excitation levels are as follows:

| Excitation | Margin, % |
|---|---|
| OBE | 80 |
| SSE | 300 |
| SSE + LOCA | 400 |

These margins were calculated as follows:

Oconee level = 0.10G

Level to obtain 150 mils = 0.40G

$$\text{Margin} = \frac{0.04 - 0.10}{0.10} \times 100 = 300\%$$

For the SSE case, for example, the allowable deformation equals 150 mils.

It is concluded that the reference fuel assembly design can withstand the horizontal contact loads.

##### *3.9.2.4.5.2 Vertical Contact Analysis*

##### *3.9.2.4.5.2.1 Guide Tube Buckling*

The specific design criterion for guide tube buckling is that the compressive load in guide tubes should not exceed 85 percent of the static Euler buckling load. The holddown spider will be allowed to yield since it serves no safety function.

The results of this analysis are presented in Figure 3-46, which shows that the maximum loads experienced by the guide tubes during a LOCA are less than the allowable guide tube loads defined in the design criteria. The margin of safety is 4 percent.

In the vertical contact analysis the effects of vertical and horizontal seismic excitation were considered. For the following reasons, however, their contributions were determined to be negligible.

1. Because of the high vertical stiffness of the vessel and internals, the core will experience essentially unamplified ground-motion in the vertical direction. The resulting forces in a fuel assembly will be less than 2 percent of those imposed by LOCA alone.
2. Horizontal seismic excitation of the fuel assembly will cause lateral displacement of the assembly. This was considered in reducing the allowable Euler buckling load of the guide tubes since the guide tubes would have some initial curvature due to horizontal assembly displacement. It was found that this small curvature of the span between spacer grids reduced the allowable buckling load of an assumed straight tube less than 2 percent.

In light of the above comments, the margin of safety quoted reflects the effects of combined LOCA and seismic loading; and seismic loads, although negligible, were considered in the vertical contact analysis.

It is concluded that the fuel assembly design can withstand a vertical LOCA contact.

*3.9.2.4.5.2.2 Upper End Spacer Grid Welds*

As shown in Figure 3-47, the spacer grid is formed of strips assembled in egg-crate fashion. The top and the bottom of each such intersection are tungsten-inert-gas welded. During a loss-of-coolant accident, the fuel assemblies contact the upper internals grid. The resultant fuel rod deceleration loads are transmitted through the end spacer grid. The load carried by the grid is limited to the slip load of the fuel rods in the grid.

The bending moment at the middle of a grid strip was calculated on the basis that the grid strip was a simply supported beam subjected to the slip load of the fuel rods. The moment divided by the grid depth is the force carried by the grid welds.

To verify this method of calculating the force on a spacer grid weld, a series of tests was conducted using portions of spacer grids loaded as shown in Figure 3-47. The grid strips were loaded to failure, and the corresponding force on the welds was compared with previous pull-test results for welds loaded only in tension. These tests indicated that the weld strength can be predicted using the analytical methods described.

The specific design criteria for the upper end spacer grid welds follows from the capability of these welds. The spacer grid welds are capable of supporting a tensile load of 225 pounds at room temperature. The comparable load at temperature (600°F) is 200 pounds. The stresses from normal LOCA and earthquake loads are limited to 85 percent of ultimate stress, reducing the 200-pound load to an allowable load of 170 pounds. The force carried by the end spacer grid welds was calculated using the method described above. For the total maximum possible fuel rod slip load, the maximum weld force during a LOCA is 150 pounds. The allowable load is 170 pounds, and the margin is 13 percent. The loads due to a LOCA or earthquake are not additive to those due to normal operation because the maximum loads are limited by the available friction loads between the end grids and the fuel rods.

For the upper end spacer grid, all major loading is caused by the axial motion of the fuel rods relative to the grid. The direction of this loading is normal to the plane of the grid as shown in Figure 9 of BAW-10035. Since each fuel rod is restrained in the axial direction by a frictional force exerted by the end spacer grid, the maximum load that any rod can exert on the grid is limited to this frictional force. This is

true regardless of the source of loading whether it be from normal operation, LOCA or earthquake. For this reason, the above conclusion regarding the addition of loads due to normal operation, LOCA or earthquake is considered valid.

It is concluded that the end spacer grid welds can withstand the vertical contact loads.

*3.9.2.4.5.2.3 End Spacer Grid Assembly*

This assembly consists of an end spacer grid, a skirt that connects the end spacer grid to the end fitting, and the end fitting. The skirt is formed by extending the 20-mil outside strips of the spacer grid and reinforcing them with a 30-mil doubler. This composite plate is attached to the end fitting with sixteen 3/8-inch-diameter screws countersunk into sixteen 1-inch-wide bosses on the end fitting, as well as butting against a shoulder on the end fitting.

The Euler critical load of this box section is calculated assuming that only the width of material in contact with the 1-inch-wide bosses on the end fitting is effective as column material; i.e., the "column" consists of sixteen 1-inch-wide strips which are assumed to be pin-ended. The remainder of the material is evaluated as cantilever springs providing lateral support at the column midheight.

The skirt and attaching bolts are evaluated for a compressive load equal to the maximum possible slip load of the fuel rods in the end spacer grid due to vertical contact at the beginning of life.

The assembly is also evaluated for transverse loads. An arbitrary and conservative lateral deflection of 1 inch from the horizontal contact analysis is used, and the moment at the bolted joint is determined.

The specific design criteria for the end spacer grid assembly is that the skirt must not buckle. The allowable load is 85 percent of the critical buckling load.

It is concluded that the end spacer grid assembly is adequate for the maximum anticipated loads as given in Table 3-25. Some minimum margins are as follows:

| Component | Margin, % |
|---|---|
| Skirt buckling | 16 |
| Bolt shear | 16 |

## 3.9.2.5 Correlations of Reactor Internals Vibration - Tests with the Analytical Results

Described in this section is fuel assembly testing performed to support the analysis in Section 3.9.2.4, "Dynamic System Analysis of the Reactor Internals Under Faulted Conditions" on page 3-143.

### 3.9.2.5.1 Frequency and Damping Tests

The fuel assembly frequency and damping values were established from several test programs in which full-sized test specimens were used. Tests were performed in air, in still water at temperatures up to 200°F, and in still and flowing water at reactor operating conditions (650°F and 2200 psi). Both displacement loading (pluck tests) and steady-state sinusoidal excitation were used.

By using the Oconee response spectra displacements vs. frequency, an estimate of the maximum amplitude for a fuel assembly was established. Tests were performed over a large range of amplitude (0

in. - 0.300 in.) and a plot of frequency vs. amplitude was obtained. Fuel assembly frequencies were not established, but were part of the data obtained as a result of the tests.

The fuel assemblies were supported both top and bottom in tests fixtures which were made to duplicate the restraint provided by the upper and lower grid plates, respectively. Further details are provided in Appendix B of BAW-10035 (Reference 17 on page 3-175).

Actual production materials were used, with the exception of the fuel, for which brass pellets were substituted to represent actual fuel weight. A list of materials is given in Table 3-24.

The description of the test data obtained and the analysis and interpretation of the results are discussed in Appendix B of 8AW-10035 (Reference 17 on page 3-175).

This extensive testing confirmed that the natural frequency of the assembly is in the low frequency range, and provided the damping values for use in the analysis. Both frequency and damping are also dependent on the amplitude of vibration; this dependence is due to fuel rod slippage in the spacer grids, and the slippage is the prime source of the damping values. The tests also established that damping increases with the coolant flow velocity owing to the effect of coolant flow on the spacer grids.

#### 3.9.2.5.2 Spacer Grid Compression Case Tests

The analysis of fuel assemblies during conditions of horizontal acceleration and contact required knowledge of certain transverse characteristics of the spacer grids. These characteristics are (1) the elastic and plastic load abilities, and (2) the amount of permanent deformation and energy that can be absorbed without interfering with control rod motion. This information was obtained by performing compression or crush tests on individual spacer grids.

Each grid was filled with simulated fuel rods and guide tubes and then mounted in a vertical plane in the tensile machine (Figure 3-48). A vertical compressive load was applied while recording the grid deflection and other significant data. During the loading, efforts were continually made to insert poison pin segments into the guide tubes. The load and the grid distortion at which this was no longer possible were recorded. These test results are corrected for temperature effects by applying a ratio of the grid material's (Inconel-718) yield strength at temperature (600°F) to its yield strength at room temperature. The results of this procedure give the initial elastic load ability for the grids, the load cycling as the horizontal rows of the grid fail, and the permanent distortion of the grid at the time when the poison pins could no longer be inserted.

These results were obtained from essentially static tests. The first two results were used as input data for the horizontal contact analysis, and the third was used as the acceptance criterion for grid deformation as though it had been obtained dynamically. The results have been checked qualitatively against the effect of dynamic loading with a drop test as described below.

#### 3.9.2.5.3 Spacer Grid Case Drop Test

The results of the compression tests depended to some extent on the mode of failure, which was transverse displacement of individual fuel rod rows. To check this failure mode qualitatively under dynamic loading conditions, a single grid loaded with short lengths of fuel rod was dropped on a solid base so as to land on its side. Only those rows nearest the impact surface were crushed, and this result supported the assumption that dynamic loading would decrease with distance from the impacting surface. Therefore, since the analysis of the static crush test assumed that the maximum impact load was applied uniformly over the three rows of spacer grid nearest the impact surface, the test indicated that the analysis was conservative.

# 3.9.3 ASME CODE CLASS 1, 2, 3 COMPONENTS, COMPONENT SUPPORTS, AND CORE SUPPORT STRUCTURES

## 3.9.3.1 Load Combinations, Design Transients and Stress Limits

### 3.9.3.1.1 Reactor Coolant System

The Reactor Coolant System is designed structurally for 2,500 psig and 650°F. The system will normally operate at 2,155 psig and 604°F. The design transients are defined in Section 3.9.1.1, "Design Transients" on page 3-141.

The number of transient cycles specified in Table 5-2 for the fatigue analysis is conservative.

Reactor Coolant System components are designated as Class 1 equipment and are designed to maintain their functional integrity during an earthquake. Design is in accordance with the seismic design bases shown below. The loading combinations and corresponding design stress criteria for internals and pressure boundaries of vessels and piping are given in the section. These are summarized in Table 3-26. A discussion of each of the cases of loading combinations follows:

Case I - Design Loads Plus Operating Basis Earthquake Loads - For this combination, the reactor must be capable of continued operation; therefore, all components excluding piping are designed to Section III of the ASME Code for Reactor Vessels. The primary piping is designed according to the requirements of USAS B31.1 and B31.7. The $S_m$ values for all components, excluding bolting, are those specified in Table N-421 of the ASME Code. The $S_m$ value for bolts are those specified in Table N-422 of the ASME Code.

Case II - Design Loads Plus Safe Shutdown Earthquake Loads - In establishing stress levels for this case, a "no-loss-of-function" criterion applies, and higher stress values than in Case I can be allowed. The multiplying factor of (1.2) has been selected in order to increase the code-based stress limits and still insure that for the primary structural materials, i.e., 304 SST, 316 SST, SA302B, SA210B, and SA106C, an acceptable margin of safety will always exist. A more detailed discussion of the adequacy of these margins of safety is given in B&W Topical Report BAW-10008, Part 1, "Reactor Internals Stress & Deflection Due to LOCA and Maximum Hypothetical Earthquake." The $S_m$ values for all components are those specified in Table N-421 of the ASME Code.

The load cases for consideration of the faulted condition are defined below.

A loss-of-coolant accident coincident with a seismic disturbance has been analyzed to assure that no loss of function occurs. In this case, primary attention is focused on the ability to initiate and maintain reactor shutdown and emergency core cooling. Two additional cases are considered as follows:

Case III - Design Loads Plus Pipe Rupture Loads - For this combination of loads, the stress limits for Case II are imposed for those components, systems, and equipment necessary for reactor shutdown and emergency core cooling.

Case IV - Design Loads Plus Safe Shutdown Hypothetical Earthquake Loads Plus Pipe Rupture Loads - Two thirds of the ultimate strength has been selected as the stress limit for the simultaneous occurrence of safe shutdown earthquake and reactor coolant pipe rupture. As in Case III, the primary concern is to maintain the ability to shut the reactor down and to cool the reactor core. This limit assures that a materials strength margin of safety of 50 percent will always exist.

The design allowable stress of Case IV loads is given in B&W Topical Report BAW-10008 for 304 stainless steel. This curve is used for all reactor vessel internals including bolts. It is based on adjusting the ultimate strength curves published by U. S. Steel to minimum ultimate strength values by using the ratio of ultimate strength given by Table N-421 of Section III of the ASME Code at room temperature to the room temperature strength given by U. S. Steel.

In Cases II, III and IV, secondary stresses were neglected, since they are self-limiting. Design stress limits in most cases are in the plastic region, and local yielding would occur. Thus, the conditions that caused the stresses are assumed to have been satisfied. See B&W Topical Report BAW-10008, Part 1, for a more extensive discussion of the margin of safety, the effects of using elastic equations, and the use of limit design curves for reactor internals.

#### *3.9.3.1.1.1 Analysis of Reactor Coolant System*

This section contains the following categories of information:

1. Pertinent information on the seismic design of the Reactor Coolant System.
2. A tabulation of reactor coolant piping stresses calculated by the static approach at the most critical locations.
3. A description of the type and location of each major component support analyzed, its design, and the seismic amplification associated with the location of the support in the building.
4. An evaluation of results tabulated in (2) above.
5. A correlation between a free-standing spacial analysis of the Nuclear Steam System and a planar analysis considering building-loop inter action.

##### *3.9.3.1.1.1.1 Scope of Analysis*

The Reactor Coolant System consists of the reactor vessel, coolant pumps, steam generators, pressurizer, and interconnecting piping. For the purpose of seismic analysis, however, the Reactor Coolant System is considered to consist of the following:

a. Reactor vessel

b. One steam generator

c. Three interconnecting reactor coolant pipes, one being the 36 in. reactor outlet and two 28 in. reactor inlet pipes.

d. One pump and motor assembly on each 28 in. inlet pipe.

e. One 10 in. pressurizer surge line.

f. One 2½ in. pressurizer spray line.

The loading combinations and corresponding design stress criteria for the pressure boundaries of both vessels and piping for the analysis are Case I and Case II as defined in Section 3.9.1.1, "Design Transients" on page 3-141.

Reactor Coolant System seismic forces are defined in terms of an acceleration spectrum. An acceleration spectrum is an envelope of expected accelerations over a range of expected natural frequencies. A family of acceleration spectrum curves for various damping values was constructed. For piping systems the spectrum curve of interest is associated with ½ percent of critical damping.

Pursuant to the applicable codes, the following additional sources of loading must be considered in the seismic design of the Reactor Coolant System:

a. Pressure

b. Dead load

c. Thermal load

*3.9.3.1.1.1.2 Description of Analytical Models*

Seismic Analysis

The seismic analysis of the Reactor Coolant System is a dynamic analysis based on the theory and basic procedures outlined in References 10 on page 3-175 and 14 on page 3-175. The Reactor Coolant System as a basic piping structure is idealized in the analytical model by a concentrated mass system or structure. In this idealization, the mass of the structure in considered to be lumped or concentrated at a certain finite number of points. The resistance to deflection is caused by elastic members having strength and stiffness but are weightless.

The dynamic response of a concentrated-mass system having multiple degrees of freedom involve first determining the frequencies and shapes of the normal modes of vibration. The dynamic response of the system to a given dynamic load is evaluated by using the frequencies, mode shapes, modal participation factors, and a dynamic load given as an acceleration-frequency spectrum.

Schematic drawings of the analytical model are shown on Figure 3-49 and Figure 3-50. The model starts at the base of the reactor vessel support skirt Point A1-160. It extends through the reactor vessel, the three main reactor coolant pipes to one steam generator, then to the base of the steam generator support skirt at point A6-101. Only one steam generator was included in the model on the basis of symmetry. The only external anchors or restraints in the system are at the base of the support skirts, points A1-160 and A6-101. No intermediate Reactor Building to Reactor Coolant System dynamic coupling is included. The significance of building coupling to Reactor Coolant System is discussed in Section 3.9.3.1.1.1.8, "Evaluation of Seismic analysis of Reactor Coolant System for Existing Configuration" on page 3-160.

The mass of the system was considered to be acting or lumped at 16 points in the following manner:

a. The reactor vessel and steam generator are each represented by two masses.

b. Each reactor coolant pump motor is represented as one mass.

c. The reactor coolant pump on Branch 23 is represented by a single mass point.

d. The 36 in. reactor coolant pipe is represented by three mass points.

e. Each 28 in. reactor coolant pipe is represented by two mass points, one on each side of the pumps.

The lumped masses are connected by eighteen elastic members as follows:

a. Both the steam generator and reactor vessel are each represented by two masses connected by two elastic members which represent the vessel skirt and shell.

b. Each pump motor is connected to the pump by one member.

c. The remainder of members are the various straight lengths and bends in the three reactor coolant pipes.

Each pump body and the upper half of the steam generator were assumed to be rigid members in the system.

The dynamic analysis is done in three steps by computer programs:

a. The first program calculates the flexibility matrix for each branch in the system where each branch begins and ends at a mass point on anchor. The basic theory and equations are shown in References 12 on page 3-175 and 13 on page 3-175 with the additional consideration of flexibility due to axial and cross shearing deformations and using elbow flexibility factors, k, from USAS B31.1. The branch flexibility matrices are referred to the system coordinate axis at point A-919. They define the displacement of three deflections and three rotations resulting from forces and moments, i.e., for six degrees of freedom per mass point.

b. The second program requires the basic input of the branch matrices, the value of all the masses and the seismic spectrum. The program first inverts each flexibility matrix to obtain a 6 x 6 stiffness matrix with the translational elements in the upper left 3 x 3 sub matrix. An overall stiffness matrix is then assembled having 6N x 6N elements were N is the number of modal branch or mass points. This results in a 96 x 96 matrix. The rows and columns are then arranged such that all the translational elements are in the upper left 3N x 3N or 48 x 48 submatrix. For this analysis, the rotational inertia is assumed to be zero; therefore, the rotational degree of freedom is eliminated and only the upper left 48 x 48 submatrix is needed. (This means the final stiffness matrix used in the motion equations represents three degrees of freedom per mass point as translations in the X, Y and Z directions of the coordinate axes. Likewise, the inertia effects are three forces at each mass point.) The stiffness matrix and mass matrix provide the information needed for the next step, the program step of calculating the eigenvalues and natural frequencies. The Wilkinson method was used for this solution. The eigenvectors (mode shapes) corresponding to each frequency are then determined. The program lastly calculates for each mode the equivalent static loads at each mass point, using the seismic acceleration response spectra. The basic theory and equations used for mode shapes, participation factors and equivalent static forces are given in Reference 10 on page 3-175. The equivalent static force at each mass point for all modes is calculated by taking the largest absolute value of the inertia load at each mass point and adding to it the root mean square of the remaining modes.

c. The third program is a piping flexibility program. It calculates the forces and moments at all the branch points using the equivalent static forces from step b.

Dead Load Analysis

The analytical model for the dead load analysis is the same as for the seismic analysis and can also be represented by Figure 3-49 and Figure 3-50. In this case, the weight of the piping is placed at the mass points as concentrated forces. The program used to calculate the system forces and moments is the piping flexibility program referred to in the previous section. The output of forces and moments were used to calculate stresses to be combined with other loadings.

Thermal Flexibility Analysis

As was the case with dead load, the analytical model used for thermal expansion analysis was the same as for the seismic analysis. Likewise, the piping flexibility program was used with input of thermal expansion effects. This model is valid for the 28 in. pipes entering the steam generator near the bottom which is anchored. The results of the thermal expansion analysis show center line horizontal deflections at this location, C-901, of 0.010 in. in the X direction and 0.008 in. in the Z direction. Therefore, the steam generator can be assumed to be horizontally restrained at this location. However, the thermal expansion motion near the top of the generator (at the LOCA restraint) is shown from the analysis to be

about 0.25 in. Since the generator will be restrained in the hot condition, this motion is excessive for a valid analysis of the 36 in. pipe which enters at the top of the generator.

An alternate thermal analysis was made of the 36 in. line from the nozzle attachment to the steam generator to the nozzle attachment to the reactor vessel. The analytical model is shown on Figure 3-51. This model also includes the 10 in. pipe surge line. The anchor point motions at "A" and "B" are on the basis of no horizontal center line movement of either the steam generator or the reactor vessel. This will meet the requirement of the restrained generator.

The program used for the analysis of the 36 in. pipe is a direct adaptation of the method described in Reference 13 on page 3-175. The output of forces and moments at branch points was used to calculate stresses to be combined with other loadings.

*3.9.3.1.1.1.3 Stress Analysis of Reactor Coolant Piping*

Stress calculations made at various locations throughout the piping system are done in accordance with the Nuclear Power Piping Code USAS B31.7. Several locations, as noted on the table of results, were analyzed in accordance with paragraph 1-705.1 of B31.7. The remaining points were analyzed in accordance with Appendix F of B31.7. Stress analysis of the pressurizer surge and spray lines is completed in accordance with ASME Code, Section III, Subarticle NB-3600. Primary and primary plus secondary stresses were calculated at each location and comparison made to 1.5 $S_m$ and 3.0 $S_m$, respectively. In addition, primary stresses resulting from safe shutdown earthquake, dead load and pressure effects were compared to the value 1.8 $S_m$ (1.2 x 1.5 $S_m$). The highest stress at any location (branch point No. 10) was found to be 27,269 psi which is below the allowable value of 33,120 psi. For the points analyzed in accordance with Paragraph 1-705.1, the loading technique is as follows: The seismic moment is evaluated by adding the largest absolute value of either horizontal (X or Z) earthquake reaction plus the rms of the remaining modes to the absolute value of the vertical earthquake reaction. This value is then added to the dead weight moment and pressure loading to calculate primary stresses. Primary plus secondary stresses are calculated by doubling the seismic loading used for primary stresses and also including thermal loadings.

For the analysis described in Appendix F of USAS B31.7, the seismic loads are combined based on Table 3-27 to attain the worst case condition. The dead weight and thermal loads are used as indicated in Table 3-27 to calculate primary and primary plus secondary stresses.

Table 3-27 through Table 3-57 give pipe data, forces and moments, and final pipe stresses for various locations throughout the piping system.

*3.9.3.1.1.1.4 Stress Evaluation of Reactor Vessel*

Stress evaluation of the reactor vessel is discussed in Section 5.2.3.3.1, "Stress Analysis" on page 5-16.

*3.9.3.1.1.1.5 Stress Evaluation of Steam Generators*

1. General

   Because the steam generator is of a straight tube-straight shell design and because of a minor difference in the coefficient of thermal expansion between Inconel and carbon steel, there exists a structural limitation on the mean temperature difference between the tubes and the shell. During normal operation of the steam generator, the tube mean temperature should not be more than 32°F higher than the shell mean temperature. The maximum calculated mean tube to shell $\Delta T$ at normal operating conditions poses no problems to the structural integrity of the reactor coolant boundary. The effect of loss of reactor coolant would impose tensile stresses on the tubes and cause slight yielding across the tubes. Such a condition would introduce a small permanent deformation in the

tubes but would in no way violate the boundary integrity. The rupture of a secondary pipe would cause the tubes to become warmer than the shell and may cause tube deformation. Blowdown tests a simulating secondary side blowdown on a 37-tube model boiler, show that although a slight buckling in the tubes occurred, there was no loss of reactor coolant.

Calculations confirm that the steam generator tube sheet will withstand the loading resulting from a loss-of-coolant accident. The basis for this analysis is a hypothetical rupture of a reactor coolant pipe resulting in a maximum design pressure differential from the secondary side of 1050 psi. Under these conditions there is no rupture of the primary to secondary boundary (tubes and tube sheet).

The maximum primary membrane plus primary bending stress in the tube sheet under these conditions is 15,900 psi across the center ligaments which is well below the ASME Section III allowable limit of 40,000 psi at 650°F. Under the condition postulated, the stresses in the primary head show only the effect of its role as a structural restraint on the tube sheet. The stress intensity at the juncture of the spherical head with the tube sheet is 14,970 psi which is well below the allowable stress limit. It can therefore be concluded that no damage will occur to the tube sheet or the primary head as a result of this postulated accident.

In regard to tube integrity under loss of reactor coolant, actual pressure tests of 5/8 in. o.d./0.034 inch wall Inconel tubing show collapse under an external pressure of 4,950 psi. This is a factor of safety of 4.7 against collapse under the 1,050 psig accidental application of external pressure to the tubes.

The rupture of a secondary pipe has been assumed to impose a maximum design pressure differential of 2,500 psi across the tubes and tube sheet from the primary side. The criterion for this accident permits no violation of the reactor coolant boundary (primary head, tube sheet, and tubes).

To meet this criterion, the stress limits delineated in the ASME Pressure Vessel Code, Section III, Paragraph N-714.2 for hydrotest limitations are applicable for the aforementioned abnormal operating circumstance. The referenced section states that the primary membrane stresses in the tube sheet ligaments, averaged across the ligament and through the tube sheet thickness, do not exceed 90 percent of the material yield stress at the operating temperature; in addition, the primary membrane plus primary bending stress in the tube sheet ligaments, averaged across the ligament width at the tube sheet surface location giving a maximum stress, does not exceed 135 percent of the material yield stress at the operating temperature.

An examination of stresses under these conditions show that for the case of a 2,500 psi design pressure differential, the stresses are within acceptable limits. These stresses together with the corresponding stress limits are given in Table 3-58.

The basic design criterion for the tubes assumes a pressure differential of 2,500 psi in accordance with Section III. Therefore, the secondary pressure loss accident condition imposes no extraordinary stress on the tubes beyond that normally expected and considered in Section III requirements.

The superimposed effect of secondary side pressure loss and safe shutdown earthquake has been considered. For this condition, the criterion is that there be no violation of the primary to secondary boundary (tube and tube sheet). For the case of the tube sheet, the safe shutdown earthquake loading will contribute an equivalent static pressure loading over the tube sheet of less than 5 psi (for vertical shock).

The effect of fluid dynamic forces on the steam generator internals under secondary steam break accident conditions has been simulated in a 37-tube laboratory boiler. Results of the tests show that reactor coolant boundary integrity is maintained under the most severe mode of secondary blowdown.

The ratio of allowable stresses (based on an allowable membrane stress of 0.9 of the nominal yield stress of the material) to the computed stresses for a design pressure differential of 2,500 psi are summarized in Table 3-59.

2. Stress Intensities and Cumulative Usage Factors

Table 3-62 lists the steam generator stress intensities at various load points due to design conditions as defined in the design specifications.

The results of the transient analysis and the determination of fatigue usage factor at the same load points are listed in Table 3-62.

3. Additional Information

Additional information discussed in BAW-10027 includes:

a. Discussion of thermal fatigue due to fluctation and shifting of the liquid-vapor interface on the tubes.

b. Stress distributions and effective elastic constants obtained under thermal inplane and transverse loadings, and analysis of tube to tube sheet complex.

c. Vibration Analyses.

*3.9.3.1.1.1.6 Reactor Coolant Pumps*

The reactor coolant pump casings are designed, fabricated, inspected and tested to meet the intent of the ASME Boiler and Pressure Vessel Code, Section III, for Class A vessels, but are not code stamped.

The reactor coolant pump casing has been completely analyzed including a dynamic analysis separately from the loop to insure that the stresses throughout the casing are below the allowable for all design conditions.

An analysis in accordance with Paragraph N-415.1 of the ASME Code was performed to determine if the pump casing required a fatigue analysis for the number of design cycles specified. This analysis showed that the pump casing bowl met all the requirements of Paragraph N-415.1. Thus a fatigue analysis was not required. However, a fatigue analysis was performed on the pump casing cover in which the worst possible stress combination was considered at the two most critical points in the cover. It was found from this analysis, with this very conservative approach, that the maximum cumulative usage factor is only 0.125 for design cycles specified for this plant.

A summary of the code allowables is listed in Table 3-60 and shown pictorially on Figure 3-53 and Figure 3-54. The reinforcement area is as defined in Paragraph N-454 of the ASME Code Section III. The stress analysis performed on the bowl and the attached nozzles showed that the stresses are within the allowable limits. Note that a factor of two was applied to the nozzle loading due to seismic reactions and when these were combined with the dead weight and thermal expansion reactions, the stress levels fell within a realistic allowable of the stress intensities shown in Table 3-60. A summary of maximum calculated stresses is given in Table 3-61.

The casing cover analysis indicates that the thermal stresses and pressure stresses on the cover are within the Section III code allowables.

There are no deviations from the applicable ASME Code requirements in the design and fabrication of the pump casings other than code stamping.

The stress analysis of the Reactor Coolant System has been reviewed to confirm the adequacy of the analysis for the revised reactor coolant pump design for Oconee 1.

The reanalysis of the Reactor Coolant System primary piping was performed considering the effects of the substitution of the Westinghouse pumps. The piping meets the requirements of USAS B31.7.

Only slight modifications to the primary piping configuration were required to incorporate the replacement pumps. No changes were made to the 36 in. i.d. hot leg piping.

The revised piping configuration was first compared to the dynamic model originally used for the dead weight and seismic analysis of the piping described in Section 3.9.3.1.1, "Reactor Coolant System" on page 3-153. Either pump has a large stiffness in comparison to the piping. The addition of the transition section increased the stiffness of that portion of the loop. The effect of the small angle elbow on overall seismic response would be negligible.

The masses used in the analysis were unchanged because the pump weights are approximately the same. Therefore, in view of the above and since the contribution of the seismic and dead load stresses to the total stress is small, the piping primary stress results presented in Section 3.9.3.1.1, "Reactor Coolant System" on page 3-153 are still valid.

However, the thermal stress portion of the analysis was redone for the revised cold leg piping, primarily because of the addition of the stainless steel transition section with its higher coefficient of thermal expansion. A model similar to Figure 3-49 and Figure 3-50 was employed. The revised primary plus secondary stresses for the cold leg are given in Table 3-66 which is comparable to Table 3-56 and meet the requirements of USAS B31.7, 1968 edition.

#### *3.9.3.1.1.1.7 Stress Analysis of Pressurizer Surge Line Piping*

Stress calculations made at various locations throughout the Surge Line were performed in accordance with the ASME Code, Section III, Subarticle NB-3600. Pursuant to the code, seismic, thermal, pressure and cyclic loadings were considered in the analysis.

The geometry, support conditions, joint and component descriptions are all shown in Figure 3-52. The diameter, thickness and material designation for the pipe is shown in Table 3-29.

The results of this analysis are presented in Table 3-57. The results indicate that the subject pipe meets all design criteria.

#### *3.9.3.1.1.1.8 Evaluation of Seismic analysis of Reactor Coolant System for Existing Configuration*

The objective of this evaluation is to show by qualitative analysis that the final stresses from an uncoupled analysis would be conservative compared to results from a building Nuclear Steam System (NSS) coupled analysis.

The dynamic seismic analysis was for a NSS uncoupled from any containment building structures. The steam generator will have lateral support for hot plant conditions and the significance of coupling was evaluated.

The effect of building coupling was evaluated from the results of a dynamic analysis on simplified models of the NSS and building secondary shield walls, as shown in Figure 3-55 and Figure 3-56. In these models the NSS weight is lumped as: The steam generator mass points 7 and 8, the 36 in. reactor coolant pipe mass points 8, 9 and 10 and the reactor vessel masses 10 and 11. The elastic members for the NSS are: steam generator support skirt No. 7; steam generator No. 8; 36 in. pipe numbers 9, 10, 11, 12, 13, 14 and 15; reactor vessel No. 16; reactor vessel support skirt No. 17. This portion of the NSS is basically the same as modeled on Figure 3-49 and Figure 3-50. The containment building secondary shield wall is represented by mass points 1 through 5 and elastic members 1 through 6. The lateral tie between the building and the NSS is elastic member 18.

These models were analyzed by a B&W computer program. This program performs a normal mode vibration calculation and applies a base motion spectra in a similar manner as the programs for the system described in Section 3.9.3.1.1.1, "Analysis of Reactor Coolant System" on page 3-154. This program is limited to single degree of freedom as translation per mass point.

Table 3-64 and Table 3-65 show a summary of the results from the program. Table 3-64 is a comparison of the absolute model summation of the inertia forces as effective static forces and deflections at each mass point.

Results from the ZY direction show that the building mass points will have an increase in effective static forces when coupled with NSS. Conversely, the NSS mass points have a decrease in effective static forces. The relative displacement between the steam generator mass points was reduced from 0.137 inches to 0.010 inches. The average relative displacement between piping mass points was reduced from 0.116 inches to 0.008 inches.

Results from the XY direction show an overall decrease in the mass point effective static forces for the NSS, with two individual points, one on the generator and one on the pipe, being larger. The average forces on the pipe masses, however do show a decrease when coupled. Similarly, the relative displacements shown a decrease as in the ZY direction.

Table 3-65 is a comparison of the internal forces and moments on the system elastic members. For the 36 in. pipe there is a marked decrease in values for coupled vs uncoupled in the ZY direction. In the XY direction, there is also an appreciable overall decrease in values with the exception of a local area near the juncture of element 10 and 11 at the tangent of the 180 degree bend. Here the coupled moment is greater than the uncoupled analysis moment. Fortuitously, this location is at a low stress area. To quantitatively evaluate this, the stresses at this location were calculated using an increase of seismic moment by the ratio of the coupled vs uncoupled moments. The stresses, when compared to the design calculations, show a change of less than one percent. This further indicates that the seismic forces at this location are relatively low.

The stiffness of the tie was selected on a preliminary basis. To evaluate the effects of increased rigidity, a computer run was made using a value 100 times stiffer for member 18. The results did not show any unfavorable changes in effective static forces or internal forces and moments on piping.

The effect of rotational spring constants at the base of the reactor vessel and steam generator were examined for the coupled system. Nominal values were selected to represent the strain in the concrete and anchor bolts due to seismic loading. Results of this analysis show that inclusion of spring constants in the dynamic model actually decrease the element moments and forces slightly.

It can be concluded that the coupling of the NSS to the building will not cause greater seismic responses or high resulting piping stresses.

*3.9.3.1.1.1.9 Summary and Conclusion*

A three dimensional model of the Reactor Coolant System was used to determine seismic mode shapes, frequencies, and inertia loads. The inertia loads along with thermal and dead load information was input to a piping flexibility program to obtain stresses and deflection. The Reactor Coolant System was considered uncoupled from any internal building structures. The resulting stresses were found to be within allowable limits. The maximum calculated piping stress for the operating basis earthquake and the safe shutdown earthquake (considering dead loads and pressure) are, respectively, 24,968 psi and 27,269 psi. The allowable stresses for these two conditions are 27,600 psi and 33,120 psi, respectively.

As was pointed out in the above paragraph, the seismic analysis for the Reactor Coolant System was based upon a free standing system, i.e., no coupling. However, the steam generator will have lateral support at the upper elevation for hot plant conditions and the significance of this coupling was evaluated. A planar model of the building secondary shield wall, steam generator, reactor vessel, and 36 in. pipe was analyzed. Results show that coupling of the Reactor Coolant System to the building will not cause greater seismic responses or higher resulting pipe stresses than a free standing system.

#### 3.9.3.1.2 Other Duke Class A, B, and C Piping

Piping which is Class A, B, or C is defined in Section 3.2, "Classification of Structures, Components, and Systems" on page 3-37. The applicable Code requirements are established in Section 3.2, "Classification of Structures, Components, and Systems" on page 3-37. The seismic requirements for this piping are defined in Section 3.2, "Classification of Structures, Components, and Systems" on page 3-37. The seismic analysis techniques are defined in Section 3.7.3, "Seismic Subsystem Analysis" on page 3-65.

#### 3.9.3.1.3 Field-Routed Piping and Instrumentation

Duke's practice is to detail the routing of all safety-related process lines regardless of size, except as follows:

1. Process piping - All main run process piping in Duke System Classifications A, B, C, D, E, and F is detailed on engineering drawings; however, items such as vents, drains, valve bypass warming lines, and pump seal water for all systems are "field run."

   Instrument impulse lines - end points and specific routing requirements of any safety-related instrument impulse lines are established and defined by Design Engineering, and the actual path is established in the field by Construction. At Oconee, the one exception and improvement is that the reactor coolant flow impulse lines were detailed and routed by Design Engineering.

2. It is not practical to limit "field run" piping to an extent greater than this for the following reasons:

   a. Obstructions to desirable routing would be difficult to determine and documentation of a precisely designed path would be lengthy, difficult to prepare, and difficult to follow.

   b. Revision to major process piping would cause changes in routing of small lines, resulting in many drawing changes without significant improvement in the final result.

   c. Sloping of impulse lines would be difficult to accomplish and document.

   Thus, field routing of small lines results in a superior job since obstructions and other design revisions are clearly visible and easier to consider while meeting design requirements as established by Design Engineering.

3. The special rigorous quality assurance measures and performance tests that will be conducted to assure satisfactory installation of field run piping and instrument impulse lines are as follows:

   a. All field engineered lines are schematically shown either on a diagrammatic, an instrumentation detail or a piping drawing such that mistakes in valving, connection termination points and materials are virtually eliminated.

   b. All field run piping requiring seismic design is reviewed after erection by appropriate Design Engineering Department personnel and applicable seismic control are detailed and forwarded to the Construction Department for installation.

   c. Except for very low pressure lines downstream of vent and drain valves and instrument impulse lines, all "field run" piping is hydrostatically tested in accordance with the requirements of the main process system.

d. An engineering surveillance program is conducted after erection to review all safety-related piping as well as non-safety related piping in the area to assure that appropriate criteria have been followed.

e. Instrument impulse line installation is inspected by an independent group on site and stamped approval and signoff are required before the system can be turned over to operating personnel.

f. Instrumentation testing programs are well defined by Duke test procedures. These tests document conclusively that the instrument loops are correctly installed and operate properly.

g. Design Engineering frequently visits the site and inspects instrumentation installations with corrective measures as necessary originating with the designer.

h. Both "field run" piping and impulse line installation work are scrutinized by three independent parties other than the erection party. They are the Construction QC Group, Operating personnel and Design Engineering.

This practice of "controlled field routing" of small piping and instrument impulse lines produces the best possible overall results. It is not practical to limit "field run" piping to a greater extent.

### 3.9.3.2 Pump and Valve Operability Assurance

Equipment pre-operational test programs are described in Section 14.2, "Tests Prior to Reactor Fuel Loading" on page 14-7.

### 3.9.3.3 Design and Installation Details for Mounting of Pressure Relief Devices

Design analysis and installation criteria for safety and relief valves located within the reactor coolant and main steam (thru main stop valves) pressure boundaries are as follows:

1. Piping and its Support-Restraint System are designed to accommodate and/or restrain the piping for both dynamic and static loadings as applicable such that stresses produced are within code allowables for the following:

   a. Dead weight effect

   b. Thermal loads and movements

   c. Seismic loads and deflections - movements

   d. Safety valve thrust and moment

   e. Maximum absolute differential movement between structures

   Applicable loadings are combined and considered as described in Section 3.2, "Classification of Structures, Components, and Systems" on page 3-37.

2. Nozzles are analyzed and appropriate reinforcement added such that code allowables stresses are maintained for:

   a. Internal pressure

   b. Safety valve thrust

   c. Safety valve moment

In particular, for the main steam lines outside the Containment, pressure relief is accomplished through the use of a single relief valve and sufficient safety valves to meet code requirements. The safety valves are set for progressive relief in intermediate steps of pressure within the allowed range of pressure settings to prevent more than two valves actuating simultaneously. Valves are located on a horizontal run of pipe

and are oriented in a manner that will produce torsion and bending in the main pipe during operation of the valves. The valves are staggered on opposite sides of the main steam line and set to relieve progressively to counterbalance the torque produced so that the maximum net torque on the piping system is essentially only that which results from the discharge of two safety valves. For conservatism, the piping system is designed to accept the net torque resulting from three safety valves operating simultaneously on the same side of the line. The piping support and restraint system is designed using shock suppressors and rigid stops to limit piping system stresses within code allowables as discussed above.

Dynamic thrust effects were analyzed for the Reactor Coolant System pressurizer relief discharge line to the Quench Tank, constituting a closed system. Stresses produced by the thrust effects were within the established Code allowables for the station. No other safety-related closed systems exist for Oconee.

## 3.9.3.4 Component Supports

### 3.9.3.4.1 Reactor Coolant System Component Supports

#### *3.9.3.4.1.1 Description of Supports*

Both the reactor vessel and steam generator are supported by cylindrical skirts rigidly attached to the vessels and bolted to the foundation by means of an integral base plate. The skirts are designed in accordance with ASME Section III and criteria stated in section 3.9.3.1.1, "Reactor Coolant System" on page 3-153 of this report. Lateral support is provided for the steam generator at the upper tube sheet level by means of a structural tie to the secondary shield wall.

The pressurizer is supported by 8 support pads spaced symmetrically around the circumference of the vessel. The pads are designed in accordance with Section III and criteria stated in section 3.9.3.1.1, "Reactor Coolant System" on page 3-153 of this report.

The reactor coolant piping is self-supporting with respect to dead weight, seismic, and thermal loading. The reactor coolant pumps are partially supported by hanger rods which are designed to support the dead weight of the pump motor, with the remainder of the dead weight of the pump being supported by the piping. To reduce seismic deflection, the pumps are supported laterally at the motor by means of hydraulic suppressors connected to the secondary shield wall.

#### *3.9.3.4.1.2 Method of Analysis*

##### *3.9.3.4.1.2.1 Calculation of Foundation Loads for Reactor Vessel and Steam Generator*

The steam generator and reactor vessel supports were designed for reaction loads from dead weight of the vessels, restrained thermal expansion of the reactor coolant piping, and seismic excitation.

Preliminary results were calculated as a part of the reactor coolant piping analysis. These seismic results indicated that the vessels act essentially independent of the piping. To examine each component in more detail, the steam generator and reactor vessel foundation loads were recalculated with each treated as an isolated component, independent of the reactor coolant piping.

The dynamic characteristics of the reactor vessel and steam generator and the loads on their supports were determined using a detailed lumped-mass dynamic model of each component. The models included one lateral degree of freedom per mass, with the discrete mass points connected by flexible beam segments. An additional rotational spring was included at the base of the models to represent the flexibility of the anchor bolts and concrete foundation beneath the vessels. In addition, the steam generator model assumed no connection to the secondary shield wall.

The seismic forces on the supports were calculated using the response spectra approach. Sufficient modes were included in the model to simulate the behavior of the actual structure. All the modes included in the model were calculated, and were combined as the square root of the sum of the squares of all the individual modal contributions.

Loads on the supports due to thermal expansion of the piping and dead weight which were calculated as a part of the piping analysis were incorporated into the support design also.

The results of foundation loads due to dead weight, thermal expansion, and seismic loadings for all major components are shown in Table 3-63.

*3.9.3.4.1.2.2 Calculation of Foundation Loads for Pressurizer*

The first mode natural frequency of the pressurizer is greater than 30 cps and the vessel can be considered rigid. For rigid systems the maximum acceleration at the point of support can be considered to act at the center of gravity of the vessel and a static approach used.

Static loads equal to 0.2 x Full Wet Weight were applied at the center of gravity of the vessel in both vertical and horizontal directions. These loads were assumed to act simultaneously.

The equivalent horizontal shear and overturning moment at the vessel support level was found and used in the design and analysis of the support. The vessel wall was analyzed for local loading, from the attached support, by means of a method developed by P. P. Bijlaard. The resulting stress intensities were compared to stress allowables specified in ASME Section III and criteria stated in Section 3.9.3.1.1, "Reactor Coolant System" on page 3-153.

The static analysis method, using 0.2g acceleration loads, is conservative. The 0.2g acceleration is greater than the accelerations given in the acceleration spectra for the various elevations of equipment supports. For the pressurizer supported at Elev. 821 ft. the spectra results give an acceleration of 0.06g for the design basis earthquake.

Loads due to thermal expansion were calculated as part of the piping analysis and included in the support design.

*3.9.3.4.1.2.3 Analysis of Reactor Vessel and Steam Generator Supports*

The reactor vessel and steam generator support skirts and support skirt flanges are designed and analyzed using procedures described in Chapter 10, Section 1, of Reference 15 on page 3-175. That procedure is used to determine the tensile stress in the anchor bolts, the bearing stress on the support skirt flange and the location of the neutral axis of bending on the bolt-flange mechanism.

The skirt-flange mechanism was statically analyzed for the applied forces and moments due to seismic loading on the vessel, considering a free-standing vessel (see Table 3-63).

The support skirt flange and foundation is assumed to be rigid. In regard to the reactor vessel, effects of anchor bolt pretension on the bending moment capacity of the support skirt were evaluated. With no anchor bolt pretension, the location of the neutral axis is found by trial and error methods so that the difference between the first moment of the bolt tension area and first moment of the flange compression area about the neutral axis is less than 5 percent of the smaller value. Increasing values of applied anchor bolt pretension result in less shift of the neutral axis.

The anchor pretension load necessary to prevent any separation of the support skirt flange from the foundation is the required load which will result in no shift of the neutral axis. In that case the neutral axis is located on the centerline of the vessel flange.

For a typical seismic load condition on the vessel, the support skirt flange was analyzed for flange bearing stress, anchor bolt loads, and location of neutral axis. Once the neutral axis was located, giving consideration to anchor bolt pretension loads, the flange, skirt, gusset mechanism was analyzed for applied tensile, compressive, and shear loads resulting from bending using methods from engineering mechanics.

The allowable stress criterion specified in Section 3.9.3.1.1, "Reactor Coolant System" on page 3-153 of this report was used where applicable.

#### 3.9.3.4.2 Supports for Other Duke Class A, B, C and F Piping

##### *3.9.3.4.2.1 Allowable Stress Criteria*

###### *3.9.3.4.2.1.1 Structural Members*

Allowable stresses are as follows:

1. Tension

   Normal $Ft = 0.6\ Fy$

   Upset $Ft = (1.33)\ (.6\ Fy) = .8\ Fy$

   Faulted $Ft = Fy$

2. Bending in Structural Members

   (Laterally Supported to preclude local compressive instability)**

   Normal $Fb = 0.6\ Fy$

   Upset $Fb = (1.33)\ (0.6\ Fy) = .8\ Fy$

   Faulted $Fb = Fy$

3. Bending in Base Plates

   Normal $Fb = 0.75\ Fy$

   Upset $Fb = (1.33)\ (0.75\ Fy) = Fy$

   Faulted $Fb = Fy$

4. Shear

   Normal $Fv = 0.4\ Fy$

   Upset $Fv = (1.33)\ (0.4\ Fy) = 0.533\ Fy$

   Faulted $Fv = (1.5)\ (0.4\ Fy) = 0.6\ Fy$

5. Compression

   Normal $Fc = Fa^*$

   Upset $Fc = 1.33\ Fa$

   Faulted $Fc = \left[1.67 - \left(\frac{0.33(Kl/r)}{Cc^*}\right)\right]Fa,$

For KI/r < Cc

Fc = 1.33 Fa, for KI/r > Cc

*See Section 1.5.1.3 AISC 6th Edition for definition.

**See Section 1.5.1.4 AISC 6th Edition for allowable extreme fiber compressive stress in bending for rolled shapes, built-up members, channels, etc., when full lateral support is not provided.

*3.9.3.4.2.1.2 Allowable Stresses for ASTM A36 Materials*

| Allowable Stress (Ksi) | | | |
|---|---|---|---|
| **Loading** | **Normal** | **Upset** | **Faulted** |
| Tension | 21.6 | 28.8 | 36.0 |
| Bending | | | |
| a. Members | 21.6 | 28.8 | 36.0 |
| b. Base Plates | 27.0 | 36.0 | 36.0 |
| Shear | 14.4 | 19.2 | 21.6 |

**Notes:**

1. Stress allowables for normal and upset load conditions are derived from the AISC Manual of Steel Construction, 6th Edition. Stress allowables for faulted load conditions are established by factoring AISC, 6th Edition allowables.
2. For stress conditions not covered in Section 3.9.3.4.2.1.1, "Structural Members" on page 3-166, the AISC allowables are utilized for normal loadings, and 133 percent of the AISC allowables shall be utilized for upset loadings.
3. No increase in stress allowables is permitted for material strain hardening and/or strain rate effects due to dynamic loadings.

The specified minimum yield stress (Fy) for ASTM A501 and ASTM A500, Grade B structural tubing is as defined in the 7th Edition of the AISC Manual of Steel Construction.

*3.9.3.4.2.1.3 Weld Stresses*

1. Tension, bending, compression, and shear on effective throat of complete penetration groove welds and normal compression and shear on effective throat of partial-penetration groove welds permissible allowables are the same as the base material.
2. All other shear:

| | |
|---|---|
| Normal | Fv = 18.0 ksi |
| Upset | Fv = (1.33) (18.0) = 24.0 ksi |
| Faulted | Fv = (1.5) (18.0) = 27.0 ksi |

**Note:**

1. The above shear allowables are based upon the use of E60XX electrodes or stronger. The normal allowable is taken from AWS Standard AWS D1.0-69 and the 7th Edition of the AISC Manual. The normal allowable is higher (18 ksi vs. 13.6 ksi) than the allowable given in the 6th Edition of the AISC Manual.

The normal weld allowables were increased 33 percent by AWS and AISC in 1969 to eliminate over conservatism required by the AISC 6th Edition. This over-conservatism was present due to design criteria for welds which were inconsistent with the remainder of the AISC Code and lack of test data. Changes in procedures or materials were not a consideration in this change. It is therefore considered appropriate to utilize the AISC 7th Edition weld allowables on Oconee Nuclear Station.

#### *3.9.3.4.2.1.4 Standard Components*

All standard components shall be limited by the recommended allowable load specified in either manufacturer's Hanger Standards, manufacturer's Load Capacity Data Sheets (LCD's) or Qualified Product Load Ratings.

#### *3.9.3.4.2.1.5 Combined Stresses In Structural Members*

Members subjected to both axial compression and uniaxial bending stresses shall comply with Section 1.6 of the AISC Manual, 6th Edition.

Members subjected to both axial tension and bending stresses shall comply with Section 1.6 of the AISC Manual, 6th Edition.

Members subjected to biaxial bending coincident with axial tension or compression shall be proportioned to satisfy the requirements of Section 1.6 of the AISC Manual, 6th Edition, in accordance with the guidance provided by Section 1.6 of the Commentary in the AISC Manual, 6th Edition.

#### *3.9.3.4.2.1.6 Bolts and Threaded Parts*

All allowable stresses are *based on unthreaded body area* of bolts and threaded parts.

A307 BOLTS

1. Tension

| | |
|---|---|
| Normal | = 14 ksi |
| Upset | = 1.33 (14) = 18.6 ksi |
| Faulted | = 1.67 (14) = 23.4 ksi |

2. Shear

| | |
|---|---|
| Normal | = 10 ksi |
| Upset | = 1.33 (10) = 13.3 ksi |
| Faulted | = 1.5 (10) = 15 ksi |

Threaded Parts of Other Steels

1. Tension

| | |
|---|---|
| Normal | = .4 Fy |

Upset = 1.33 (.4 Fy) = .53 Fy

Faulted = 1.67 (.4 Fy) = .67 Fy

2. Shear

Normal = .3 Fy

Upset = 1.33 (.3 Fy) = .4 Fy

Faulted = 1.5 (.3 Fy) = .45 Fy

Stress allowables for other bolts are given in Section 1.5.2, Table 1.5.2.1 of the AISC Manual, 6th Edition. The following factors shall be applied to normal allowable stresses:

| | Tension | Shear |
|---|---|---|
| Normal | 1.0 | 1.0 |
| Upset | 1.33 | 1.33 |
| Faulted | 1.67 | 1.5 |

For combined shear and tension, refer to Section 1.6.3 of the AISC Manual, 6th Edition.

*3.9.3.4.2.2 Snubbers*

4 Piping systems designed to resist seismic forces have been restrained by steel supports capable of
4 withstanding these seismic forces. Snubbers are used at locations where restraints are necessary based on
4 piping stress analysis, but thermal movement of the pipe must not be constrained. Performance selection
4 is based on manufacturer's load capacity data and the requirement that the allowable travel of the snubber
4 exceed the calculated pipe thermal travel. The midpoint of the pipe thermal travel is set at approximately
4 the mid-point of the snubber travel range with hot and cold settings established accordingly. In systems
4 where it was necessary to use hydraulic or mechanical snubbers to resist seismic forces, the mechanical
4 action associated with the snubber makes it possible to consider them as restraints against pipe whipping
4 (see Section 3.6, "Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping"
4 on page 3-59).

4 Duke Power Company specifies a margin of zero between design requirements and purchase requirements
4 because design loads are determined by detailed computerized piping analysis. In most cases, a margin
4 does exist between the design load and the maximum allowable design load of the suppresser supplied
4 since:

4 1. Suppressers are manufactured for a relatively small number of load ranges; therefore, each suppresser
4 size covers many possible loadings.

4 2. Suppressers supplied for the Oconee Nuclear Station clearly envelope the design load required for the
4 particular restraint application.

4 Prior to at their installation at the Oconee Nuclear Station, all snubbers are functional tested on a
4 specifically designed test stand to insure they meet design criteria. Hydraulic snubbers are tested for
4 activation velocity and bleed rate. Mechanical snubbers are tested for drag and acceleration rate.

4 Visual inspections are performed on all hydraulic and mechanical snubbers on regular intervals to identify
4 those that are damaged, degraded, or inoperable as caused by physical means, leakage, corrosion, or
4 environmental exposure. The inspection interval is based upon the previous inspection interval and the
4 number of unacceptable snubbers found during that interval. The interval between inspections will not be
4 greater than 48 months.

4 To verify that a snubber can operate within specific performance limits, Oconee Nuclear Station performs
4 functional testing that involves removing the snubber and testing it on a specifically-designed test stand.
4 As on installation, hydraulic snubbers are tested for activation velocity and bleed rate, and mechanical
4 snubbers are tested for drag and acceleration rate. Ten percent of the total snubber population are tested
4 during each refueling outage. Oconee Nuclear Station separates the snubber population into hydraulic
4 and mechanical and pulls a minimum 10% sample from each group. For each snubber that does not
4 meet the functional test acceptance criteria, an additional minimum 10% sample of that snubber type will
4 be tested until none are found inoperable or all the snubbers of that type have been functionally tested.
4 Functional testing in this manner provides a 95% confidence level that 90% to 100% of the snubbers
4 operate within the specified acceptance limits.

4 *3.9.3.4.2.2.1 Hydraulic Snubbers*

4 When a seismic event acts on a system that uses a hydraulic snubber to resist the seismic forces, it causes
4 the piston rod of the snubber to move faster than the activation threshold velocity of that snubber. When
4 this happens, a differential pressure is generated on the valve that allows fluid to flow from one end of the
4 snubber cylinder to the other and the valve closes. With this by-pass valve closed, the snubber acts as a
4 near rigid structural member, thus limiting any further movement of the pipe at the point of attachment.
4 A by-pass or bleed orifice between the two ends of the cylinder prevents the snubber from exceeding its'
4 rated capacity and allows a gradual pressure drop even under sustaining load against the closed by-pass
4 valve. A hydraulic snubber resists seismic forces by limiting velocity.

4 The design data for the hydraulic shock and sway suppressers used on Class I piping systems at the
4 Oconee Nuclear Station are summarized in the charts below.

4 Grinnell Hydraulic Snubbers

4

| | | | Activation Threshold | | |
|---|---|---|---|---|---|
| Size Bore | (In.) Stroke | Acceleration | Velocity In./Min. | Normal* Load(lbs) | One Time Load(lbs)* |
| 1½ | 5 | Not | 8 | 3,000 | 4,000 |
| 1 1/2 | 10 | Applicable | 8 | 1,100 | 1,500 |
| 2 1/2 | 5 and 10 | | 8 | 12,500 | 25,700 |
| 3 1/4 | 5 and 10 | Insensitive | 8 | 21,000 | 43,500 |
| 4 | 5 and 10 | To Acceleration | 8 | 32,000 | 66,000 |
| 5 | 5 and 10 | | 8 | 50,000 | 103,000 |
| 6 | 5 and 10 | | 5 | 72,000 | 148,000 |
| 8 | 5 | | 3 | 130,000 | |

4 *Actual Allowable load may be less than specified depending on length of overall assembly.

4 Pre-installation tests of the hydraulic suppressors were performed by the vendor. The testing of the
4 snubber valves varies according to valve type. All shock suppressor assemblies were subjected to the same
4 testing. A description of these tests is provided below:

4 **Shock Suppressor Assembly**

4 After assembly, each completed shock suppressor was mounted in a fill and test facility. During the fill
4 and purge operations, this equipment verified that the suppressor rod was free to move through its full
4 stroke. After filling, the equipment verified that the suppressor would snub and that it was free of air (i.e.,
4 movement to snub was not excessive).

4 Snubber Valve - Test 1

4 a. All parts were visually inspected for cleanliness and for complete deburring.

4 b. After assembly, the valve was placed in flow test facility. Locking velocity was set to the
4 required value; then, after locking, bleed rate under load was set to the required value.

4 c. The test uses the fluid with which the snubbers were filled. Still wet with this fluid, the valves
4 were capped for stock.

4 Snubber Valve - Test 2

4 a. All parts were inspected for cleanliness and for complete deburring.

4 b. Valve was assembled without springs and subject to low flow, high pressure test. This test
4 verified that the pool would move freely to the closed position in each direction and that
4 bypass flow in the closed position was within acceptance limits.

4 c. Following this test, the springs were installed and the valves were prepared for stock.

4 LISEGA HYDRAULIC SNUBBERS

| SNUBBER TYPE (1) | STROKE (2) | NORMAL LOAD (2) | REACTION VELOCITY | BYPASS VELOCITY |
|---|---|---|---|---|
| 30185x | 4 inches | 675 lbs. | 4.7 - 14.2 ipm | .47 - 4.7 ipm |
| 30385x | 4 inches | 1800 lbs. | 4.7 - 14.2 ipm | .47 - 4.7 ipm |
| 30395x | 8 inches | 1800 lbs. | 4.7 - 14.2 ipm | .47 - 4.7 ipm |
| 30425x | 5-7/8 inches | 4000 lbs. | 4.7 - 14.2 ipm | .47 - 4.7 ipm |
| 30435x | 11-3/4 inches | 4000 lbs | 4.7 - 14.2 ipm | .47 - 4.7 ipm |
| 30525x | 5-7/8 inches | 10350 lbs | 4.7 - 14.2 ipm | .47 - 4.7 ipm |
| 30535x | 11-3/4 inches | 10350 lbs. | 4.7 - 14.2 ipm | .47 - 4.7 ipm |
| 30625x | 5-7/8 inches | 22450 lbs. | 4.7 - 14.2 ipm | .47 - 4.7 ipm |
| 30635x | 11-3/4 inches | 22450 lbs. | 4.7 - 14.2 ipm | .47 - 4.7 ipm |
| 30725x | 5-7/8 inches | 44900 lbs. | 4.7 - 14.2 ipm | .47 - 4.7 ipm |
| 30735x | 11-3/4 inches | 44900 lbs. | 4.7.- 14.2 ipm | .47 - 4.7 ipm |
| 30825x | 5-7/8 inches | 78600 lbs. | 4.7 - 14.2 ipm | .47 - 4.7 ipm |
| 30835x | 11-3/4 inches | 78600 lbs. | 4.7 - 14.2 ipm | .47 - 4.7 ipm |
| 30925x | 5-7/8 inches | 123500 lbs. | 4.7 - 14.2 ipm | .47 - 4.7 ipm |

| SNUBBER TYPE (1) | STROKE (2) | NORMAL LOAD (2) | REACTION VELOCITY | BYPASS VELOCITY |
|---|---|---|---|---|
| 30935x | 11-3/4 inches | 123500 lbs. | 4.7 - 14.2 ipm | .47 - 4.7 ipm |

Notes:

(1) These are the model numbers for the stock Lisega snubbers. The model numbers used at the Oconee Nuclear Station are usually followed by a suffix showing that it is a replacement for a particular brand of snubber. (i.e. 303856RF1 is a type 3038 snubber, nuclear specification (5), design year 1986, that replaces a PSA-1 mechanical snubber with a flanged end.)

(2) Snubber stroke can be modified by the manufacturer, at the request of the purchaser, and this may effect the normal load of the snubber.

(3) The 'x' on the end of the snubber type is an abbreviation of the design year (i.e. 6 = 1986)

Lisega has been audited by ASME to certify them to supply Component Standard Supports manufactured without welding in accordance with ASME Section III, Subsection NF.

Lisega has performed extensive testing on the Viton seals in their hydraulic snubbers. The objective of this aging test was to demonstrate the suitability of Viton as a seal material for use in an environment typical of a Nuclear Power Station. The test exposed the seals to a combination of heat, radiation, and extensive dynamic cycling. The tests performed are described below.

**A.** The test seals were subjected to ionising radiation of $1 \times 10^7$ rads installed in the snubbers in the irradiated condition.

**B.** The following functional tests were performed:

- **B.1.** Measurement of drag force, tension & compression
- **B.2.** Response speed, tension & compression
- **B.3.** Dynamic functional behavior

**C.** The specimens were exposed to a continuous temperature of 175°C for 480 hours in a hot air circulation furnace. This exposure is equivalent to approximately 8 years aging.

**D.** Functional tests were performed per Para. B.

**E.** The following continuous dynamic loadings were imposed on all specimens:

- **E.1.** 3,600 cycles at 100% of rated load.
- **E.2.** 43,200 cycles at 50% of rated load.
- **E.3.** 273,600 cycles at 10% of rated load.

**F.** Functional tests were performed per Para. B. (less tests per B.3.)

**G.** Again the specimens were exposed to a continuous temperature of 175 C for 480 hours in a hot air circulating furnace. This equals an additional 8 years of ageing (total 16 years).

**H.** A fourth functional test sequence was performed per Para. B.

**I.** Continuous cyclic loading was again performed after the second ageing as follows:

- **I.1.** 3,600 cycles at 100% of rated load.

4 I.2. 43,200 cycles at 50% of rated load.

4 I.3. 273,600 cycles at 10% of rated load.

4 J. A final functional test sequence was performed per Para. B.

4 At the end of the test sequence, all specimens were disassembled, and the seals were visually inspected for
4 signs of wear.

4 SUMMARY & CONCLUSION

4 Post test visual examination of the seals revealed no signs of wear, although some weeping of hydraulic
4 fluid was noted. There were no performance deviations or anomolies due to seal failure noted throughout
4 the entire test sequence.

4 The successful completion of this extensive mechanical and environmental ageing process has proven that
4 seals manufactured from Viton materials, in conjunction with proper surface finishes and working
4 tolerances as designed in the LISEGA Hydraulic Snubber, will :

4 1. Successfully withstand the environmental conditions typical of those found in a Nuclear Power
4 Station.

4 2. Provide excellent resistance to wear due to excessive dynamic loadings.

4 3. Exhibit an effective service life exceeding 16 years under normal operating conditions.

4 *3.9.3.4.2.2.2 Mechanical Snubbers*

4 When seismic forces in a system are resisted using mechanical snubbers, the mechanical snubber translates
4 linear movement between the system and the support structure into rotational motion within the snubber.
4 The snubber's telescoping cylinder is attached to the fixed support cylinder by a screw and nut assembly.
4 Relative motion between the two causes the screw shaft to turn which causes an inertia mass to turn.
4 The torque required to start the rotary motion of the snubber internals limits the rate of acceleration of
4 the attached pipe. A mechanical snubber resists forces by limiting acceleration.

4 The design data for mechanical snubbers used on Class I piping systems at the Oconee Nuclear Station is
4 summarized in the chart below.

4 Pacific Scientific Mechanical Snubbers

4 Allowable Loads @ 300°F

4
4

| | SIZE | STROKE (IN>) | ACCELERA-TION LIMIT | NORMAL* LOAD (LBS) | ONE TIME* LOAD(LBS.) |
|---|---|---|---|---|---|
| | 1/4 | 4 | 0.64ft/sec² | 350 | 590 |
| | 1/2 | 2.5 | 0.64ft/sec² | 650 | 1,040 |
| STD. | 1 | 4 | 0.64ft/sec² | 1,500 | 2,300 |
| STROKE | 3 | 5 | 0.64ft/sec² | 6,000 | 11,520 |
| | 10 | 6 | 0.64ft/sec² | 15,000 | 23,600 |
| | 35 | 6 | 0.64ft/sec² | 50,000 | 91,000 |
| | 100 | 6 | 0.64ft/sec² | 120,000 | 180,000 |

| | SIZE | STROKE (IN>) | ACCELERA-TION LIMIT | NORMAL* LOAD (LBS) | ONE TIME* LOAD(LBS.) |
|---|---|---|---|---|---|
| EXT. STROKE | 1 | 8 | 0.64ft/sec$^2$ | 1,487 | 2,200 |
| | 3 | 10 | 0.64ft/sec$^2$ | 6,000 | 11,520 |
| | 10 | 12 | 0.64ft/sec$^2$ | 14,400 | 22,032 |

*Actual allowable load may be less than specified depending on angular displacement of load path with end bracket.

## 3.9.4 CONTROL ROD DRIVE SYSTEMS

The Control Rod Drive Mechanism is described in Section 4.5.3, "Control Rod Drives" on page 4-59.

## 3.9.5 REACTOR PRESSURE VESSEL INTERNALS

Reactor pressure vessel internals are described in Section 4.5, "Reactor Materials" on page 4-49.

## 3.9.6 REFERENCES


1. Porse, L., "Reactor Vessel Design Considering Radiation Effects," *ASME Paper No. 63-WA-100.*
2. Pellini, W. S. and Puzak, P. P., "Fracture Analysis Diagram Procedures for the Fracture-Safe Engineering Design of Steel Structures," *Welding Research Council Bulletin 88, May 1963.*
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12. Chen, L. H., "Piping Flexibility Analysis by Stiffness Matrix," ASME *Journal of Applied Mechanics*, December 1959.
13. "Design of Piping System," The M. W. Kellogg Company, Second Edition, 1956.
14. "Nuclear Reactors and Earthquakes," *TID-7024*, Chapters 5 and Appendix E, August 1963.
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16. Topical Report: "Reactor Internals Stress and Deflection Due to Loss-of Coolant Accident and Maximum Hypothetical Earthquake." Babcock and Wilcox *BAW-10008, Part 1, Rev. 1.* June, 1970.
17. DeMars, R. V. and Steinke, R. R. "Fuel Assembly Stress and Deflection Analysis for Loss-of-Coolant Accident and Seismic Excitation (Non-proprietary Version of BAW-10008, Part 2, Rev. 1)". Babcock and Wilcox. *BAW-10035*, January, 1972.
18. Thoren, D. E. and Harris, R. J., "Prototype Vibration Measurement Program for Reactor Internals, 177-Fuel Assembly Plant." Babcock and Wilcox. *BAW-10038*, September, 1972.

# 3.10 SEISMIC QUALIFICATION OF INSTRUMENTATION AND ELECTRICAL EQUIPMENT

This section describes the seismic considerations applied to instrumentation and electrical equipment during the original design of the Oconee Nuclear Station as well as in modifications to the station after issuance of the operating license.

## 3.10.1 SEISMIC QUALIFICATION CRITERIA

The seismic design basis for instrumentation and electrical equipment is that the electrical devices considered essential in performing Reactor Protection and Engineered Safeguards functions and in providing emergency power shall be designed to assure that they will not lose their capability to perform intended safety functions during and following the design basis event (MHE). This basic criteria has remained unchanged since the issuance of the operating license; however, the seismic qualification techniques and documentation requirements for various plant modifications have in many instances followed the advances in the state of the art.

The specific equipment included in the scope identified above including the associated seismic qualification documentation reference is provided in Table 3-68.

## 3.10.2 METHODS AND PROCEDURES FOR QUALIFYING INSTRUMENTATION AND ELECTRICAL EQUIPMENT

In order to meet the seismic design objectives defined in Section 3.10.1, Seismic Qualification Criteria, the following seismic evaluation methods were employed consistent with the applicable licensing commitment.

Testing

Devices may be qualified by either shaker or impact tests. A certification of the test results or a copy of the test results are required. Additionally, a manufacturer's certification that a certain type of equipment would withstand the seismic conditions is acceptable based on previous testing/experience with similar equipment.

Analysis

Devices may also be qualified by analytical methods. For example, one evaluation method involves calculating/determining the natural frequency of the device, entering the appropriate response spectra damping curves, and determining the corresponding amplification factor. The device is then evaluated using this "G" loading value. Alternatively, the devices may be evaluated without calculating/determining its natural frequency by using the peak amplification factor from the appropriate response spectra damping curve to determine the "G" loading.

# 3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

## 3.11.1 EQUIPMENT IDENTIFICATION AND ENVIRONMENTAL CONDITIONS

Duke has a program in place for environmental qualification of safety-related electrical equipment inclusive of equipment required to achieve a safe shutdown. Environmental effects resulting from the postulated design basis accidents documented in Chapter 15, "Accident Analyses" on page 15-1 have been considered in the qualification of electrical equipment which is covered by this program. This program has been reviewed and approved by NRC (Reference 2 on page 3-181).

### 3.11.1.1 Equipment Identification

Safety-related electrical equipment that is required to perform a safety function(s) in a postulated harsh environment is identified in Duke Power Company's response to NRC IE Bulletin 79-01B (Reference 1 on page 3-181).

Safety-related mechanical equipment including design information is identified in Section 3.2.2, "System Quality Group Classification" on page 3-38.

### 3.11.1.2 Environmental Conditions

The postulated harsh environmental conditions resulting from a LOCA or HELB inside the Reactor Building and a HELB outside the Reactor Building are identified and discussed in Duke Power Company's response to NRC IE Bulletin 79-01B (Reference 1 on page 3-181).

2 The environmental parameters that compose the overall worst-case containment environment are as
2 follows:

2 Containment Temperature: Time history as shown in Figure 15-71 for the Design Basis Accident
2 (DBA), a 5.0 $ft^2$ hot leg break.

2 Containment Pressure: Time history as shown in Figure 15-56 for the largest (14.1 $ft^2$) hot leg break.

2 Relative Humidity: 100%

2 Radiation: Total integrated radiation dose for the equipment location includes the 40 year normal
2 operating dose plus the appropriate accident dose based on equipment operability requirements. The
2 bases for determining the containment radiation environment are discussed in Chapter 12, "Radiation
2 Protection" on page 12-1.

2 Chemical Spray: Boric acid spray resulting from mixing in the containment sump with borated water
2 from the borated water storage tank. Refer to Section 6.2.2, "Containment Heat Removal Systems"
2 on page 6-22 for additional information on chemical spray.

## 3.11.2 QUALIFICATION TEST AND ANALYSIS

Safety-related equipment identified in Section 3.11.1.1, "Equipment Identification" is qualified by test and/or analysis. The method of qualification for this Class 1E equipment is identified in Duke Power Company's response to NRC IE Bulletin 79-01B (Reference 1 on page 3-181).

## 3.11.3 QUALIFICATION TEST RESULTS

The results of the qualification tests and/or analyses for the electrical equipment identified in Section 3.11.1.1, "Equipment Identification" are presented in the qualification documentation references identified in Duke Power Company's response to NRC IE Bulletin 79-01B (Reference 1 on page 3-181). Additionally, a summary of the qualification results is also presented in the bulletin response.

## 3.11.4 LOSS OF VENTILATION

The control area air conditioning and ventilation systems (Section 9.4.1, "Control Room Ventilation" on page 9-53) are conservatively designed to provide a suitable environment for the control and electrical equipment. In addition, redundant air conditioning and ventilation equipment is provided, as summarized below, to assure that no single failure of an active component within these systems will prevent proper control area environmental control.

1. Two 100 percent capacity supply fans with filter banks and chilled water coils.
2. Two 100 percent capacity chillers.
3. Two 50 percent capacity outside air booster fans.

2

2 The Station Blackout scenario involves a four hour loss of ventilation to the control area. Assuming the
2 non-essential loads are manually stripped within the first 30 minutes of the event, and the initial ambient
2 temperatures outlined in the Selected Licensee Commitments Manual, Section 16.8.1 are not exceeded,
2 analysis has shown that the following temperatures would not be exceeded:

| | |
|---|---|
| 2 Control rooms | 120°F |
| 2 Cable rooms | 137°F |
| 2 Electrical equipment rooms | 115°F |
| 2 I&C Battery rooms | 107°F |

2 The above temperatures are within the specifications of the control room habitability requirements of
2 10CFR 50.63, and within the operating temperature limits of the equipment required to operate during
2 the scenario.

## 3.11.5 ESTIMATED CHEMICAL AND RADIATION ENVIRONMENT

The estimated chemical and radiation environments at Oconee are discussed in Duke Power Company's response to NRC IE Bulletin 79.01B (Reference 1 on page 3-181). Additional information regarding chemical and radiation conditions is presented in Section 6.5, "Fission Product Removal and Control Systems" on page 6-55 and in Chapter 12, "Radiation Protection" on page 12-1, respectively.

## 3.11.6 REFERENCES

1. Oconee Nuclear Station Response to IE Bulletin 79-OlB, as revised, including Response to NRC Equipment Qualification Safety Evaluation Report.
2. Letter from J. F. Stolz (NRC) to H. B. Tucker (Duke) dated March 20, 1985.

   Subject: Safety Evaluation Report on Environmental Qualification of Electrical Equipment Important to Safety.

**THIS IS THE LAST PAGE OF THE CHAPTER 3 TEXT PORTION.**

**Table 3-23. Auxiliary Building Loads and Conditions**

| AREA | CONDITIONS | |
|---|---|---|
| Control Room | A,B,C,D,E | Blow out panels designed to relieve 3 psi differential pressure |
| Cable Room | A,B,C,D,E | |
| Electrical Equipment Room | A,B,C,D,E | |
| Spent Fuel Pool | A,B,C,D,E | Blow out panels designed to relieve 3 psi differential pressure |
| Spent Fuel Storage Racks | A,D | Inherently resistant to wind loads |
| Spent Fuel Handling Crane | A,D,E | Inherently resistant to wind loads. Hold down device provided |
| Penetration Room Frames | A,B,D | Physical separation provided for missile protection |
| Cable Shaft | A,B,C,D,E | |
| Elevator Steel Shaft | A,D | |
| Main Steam Pipe Supports | A,B,D | |
| Hot Machine Shop | A,D | |
| Balance of Auxiliary Building | A,B,D | Frame designed for B, but not external walls above grade. Areas below grade are inherently protected against missiles in C and E. |

A = All normal dead, equipment, live, and wind loads due to 95 mph wind.
B = Normal dead and equipment loads plus tornado wind load due to 300 mph wind.
C = Tornado missiles of (1) 8 in. diameter x 12 ft. long piece of wood, 200 pounds, 250 mph, and (2) 2,000 pound automobile, 100 mph, 20 sq. ft. impact area, for 25 ft. above grade.
D = Normal dead and equipment loads plus maximum hypothetical earthquake loads.
E = Turbine-generator missile, 5,944 pounds, 502 fps, kinetic energy of 23.25 x $10^6$ ft.-lbs., side on impact area of 8.368 sq. ft. and end on impact area of 3.657 sq. ft.

**Table 3-24. Fuel Assembly Materials**

| Component | Material |
|---|---|
| Guide tubes | Zircaloy-4 |
| Spacer sleeves | Zircaloy-4 |
| Fuel rod cladding | Zircaloy-4 |
| Instrument tube | Zircaloy-4 |
| Spacer grids | Inconel-718 |
| Holddown spring | Inconel X-750 |
| Guide tube nuts | Type 304 stainless steel |
| End grid assembly screws | Type 304 stainless steel |
| Holddown spider | Stainless steel, grade CF-3M |
| End Fittings | Stainless steel, grade CF-3M |

**Table 3-25. Fuel Assembly Loads and Permanent Deflection Limits and Analysis Results**

| | Calculated Deflection or Load | Allowable Deflection or Load | Margin % |
|---|---|---|---|
| Horizontal contact analysis spacer grid permanent deformation, in. | | | |
| OBE | 0.000 | 0.000 | 80(a) |
| SSE | 0.010 | 0.150 | 400(a) |
| SSE + LOCA | 0.012 | 0.150 | 300(a) |
| Vertical contact analysis, lb | | | |
| Guide tube buckling | 5,400 | 5,588 | 4 |
| Upper end spacer grid welds | 150 | 170 | 13 |
| End spacer grid assembly - buckling | 11,000 | 12,800 | 16 |
| End spacer grid bolts - shear | 690 | 800 | 16 |

**Note:**

(a)Calculated from the relation

$$\text{Margin} = \frac{\text{level to obtain limit - Duke level}}{\text{Duke level}} \text{ X } 100.$$

**Table 3-63. Foundation Loads for Major Components**

| Component | Load Description | Horizontal Force, kips | Vertical Force, kips | Overturning Moment, ft-kips | Twisting Moment ft-kips | |
|---|---|---|---|---|---|---|
| Reactor Vessel | Reactor Coolant Piping - Thermal Expansion | 40 | | -280 | 1,200 | 135 |
| | Dead Load | -- | | 2,120 | -- | -- |
| | Seismic Horizontal + Vertical | 275 | | 180 | 8,300 | 2,960 |
| Steam Generator | Reactor Coolant Piping - Thermal Expansion | 365 | | 135 | -- | 20 |
| | Steam and Feedwater Piping Thermal Expansion Load | 20 | | 20 | -- | -- |
| | Dead Load | -- | | 1,845 | -- | -- |
| | Seismic Horizontal + Vertical | 175 | | 336 | 7,920 | 735 |
| Pressurizer | Surge and Spray Line Thermal Expansion Load | 10 | | -1 | 115 | 30 |
| | Dead Load | -- | | 390 | -- | -- |
| | 0.2 g Seismic Load in Any Direction | 90 | | 470 | 655 | 30 |

2 **Table 3-68 (Page 1 of 6). Electrical Equipment Seismic Qualification**

| | Equipment Identification | Seismic Qualification Documentation Reference |
|---|---|---|
| 1. | Reactor Protective System Cabinets/Components | BAW-10003A, Rev. 4 (OM 311-0490)<br>5 BWNT 51-12353 53-01 (OM 2201-C-0929) STAR<br>5 Qualification Test Report Summary<br>5 BWNT 51-1235173-02 (OM 2201-C-0930) STAR AVIM<br>5 Qualification Test Report Summary |
| 2. | Engineered Safeguards Protective Cabinets/Components | BAW-10003A, Rev. 4 (OM 311-0490) |
| 3. | Reactor Protective System Sensors | |
| | 1. RC Pressure Transmitters (NR) | Rosemount Report 2758&127516 |
| | 2. RC Temperature RTD's | 4 &D8400102 |
| | 3. RC Flow Transmitters | 4 also B & W 58-0261-00 |
| | 4. RB Pressure Switches | 4 Rosemount Report 1177117A, and B & W 58-0082-00 |
| | 5. RCP Power Monitors | 4 B & W 58-0081-00<br>4 and Rosemount Report D8400102<br>Herron Lab Report F-7040, and B & W 58-0080-00 (OM 360-0010)<br>Duke/Exide Test Report PH58644 |
| 4. | Engineered Safeguards Protective System Sensors | 4 Rosemount Test Report D830040(OM-0267.A-0114)<br>4 ITT-Barton Test Report R3-764-9 (OM-0267.A-0041)<br>4 ASCO Test Report AQR-101083 (OM-0267.A-0050) |
| | 1. RC Pressure Transmitters (WR) | |
| | 2. RB Pressure Transmitters | |
| | 3. RB Pressure Switches | |
| 5. | 4160 VAC Station Auxiliary Switchgear (1TC, 1TD, 1TE; 2, 3) | ITE Report No. R-8793, and<br>Gould Report No. 33-53719-SS (OM 302-0617) |
| 6. | 600 VAC Load Centers (1X8, 1X9, 1X10; 2, 3) | Gould Report No. 33-53729-SSA (OM 301-0079) |
| 7. | Motor Control Centers (1XS1, 1XS2, 1XS3; 2; 3) | A. O. Smith Corp. "Simulated Seismic Testing - Bulletin 6200 Motor Control Centers" (OM 308-372) |
| 8. | DC Distribution Centers (1DCA, 1DCB; 2; 3) | A. O. Smith Corp. "Simulated Seismic Testing - Bulletin 6200 Motor Control Centers" (OM 308-372) |

2 **Table 3-68 (Page 2 of 6). Electrical Equipment Seismic Qualification**

| | Equipment Identification | Seismic Qualification Documentation Reference |
|---|---|---|
| 9. | AC Panelboards (1KVIA, 1KVIB, 1KVIC, 1KVID; 2; 3) | Wyle Lab Report 42729-1 |
| 10. | DC Panelboard (1DIA, 1DIB, 1DIC, 1DID; 2; 3) | Wyle Lab Report 42729-1 |
| 11. | Control Batteries/Racks (1CA, 1CB; 2; 3) | Exide Power Systems Div. "Seismic Test of F.T.C. Cells" and Qualification Report for Vital Control Batteries and Racks (OM 320-145) |
| 12. | Battery Chargers (1CA, 1CB, 1CS; 2; 3) | Exide Power Systems Div. "Seismic Test of UPC-130-3-60" |
| 4 12A. | Battery Chargers (1CA, 1CB, 1CS) | Wyle Lab Report 43185-2 (OM 346-0105-1) |
| 13. | Inverters (1DIA, 1DIB, 1DIC, 1DID; 2; 3) | Exide Power Systems Div. "Seismic Test of 120/9.3 F1" |
| 14. | Isolating Diode Assemblies (1ADA, 1ADB, 1ADC, 1ADD; 2; 3) | Exide Power Systems Div. "Seismic Test of Diode Monitors" |
| 15. | Oconee Main Control Boards | Wyle Lab Report WR 73-1 (OM 1393-0008) |
| 16. | Engineered Safeguards Terminal Cabinets | Wyle Lab Report WR 73-1 (OM 1393-0008) |
| 17. | Emergency Power Switching Logic Cabinets | Wyle Lab Report WR 73-1 (OM 1393-0008) |
| 18. | Oconee Unit Boards | Wyle Lab Report WR 73-1 (OM 1393-0008) |
| 19. | Oconee Vertical Boards | Wyle Lab Report WR 73-1 (OM 1393-0008) |
| 20. | Oconee Auxiliary Boards | Wyle Lab Report WR 73-1 (OM 1393-0008) |
| 21. | Keowee Emergency Start Cabinets | Wyle Lab Report WR 73-1 (OM 1393-0008) |
| 22. | Keowee Control Boards | Wyle Lab Report WR 73-1 (OM 1393-0008) |
| 23. | Keowee Miscellaneous Terminal Cabinets | Wyle Lab Report WR 73-1 (OM 1393-0008) |
| 24. | Keowee Main Turbine - Generators | ONS Emergency Power Source Seismic Evaluation Technical Position Paper |
| 25. | Keowee - Oconee Underground Power Circuit | ONS Emergency Power Source Seismic Evaluation Technical Position Paper |

2 **Table 3-68 (Page 3 of 6). Electrical Equipment Seismic Qualification**

| | Equipment Identification | Seismic Qualification Documentation Reference |
|---|---|---|
| 26. | Keowee Logic Cabinets | Wyle Lab Report WR 73-1 (OM 1393-0008) |
| 3 27. | Keowee 125 VDC Battery Chargers | OM 320-0167 |
| 28. | Keowee 125 VDC Battery/Racks | Keowee Battery Environmental Qualification Report (KM 320-16) |
| 29. | Keowee 125 VDC Distribution Centers | A. O. Smith Corp., "Simulated Seismic Testing-Bulletin 6200 Motor Control Center" (OM 308-372) |
| 3 30. | 230 KV Swyd Battery Chargers | OM 320-0167 |
| 2 31. 2 | 230 KV Swyd Control Batteries | C & D Charter Power Systems Report Number QR-27189-01 (OM-320-163) |
| 32. | 230 KV Swyd Distribution Centers | A. O. Smith Corp., "Simulated Seismic Testing-Bulletin 6200 Motor Control Centers" (OM 308-372) |
| 33. | 230 KV Swyd Panelboards | Wyle Lab Report 42729-1 |
| 34. | Oconee/Keowee Overhead Power Path Equipment | |
| | a. Keowee Main Stepup Transformer | G. E. letter to R. S. Thompson, 09-06-76, and G. E. letter to J. E. Stoner, 04-03-77 (K-301) |
| | b. Oconee Startup Transformers | G. E. letter to R. S. Thompson, 09-06-76, and G. E. letter to J. E. Stoner, 04-03-77 (OS-83-B) |
| | c. 230 KV Disconnect Switches | ITE letter & Attachment to R. S. Thompson, 08-26-76, (OS-96-C) and OSC-926 |
| | d. 230 KV Power Circuit Breakers | R. B. Priory letter to J. E. Stoner, 03-21-78, (OS-96), J. G. Hester letter to J. E. Stoner, 03-24-78, (OS-96) 5 and Wyle Lab Report 43852-1 (OM 323-0313-001) |
| | e. 230 KV Swyd. Coupling Capacitor | G. E. letter & Attachments to J. C. Papaspyrou, 08-18-76, (OS-96-D) and OSC-926 |
| | f. 230 KV Swyd. Lightning Arrestors | G. E. letter & Attachments to J. C. Papaspyrou, 08-06-76, (OS-96-E), and OSC-926 |

2 **Table 3-68 (Page 4 of 6). Electrical Equipment Seismic Qualification**

| Equipment Identification | Seismic Qualification Documentation Reference |
|---|---|
| g. 230 KV Swyd. DC Distribution Centers | A. O. Smith Corp., "Simulated Seismic Testing-Bulletin 6200 Motor Control Centers" (OM 308-372) |
| h. 230 KV Swyd. DC Panelboards | ITE letter & Attachments to J. E. Stoner, 08-16-76, (OS-89) |
| i. 230 KV Swyd. Control Batteries/Racks | C & D letter to C. J. Wylie, 09-02-76 (OS-93) |
| 3 j. 230 KV Swyd. Battery Chargers | OM 320-0167 |
| k. 230 KV Swyd. Relay House Lighting System (Anchoring Only) | J. P. Bultman letter to J. E. Stoner, 09-23-76, (OS-89) |
| l. 230 KV Swyd. Relay Panels/Equipment | Wyle Lab Report WR 76-17 (OM 393-0006) |
| 35. AC Control Rod Drive Breaker Cabinet | Oconee SMR ON-0376, B&W Report LR:74:6383-01:4 |
| 36. Standby Shutdown Facility | |
| a. Control Console | Wyle Lab Report 45676-1 (OM 1393-0013) |
| b. Miscellaneous Equipment and Interconnecting Cabinets | Wyle Lab Report 45676-1 (OM 1393-0013) |
| c. Diesel Generator | Flight Dynamics Inc. Report No. A-11-80 (OM 351-0206) |
| d. 4160 VAC Switchgear | Gould Report No. 33-53566-SS(OM-302-0615) |
| 2 e. 600 VAC Motor Control Centers | GTE Seismic Report (OM 308-0361-001, -002, and 003) |
| 2 f. 208 VAC Motor Control Centers | GTE Seismic Report (OM 308-0361-001, -002, and -003) |
| g. 120 VAC/125 VDC Panelboards | Square-D Report No. 8998-10.09-L31 (ESSEM 1X-A-6) |
| h. 600 VAC Load Centers | Gould Report No. 33-53729-SSA (OM301-80) |
| i. Inverters | Wyle Lab Report 58456-1 (OM 346-0095) |
| j. Battery Chargers | Wyle Lab Report 45278-1 (OM 320-111) and 46636-1 (OM 320-121) |
| k. Voltage Regulators | Wyle Lab Report 44741-1 (OM 352-0012) |

2 **Table 3-68 (Page 5 of 6). Electrical Equipment Seismic Qualification**

| | Equipment Identification | Seismic Qualification Documentation Reference |
|---|---|---|
| | l. Control Batteries/Racks | Wyle Lab Report 45001-1 (OM 320-109) |
| | m. Environmental Chamber and Resistor Cabinet | Wyle Lab Report 46065-1 (OM 393-0004) |
| | n. SSF Transmitters | Wyle Lab Report 45304-1 (OM 1393-0006) |
| | 37. TMI Action Item Additions | |
| | a. Reactor Building High Range Radiation Monitors | Victoreen Report No. 950-301 (OM 360-35) |
| 5 5 5 | b. Anticipatory Reactor Trip Pressure Switches and RPS Logic Equipment | B & W Report No. BWNP-20210-1 (OM-304-0001, OM-2304-0001) or Static-O-Ring Report Nos. 9058-102 (OM-267A-0124) and 9058-104 (OM-267-1284) |
| | c. Hydrogen Analyzer Control Panel (Duke Portion) | Wyle Lab Report No. 45477-1 (OM 1393-0009) |
| | d. Post-Accident Monitoring Recorders | Wyle Lab Report WR-80-48, Rev. 1 (OM 1393-0012) |
| | e. Post-Accident Monitoring Indicators | Wyle Lab Report WR-80-48, Rev. 1 (OM 1393-0012) |
| 5 5 | f. Emergency Feedwater Initiation Pressures Switches | Custom Component Switches, Inc. Report No. QTR 604-01 (CG 3008.02-01, CG 3008.02-06) or Static-O-Ring Report Nos. 9058-102 (OM-267A-0124) and 9058-104 (OM-267-1284) |
| | g. Emergency Sump Level Transmitters | Wyle Lab Report 45700-1 (OM 360-38) |
| | h. Normal and Emergency Sump Level Transmitters | FCI Test Report No. 708143 (OM 267-0762) |
| 2 | i. RB Pressure Transmitters | RMT Report No. D8400102 Rev. B (OM-267-0969) |
| 4 | j. Post Accident Sampling Solenoid Valves (Air) | Valcor Test Report QR-70900-65 (CNM-1210.04-0394) Valcor Test Report QR-52600-5940-2 (OM 360-34) |
| | k. Post Accident Sampling Solenoid Valves (Liquid) | Target Rock Report No. 2375 (OM 360-32) |
| | l. High Point Vent System Solenoid Valves | Target Rock Report No. 2375 (OM-360-32) |

2 **Table 3-68 (Page 6 of 6). Electrical Equipment Seismic Qualification**

| | Equipment Identification | Seismic Qualification Documentation Reference |
|---|---|---|
| | m. RVLIS (Reactor Vessel Level Cabinets Instrumentation System) | Westinghouse Reports WCAP-8687 EQTR-E53A (OM-311.B-24), EQDP-ESE-4 (OM-311.B-25), WCAP8687 EQTR-E04A (OM-311.B-26), WCAP8687 EQAR-E61B (OM-311.B-32), WCAP8687 EQTR-E02A (OM-311.B-35) and E04A-ADD1 (OM-311.B-40). |
| | n. OTSG Level Control System Cabinets | Wyle Lab Report No. 44662-1 (OM 393-0001). |
| 38. | Reactor Coolant Pump Monitor Cabinet | Rochester Instrument Systems SN 909335 (OM 393-0007) |

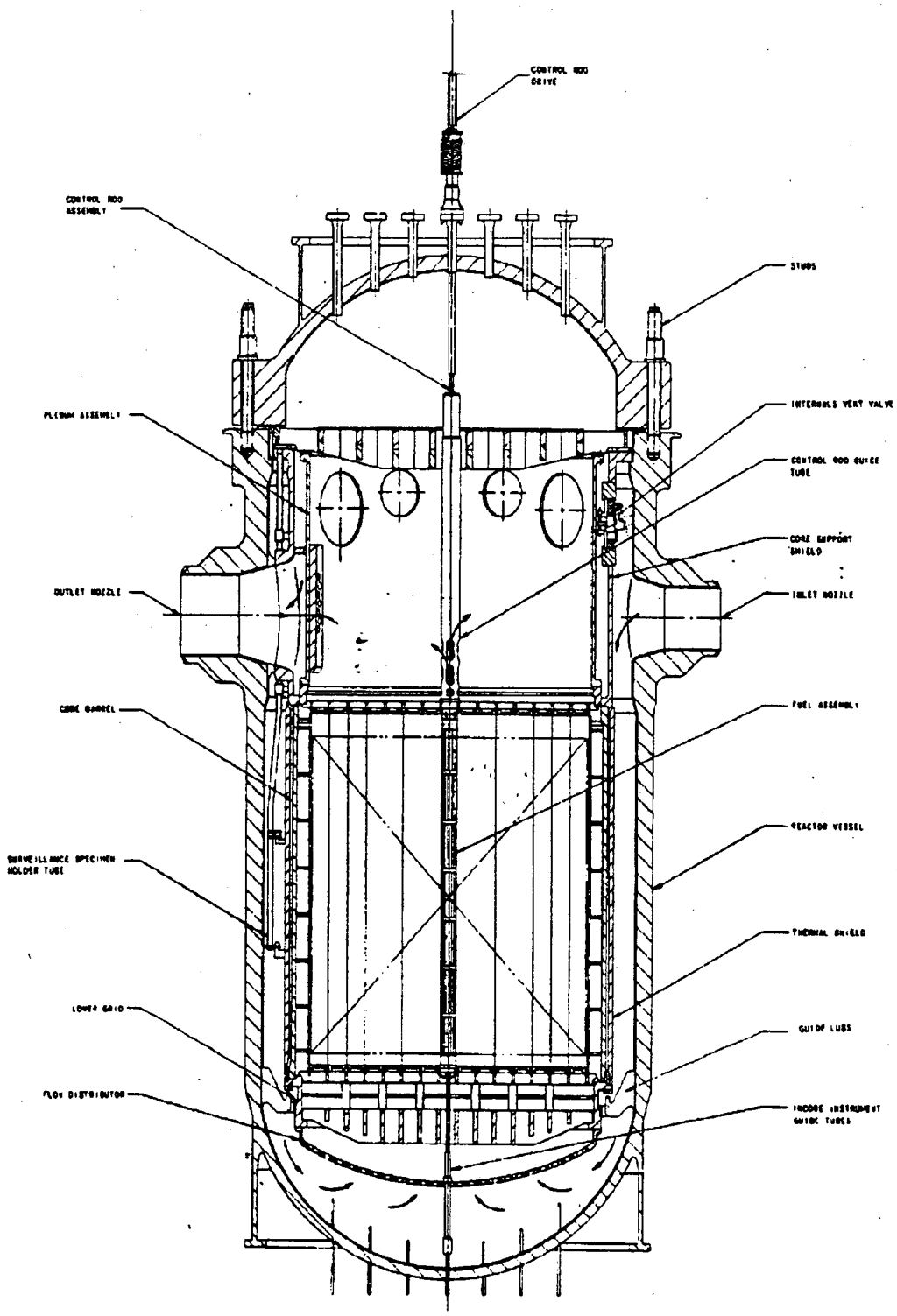


Figure 3-39.
Reactor Vessel and Internals

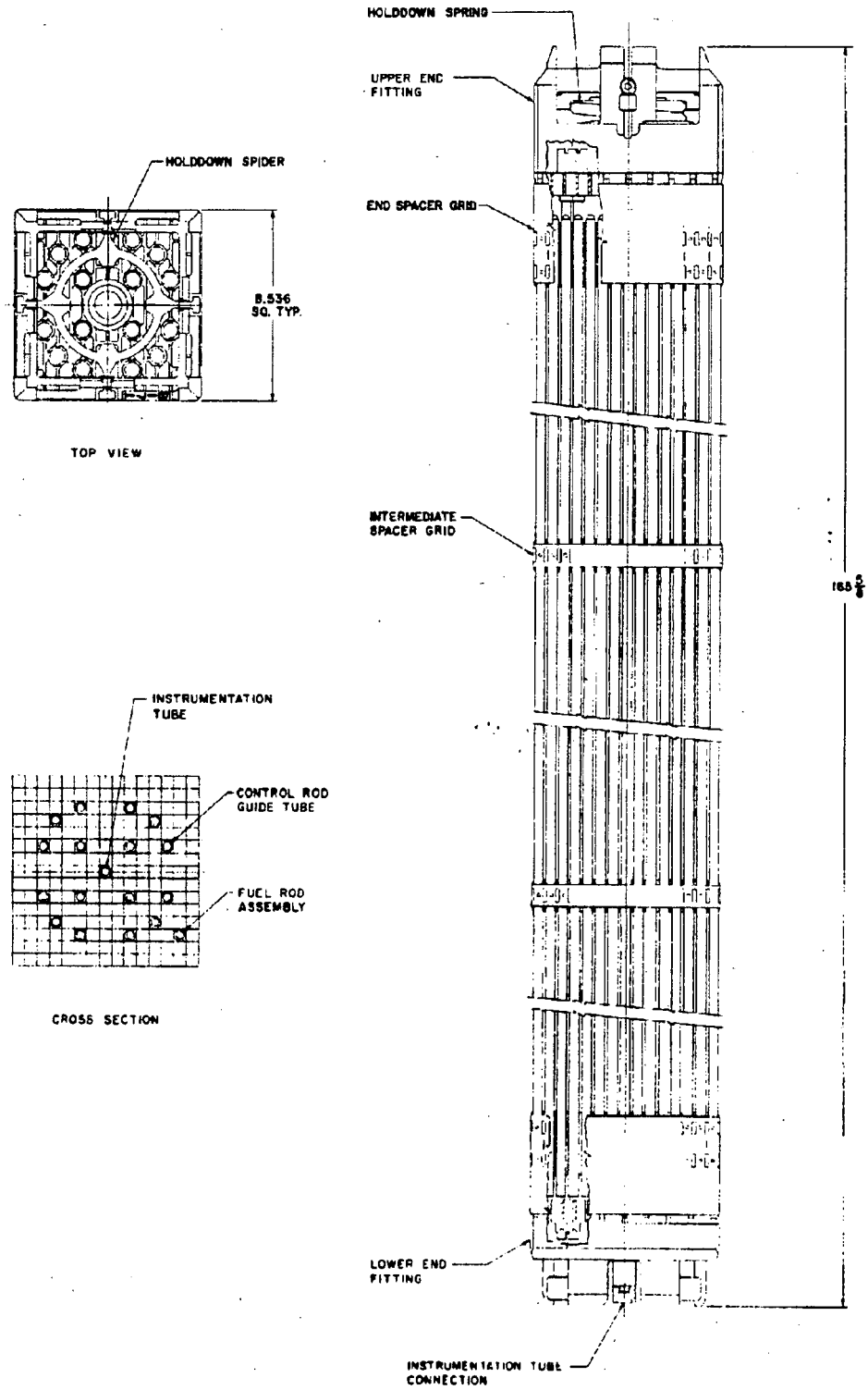


Figure 3-40.
Canless Fuel Assembly

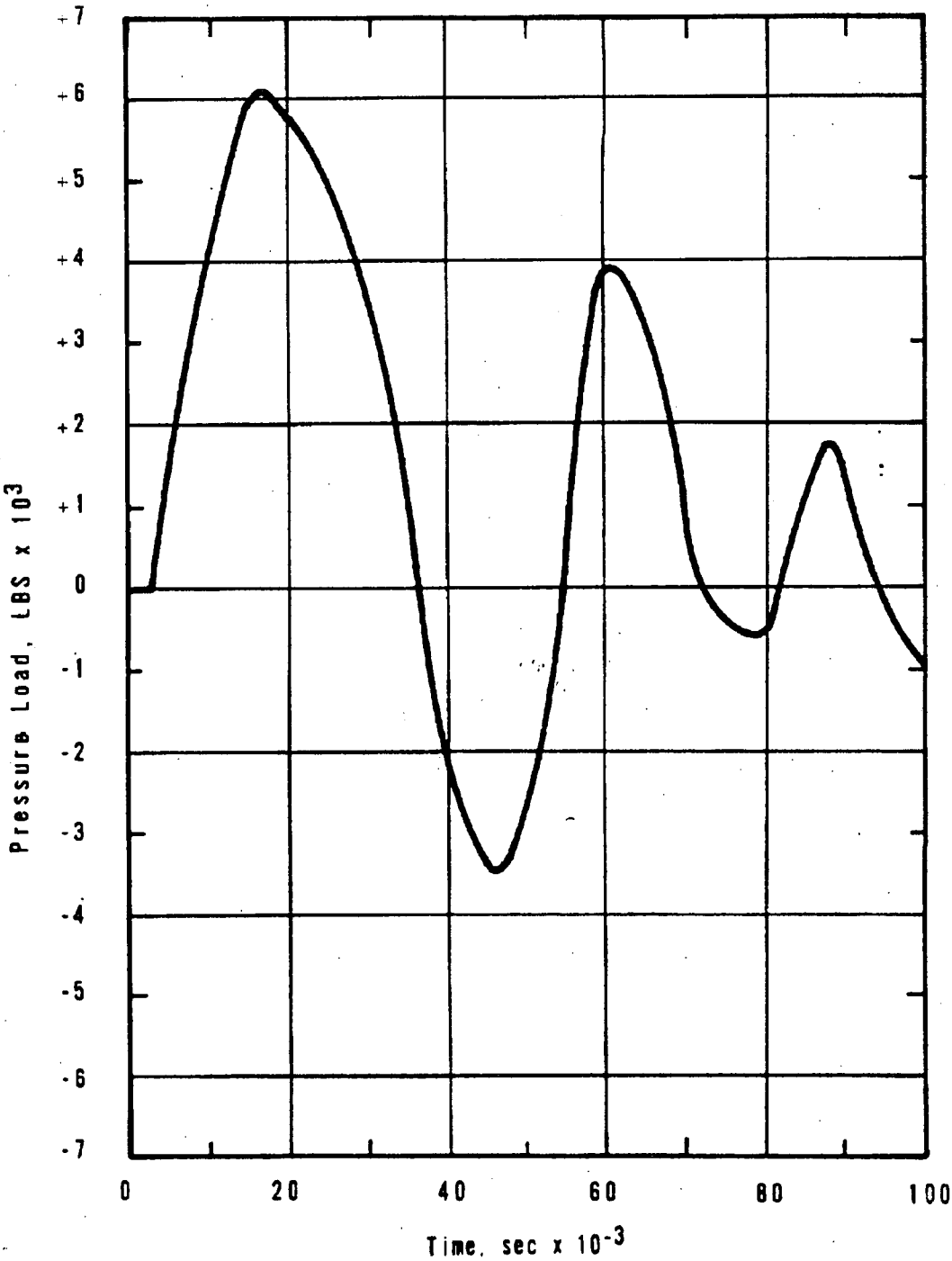


Figure 3-41.
Vertical Contact Loading Curve

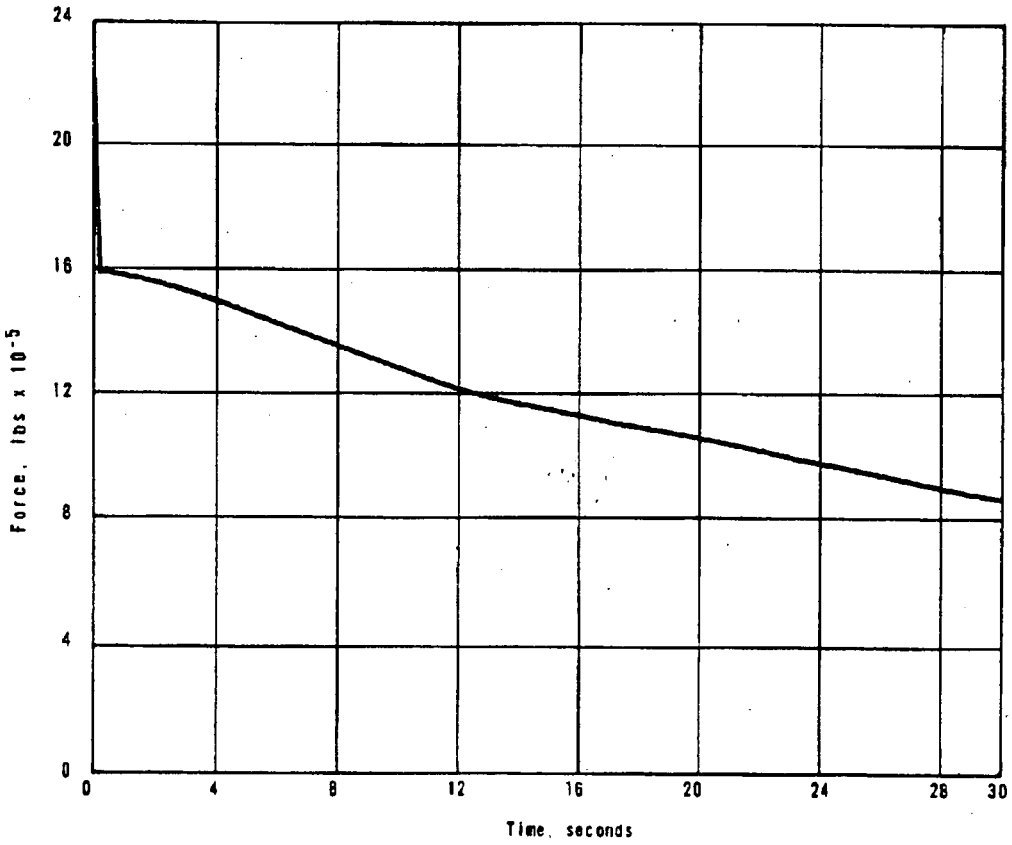


Figure 3-42.
Thrust-time Curve for Circumferential or Longitudinal Break of 36 Inch-ID Pipe

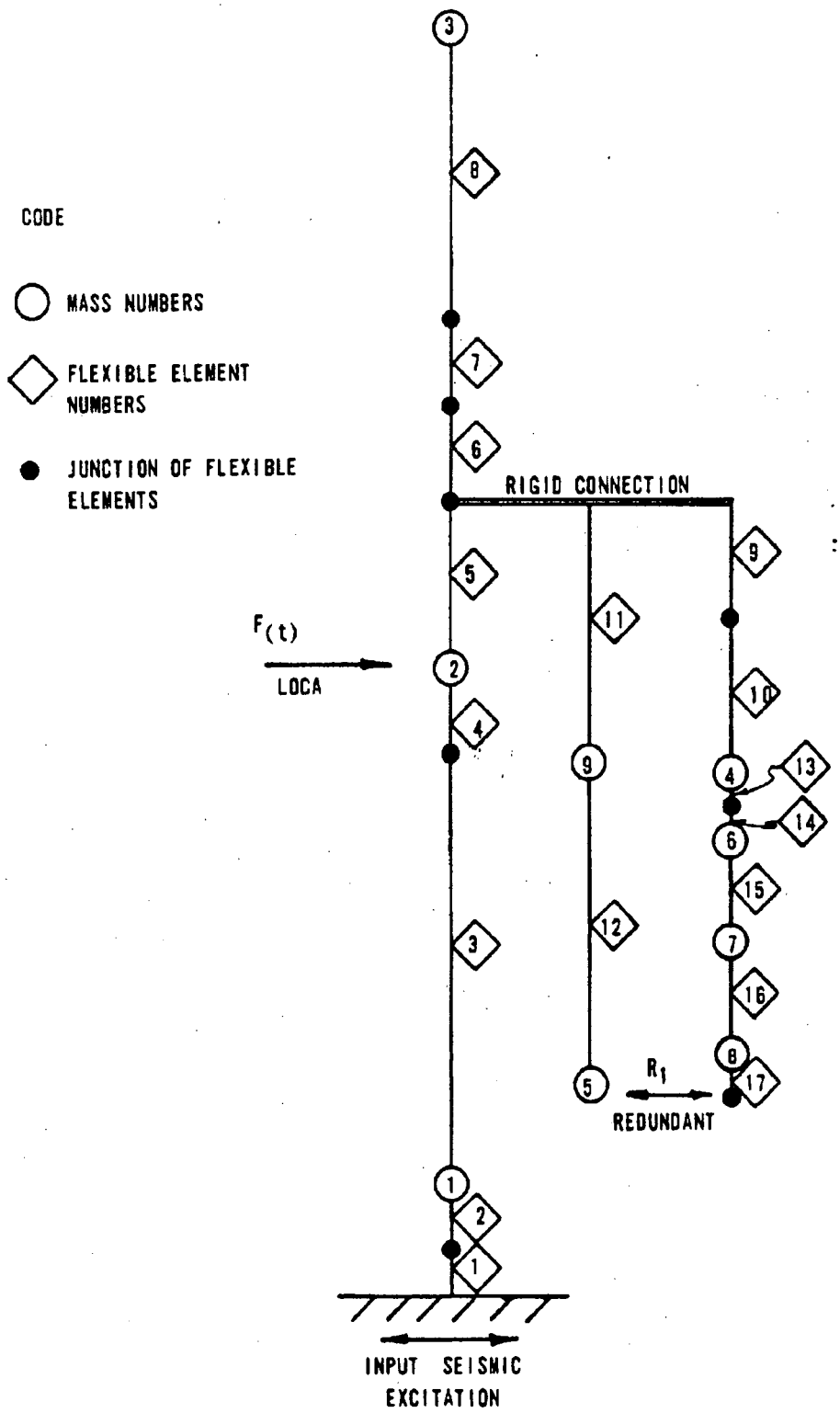


Figure 3-43.
First Segment Model

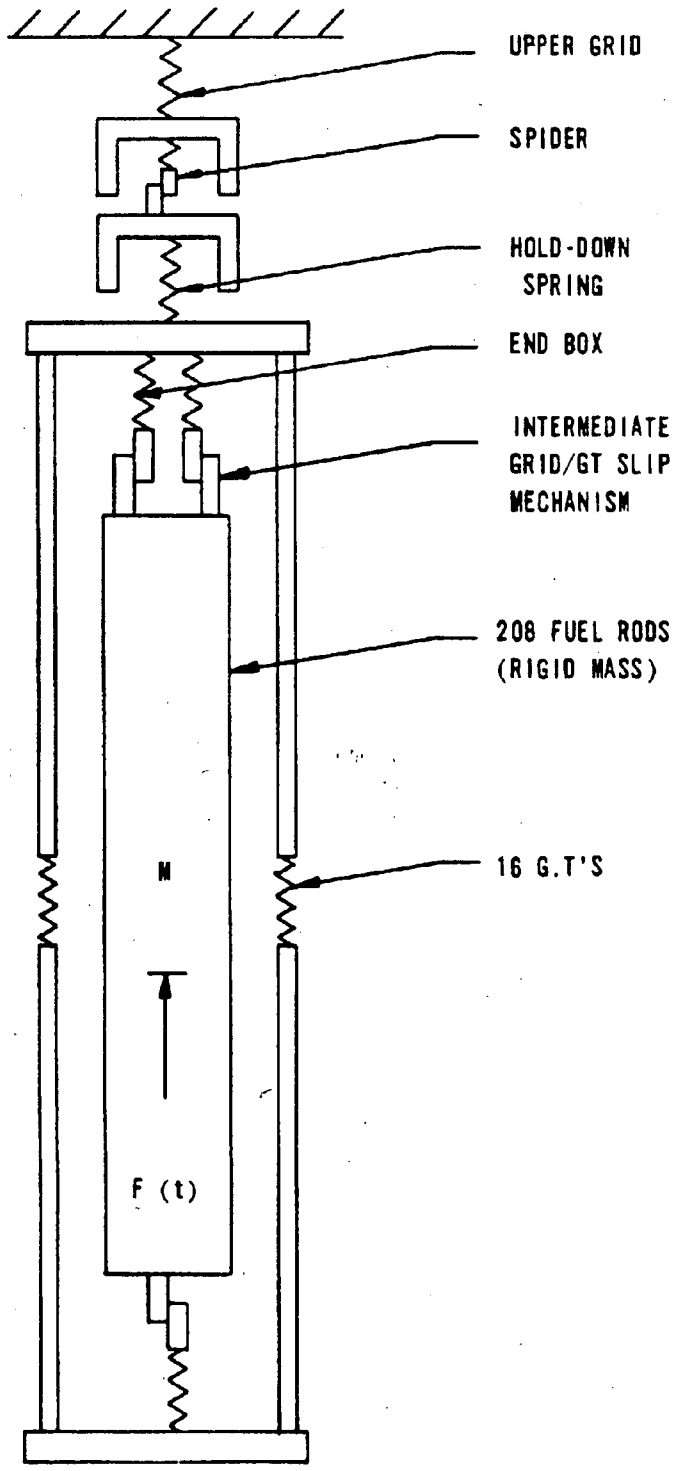


Figure 3-44.
Fuel Assembly Contact Model

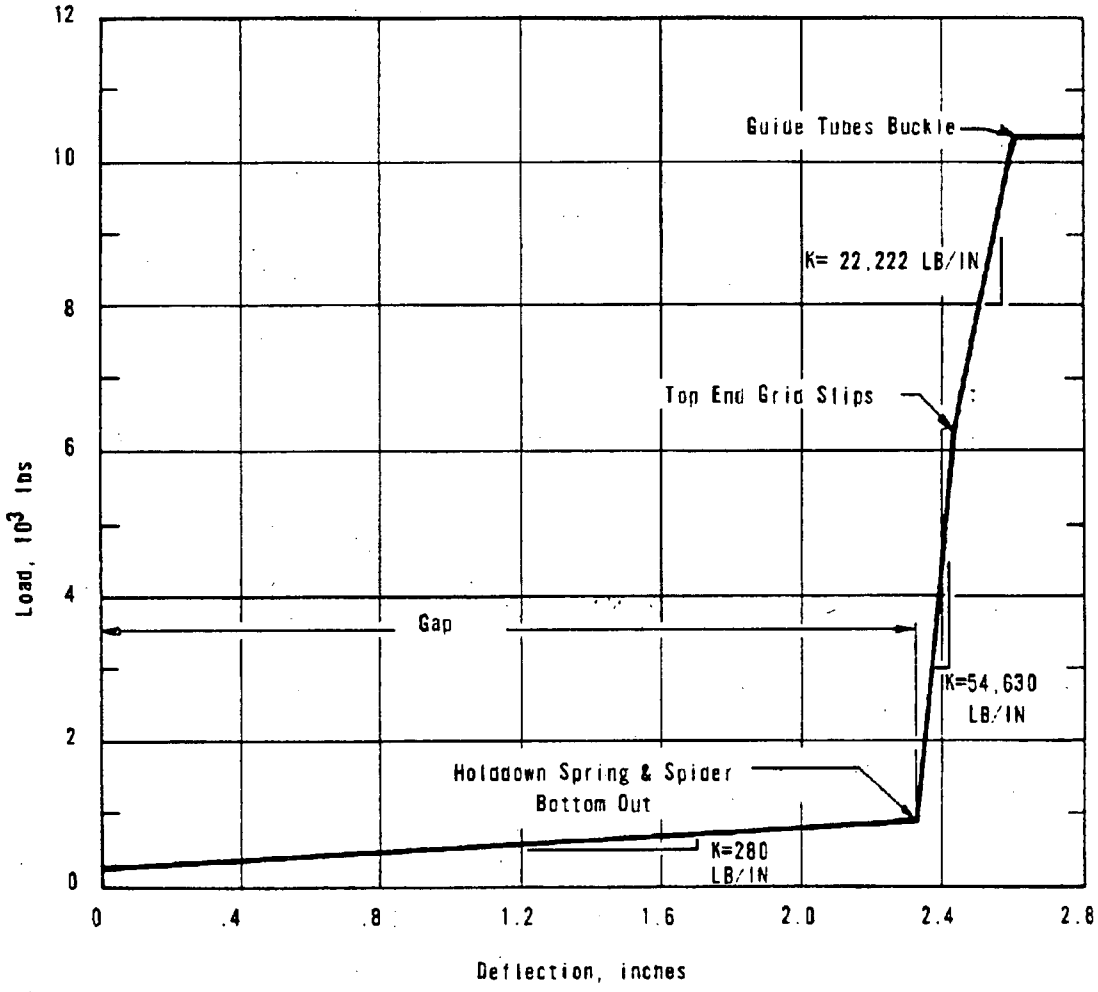


Figure 3-45.
Beginning-of-Life Spring Curve

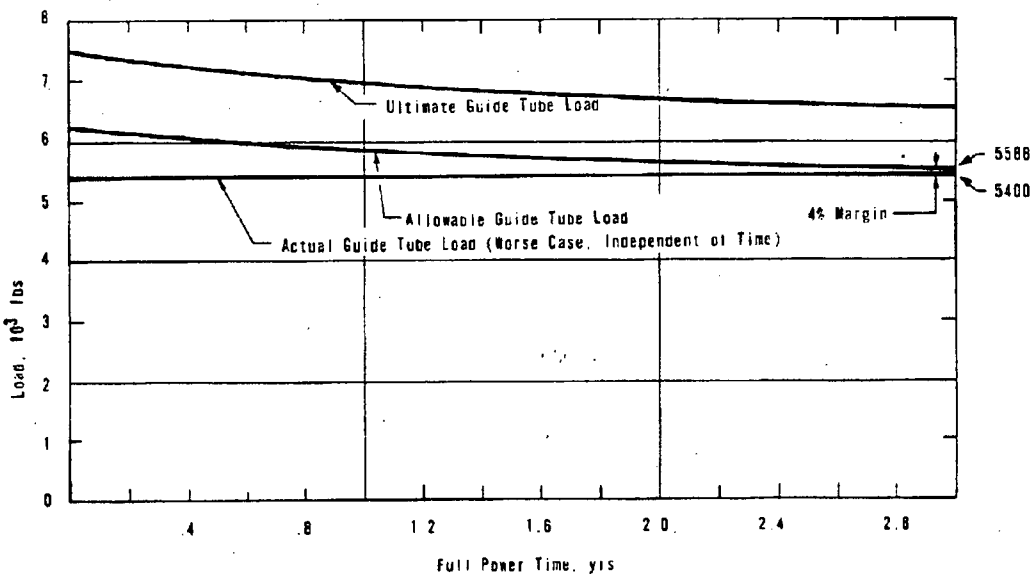


**Figure 3-46.**
**Vertical Contact Analysis Results**

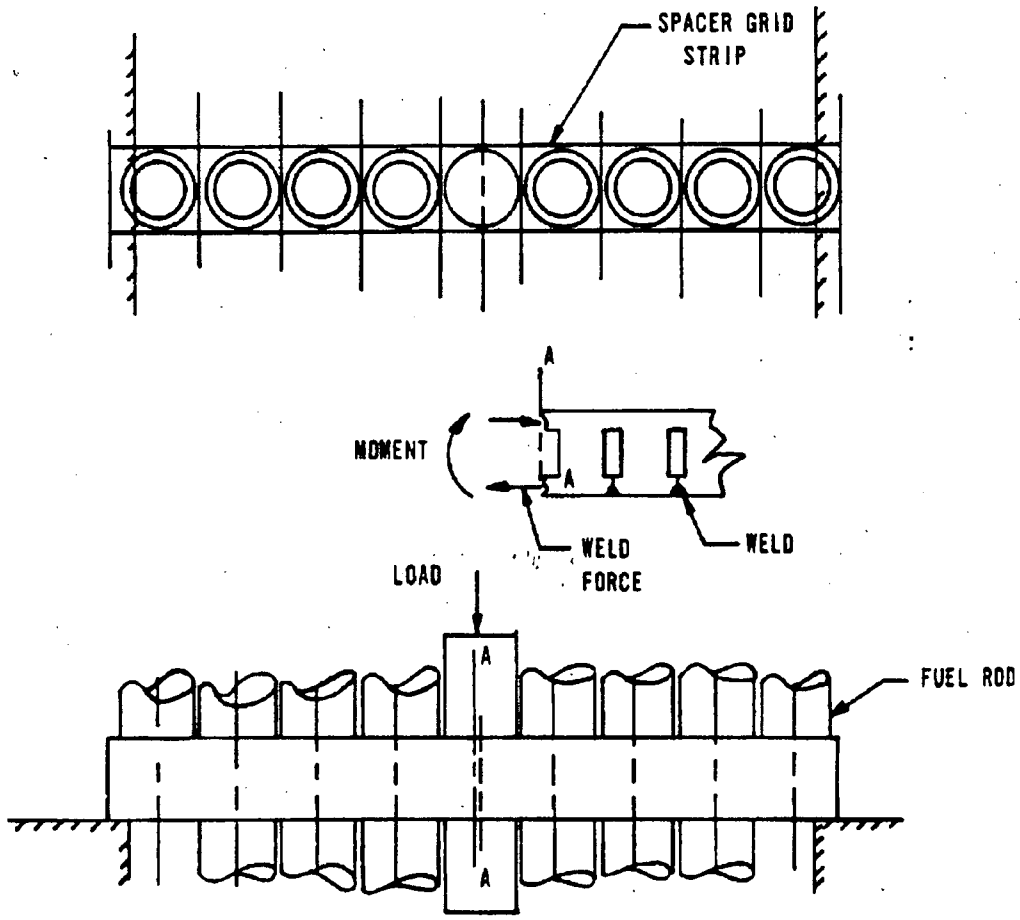


Figure 3-47.
Spacer Grid Weld Tests

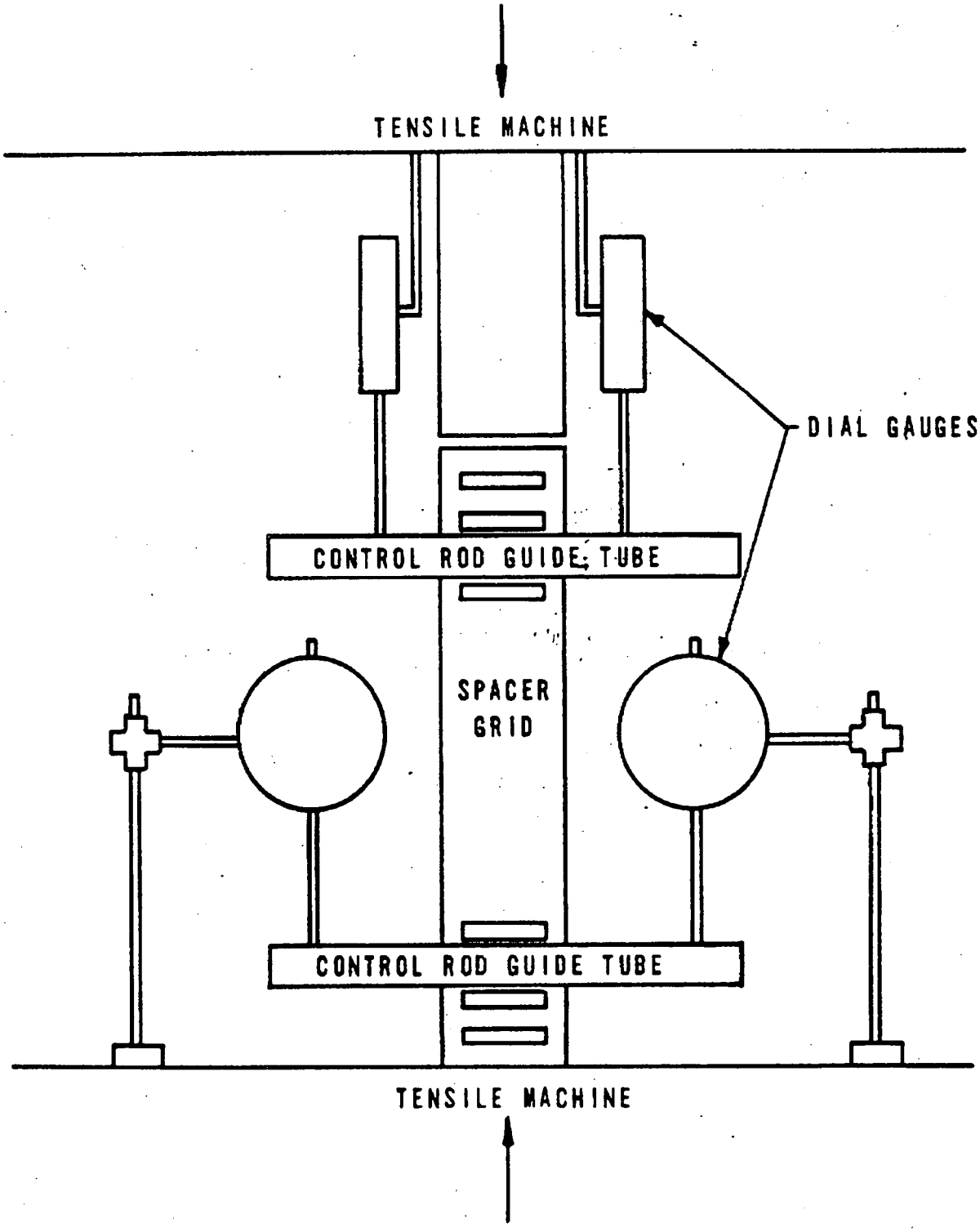


Figure 3-48.
Spacer Grid Compression Test Mount

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# CHAPTER 4. REACTOR

# 4.1 SUMMARY DESCRIPTION

The reactor is a pressurized water reactor and is functionally comprised of the reactor internals, fuel system, and control rod drives. The fuel system consists of the fuel assemblies and control components.

The major functions of the reactor internals are to support the core, maintain fuel assembly alignment, and direct the flow of reactor coolant.

The fuel system is designed to operate at 2,568 MWt with sufficient design margins to accommodate transient operation and instrument error without damage to the core and without exceeding limits for the Reactor Coolant System (RCS). The fuel system is designed to meet the performance objectives within the limits of design and operation specified in Sections 4.2, "Fuel System Design" on page 4-5, 4.3, "Nuclear Design" on page 4-19, and 4.4, "Thermal and Hydraulic Design" on page 4-41.

The fuel assembly is designed for structural adequacy and reliable performance during core operation. This includes steady-state and transient conditions under the combined effects of pressure, temperature, hydraulic forces, and irradiation. The fuel assembly is mechanically compatible with the reactor internals and control components. In addition to incore operation, the fuel assembly must be designed for handling, shipping, and storage to assure that the fuel assembly maintains its dimensional and structural integrity. Section III of the ASME Boiler and Pressure Vessel Code serves as a guide for fuel assembly and reactivity control component analysis.

The fuel assembly thermal-hydraulic operating characteristics have been determined and found to be compatible with design limits. Power peaks are controlled during transients so that no fuel melting occurs. The minimum core DNB ratio at the design overpower is well above the design limit. Although net steam generation occurs in the hottest core channels at the design overpower, hydraulic stability analyses have shown that no flow oscillations will occur.

The control components (control rod assemblies, axial power shaping rod assemblies, and burnable poison rod assemblies) are designed to perform their functions in controlling the reactor.

Core reactivity is controlled by control rod assemblies (CRAs), axial power shaping rod assemblies (APSRAs), burnable poison rod assemblies (BPRAs) and soluble boron in the coolant. Sufficient CRA worth is available to shut the reactor down with at least 1% $\Delta k/k$ subcritical margin in the hot condition at any time during the life cycle with the most reactive CRA stuck in the fully withdrawn position. Equipment is provided to add soluble boron to the reactor coolant to ensure a similar shutdown capability when the reactor is cooled to ambient temperatures.

The reactivity worth of a CRA and the rate at which reactivity can be added are limited to ensure that credible reactivity accidents cannot cause a transient capable of damaging the RCS or causing significant fuel failure.

The control rod guide path is designed to ensure that the control assemblies will not disengage from the fuel assembly guide tubes during operation. Guidance is provided by close-tolerance indexing of the fuel assembly upper end fitting with the upper grid rib section.

# 4.2 FUEL SYSTEM DESIGN

The fuel system consists of fuel assemblies and control components which are designed to the bases described in Section 4.2.1, "Design Bases - Fuel System Design" and Section 4.2.2, "Description - Fuel System Design" on page 4-8.

## 4.2.1 DESIGN BASES - FUEL SYSTEM DESIGN

The fuel is designed to meet the performance objectives specified in Section 4.2.1.1, "Fuel System Performance Objectives" without exceeding the limits of design and operation specified in Section 4.2.1.2, "Limits."

### 4.2.1.1 Fuel System Performance Objectives

The core is designed to operate at 2568 MWt (rated power) with sufficient design margins to accommodate transient operation and instrument error without damage.

The fuel rod cladding is designed to maintain its integrity for the anticipated operating transients throughout core life. The effects of gas release, fuel dimensional changes, and corrosion- or irradiation-induced changes in the mechanical properties of cladding are considered in the design of fuel assemblies.

### 4.2.1.2 Limits

#### 4.2.1.2.1 Nuclear Limits

5 The core has been designed to the following nuclear limits and capabilities, all of which are intended to
5 preserve the integrity of the fuel assemblies:

5 1. The core will have sufficient reactivity to produce the design power level and lifetime without
5 exceeding the control capacity or shutdown margin.

5 2. Fuel assemblies have been designed for the maximum burnups shown in Table 4-2. If they are not
5 bounded, acceptable reanalyses shall be performed.

5 3. Power histories must be bounded by those assumed within generic mechanical and thermal hydraulic
5 (fuel assembly) analyses. If they are not bounded, acceptable reanalyses shall be performed.

5 4. The maximum feed fuel enrichment is constrained by the maximum allowed in the Technical
5 Specifications (Spent Fuel Pool storage requirements).

5 5. Values of important core safety parameters predicted for the cycle have been verified to be
5 conservative with respect to their values assumed in the Chapter 15 safety/accident (and any other
5 pertinent) analyses. If they are not conservative, acceptable reanalyses shall be performed.

5 Controlled reactivity insertion rates due to a single CRA group withdrawal shall be limited to a
5 maximum value assumed within the Chapter 15 Rod Withdrawal Accident at Rated Power, and
5 within the Chapter 15 Startup Accident. Controlled reactivity insertion rates due to soluble boron
5 removal shall be limited to a maximum value assumed within the Chapter 15 Moderator Dilution
5 Accident.

The power Doppler and moderator temperature coefficients at power will be negative. However, as described within Chapter 15, the control system is capable of compensating for reactivity changes resulting from either positive or negative nuclear coefficients.

6. Reasonable and permissive reactor control and maneuvering procedures during nominal operation and during transients will not produce peak-to-average power distributions greater than those listed in Table 4-1. This, along with criteria 7 and 8, below, preserves the LOCA linear heat rate, linear heat rate to melt (LHRTM), and DNBR limits.

7. Part length axial power shaping rods (APSRs) are to be utilized to allow the shaping of power axially in the core, thereby thwarting any tendency towards axial instability resulting from a redistribution of xenon.

   To preclude the possiblity of azimuthal instability resulting from a redistribution of xenon, the highest moderator temperature coefficient assumed within Chapter 15 safety/accident analyses must be bounded by the threshold listed within Table 4-7.

8. Technical Specification limits of specified operating parameters (quadrant power tilt, power imbalance, and control rod insertion), and on reactor protective system trip setpoints (power imbalance) after allowance for appropriate measurement tolerances should have adequate margin from design limits of these parameters during operational conditions throughout the cycle such that sufficient operating flexibility is retained for the fuel cycle.

#### 4.2.1.2.2 Reactivity Control Limits

The control system and operational procedures will provide adequate control of the core reactivity and power distribution. The following control limits and capabilities shall be:

1. A control system consisting of part length axial power shaping rods (APSRs) shall be provided to control the core axial power distribution.

2. A shutdown margin of at least 1.0% $\Delta\rho$ shall be maintained throughout core life with the most reactive CRA stuck in the fully withdrawn position.

3. CRA withdrawal rate (as listed within Chapters 7 and 15) shall limit the maximum reactivity insertion rate to that assumed within the Chapter 15 Rod Withdrawal Accident at Rated Power, and within the Chapter 15 Startup Accident.

4. Boron dilution rate (as listed within Chapter 15) shall limit the maximum reactivity insertion rate to that assumed within the Chapter 15 Moderator Dilution Accident.

5. A control rod shall not be misaligned from the group average by the value listed within the Technical Specifications, and constrained within Chapters 7 and 15 (Control Rod Misalignment Accident). Except during the startup physics test program, operating rod overlap shall be within the bounds listed within the Technical Specifications, and constrained within Chapters 7 and 15 (Startup Accident).

6. Maximum boron (hot full power, or otherwise) will be constrained by those assumed within Chapter 15 or Technical Specifications. Sufficient soluble boron shall be available within the control system equipment (BWST, CBAST, and CFT) to ensure a 1.0% $\Delta\rho$ shutdown capability with the most reactive CRA stuck in the fully withdrawn position when the reactor is cooled to ambient temperatures.

7. There are no design constraints on BPRA poison enrichment or number of BPRA assemblies, except for those inferred by the peak-to-average power distributions constraints listed within Table 4-1, by Chapter 15 constraints, by Technical Specifications constraints (such as moderator temperature coefficient), or by the limiting core bypass flow assumed within thermal hydraulic analyses.

5 For more detail, refer to Section 4.3, "Nuclear Design" on page 4-19.

#### 4.2.1.2.3 Thermal and Hydraulic Limits

The reactor core is designed to meet the following limiting thermal and hydraulic conditions:

5 1. The fuel pin must be designed so that the maximum fuel temperature does not exceed the fuel melting
5 limit at any time during core life. The TACO3 computer program is used to verify heat rate capacity
5 (Reference 4 on page 4-17).

5 2. The minimum allowable DNBR during steady-state operation and anticipated transients for
5 Mark-B4Z and higher fuel is 1.18 with the BWC correlation (Reference 6 on page 4-17).

3. Although generation of net steam is allowed in the hottest core channels, flow stability is required during all steady-state and operational transient conditions.

By preventing a departure from nucleate boiling (DNB), neither the cladding nor the fuel is subjected to excessively high temperatures.

For more detail refer to Section 4.4, "Thermal and Hydraulic Design" on page 4-41.

#### 4.2.1.2.4 Mechanical Limits

Fuel assemblies are designed for structural adequacy and reliable performance during core operation, handling, and shipping. Design criteria for core operation include steady state and transient conditions under combined effects of flow induced vibration, temperature gradients, and seismic disturbances.

Spacer grids, located along the length of the fuel assembly, position fuel rods in a square array, and are designed to maintain fuel rod spacing during core operation, handling, and shipping. Spacer-grid to fuel-rod contact loads are established to minimize fretting, but to allow axial relative motion resulting from fuel rod irradiation growth and differential thermal expansion.

The fuel assembly upper end fitting is indexed to the plenum assembly by the upper grid rib section immediately above the fuel assemblies to assure proper alignment of the fuel assembly guide tubes to the control rod guide tube. The guidance of the control rod assembly and axial power shaping rod assembly is designed such that these assemblies will never be disengaged from the fuel assembly guide tubes during operation.

1. Section III of the ASME Boiler and Pressure Vessel Code is used as a guide in classifying the stresses into various categories and combining these stresses to determine stress intensities. The following limits apply to the fuel cladding stress analysis.

   a. The stress intensity value of the primary membrane stresses in the fuel rod cladding that are not relieved by small material deformation of the cladding shall not exceed the lesser of 2/3 the minimum unirradiated yield strength or 1/3 the minimum unirradiated ultimate tensile strength.

   b. All fuel cladding stresses (primary and secondary) shall not exceed the lessor of 2.0 times the unirradiated yield strength or 1.0 times the unirradiated ultimate tensile strength for Condition I and II occurrences, as permitted by the ASME Boiler and Pressure Vessel Code. Refer to Section 4.2.3.1.1, "Clad Stress and Strain" on page 4-11 for the Duke clad stress and strain methodology.

2. Strain limits for this stress condition are established based on low cycle fatigue techniques, not to exceed 90 percent of the material fatigue life. Evaluation of cyclic loading is based on conservative estimates of the number of cycles to be expected. An example of this type of stress is the thermal stress resulting from thermal gradients across the cladding thickness.

3. Cladding uniform strain is limited to a maximum of 1.0 percent.

4. Cladding Collapse

The digital computer code CROV (Reference 1 on page 4-17) is used to demonstrate, for each fuel cycle design, that the effective full power hours (or equivalent burnup) to complete cladding collapse is greater than the incore residence time.

5. Fuel Thermal Analysis

The digital computer code TACO3 (Reference 4 on page 4-17) is used to ensure that fuel performance is satisfactory. Specifically the centerline temperature is maintained below fuel melt limits and end of life pin pressure is maintained below the value which would cause clad lift off.

Refer to Section 4.2.3.1.3, "Fuel Thermal Analysis" on page 4-13 for design evaluations of the fuel thermal analyses.

## 4.2.2 DESCRIPTION - FUEL SYSTEM DESIGN

The complete core has 177 fuel assemblies which are arranged in the approximate shape of a cylinder. All fuel assemblies are similar in mechanical construction, and are mechanically interchangeable in any core location. The reactivity of the core is controlled by 61 control rod assemblies (CRAs) and 8 axial power shaping rod assemblies (APSRAs), a variable number of burnable poison rod assemblies (BPRAs), and soluble boron in the coolant. APSRAs are identical in physical configuration to the CRAs but have absorber material only in the lower portion of the rods. Burnable poison rod assemblies (Figure 4-1) are installed in selected fuel assemblies not containing an APSRA or a CRA. The burnable poison rod assemblies (BPRAs) assure a negative moderator temperature coefficient through core lifetime. The mechanical and geometric configuration of the CRAs and BPRAs permit full interchangeability in any fuel assembly. Currently, the APSRs are not fully interchangeable. The APSR design for fuel assemblies with helical hold down springs (MK B-4 to MK B-9) is different from fuel assemblies with cruciform hold down springs (MK B-10 and MK B-10F).

Important core design, thermal, and hydraulic characteristics are tabulated in Table 4-1, and fuel assembly component materials are presented in Table 4-2.

### 4.2.2.1 Fuel Assemblies

#### 4.2.2.1.1 General

The fuel assembly designs shown in Figure 4-2, Figure 4-3, Figure 4-36, and Figure 4-37, are typical of the designs used in Oconee 1, 2 and 3 and are of the canless type where the eight spacer grids, the end fittings, and the guide tubes form the basic structure. Fuel rods are supported at each spacer grid by contact points integral with the wall of the cell boundary. For the MK B-4, MK B-5, MKB-4Z, and MK B-5Z designs, the guide tubes are permanently attached to the upper end and lower end fittings. Introduction of the reconstitutable top nozzle on the MK B-6, MK B-7, MK B-8, MK B-9, MK B-10, and MK B-10F designs eliminates permanent attachment of the guide tubes to the upper end fitting. Use of similar material in the guide tubes and fuel rods results in minimum differential thermal expansion.

The fuel is sintered low-enriched uranium dioxide cylindrical pellets. The pellets are clad in Zircaloy-4 tubing and sealed by Zircaloy-4 end caps, welded at each end. The clad, fuel pellets, end caps, and fuel support components form a "fuel rod." Two hundred and eight fuel rods, sixteen control rod guide tubes, one instrumentation tube assembly, six segmented spacer sleeves, eight spacer grids, and two end fittings make up the basic "Fuel Assembly" (Figure 4-2, Figure 4-3, Figure 4-36, and Figure 4-37). The guide tubes, spacer grids, and end fittings form a structural cage to arrange the rods and tubes in a 15 x 15 array. The center position in the assembly is reserved for instrumentation. Control rod guide tubes are located

in 16 locations of the array. Fuel assembly components, materials, and dimensions are tabulated in Table 4-2.

At Oconee, several variations of the basic fuel assembly are used; Mark B-4, Mark B-4Z, Mark B-5, Mark B-5Z, Mark B-6, Mark B-7, Mark B-8, Mark B-9, Mark B-10, and MK B-10F. The Mk B-4 assembly is a 15 x 15 Inconel grid assembly and has been used successfully for many years in all units. Mk B-5 is identical to the Mk B-4 except that its upper end fitting has been modified to provide a positive holddown capability of fixed core components without a retainer assembly installed. This type was first implemented in the Oconee 3 Cycle 7 Reload. The Mk B-4Z and Mk B-5Z assemblies differ from the Mk B-4 and Mk B-5 assemblies in grid material composition of the intermediate grids. The Z-grid assemblies are made of Zircaloy-4 material (with minor changes made to the assembly to accommodate the differences in grid designs).

The MK B-6, MK B-7, and MK B-8 designs are shown on Figure 4-3. The MK B-9 design is shown in Figure 4-36. The MK B-10 and MK B-10F designs are shown in Figure 4-37. The MK B-6 is similar in design to the MK B-5Z except for the addition of the removable upper end fitting and deletion of the top grid skirt. The MK B-6 reconstitutable top nozzle features the use of locking cups on the guide tube nuts to enable removal of the upper end fitting for purposes of reconstitution. The MK B-7 is similar to the MK B-6, except that the lower end fitting has been shortened and also the cladding length and shoulder gap dimensions have increased with respect to the MK B-6. The MK B-8 is identical to the MK B-7, however, the lower end cap was lengthened and the bottom spacer grid was lowered to increase resistance to coolant-borne debris. The MK B-9 is similar to the MK B-8 except the fuel pellet ID is increased and the active fuel length is decreased so that the overall kg U loading remains the same. In addition, the MK B-9 has a skirtless LEG and removable LEF to facilitate fuel rod reconstitution from the bottom of the fuel assembly. The MK B-10 is similar to the MK B-9 except the HDS configuration has a cruciform shape as opposed to the traditional Helical. The MK B-10 also contains a modified UEF locknut design to expedite crimping and verification of positive locking device. For the Oconee 2 Cycle 15 reload, the MK B-10 fuel rod design incorporated a 6 inch enriched axial blanket at the top and bottom of the fuel stack. The MK-B10F contains larger diameter pellets, a change in the pellet dish design from a truncated cone to a spherical dish, lower initial prepressure, axial blankets, and the bottom plenum spring was removed.

#### 4.2.2.1.2 Fuel Rod

The fuel rod consists of fuel pellets, cladding, fuel support components, and end caps. All fuel rods are internally pressurized with helium.

The pellets are manufactured by cold pressing enriched uranium dioxide powder into cylinders with edge chamfers and dish at each end and then sintering to obtain the desired density and microstructure. After sintering, the pellets are centerless ground to the required diametrical dimensions.

There are spring spacers located both above and below the pellet stack in the MK B-4 to MK B-10 fuel rod designs. Both springs are designed to accommodate maximum thermal expansion of the fuel column without being deflected beyond solid height. The lower spring is much stiffer by design, so the fuel column preload, thermal expansion and irradiation expansion principally compresses the upper spring. The MK B-10F fuel rod does not contain a bottom plenum spring.

Zircaloy spacers are located between the fuel pellets and the spring spacers to provide thermal insulation and separation in the MK B-4 to MK B-7 fuel rod designs. The MK B-8 through MK B-10F fuel rod designs do not use Zircaloy spacers.

4 Beginning with Oconee 2 Cycle 15, the fuel rods will contain a 6 inch axial blanket at the top and bottom
4 of the fuel stack. The pellets in the axial blanket will have a lower enrichment and will be slightly longer
4 than those in the active fuel region. The function, behavior, and analysis of fuel rods containing axial
4 blankets will be the same as non-blanket fuel rods.

Fission gas generated in the fuel is released into pellet voids, the radial gap between the pellets and the
5 cladding, and into the plenum spring space. Fuel rod data are given in Table 4-2, and a typical fuel rod is
5 shown in Figure 4-4.

#### 4.2.2.1.3 Spacer Grids

Spacer grids are constructed from strips which are slotted and fitted together in "egg crate" fashion. Each grid has 32 strips, 16 perpendicular to 16, which forms the 15 x 15 lattice. The square walls formed by the interlaced strips provide support for the fuel rods in two perpendicular directions. Contact points on the walls of each square opening are integrally punched in the strips. On each of the two end spacer grids, the peripheral strip is extended and rigidly attached to the respective end fitting. The grid extension (skirt)
5 exists only on the bottom grid in the MK B-6, MK B-7, and MK B-8 designs. The MK B-9 to MK
5 B-10F designs do not have grid skirts.

#### 4.2.2.1.4 Lower End Fittings

The lower end fitting positions the assembly in the lower grid rib section. The lower ends of the fuel rods rest on the grillage of the lower end fitting. Penetrations in the lower end fitting are provided for attaching the control rod guide tubes and for access to the instrumentation tube assembly. The MK B-5Z, MK
5 B-6, MK B-7, MK B-8, MK B-9, MK B-10, and MK B-10F lower end fittings are of an anti-straddle
5 design which will prevent the fuel assembly from being improperly seated on the lower grid assembly.
5 The MK B-9, MK B-10, and MK B-10F LEF are removable to facilitate fuel rod reconstitution from the
1 bottom of the fuel assembly.

#### 4.2.2.1.5 Upper End Fitting

The upper end fitting positions the upper end of the fuel assembly in the upper grid rib section and provides means for coupling the handling equipment. An identifying number on each upper end fitting provides positive identification.

Attached to the upper end fitting is a holddown spring. This spring provides a positive holddown margin to oppose hydraulic forces resulting from the flow of the primary coolant.

Penetrations in the upper end fitting grid are provided for the guide tubes.

1 MK B-5 and earlier fuel assembly designs do not allow for removal of the UEF. The MK B-6, MK B-7,
5 MK B-8, MK B-9, MK B-10, and MK B-10F assemblies do allow for the removal of the UEF in order
5 to perform fuel assembly reconstitution.

#### 4.2.2.1.6 Guide Tubes

The Zircaloy guide tubes provide continuous guidance for the control rod assemblies when inserted in the fuel assembly and provide the structural continuity for the fuel assembly. Welded to each end of a guide tube are flanged and threaded sleeves, which secure the guide tubes to each end fitting by lock-welded nuts
5 (except in the case of the MK B-6 through MK B-10F designs, the upper guide tube nut is held secure by
5 a crimped locking cup). Transverse location of the guide tubes is provided by the spacer grids. The MK
5 B-9, MK B-10, and MK B-10F designs incorporate smaller guide tube holes so that more coolant flows
1 alongside the fuel rods. The smaller holes allow less coolant inside the guide tubes.

#### 4.2.2.1.7 Instrumentation Tube Assembly

This assembly serves as a channel to guide, position, and contain the in-core instrumentation within the fuel assembly. The instrumentation probe is guided up through the lower end fitting to the desired core elevation. It is retained axially at the lower end fitting by a retainer sleeve.

#### 4.2.2.1.8 Spacer Sleeves

The spacer sleeve fits around the instrument tube between spacer grids and prevent axial movement of the spacer grids during primary coolant flow through the fuel assembly.

## 4.2.3 DESIGN EVALUATION - FUEL SYSTEM DESIGN

This subsection contains a description of the fuel system design evaluation and is primarily a mechanical evaluation.

Nuclear design evaluation is contained within Section 4.3.3, "Nuclear Evaluation" on page 4-24. Thermal hydraulic design evaluation is presented in Section 4.4.3, "Thermal and Hydraulic Evaluation" on page 4-42.

### 4.2.3.1 Fuel Rod

The basis for the design of the fuel rod is discussed in Section 4.2.1, "Design Bases - Fuel System Design" on page 4-5. Materials testing and actual operation in reactor service with Zircaloy cladding have demonstrated that Zircaloy-4 material has sufficient corrosion resistance and mechanical properties to maintain the integrity and serviceability required for design burnup.

If fuel assemblies are damaged or develop leaking fuel rods, the fuel assemblies can be reconstituted in order to replace damaged rods. The primary replacement is a fuel rod that contains pellets of naturally enriched uranium dioxide ($UO_2$). Aside from enrichment, this rod is similar in design and behavior as a standard fuel rod and is analyzed using standard approved methods. If grid damage exists, solid filler rods made of stainless steel or Zircaloy could be used as a replacement. A maximum of 10 such filler rods can be substituted into a single fuel assembly.

The NRC has approved Duke's reconstitution topical report (Reference 7 on page 4-17). This report details the methodology and guidelines Duke Power Company will use to support fuel assembly reconstitution with filler rods. This methodology ensures acceptable nuclear, mechanical, and thermal-hydraullic performance of reconstituted fuel assemblies.

#### 4.2.3.1.1 Clad Stress and Strain

The following descriptions summarize the analyses of fuel rod cladding stress and strain for reload fuel cycle designs, as performed by Duke. Duke employs TACO3 (Reference 4 on page 4-17), in conjunction with its own approved methodology (Reference 2 on page 4-17).

1. Cladding Stress Analysis

   The cladding stresses for a new fuel cycle design are bounded by a conservative design analysis that uses Section III of the ASME Boiler and Pressure Vessel Code as a guide in classifying the stresses into various categories, assigning appropriate limits to these categories, and combining these stresses to determine stress intensity. Each new fuel cycle design is assessed to determine if reanalysis is required. The stress analysis is very conservative, and reanalysis should not be required for standard Mark B reloads.

The fuel rod stress analysis considers those stresses that are not relaxed by small material deformation, and this analysis complies with the following design criteria:

- All fuel cladding stresses (primary and secondary) shall not exceed the lesser of 2.0 times the minimum unirradiated yield strength or 1.0 times the minimum unirradiated ultimate tensile strength for condition I and II occurrences, per Section III of the ASME Boiler and Pressure Vessel Code.
- The stress intensity value of the primary membrane stresses in the fuel rod cladding, which are not relieved by small material deformation of the cladding, shall not exceed the lesser of:
  1. one-third of the specified minimum ultimate tensile strength at room temperature
  2. one-third of the minimum ultimate tensile strength at operating temperature
  3. two-thirds of the specified minimum yield strength at room temperature
  4. two-thirds of the minimum yield strength at operating temperature

In performing the stress analysis, all the loads are selected to represent the worst case loads and are then combined. This represents a conservative approach since they cannot occur simultaneously. This insures that the worst conditions for condition I and II events are satisfied. In addition, these input parameters were chosen so that they conservatively envelope all Mk-B design conditions.

5 The primary membrane stresses result from pressure loading. Stresses resulting from creep ovalization are addressed in the creep collapse analysis.

5 The internal fuel rod pressure can exceed system pressure by a proprietary value during condition I and II
5 occurrences (at coolant temperatures greater than 525°F). Also, during a cooldown between 525°F and
5 425°F, the fuel rod is permitted to operate in tension, with a tensile hoop stress less than 20,000 psi. Cooldown limits on reactor coolant system pressure and reactor coolant system temperature prohibit
5 tensile stresses greater than 20,000 psi. Cladding tensile stresses are also addressed at cold (room temperature) conditions at BOL.

The minimum internal fuel rod pressure at HZP conditions is combined with the maximum design system pressure during a transient to simulate the maximum compressive pressure differential across the cladding. The worst case compressive pressure loads are combined with the other worst case loads. These are described below:

- The maximum grid loads will occur at BOL. During operation, the contact force will relax with time due to fuel rod creep-down and ovalization as well as grid spring relaxation.
- Conservative cladding dimensions with regard to stress.
- The maximum radial thermal stress will occur at the maximum rated power (power level corresponding to centerline fuel melt). This stress cannot physically occur at the same time the maximum pressure loading occurs, but is assumed to do so for conservatism. (Maximum cladding temperature gradient is combined with minimum pin pressure.)
- Ovality bending stresses are calculated at BOL conditions. A linear stress distribution is assumed. The creep collapse analysis calculates the stress increase with time and ovalization.
- Flow induced vibration and differential fuel rod growth stresses are also addressed.

Resulting stresses meet the above criteria for both primary membrane and primary plus secondary stress intensities.

2. Cladding Strain Analysis

The limit on transient cladding strain is that uniform total strain of the cladding should not exceed 1.0%.

Duke performs a generic strain analysis using TACO3 to ensure that the strain criterion is not exceeded. For each reload cycle, the generic strain power history is compared to the predicted power history in the final fuel cycle design. If the generic power history is violated, cladding strain is re-analyzed using a new bounding power history.

Maximum tensile elastic and plastic strain occurs at the clad inside diameter. Clad strain is calculated as:

$$\text{Clad Strain\%} = \frac{\text{Clad ID}_{\text{transient}} - \text{Clad ID}_{\text{transient beginning}}}{\text{Clad ID}_{\text{transient beginning}}} \text{x}100$$

where the clad ID prior to and after a power ramp (transient) is calculated by TACO3 using the methodology explained in references 2 and 4.

3. End of Life Pressures

An analysis is performed to demonstrate that the internal pin pressure does not exceed a value that would cause: (1) the fuel-clad gap to increase due to outward cladding creep during steady-state operation and, (2) extensive DNB propagation to occur. (Section 4.2.3.1.3, "Fuel Thermal Analysis.")

#### 4.2.3.1.2 Cladding Collapse

Cladding creepdown under the influence of external (system) pressure is a phenomenon that must be evaluated during each reload fuel cycle design to ensure that the most limiting fuel rod does not exceed the cladding collapse exposure limit. Cladding creep is a function of neutron flux, cladding temperature, applied stress, cladding thickness, and initial ovality. Acceptability of a fuel cycle design is demonstrated by comparing the power histories of all the fuel assemblies against the generic assembly power history used in existing design analyses. Changes in pellet or cladding design are also evaluated against previously analyzed fuel rod geometries and a reanalysis is performed if necessary.

The CROV (Reference 1 on page 4-17) computer code calculates ovality changes in the fuel rod cladding due to thermal and irradiation creep and is used to perform the fuel rod creep collapse analysis when required. CROV predicts the conditions necessary for collapse and the resultant time to collapse. Conservative inputs to the CROV cladding collapse analysis include the use of minimum cladding wall thickness and maximum initial ovality (conservatively assumed to be a uniform oval tube), as allowed by manufacturing specifications or batch specific as-built tolerance limits. Other conservatisms included are minimum backfill pressure and zero fission gas release. Internal pin pressure and cladding temperatures, input to CROV, are calculated by TACO3 using a (conservative) generic radial power history, and a typical axial flux shape.

The conservative fuel rod geometry and conservative power history are used to predict the number of EFPH (or equivalent burnup) required for complete cladding collapse. To demonstrate acceptability, the maximum expected residence time of the cycle is compared against the EFPH (or equivalent burnup) required for complete collapse. All operating cores must meet this criterion.

#### 4.2.3.1.3 Fuel Thermal Analysis

Duke Power Company is performing its own reload design analyses per the approved methodology of Reference 2 on page 4-17. Duke currently uses the TACO3 fuel pin performance code. The following

paragraphs summarize the methods that are used by Duke in performing its Oconee reload fuel temperatures, end of life pin pressure, and ECCS analysis interface criteria analyses, using TACO3.

1. Fuel Pin Pressure Analysis

The pin pressure limit is intended to preserve the fuel-clad heat transfer characteristics by preventing clad liftoff. This limit provides reasonable assurance that: (1) excessive fuel temperatures, (2) excessive internal gas pressures due to fission gas release, and (3) excessive cladding stresses and strains are prevented.

The maximum allowable pin burnup is based on whichever of the following conditions occurs first:

A. Maximum Internal Pin Pressure: The fuel rod internal pressure is limited to a proprietary value above the nominal system pressure.

B. Clad Liftoff Limit: Clad liftoff occurs when the clad's outward creep rate exceeds the pellet's swelling rate. Clad liftoff is based on the ratio of cladding diametral strain rate divided by the fuel diametral strain rate at each axial elevation. Fuel-clad liftoff occurs when this ratio is $\geq$ 1.0 at any axial elevation where the local LHR is $\geq$ 3.0 kw/ft.

Duke performs a generic pin pressure analysis using the methodology described in Reference 2 on page 4-17. For each reload cycle, the generic power history is compared to the predicted power history in the final fuel cycle design. If the generic power history is violated, the EOL pin pressure is re-calculated using a new bounding power history.

2. Linear Heat Rate Capability

The fuel cannot exceed the temperature which would cause it to melt. Linear Heat Rate to Melt (LHRTM) limits are used to determine core protection limits which ensure that fuel melting will not occur. Duke performs a generic LHRTM analysis using the methodology described in Reference 2 on page 4-17.

TACO3 reduces the best estimate fuel temperature by a proprietary value which is based on comparison with measured data that inherently includes the effects of manufacturing variations, code predictions, transient fission gas release, and cladding oxide formation.

For each reload cycle, the generic power history is compared to the predicted power history in the final fuel cycle design. If the generic power history is violated, LHRTM is re-analyzed using a new bounding power history.

3. ECCS Analysis Interface Criteria

Duke reviews each batch of fuel and the fuel cycle design for compatibility with the vendor's fuel rod thermal analysis inputs to the ECCS analysis. Review criteria have been developed by Duke and have been reviewed and approved by the vendor.

Should the fuel rod thermal analysis inputs for a specific cycle lie outside the vendor's generic analysis, Duke will reperform the fuel rod thermal analysis to ensure that the results remain bounded by the results of the vendor's generic analysis. In the unlikely event that the cycle specific thermal analysis results (fuel temperature and pin pressure) are more limiting than the vendor's generic analysis, either the fuel cycle design must be modified or the vendor must resolve the concern within the vendor's ECCS analysis. Responsibility for identification of incompatibility and resolution lies with Duke.

## 4.2.4 FUEL ASSEMBLY, CONTROL ROD ASSEMBLY, AND CONTROL ROD DRIVE MECHANICAL TESTS AND INSPECTION

To demonstrate the mechanical adequacy and safety of the fuel assembly, control rod assembly (CRA), and control rod drive, a number of functional tests have been performed.

### 4.2.4.1 Prototype Testing

A full-scale prototype fuel assembly, CRA, and control rod drive have been tested in the Control Rod Drive Line (CRDL) Facility located at the B&W Research Center, Alliance, Ohio (Reference 3 on page 4-17). This full-sized loop is capable of simulating reactor environmental conditions of pressure, temperature, and coolant flow. To verify the mechanical design, operating compatibility, and characteristics of the entire control rod drive fuel assembly system, the drive was stroked and tripped approximately 200 percent of the expected operating life requirements.

A portion of the testing was performed with maximum misalignment conditions. Equipment was available to record and verify data such as fuel assembly pressure drop, vibration characteristics, and hydraulic forces and to demonstrate control rod drive operation and verify scram times. All prototype components were examined periodically for signs of material fretting, wear, and vibration/ fatigue to insure that the mechanical design of the equipment met reactor operating requirements.

The Type C prototype drive mechanism used on Oconee 3 was tested at Diamond Power Specialty Corporation, Lancaster, Ohio (Reference 3 on page 4-17). This consisted of component testing, a 100 percent misalignment life test (equivalent to 20 year operation), and motor performance tests. Throughout these tests the drive components were examined for material fretting, wear and vibrational fatigue.

### 4.2.4.2 Model Testing

Many functional improvements have been incorporated in the design of the fuel assembly as a result of model tests. For example, the spacer grid to fuel rod contact area was fabricated to ten times reactor size and tested in a loop simulating the coolant flow Reynolds number of interest. Thus, visually, the shape of the fuel rod support areas was optimized with respect to minimizing the severity of flow vortices and pressure drop. A 9-rod (3 x 3) assembly using stainless steel spacer grid material has been tested at reactor conditions (640°F, 2,200 psi, 13 fps coolant flow) for 210 days. Two full sized canned fuel assemblies with stainless steel spacer grids have been tested at reactor conditions, one for 40 days and the other for 22 days. A prototype canless fuel assembly using Inconel 718 spacer grids has been tested for approximately 90 days, approximately half of that time at reactor conditions. The principal objectives of these tests were to evaluate fuel assembly and fuel rod vibration and/or fretting wear resulting from flow-induced vibration. Vibratory amplitudes have been found to be very small, and, with the exception of a few isolated instances which are attributed to pretest spacer grid damage, no unacceptable wear has been observed.

### 4.2.4.3 Component and/or Material Testing

#### 4.2.4.3.1 Fuel Rod Cladding

5 Refer to Appendix B of Reference 5 on page 4-17 for a detailed report of externally pressurized fuel rod creep collapse tests.

#### 4.2.4.3.2 Fuel Assembly Structural Components

The structural characteristics of the fuel assemblies which are pertinent to loadings resulting from normal operation, handling, earthquake, and accident conditions are investigated experimentally in test facilities such as the CRDL Facility. Structural characteristics such as natural frequency and damping are determined at the relatively high (up to approximately 0.300 in.) amplitude of interest in the seismic and LOCA analyses. Natural frequencies and amplitudes resulting from flow-induced vibration are measured at various temperatures and flow velocities, up to reactor operating conditions.

#### 4.2.4.3.3 B&W Fuel Surveillance Program

B&W conducts various test programs aimed at obtaining fundamental engineering data on fuel and control components for design, manufacturing, and licensing support. The extensive previous operating history and detailed fuel surveillance confirms the basic soundness of the B&W fuel design. The operation of all B&W fuel will continue to be closely monitored to ensure continued safe and reliable fuel performance.

### 4.2.4.4 Control Rod Drive Tests and Inspection

#### 4.2.4.4.1 Control Rod Drive Developmental Tests

The testing and development program for the roller nut drive has been completed. The prototype drive was tested at the B&W Research Center at Alliance, Ohio. Wear characteristics of critical components have indicated that material compatibility and structural design of these components would be adequate for the design life of the mechanism. The trip time for the mechanism as determined under test conditions of reactor temperature, pressure, and flow was well within the specification requirements.

The Type C prototype drive was tested at the Diamond Power Specialty Corporation, Lancaster, Ohio (Reference 3 on page 4-17).

## 4.2.5 REFERENCES


1. T. Miles, D. Mitchell, G. Meyer, and L. Hassenpflug, Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, B&W, *BAW-10084P-A*, Rev. 3, Lynchburg, Va., July 1995.
2. DPC-NE-2008, Duke Power Company Fuel Mechanical Reload Analysis Methodology using TACO3.
3. J. T. Williams, R. E. Harris, and John Ficor, Control Rod Drive Mechanism Test Program, Revision 3, B&W, *BAW-10029A*, Rev. 3, Lynchburg, Va., August 1976.
4. TACO3 Fuel Pin Thermal Analysis Code, *BAW-10162P-A,* Lynchburg, VA, November 1989.
5. A.F.J. Eckert, H.W. Wilson, and K.E. Yoon, Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, B&W, BAW-10084P-A, Rev. 2, Lynchburg, VA, October 1978.
6. Correlation of 15 x 15 Geometry Zircaloy Grid Rod Bundle CHF Data with the BWC Correlation, B&W, *BAW-10143, Part 2,* Lynchburg, Va., March 1980.
7. DPC-NE-2007P-A, Duke Power Company Fuel Reconstitution Analysis Methodology, October 1995.

# 4.3 NUCLEAR DESIGN

The reactor core is designed to operate at 2568 MWt with sufficient nuclear design margins to accommodate transient operation without damage to the core. The core design characteristics are given in Table 4-1.

Core reactivity is controlled by control rod assemblies (CRA), soluble boron in the coolant, and burnable poison rod assemblies (BPRA). Sufficient CRA worth is available to shut down the reactor with at least a 1% $\Delta k/k$ subcritical margin in the hot condition at any time during the cycle with the most reactive CRA stuck in the fully withdrawn position. Equipment is provided to add soluble boron to the reactor coolant to ensure a similar shutdown capability when the reactor is cooled to ambient temperatures.

The reactivity worth of a CRA and the rate at which reactivity can be added are limited to ensure that credible reactivity accidents cannot cause a transient capable of damaging the RCS or causing significant fuel failure.

## 4.3.1 DESIGN BASES - NUCLEAR DESIGN

5 The core has been designed to the following nuclear limits and capabilities, all of which are intended to
5 preserve the integrity of the fuel assemblies:

5 1. The core will have sufficient reactivity to produce the design power level and lifetime without
5 exceeding the control capacity or shutdown margin.

5 2. Fuel assemblies have been designed for the maximum burnups shown in Table 4-2. If they are not
5 bounded, acceptable reanalyses shall be performed.

5 3. Power histories must be bounded by those assumed within generic mechanical and thermal hydraulic
5 (fuel assembly) analyses. If they are not bounded, acceptable reanalyses shall be performed.

5 4. The maximum feed fuel enrichment is constrained by the maximum allowed in the Technical
5 Specifications (Spent Fuel Pool storage requirements).

5 5. Values of important core safety parameters predicted for the cycle have been verified to be
5 conservative with respect to their values assumed in the Chapter 15 safety/accident (and any other
5 pertinent) analyses. If they are not conservative, acceptable reanalyses shall be performed.

5 Controlled reactivity insertion rates due to a sinle CRA group withdrawal shall be limited to a
5 maximum value assumed within the Chapter 15 Rod Withdrawal Accident at Rated Power, and
5 within the Chapter 15 Startup Accident. Controlled reactivity insertion rates due to soluble boron
5 removal shall be limited to a maximum value assumed within the Chapter 15 Moderator Dilution
5 Accident.

5 The power Doppler and moderator temperature coefficients at power will be negative. However, as
5 described within Chapter 15, the control system is capable of compensating for reactivity changes
5 resulting from either positive or negative nuclear coefficients.

5 6. Reasonable and permissive reactor control and maneuvering procedures during nominal operation and
5 during transients will not produce peak-to-average power distributions greater than those listed in
5 Table 4-1. This, along with criteria 7 and 8, below, preserves the LOCA linear heat rate, linear heat
5 rate to melt (LHRTM), and DNBR limits.

5 7. Part length axial power shaping rods (APSRs) are to be utilized to allow the shaping of power axially
5 in the core, thereby thwarting any tendency towards axial instability resulting from a redistribution of
5 xenon.

5 To preclude the possiblity of azimuthal instability resulting from a redistribution of xenon, the highest
5 moderator temperature coefficient assumed within the Chapter 15 safety/accident analyses must be
5 bounded by the threshold listed within Table 4-7.

5 8. Technical Specification limits of specified operating parameters (quadrant power tilt, power imbalance,
5 and control rod insertion), and on reactor protective system trip setpoints (power imbalance) after
5 allowance for appropriate measurement tolerances should have adequate margin from design limits of
5 these parameters during operational conditions throughout the cycle such that sufficient operating
5 flexibility is retained for the fuel cycle.

## 4.3.2 DESCRIPTION - NUCLEAR DESIGN

A summary of the nuclear characteristics of the core is given in Table 4-3.

### 4.3.2.1 Excess Reactivity

The Oconee reactor cores are designed with sufficient excess reactivity to yield the desired cycle length. This excess reactivity is controlled by soluble boron, burnable poison rod assemblies (BPRA), and control rod assemblies (CRA).

5 Generally, the nuclear designer makes an engineering trade-off between soluble boron and burnable poison rods to assure that the BOC moderator coefficient for power levels above 95 percent Hot Full Power (HFP) is nonpositive. Table 4-4 shows a typical eighteen month fuel cycle's excess reactivity at various conditions.

Table 4-5 shows the k-effective calculated for a single fuel assembly. The minimum critical mass, with and without xenon and samarium poisoning, may be specified as a single assembly or as multiple assemblies in various geometric arrays. The unit fuel assembly has been investigated for comparative purposes. A single cold, clean assembly containing a maximum probable enrichment of 3.5 weight per cent is subcritical. Two assemblies side-by-side are supercritical under these conditions.

### 4.3.2.2 Reactivity Control

The excess reactivity is controlled by a combination of soluble boron, lumped burnable poison, and control rods. Long term decreases in reactivity caused by fuel burnup are offset by decreases in soluble boron concentration and decreases in burnable poison worth. Short term reactivity effects are controlled by changes in control rod position.

Soluble Boron

Figure 4-5 illustrates a typical variation of soluble boron versus cycle length of an eighteen month fuel cycle. The change in boron concentration accounts for depletion of the fuel and is also a function of the BPRA loading and burnout.

Burnable Poison Rod Assemblies (BPRAs)

Figure 4-6 shows a typical burnable poison loading and enrichment scheme for an eighteen month fuel cycle. The BPRAs burnout as the fuel depletes and at end of cycle have a small residual reactivity effect caused by structural materials and water displacement effects.

The BPRA loadings and placement are chosen to shape radial power peaks and to decrease initial soluble boron concentration to a level where the BOC moderator temperature coefficient is non-positive. Since the BPRA assemblies are located in the control rod guide tubes, they cannot be placed in rodded locations. In addition, they will usually be in fresh fuel assemblies. See Section 4.5.2.4, "Burnable Poison Rod Assembly (BPRA)" on page 4-58 for a physical description of the BPRAs. See the appropriate reload design change report for actual BPRA loadings for any particular cycle.

Control Rod Assemblies

Oconee has 61 full length control rods assigned to seven control rod groups (1 to 7). Groups 1 to 4 are designated safety banks and are maintained out of the core above HZP. Groups 5 to 7 are designated control banks and may be inserted to pre-established limits shown in the Technical Specifications between HZP and HFP.

A typical control rod pattern is shown in Figure 4-7. The groupings of control rods into the various rod groups can vary with reload cycle and reference to the appropriate reload design change report should be made for the particular pattern being used for a particular cycle. In addition to being able to shut the reactor down, full length control rods are used to control reactivity changes caused by power level changes, transient xenon, and small periodic boron dilution changes.

Oconee has 8 Axial Power Shaping Rods (APSRs) which are always assigned to Group 8. These rods do not insert upon reactor trip and are used for axial power shaping and can be used to damp axial xenon oscillations.

### 4.3.2.3 Reactivity Shutdown Analysis

The ability to shut down the core from any operating condition by 1% $\Delta\rho$ is a Technical Specification requirement. This is accomplished by analytical calculations during the reload design and rod index limits are set such that at least a 1% $\Delta\rho$ shutdown margin is available for a trip from any allowable operating condition.

Table 4-6 illustrates a shutdown margin calculation for a typical Oconee fuel cycle. Conservatisms include a worth reduction penalty for control rod burnup and a 10 percent rod worth uncertainty. The flux redistribution effect is included if the power deficit was calculated with a two-dimensional code. This item does not need to be shown in a shutdown margin table if a three-dimensional calculation of power deficit was performed.

A detailed discussion of the calculation of the remaining parameters in Table 4-6 can be found in Reference 1 on page 4-39 and Reference 2 on page 4-39.

For the shutdown margin calculation for a particular reload cycle refer to the bases behind the appropriate reload design change report.

### 4.3.2.4 Reactivity Coefficients

Reactivity coefficients form the basis for studies involving normal and abnormal reactor operating conditions. These coefficients have been investigated as part of the analysis of this core and are described below as to function and overall range of values.

#### 4.3.2.4.1 Doppler Coefficient

The Doppler coefficient reflects the change in reactivity as a function of fuel temperature. The Doppler coefficient of reactivity is due primarily to Doppler broadening of the U-238 resonances with increasing fuel temperature. A rise in fuel temperature results in an increase in the effective absorption cross section of the fuel and a corresponding reduction in neutron production. A typical range for the Doppler coefficient under operating conditions would be $-1.1 \times 10^{-5}$ to $-1.7 \times 10^{-5}$ $(\Delta\rho)$/deg F.

#### 4.3.2.4.2 Moderator Void Coefficient

The moderator void coefficient relates the change in neutron multiplication to the presence of voids in the moderator. The expected range for the void coefficient is shown in Figure 4-8.

#### 4.3.2.4.3 Moderator Pressure Coefficient

The moderator pressure coefficient relates the change in moderator density, resulting from a reactor coolant pressure change, to the corresponding effect on neutron production. This coefficient is opposite in sign and considerably smaller when compared to the moderator temperature coefficient. A typical range of pressure coefficients over a life cycle would be $-1.4 \times 10^{-7}$ to $+3 \times 10^{-6}$ $(\Delta\rho)$/psi.

#### 4.3.2.4.4 Moderator Temperature Coefficient

The moderator temperature coefficient relates a change in neutron multiplication to the change in reactor coolant temperature. Reactors using soluble boron as a reactivity control have a less negative moderator temperature coefficient than do cores controlled solely by movable or fixed CRA. The major temperature effect on the coolant is a change in density. An increasing coolant temperature produces a decrease in water density and an equal percentage reduction in boron concentration. The boron concentration change results in a positive reactivity component by reducing the absorption in the coolant. The magnitude of this component is proportional to the total reactivity held by soluble boron. Distributed poisons (burnable poison rods or inserted control rods) have a negative effect on the moderator coefficient for a system with 1200 ppm boron and no rods inserted. Depending on the core size, core loading, and power density, a plant may or may not require additional distributed poisons to yield the appropriate moderator temperature coefficient as determined by the safety analysis and the stability analysis of the core. An example of this, as pertaining to the first cycle, is illustrated in Table 4-7.

Items 4d on page 4-8 and 6 on page 4-8 in Table 4-7 above reflect three dimensional calculations using thermal feedback. These coefficients are more negative than the two-dimensional isothermal values previously calculated and shown. It is seen from comparison (Table 4-7, Table 4-8, Table 4-9) that three-dimensional spatially distributed effects are important in the determination of reactivity coefficients.

The three-dimensional PDQ07 calculation with thermal feedback was also used to calculate for Oconee 1 Cycle 1 the change in spatially dependent moderator coefficient for changes in inlet, outlet, and core average moderator temperature ($°F_m$), as shown in Table 4-8.

The Oconee reactors operate on a constant core average moderator temperature with both inlet and outlet temperature changing with power level. The core average moderator temperature as seen by the control system is defined to be

$$T_m = \frac{T_{in} + T_{out}}{2}$$

The BOL distributed temperature moderator coefficients for different reactor power levels are presented in Table 4-9 for Oconee 1, Cycle 1, and for a typical reload cycle with three dimensional codes PDQ07 and

SIMULATE-3P, respectively, and both with thermal feedback. These coefficients were found by changing both inlet and outlet temperatures. Criticality in each case was attained by appropriate control rod insertion for Oconee 1, Cycle 1, and by boron for the typical reload cycle.

The moderator temperature coefficient was also calculated for the equilibrium xenon condition at the beginning of the fuel cycle. The calculation assumed 2.1% $\Delta\rho$ in control rods for Oconee 1, Cycle 1; boron search was used for a typical reload cycle. The 100% power moderator coefficient varied in the manner shown in Table 4-10.

The EOL coefficient was calculated for a change in both the inlet and outlet temperatures with a boron concentration of 17 ppm. The coefficient for 100% power was found to be:

$$\alpha_m = -2.8 \times 10^{-4} \frac{\Delta\rho}{°F_m}$$

This, then, is the "rods out" moderator coefficient at the end of the first fuel cycle for Oconee 1, Cycle 1.

The coefficients reported in Table 4-8 and Table 4-9 (Oconee 1, Cycle 1) are for a core containing 2.1 percent $\Delta\rho$ in control rods. A "rods out" calculation for the beginning of life moderator conditions in Item 2, Table 4-8 was performed as a basis for comparison and the result was

$$\alpha_m = +0.52 \times 10^{-4} \Delta\rho/°F_m$$

An examination of the data in Table 4-7 shows that the limiting factor on a moderator coefficient is the value used during Oconee 1, Cycle 1 safety analysis, i.e., $+0.9 \times 10^{-4}\ \Delta\rho/°F_m$. The margin between this value and the nominal calculated value of $+0.27 \times 10^{-4}\ \Delta\rho/°F_m$ is considered adequate to cover uncertainties.

#### 4.3.2.4.5 Power Coefficient

The power coefficient, $\alpha p$, is the fractional change in neutron multiplication per unit change in core power level. A number of factors contribute to $\alpha p$, but only the moderator temperature coefficient and the Doppler coefficient contributions are significant. The power coefficient can be written as:

$$\alpha_p = \alpha_m \frac{\partial T_m}{\partial P} + \alpha_f \frac{\partial T_f}{\partial P}$$

where:

$\alpha_m$ = moderator temperature coefficient

$\alpha_f$ = fuel Doppler coefficient

$\frac{\partial T_m}{\partial P}, \frac{\partial T_f}{\partial P}$ = change in moderator and fuel temperature per unit change in core power.

Power coefficients were calculated for Oconee 1, Cycle 1 and for a typical reload cycle at BOL (time zero) at various power levels. For Oconee 1, Cycle 1, a boron concentration of 1200 ppm was used for all power levels, and criticality was achieved with control rods. For a typical reload cycle, boron search was used for all power levels, and criticality was achieved by boron. The three-dimensional codes PDQ07 and SIMULATE-3P, both with thermal feedback, were used to include the effects of spatially distributed fuel and moderator temperatures.

5 The results are presented in Table 4-11.

#### 4.3.2.4.6 pH Coefficient

Currently, there is no definite correlation which will permit prediction of pH reactivity effects. Some of the parameters needing correlation are the effects relating pH reactivity change for various operating reactors, pH effects versus reactor operating time at power, and changes in effects with varying clad, temperature, and water chemistry. Yankee, Saxton, and Indian Power Station 1 have experienced reactivity changes at the time of pH changes, but there is no clear-cut evidence that pH is the direct reactivity influencing variable without considering other items such as clad materials, fuel assembly crud deposition, system average temperature, and prior system water chemistry.

The pH characteristic of this design is shown below in Table 4-12 where the cold values are measured and the hot values are calculated.

Saxton experiments (Reference 3 on page 4-39) have indicated a pH reactivity effect of 0.0016 $\Delta\rho/\Delta pH$ unit change with and without local boiling in the core. Considering system makeup rate of 35,000 lb/h and the core in the hot condition with 1,200 ppm boron in the coolant, the corresponding changes in pH are 0.02 pH units per hour for boron dilution and 0.05 pH units per hour for $^{7}Li$ dilution (starting with 0.5 ppm $^{7}Li$). Applying the pH worth value quoted above from Saxton, the total reactivity insertion rate for the hot condition is $3.1 \times 10^{-8}$ $\Delta\rho$/sec. This insertion rate or reactivity can be easily compensated by the operator or the automatic control system.

### 4.3.2.5 Reactivity Insertion Rates

5 Figure 4-10 displays the typical integrated rod worth of three overlapping rod banks as a function of
5 distance withdrawn. The indicated groups are those used in the core during power operation. Using an
5 assumed nominal of 1.2% $\Delta\rho$ CRA groups and an assumed 30 in./min CRA drive speed in conjunction
5 with the reactivity response given in Figure 4-10 yields a maximum reactivity insertion rate of $1.09 \times 10^{-4}$
5 $(\Delta\rho)$/sec. The maximum reactivity insertion rate for soluble boron removal, using an assumed boron
5 dilution rate of 500 GPM, is $0.16 \times 10^{-4}$ $(\Delta\rho)$/sec.

### 4.3.2.6 Power Decay Curves

Figure 4-11 displays the beginning-of-life power decay curves for the CRA worths corresponding to the 1 percent hot shutdown margin with and without a stuck rod. The power decay is initiated by the trip of the CRA with a 300 msec delay from initiation to start of CRA motion. The time required for insertion of a CRA 2/3 of the distance into the core is 1.4 sec.

## 4.3.3 NUCLEAR EVALUATION

5 The nuclear evaluation for a fuel cycle design is composed of the preliminary fuel cycle design, the final
5 fuel cycle design, safety analysis physics parameters, maneuvering analysis, core operating limits (Technical
5 Specifications) calculation, final core loading map calculation, and core monitoring parameters calculation.

5 The preliminary fuel cycle design determines the number and enrichment of the fresh fuel to be inserted
5 for a given cycle.

5 The final fuel cycle design uses the models discussed in Section 4.3.3.1, "Analytical Models" on page 4-27
5 to optimize the placement of fresh and burned fuel assemblies, control rod groupings, and BPRA (if any)
5 to result in an acceptable fuel design. It must meet the following current design criteria with appropriate
5 reductions to account for calculational uncertainties:

1. Operate to the scheduled end-of-cycle (EOC) plus ten days, with a minimum boron (typically 10 ppmb) remaining at the EOC midpoint.
2. The U235 fuel enrichment must be bounded by that listed within the Technical Specifications (Spent Fuel Pool storage requirements).
3. Maximum pin burnup must be bounded by the appropriate limit for a fuel type.
4. Maximum assembly average burnup must be bounded by the appropriate limit for a fuel type.
5. The power histories must be bounded by those used in generic analyses.
6. For the current bypass flow assumptions, the typical number of 44 BPRAs gives sufficient margin.

During the safety analysis physics parameters, a number of physics parameters are calculated and are verified as conservative with respect to those assumed within the Chapter 15 safety/accident analyses. These include, but are not limited to, the following:

1. Moderator temperature coefficient
2. Doppler coefficient
3. Ejected rod worth
4. Dropped rod worth
5. Total/maximum CRA group worth
6. Kinetics parameters
7. Shutdown margin
8. Maximum reactivity insertion rates (due to controlled rod withdrawal and boron dilution)
9. Differential boron worth
10. Boron concentrations

The purpose of a maneuvering analysis is to generate three dimensional power distributions, rod positions, and imbalances for a variety of reasonable and permissive rod positions, xenon distributions, and power levels. The maneuvering analysis can be described as four discrete phases. The first is the nominal fuel cycle depletion performed at a nominal rod index (typically, rod index = 292 and APSRs at 35% withdrawn) to establish a fuel depletion history. The second is the power maneuver performed at BOC (4 EPFD), at EOC (with appropriate adjustments to ensure critical conditions), and at least one other point in between; APSRs are positioned as necessary to maintain xenon control and to maintain predetermined imbalance limits. The third is to perform control rod and APSR scans at the most severe times of the power maneuver. The fourth step is to perform selected control rod and APSR scans at various nominal depletion steps. Each of these phases involves running multiple three dimensional cases and generation of three dimensional power distributions, rod positions, and imbalances for each case. The data is processed by utility codes to calculate margins to LHRTM, DNBR, and LOCA limiting criteria, and to produce 'fly-speck' plots, Application of appropriate calculational conservatisms are described within Reference 2 on page 4-39. Note that the derivations of the LHRTM, DNBR, and LOCA limiting criteria have been bounded by limiting power distribution listed within Table 4-1.

In addition, the initial rod positions assumed within the following safety parameters must be bounded by the rod insertion limits determined during the maneuvering analysis:

1. Shutdown margin at HZP, BOC to EOC $\geq$ 1.0% $\Delta\rho$ (with the most reactive CRA stuck in the fully withdrawn position).

2. Maximum ejected rod worth at HZP, NoXe, BOC and EOC, as bounded by that assumed within the Chapter 15 Rod Ejection Accident.
3. Maximum ejected rod worth at HFP, EqXe, BOC and EOC, as bounded by that assumed within the Chapter 15 Rod Ejection Accident.
4. Maximim dropped rod worth at HFP, NoXe, EOC, as bounded by that assumed within the Chapter 15 Control Rod Misalignment Accident.
5. Maximum dropped rod worth HFP, EqXe, EOC, as bounded by that assumed within the Chapter 15 Control Rod Misalignment Accident.

During core operating limits calculation, data from 'fly-speck' plots generated during the maneuvering analysis are used to set limits on operational alarm setpoints (quadrant power tilt, control rod insertion, power-imbalance), and reactor protective system trip setpoints (power-imbalance). In addition, limits on control rod insertion based on the shutdown margin and required boron concentrations within the control system equipment are developed (or retrieved from appropriate sources). These limits are chosen such that sufficient operating flexibility is retained for the fuel cycle, while maintaining sufficient margin to design and safety criteria. Thse limits are set according to the allowances for appropriate measurement tolerances and uncertainties, which include, but are not limited to the following:

1. In-core detector system (observability and variability) and in-core monitoring software uncertainties
2. Out-of-core to In-core calibration/correlation uncertainty
3. Control rod position uncertainties
4. Flux-flow ratio adjustment
5. Reactor protective system hardware uncertainties
6. Boron concentration and volume uncertainties

The following Technical Specification limit is presumed as being met by the startup physics test program criteria for moderator temperature coefficient (which must be less than $+0.5 \times 10^{-4}$ $\Delta\rho$/deg F):

1. Moderator temperature coefficient $\leq$ 0.0 at > 95% hot full power.

Table 4-9 for a typical reload design, generated with SIMULATE-3P, shows that this presumption is valid. The moderator temperature coefficient (MTC), when changing conditions from BOC, HZP, NoXe, and ARO to BOC, 95%fp, NoXe, and ARO, becomes negative; note that the 95% no xenon condition is conservative. Given the change, and startup physics test program criteria of $+0.5 \times 10^{-4}$ $\Delta\rho$/deg F, the MTC at 95%fp is negative (approximately $-0.20 \times 10^{-4}$ $\Delta\rho$/deg F).

During final core loading map calculation, placement of fuel assemblies and related core components in a reload core are determined. Special considerations such as even distributions of fresh fuel loadings and BPRA poison loadings are taken into account to minimize the possibility of an asymmetric or tilted core which would perturb the assumptions and predictions made during the fuel cycle design process.

During core monitoring parameters calculation, certain physics parameters are calculated to enable an orderly and safe startup of the cycle, to perform the startup physics test program, and to perform corefollow calculations. Other physics parameters are used to update the in-core monitoring software residing within the plant process computer. The in-core monitoring software monitors the quadrant tilt, power imbalance, and rod positions, and actuates alarms if these parameters violate the operational limits. Periodic calculations are also done to verify the existence of the 1.0% $\Delta\rho$ shutdown margin, and to check predictions versus measured data. As such, monitoring of core performance during cycle operation confirms the validity of predictions and ensures that design and safety criteria are satisfied.

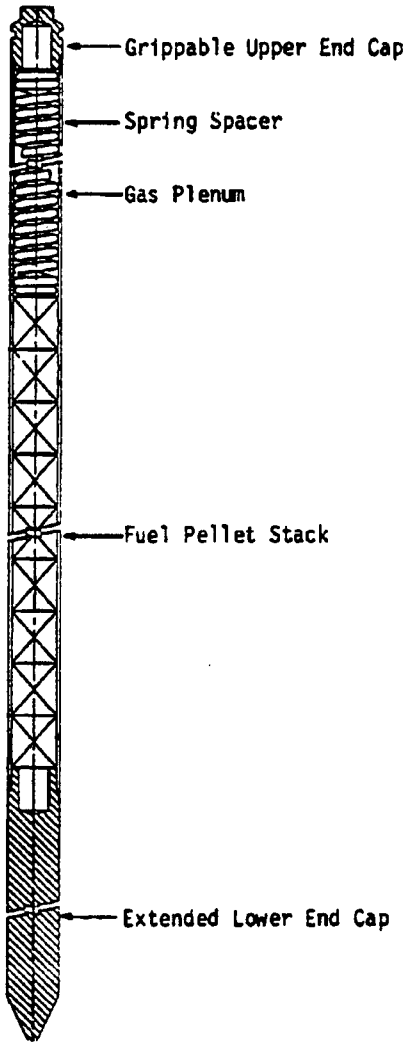


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Figure 4-4.
Pressurized Fuel Rod

B & W FUEL ASSEMBLY MARK B–6, B–7, B–8

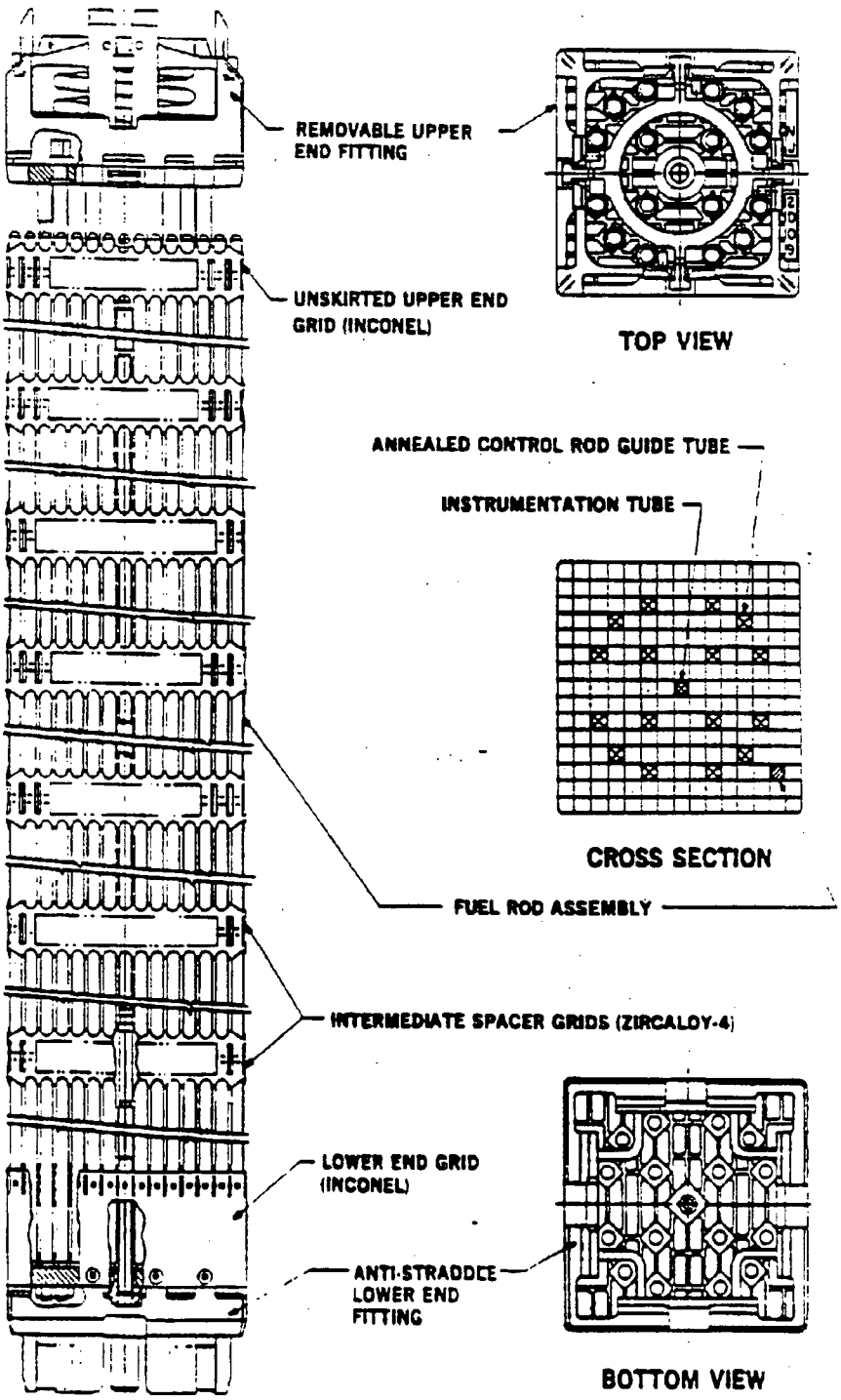


Figure 4-3.
Fuel Assembly (Typical)

**Table 4-19. Axial Power Shaping Rod Assembly Data**

| Item | Data |
|---|---|
| Gray APSR Design | |
| Number of Axial Power Shaping Rod Assemblies | 8 |
| Number of Axial Power Shaping Rods per Assembly | 16 |
| Outside Diameter of Axial Power Shaping Rod, in. | 0.440 |
| Cladding Thickness, in. | 0.027 |
| Cladding Material | Type 304 SS, Stainless Steel Cold-Worked |
| Plug Material | Type 304 SS, Annealed |
| Poison Material | Inconel - 600 |
| Spider Material | SS Grade CF3M |
| Female Coupling Material | Type 304 SS, Annealed |
| Length of Poison Section, in. | 63 |
| Stroke of Control Rod, in. | 139 |

**Table 4-20. Burnable Poison Rod Assembly Data**

| Item | Data |
|---|---|
| Number of Burnable Poison Rods per Assembly | 16 |
| Outside Diameter of Burnable Poison Rod, in. | 0.430 |
| Cladding Thickness, in. | 0.035 |
| Cladding Material | Zircaloy-4, Cold Worked |
| End Cap Material | Zircaloy-4, Annealed |
| Poison Material | $B_4C$ in. $Al_2O_3$ |
| Length of Poison Section, in. | 126 |
| Spider Material | SS Grade CF3M |
| Coupling Mechanism Material | Type 304 SS, Annealed |

0 **Table 4-14. Typical Thermal-Hydraulic Design Conditions**

| Oconee Unit | Typical MK-BZ |
|---|---|
| Power level, MWt | 2568 |
| System pressure, psia | 2200 |
| Reactor coolant flow, % design flow | 107.5 |
| Vessel inlet coolant temp, 100% power, °F | 555.8 |
| Vessel outlet coolant temp, 100% power, °F | 602.2 |
| Ref design axial flux shape | 1.5 cos |
| Ref design radial-local power peaking factor | 1.71 |
| Active fuel length, in. | 141.75 |
| Average heat flux, 100% power, $10^3$ Btu/hr-ft$^2$ | 176[a] |
| CHF correlation | BWC |
| CHF correlation limit | 1.18 |
| Hot channel factors | |
| Power Peaking Factor ($F_q$) | 1.013 |
| Flow area | 0.97 |

**Note:**

(a) Based on densified length of 141.75 in.

**Table 4-2 (Page 2 of 2). Fuel Assembly Components**

| | Item | Material | Dimensions (In) |
|---|---|---|---|
| 1 | MK B-4Z | | > 47,000 |
| 1 | MK B-5, B-5Z | | > 50,000 |
| 5 | MK B-6, B-7, B-8, B-9, B-10, MK B-10 Axial Blanket, MK B-10F | | > 50,000 |

**Note:**

(1) Typical geometry. Batch specific geometry is reported in individual reload reports.

**Table 4-3. Nuclear Design Data**

| | | Oconee I | Oconee II | Oconee III |
|---|---|---|---|---|
| | Fuel Assembly Volume Fractions | | | |
| | Fuel | 0.303 | 0.303 | 0.303 |
| | Moderator | 0.580 | 0.580 | 0.580 |
| | Zircaloy | 0.102 | 0.102 | 0.102 |
| | Stainless Steel | 0.003 | 0.003 | 0.003 |
| | Void | 0.012 | 0.012 | 0.012 |
| | | 1.000 | 1.000 | 1.000 |
| 5 | Total $UO_2$ (Metric Tons) | | | |
| 5 | First Cycle | 94.1 | 93.1 | 93.1 |
| 5 | Equilibrium | 97.9 | 97.9 | 97.9 |
| | Core Dimensions, in. | | | |
| | Equivalent Diameter | 128.9 | 128.9 | 128.9 |
| 5 | Nominal Active Height | 142.3 | 142.3 | 142.3 |
| | Unit Cell $H_2O$ to U Atomic Ratio (Fuel Assembly) | | | |
| | Cold | 2.85 | 2.88 | 2.88 |
| | Hot | 2.04 | 2.06 | 2.06 |
| | Full-Power Lifetime, Days | | | |
| | First Cycle | 309 | 440 | 479 |
| 5 | Equilibrium Cycle | 480 | 480 | 480 |
| | Fuel Irradiation, MWD/MTU | | | |
| | First Cycle Average | 9,569 | 13,774 | 14,995 |
| 5 | Equilibrium Cycle Average | 14,290 | 14,290 | 14,290 |
| | Fuel Loading, wt% U-235 | | | |
| | Core Average First Cycle | 2.10 | 2.62 | 2.56 |
| | First Reload Average | 3.15 | 2.64 | 2.54 |
| 5 | Typical Core Average Equilibrium Cycle | 3.70 | 3.70 | 3.70 |
| | Control Data | | | |
| | Control Rod Material | Ag-In-Cd | Ag-In-Cd | Ag-In-Cd |
| | Number of Full Length CRA's | 61 | 61 | 61 |
| | APSR Material | INC-600 | INC-600 | INC-600 |
| | Number of APSR's | 8 | 8 | 8 |
| | Control Rod Cladding Material | SS 304 | SS 304 | SS 304 |

**Table 4-2 (Page 1 of 2). Fuel Assembly Components**

| | Item | Material | Dimensions (In) |
|---|---|---|---|
| | Fuel Rod: | | |
| 1 | Fuel | $UO_2$ Sintered Pellets | |
| 1 | | MK B-8 and earlier | 0.3686 Dia. |
| 3 | | MK B-9 and B-10 | 0.3700 Dia. |
| 5 | | MK B-10F | 0.3735 Dia. |
| 5 | Fuel Clad | | |
| 5<br>5 | MK B-10 & earlier | Zircaloy-4 | 0.430 OD x 0.377 ID<br>x 153 - 1/8 long |
| 5<br>5 | MK B-10F | Zircaloy-4 | 0.430 OD x 0.380 ID<br>x 150.82 long |
| | Fuel Rod Pitch | | 0.568 |
| 1 | Active Fuel Length | | |
| 1 | | MK B-8 and earlier | 141.8 |
| 4<br>4 | | MK B-9 and B-10<br>(Non-blanket) | 140.6 |
| 4 | | MK B-10 Axial Blanket | 140.7 |
| 5 | | MK B-10F | 142.3 |
| | Fuel Assembly: | | |
| | Fuel Assembly Pitch | | 8.587 |
| | Overall Length | | 165-5/8 |
| | Control Rod Guide Tube | Zircaloy-4 | 0.530 OD x 0.016 wall |
| | Instrumentation Tube | Zircaloy-4 | 0.493 OD x 0.441 ID |
| | End Fittings | Stainless Steel<br>(Castings) | |
| 5 | End Spacer Grid | | |
| 5<br>5 | MK B-8 & earlier | Inconel-718 Strips | 0.020 thick exteriors<br>0.016 thick interiors |
| 5<br>5 | MK B-9 to MK B-10F | Zircaloy-4 | 0.020 thick exteriors<br>0.018 thick interiors |
| 5 | Intermediate Spacer Grid | Zircaloy-4 | 0.021 thick exteriors<br>0.018 thick interiors |
| | Spacer Sleeve | Zircaloy-4 | 0.554 OD x 0.502 ID |
| | Fuel Assembly Design: | | Maximum Assembly Burnup (MWD/MTU) |
| | MK B-4 | | > 43,000 |

**Table 4-1 (Page 3 of 3). Core Design, Thermal, and Hydraulic Data**

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**Note:**

5 1. Typical results only.
0 2. Based on reference peaking and densified length of 141.75 inches as applicable.
0 3. Based on 107.5 percent design flow and 9.0 percent bypass flow as applicable.

**Table 4-1 (Page 2 of 3). Core Design, Thermal, and Hydraulic Data**

| | | |
|---|---|---|
| 4 | MK B-10 Axial Blanket | 140.7 |
| 5 | MK B-10F | 142.3 |
| | Density, % of Theoretical | |
| 5 | MK B-10 and earlier | 95.0 |
| 5 | MK B-10F | 96.0 |
| | Heat Transfer and Fluid Flow at Rated Power | |
| | Total Heat Transfer Surface in Core, $ft^2$ | 48,608 (2) |
| | Average Heat Flux, Btu/hr-$ft^2$ | $175 \times 10^3$ (2) |
| | Maximum Heat Flux, Btu/hr-$ft^2$ | $450 \times 10^3$ (2) |
| | Average Power Density in Core, kW/$\ell$ | 85.31 (2) |
| | Average Thermal Output, kW/ft of Fuel Rod | 5.8 (2) |
| | Maximum Thermal Output, kW/ft of Fuel Rod | 14.9 (2) |
| 0 | Average Core Fuel Temperature, °F | 1240 (2) |
| 0 | Total Reactor Coolant Flow, lb/hr | $139.6 \times 10^6$ (3) |
| | Core Flow Area (Effective for Heat Transfer), $ft^2$ | 49.19 |
| 0 | Core Coolant Average Velocity, fps | 16.22 (3) |
| | Power Distribution | |
| | Maximum/Average Power Ratio, Radial x Local ($F_{\Delta h}$ Nuclear) | 1.71 |
| | Maximum/Average Power Ratio, Axial ($F_z$ Nuclear) | 1.5 |
| | Overall Power Ratio ($F_q$ Nuclear) | 2.57 |
| | Power Generated in Fuel and Cladding, % | 97.3 |
| | Hot Channel Factors | |
| 4 | Power Peaking Factor ($F_Q$) | 1.013 |
| | Flow Area Reduction Factor ($F_A$) | |
| 0 | Unit/CRGT Bundle Cells | 0.98 |
| 0 | IGT Bundle Cells | 0.97 |
| 4 | | |
| | Hot Spot Maximum/Average Heat Flux Ratio | |
| | ($F_q$ nuc and mech) | 2.63 |
| | DNB Data | |
| | Design Overpower (% Rated Power) | 112 |
| 0 | | |
| 4 | Limiting DNB Ratio (BWC) Mark - BZ Fuel | 1.18 |

# APPENDIX 4. CHAPTER 4 TABLES AND FIGURES

**Table 4-1 (Page 1 of 3). Core Design, Thermal, and Hydraulic Data**

| | | |
|---|---|---|
| | Reactor | |
| | Rated Heat Output, MWt | 2,568 |
| | Vessel Coolant Inlet Temperature, °F | 555.6 |
| | Vessel Coolant Outlet Temperature, °F | 602.4 |
| | Core Outlet Temperature, °F | 606.2 |
| | Core Operating Pressure, psig | 2,185 |
| | Core and Fuel Assemblies | |
| | Total Number of Fuel Assemblies in Core | 177 |
| | Number of Fuel Rods per Fuel Assembly | 208 |
| | Number of Control Rod Guide Tubes per Assembly | 16 |
| | Number of In-Core Instrumentation Positions per Fuel Assembly | 1 |
| | Fuel Rod Outside Diameter, in. | 0.430 |
| | Clad Thickness, in. | |
| 5 | MK B-10 and earlier | 0.0265 |
| 5 | MK B-10F | 0.0250 |
| | Fuel Rod Pitch, in. | 0.568 |
| | Fuel Assembly Pitch Spacing, in. | 8.587 |
| | Unit Cell Metal/Water Ratio (Volume Basis) | 0.82 |
| | Clad Material | Zircaloy-4 (Cold Worked) |
| | Fuel | |
| | Material | $UO_2$ |
| | Form | Dished-End, Cylindrical Pellets |
| 1 | Pellet Diameter, in. | |
| 1 | MK B-8 and earlier | 0.3686 |
| 3 | MK B-9 and B-10 | 0.3700 |
| 5 | MK B-10F | 0.3735 |
| 1 | Active Length, in. | |
| 1 | MK B-8 and earlier | 141.8 |
| 4 | MK B-9 and B-10 (Non-blanket) | 140.6 |

## 4.5.5 REFERENCES

1. E. O. Hooker, H. J. Fortune, "Design of Reactor Internals and Incore Instrument Nozzles for Flow Induced Vibrations," B&W, *BAW-10051*, Lynchburg, VA., September 1972.
2. "Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake," B&W, *BAW-10008, Part 1, Revision 1*, Lynchburg, VA., June 1970.
3. R. V. DeMars, R. R. Steinke, "Fuel Assembly Stress and Deflection Analysis for Loss-of-Coolant Accident and Seismic Excitation," B&W, *BAW-10035*, Lynchburg, VA., January 1972.
4. "Internals Vent Valve Evaluation," B&W, BAW-10005, Revision 1, Lynchburg, VA., June 1970.
5. James T. Williams, R. E. Harris, John Ficor, "Control Rod Drive Mechanism," B&W, *BAW-10029A*, *Revision 3*, Lynchburg, VA., August 1976.

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operation, the most significant indication of such a failure would be a change in the free-disc motion as a result of altered rotational clearances.

1 **4.5.4.2.4 Handling Test**

A prototype valve assembly was successfully installed and removed remotely in a test stand to confirm the adequacy of the vent valve handling tool.

1 **4.5.4.2.5 Closing Force Test**

A 1/6 scale model valve disc closing force (excluding gravity) test is described in Section 4.4.4, "Thermal and Hydraulic Tests and Inspection" on page 4-44.

1 **4.5.4.2.6 Vibration Testing**

The full-size prototype valve's response to vibration was determined experimentally to verify prior analytical results which indicated that the valve disc would not move relative to the body seal face as a result of vibration caused by transmission of core support shield vibrations. The prototype valve was mounted in a test fixture which duplicated the method of valve mounting in the core support shield. The test fixture with valve installed was attached to a vibration test machine and excited sinusoidally through a range of frequencies which encompassed those which may reasonably be anticipated for the core support shield during reactor operation. The relative motion between the valve disc and seat was monitored and recorded during test. The test results indicated that there was no relative motion of the valve to its seat for conditions simulating operating conditions. After no relative motion was observed or recorded during test, the valve disc was manually forced open during test to observe its response. The disc closed with impact on its seat, rebounded open and reseated without any adverse affects to valve seal surfaces, characteristics, or performance. From this oscillograph record, the natural frequency of the valve disc was conservatively calculated as approximately 1500 cps; whereas, the range of frequencies for the Oconee system (including internals components) has been established as 15 to 160 cps.

These frequencies are separated by an ample margin to conclude that no relative motion between the valve disc and its seal will occur during normal reactor operation.

Each production valve will be subjected to tests (2) and (3) above except that no additional analysis will be performed in conjunction with test (2).

The valve disc, hinge shaft, shaft journals (bushings), disc journal receptacles, and valve body journal receptacles are designed to withstand without failure the internal and external differential pressure loadings reresulting from a loss-of-coolant accident. These valve materials will be nondestructively tested and accepted in accordance with the ASME Code III requirements for Class A vessels as a reference quality level.

During scheduled refueling outages after the reactor vessel head and the internals plenum assembly have been removed, the vent valves are accessible for visual and mechanical inspection. A hook tool is provided to engage with the valve disc exercise lug described in Item 7 on page 4-53 With the aid of this tool, the valve disc will be manually exercised to evaluate the disc freedom. The hinge design incorporates special features, as described in Item 7 on page 4-53, minimize the possibility of valve disc motion impairment during its service life. With the aid of the hook tool, the valve disc can be raised and a remote visual inspection of the valve body and disc sealing faces can be performed for evaluation of observed surface irregularities.

Remote installation and removal of the vent valve assemblies if required is performed with the aid of the vent valve handling tool which includes unlocking and operating features for the retaining ring jackscrews.

An inspection of hinge parts is not planned until such time as a valve assembly is removed because its free-disc motion has been impaired. In the unlikely event that a hinge part should fail during normal

#### 4.5.4.1.4 Visual (5X Magnification) Examination

This examination is performed in accordance with and results accepted on the basis of a B&W Quality Control Specification which complies with NAV-SHIPS 250-1500-1. Each entire weld pass and adjacent base metal are inspected prior to the next pass from the root to and including the cover passes.

1. Partial penetration non-radiographically or non-ultrasonically feasible structural weld joints are 100 percent inspected to the above specification.
2. Partial or full penetration attachment weld joints for nonstructural materials or parts are 100 percent inspected to the above specification.
3. Partial or full penetration weld joints for attachment of mechanical devices which lock and retain structural fasteners.
4. Personnel conducting these examinations are trained and qualified.

After completion of shop fabrication, the internals components are shopfitted and assembled to final design requirements. The assembled internals components undergo a final shop fitting and alignment of the internals with the "as built" dimensions of the reactor vessel. Dummy fuel and CRAs are used to insure that ample clearances exist between the fuel and internals structures guide tubes to allow free movement of the CRA throughout its full stroke length in various core locations. Fuel assembly mating fit is checked at all core locations. The dummy fuel and CRAs are identical to the production components except that they are manufactured to the most adverse tolerance space envelope, and they contain no fissionable or absorber materials.

All internal components can be removed from the reactor vessel to allow inspection of all vessel interior surfaces. Internals components surfaces can be inspected when the internals are removed to the canal underwater storage location.

### 4.5.4.2 Internals Vent Valves Tests and Inspection

The internals vent valves are designed to relieve the pressure generated by steaming in the core following a LOCA so that the core will remain sufficiently cooled. The valves were designed to withstand the forces resulting from rupture of either a reactor coolant inlet or outlet pipe. To verify the structural adequacy of the valves to withstand the pressure forces and perform the venting function, the following tests were performed:

1 #### 4.5.4.2.1 Hydrostatic Testing

A full-size prototype valve assembly (valve disc retaining mechanism and valve body) was hydrostatically tested to the maximum pressure expected to result during the blowdown.

1 #### 4.5.4.2.2 Frictional Load Tests

Sufficient tests were conducted at zero pressure to determine the frictional loads in the hinge assembly, the inertia of the valve disc, and the disc rebound resulting from impact of the disc on the seat so that the valve response to cyclic blowdown forces may be determined analytically.

1 #### 4.5.4.2.3 Pressure Testing

A prototype valve was pressurized to determine the pressure differential required to cause the valve disc to begin to open. A determination of the pressure differential required to open the valve disc to its maximum open position was simulated by mechanical means.

The effects of internals misalignment was evaluated on the basis of the test results from the CRDL tests described in Section 4.2.4.4, "Control Rod Drive Tests and Inspection" on page 4-16. These test results, correlated with the internals guide tube design, insure that the CRA can be inserted at specified rates under conditions of maximum misalignment.

Internals shop fabrication quality control tests, inspection, procedures, and methods are similar to those for the pressure vessel described in Section 5.2.3.11, "Quality Assurance" on page 5-32. The internals surveillance specimen holder tubes and the material irradiation program are described in Section 5.2.3.13, "Reactor Vessel Material Surveillance Program" on page 5-35.

A listing is included herewith for all internals nondestructive examinations and inspections with applicable codes or standards applicable to all core structural support material of various forms. In addition, one or more of these examinations are performed on materials or processes which are used for functions other than structural support (i.e. alignment dowels, etc.) so that virtually 100 percent of the completed internals materials and parts are included in the listing. Internals raw materials are purchased to ASME Code Section II or ASTM material specifications. Certified material test reports are obtained and retained to substantiate the material chemical and physical properties. All internals materials are purchased and obtained to a low cobalt limitation. The ASME Code Section III, as applicable for Class A vessels, is generally specified as the requirement for reference level nondestructive examination and acceptance. In isolated instances when ASME III cannot be applied, the appropriate ASTM Specifications for non-destructive testing are imposed. All welders performing weld operations on internals are qualified in accordance with ASME Code Section IX applicable Edition and Addenda. The primary purpose of the following list of non-destructive tests is to locate, define, and determine the size of material defects to allow an evaluation of defect, acceptance, rejection, or repair. Repaired defects are similarly inspected as required by applicable codes.

#### 4.5.4.1.1 Ultrasonic Examination

1. Wrought or forged raw material forms are 100 percent inspected throughout the entire material volume to ASME III, Class A.
2. Personnel conducting these examinations are trained and qualified.

#### 4.5.4.1.2 Radiographic Examination (includes X-ray or radioactive sources)

1. Cast raw material forms are 100 percent inspected to ASME III Class A or ASTM.
2. All circumferential full penetration structural weld joints which support the core are 100 percent inspected to ASME III Class A.
3. All radiographs are reviewed by qualified personnel who are trained in their interpretation.

#### 4.5.4.1.3 Liquid Penetrant Examination

1. Cast form raw material surfaces are 100 percent inspected to ASME III Class A or ASTM.
2. Full penetration non-radiographic or partial penetration structural welds are inspected by examination of root, and cover passes to ASME III Class A.
3. All circumferential full penetration structural weld joints which support the core have cover passes inspected to ASME III Class A.
4. Personnel conducting these examinations are trained and qualified.

5. Lead Screw

   Refer to Section 4.5.3.1.4, "CRDM Subassemblies" on page 4-60.

6. Rotor Assembly

   Refer to Section 4.5.3.1.4, "CRDM Subassemblies" on page 4-60.

7. Torque Tube and Torque Taker

   The torque tube is a separate tubular assembly containing a key that extends the full length of the leadscrew travel. The tube assembly is secured in elevation and against rotation at the lower end of the closure assembly by a retaining ring, keys and the insert closure. The lower end of the torque tube houses the snubber assembly and is the down stop. The leadscrew contacts the insert closure assembly for the upper mechanical stop.

   The torque taker assembly consists of the position indicator permanent magnet, the snubber piston and a positioning keyway. The torque taker assembly is attached to the top of the leadscrew and has a keyway that mates with the key in the torque tube to provide both radial and tangential positioning of the leadscrew.

8. Snubber Assembly

   The total snubber assembly is composed of a piston that is the lower end of the torque taker assembly and a snubber cylinder and belleville spring assembly which is attached to the lower end of the torque tube. The snubber cylinder is closed at the bottom by the snubber bushing and leadscrew. The snubber cylinder has a twelve-inch active length in which the free-fall tripped leadscrew and control rod assembly is decelerated without applying greater than ten times gravitational force on the control rod. The damping characteristics of the snubber is determined by the size and position of a number of holes in the snubber cylinder wall and the leakage at the snubber piston and bushing. Leakage reduction at the snubber piston and bushing can only be reduced to a minimum amount caused by practical operating clearances. Therefore, at the end of the snubbing stroke, there is kinetic energy from a five foot per second impact velocity that is absorbed by the belleville spring assembly by a slight instantaneous overtravel past the normal down stop.

9. Lead Screw Guide

   Refer to Section 4.5.3.1.4, "CRDM Subassemblies" on page 4-60.

10. Position Indications

    Refer to Section 4.5.3.1.4, "CRDM Subassemblies" on page 4-60.

11. Motor Tube Design Criteria

    Refer to Section 4.5.3.1.4, "CRDM Subassemblies" on page 4-60.

The Type C mechanism is presented in Figure 4-33 and is described in Reference 5 on page 4-69.

## 4.5.4 INTERNALS TESTS AND INSPECTIONS

### 4.5.4.1 Reactor Internals

The hydraulic design of the upper and lower plena of the internals is evaluated and guided by the results from the 1/6 scale model flow test which is described in Section 4.4.4, "Thermal and Hydraulic Tests and Inspection" on page 4-44. These test results have guided the design to obtain minimum flow maldistribution, and the test data allowed verification of vessel flow and pressure drop.

Quality standards relative to material selection, fabrication, and inspection are specified to insure safety function of the housings essential to accident prevention. Materials conform to ASTM or ASME, Section II, Material Specifications. All welding shall be performed by personnel qualified under ASME Code, Section IX, Welding Qualifications. These design and fabrication procedures establish quality assurance of the assemblies to contain the reactor coolant safely at operating temperature and pressure.

In the highly unlikely event that a pressure barrier component or the control rod drive assembly does fail catastrophically, i.e., ruptured completely, the following results would ensue:

a. Control Rod Drive Nozzle

The assembly would be ejected upward as a missile until it was stopped by the missile shield over the reactor. This upward motion would have no adverse effect on adjacent assemblies.

b. Motor Tube

The failure of this component anywhere above the lower flange would result in a missile-like ejection into the missile shielding over the reactor. This upward motion would have no adverse effect on adjacent mechanisms.

12. Axial Power Shaping Rod Drive

For actuating the partial length control rods which maintain their set position during a reactor-trip of the shim safety drive, the CRDM is modified so that the roller nut assembly will not disengage from the lead screw on a loss of power to the stator. Except for this modification, the shim drives and the axial power shaping rod drives are identical.

## 4.5.3.2 Type C Mechanisms

### 4.5.3.2.1 Shim Safety Drive Mechanism

The Type C shim safety drive mechanism consists of a motor tube which houses a torque tube, a leadscrew, its rotor assembly, and a snubber assembly. The top end of the motor tube is closed by a closure and vent assembly. An external motor stator surrounds the motor tube (a pressure housing) and position indication switches are arranged outside the motor tube extension.

### 4.5.3.2.2 CRDM Subassemblies

Those parts of the Type C CRDM subassemblies which are different from the Type A CRDM (Section 4.5.3.1.4, "CRDM Subassemblies" on page 4-60) are described below:

1. Motor Tube

That portion of the motor tube wall between the rotor assembly and the stator is constructed of martensitic stainless steel.

2. Motor

The stator is a 48-slot four-pole arrangement with water cooling coils in the outside casing. The stator is varnish impregnated after winding to establish a sealed unit.

3. Plug and Vent Valve

Refer to Section 4.5.3.1.4, "CRDM Subassemblies" on page 4-60.

4. Actuator

Refer to Section 4.5.3.1.4, "CRDM Subassemblies" on page 4-60.

the upper end of the motor tube extension. The lower end of the tube assembly supports the buffer and is the down stop. A set of indexing serrations mate to prevent rotation and orient the torque extension tube with the motor tube below the cap and vent valve assembly. An integral shoulder at the top of the tube rests against a step in the motor tube inside diameter to provide a vertical support.

The torque taker assembly consists of the position indicator permanent magnet, the buffer piston, and a positioning key. The torque taker key fixed at the top of the lead screw is mated with the torque extension tube keyway to provide both radial and tangential positioning of the lead screw.

8. Buffer

The buffer assembly is capable of decelerating the translating mass from the unpressurized terminal velocity to zero velocity without applying greater than ten times the gravitational force on the control rod. The water buffer consists of a piston fixed to the top end of the screw shaft and a cylinder which is fixed to the lower end of the torque extension tube. Twelve inches above the bottom stop, the piston at the top of the screw enters the cylinder. Guiding is accomplished because the piston and torque key are in a single part, and cylinder and keyway are in a single mating part. As the piston travels into the cylinder, water is driven into the center of the lead screw through holes in the upper section which produce the damping pressure drop. The number of holes presented to the buffer chamber is reduced as the rod moves into the core, so that the damping coefficient increases as the velocity reduces, thereby providing an approximately uniform deceleration. A large helical buffer spring is used to take the kinetic energy of the drive line at the end of the water buffer stroke. The buffer spring accepts a five foot per second impact velocity of the drive line and control rod with an instantaneous overtravel of one inch past the normal down stop. The inclusion of this buffer spring permits practical clearances in the water buffer.

9. Lead Screw Guide

The lead screw guide bushing acts as a primary thermal barrier and as a guide for the screw shaft. As a primary thermal barrier, the bushing allows only a small path for free convection of water between the mechanism and the closure head nozzle. Fluid temperature in the mechanism is largely governed by the flow of water up and down through this bushing. The diametrical clearance between screw shaft and bushing is large enough to preclude jamming the screw shaft and small enough to hold the free convection to an acceptable value. In order to obtain trip travel times of acceptably small values, it is necessary to provide an auxiliary flow path around the guide bushing. The larger area path is necessary to reduce the pressure differential required to drive water into the mechanism to equal the screw displacement. The auxiliary flow paths are closed for small pressure differentials (several inches of water) by ball check valves which prevent the convection flow but, open fully during trip.

10. Position Indications

Two methods of position indication are provided: an absolute position indicator and a relative position indicator. The absolute position transducer consists of a series of magnetically operated reed switches mounted in a tube parallel to the motor tube extension. Each switch is hermetically sealed. Switch contacts close when a permanent magnet mounted on the upper end of the lead screw extension comes in close proximity. As the lead screw (and the control rod assembly) moves,
2 switches operate sequentially producing an analog voltage proportional to position. Additional reed switches are included in the same tube with the absolute position transducer to provide full withdrawal and insertion signals. The relative position indicator consists of a small pulse-stepping
2 motor driving a potentiometer that generates a signal proportional to the position demand for the rod
2 as indicated by the number of power pulses received by the rod drive motor.

11. Motor Tube Design Criteria

The motor tube design complies with Section III of the ASME Boiler and Pressure Vessel Code for a Class A vessel. The operating transient cycles, which are considered for the stress analysis of the reactor pressure vessel, are also considered in the motor tube design.

The motor tube is a three-piece welded assembly designed and manufactured in accordance with the requirements of the ASME Code, Section III, for Class A nuclear pressure vessel. Materials conform to ASTM or ASME, Section II, Material Specifications. All welding shall be performed by personnel qualified under ASME Code, Section IX, Welding Qualifications. The motor tube wall between the rotor assembly and the stator is constructed of magnetic material to present a small air gap to the motor. This region of the motor tube is of low alloy steel clad on the inside diameter with stainless steel or with Inconel. The upper end of the motor tube functions only as a pressurized enclosure for the withdrawn lead screw and is made of stainless steel transition welded to the upper end of the low alloy steel motor section. The lower end of the low alloy steel tube section is welded to a stainless steel machined forging which is flanged at the face which contacts the vessel control rod nozzle. Double gaskets, which are separated by a ported test annulus, seal the flanged connection between the motor tube and the reactor vessel.

2. Motor

The motor is a synchronous reluctance unit with a slip-on stator. The rotor assembly is described in 6 below. The stator is a 48-slot four-pole arrangement with water cooling coils wound on the outside of its casing. The stator is encapsulated after winding to establish a sealed unit. It is six phase star-connected for operation in a pulse-stepping mode and advances 15 mechanical degrees per step. The stator assembly is mounted over the motor tube housing as shown in Figure 4-34.

3. Plug and Vent Valve

The upper end of the motor tube is closed by a closure insert assembly containing a vapor bleed port and vent valve. The vent valve and insert closure have double seals. The insert closure is retained by a closure nut which is threaded to the inside of the motor tube. The sealing for the closure is applied by jackscrews threaded through the closure nut.

4. Actuator

The actuator consists of the translating lead screw, its rotating nut assembly, and the torque taker assembly on the screw. The actuator lead screw travel is 139 inches.

5. Lead Screw

The lead screw has a lead of 0.750 in. The thread is double lead with a single pitch spacing of 0.375 in. Thread lead error is held to close tolerances for uniform loading with the roller nut assemblies. The thread form is a modified ACME with a blank angle that allows the roller nut disengage without lifting the screw.

6. Rotor Assembly

The rotor assembly consists of a ball bearing supported rotor tube carrying and limiting the travel of a pair of scissors arms. Each of the two arms carry a pair of ball bearing supported roller (nut) assemblies which are skewed at the lead screw helix angle for engagement with the lead screw. The current in the motor stator (two of a six winding stator) causes the arms that are pivoted in the rotor tube to move radially toward the motor tube wall to the limit provided thereby engaging the four roller nuts with the centrally located lead screw. Also, four separating springs mounted in the scissor arms keep the rollers disengaged when the power is removed from the stator coils. A second radial bearing mounted to the upper end of the rotor tube has its outer race pinned to both scissor arms thereby synchronizing their motion during engagement and disengagement. When a three phase rotating magnetic field is applied to the motor stator, the resulting force produces rotor assembly rotation.

7. Torque Extension Tube and Torque Taker

The torque extension tube is a separate tubular assembly containing a keyway that extends the full length of the lead screw travel. The tube assembly is supported against rotation and in elevation by

Continuous position indication, as well as an upper and lower position limit indication, shall be provided for each CRDM. The accuracy of the position indicators shall be consistent with the tolerance set by reactor safety analysis.

5. Drive Speed

   The control rod drive control system shall provide a single uniform mechanism speed. The drive controls, or mechanism and motor combination, shall have an inherent speed limiting feature. The speed of the mechanism shall be 30 in./min for both insertion and withdrawal. The withdrawal speed shall be limited to not exceed 25 percent overspeed in the event of speed control fault.

6. Mechanical Stops

   Each CRDM shall have positive mechanical stops at both ends of the stroke or travel. The stops shall be capable of receiving the full operating force of the mechanisms without failure.

7. Control Rod Positioning

   The control rod drives shall provide for controlled withdrawal or insertion of the control rods out of, or into, the reactor core to establish and hold the power level required.

#### 4.5.3.1.2 Additional Design Criteria

The following criterion is applicable only to the mechanisms which actuate control rod assemblies: The shim safety drives are capable of rapid insertion or trip for emergency reactor conditions.

#### 4.5.3.1.3 Shim Safety Drive Mechanism

The shim safety drive mechanism consists of a motor tube which houses a lead screw and its rotor assembly, and a buffer. The top end of the motor tube is closed by a closure and vent assembly. An external motor stator surrounds the motor tube (a pressure housing) and position indication switches are arranged outside the motor tube extension.

The control rod drive output element is a non-rotating translating lead screw coupled to the control rod. The screw is driven by separating anti-friction roller nut assemblies which are rotated magnetically by a motor stator located outside the pressure boundary. Current impressed on the stator causes the separating roller nut assembly halves to close and engage the lead screw. Mechanical springs disengage the roller nut halves from the screw in the absence of a current. For rapid insertion, the nut halves separate to release the screw and control rod, which move into the core by gravity. A hydraulic buffer assembly within the upper housing decelerates the moving CRA to a low speed a short distance above the CRA full-in position. The final CRA deceleration energy is absorbed by the down-stop buffer spring. The CRDM is a totally sealed unit with the roller nut assemblies magnetically driven by the stator coil through the motor tube pressure housing wall. The lead screw assembly is connected to the control rod by a bayonet type coupling. An anti-rotation device (torque taker) prevents rotation of the lead screw while the drive is in service. A closure and vent assembly is provided at the top of the motor tube housing to permit access to couple and release the lead screw assembly from the control rod. The top end of the lead screw assembly is guided by the buffer piston and its guide. Two of the six phase stator housing windings are energized to maintain the control rod position when the drive is in the holding mode.

#### 4.5.3.1.4 CRDM Subassemblies

The CRDM is shown in Figure 4-34 and Figure 4-35. Subassemblies of the CRDM are described as follows:

1. Motor Tube

The burnable poison rod is clad in cold-worked Zircaloy-4 tubing and Zircaloy-4 upper and lower end pieces. The end pieces are welded to the tubing to form a water and pressure-tight container for the absorber material. The Zircaloy-4 tubing provides the structural strength of the burnable poison rods.

In addition to their nuclear function, the BPRA also serve to minimize guide tube bypass coolant flow. Pertinent data on the BPRA is shown in Table 4-20.

The burnable poison rods are designed to withstand all operating loads including those resulting from hydraulic forces and thermal gradients. The ability of the burnable poison rod clad to resist collapse due to the system pressure and internal pressure has been demonstrated by an extensive test program on cold-worked Zircaloy-4 tubing (Section 4.2.4.3.1, "Fuel Rod Cladding" on page 4-15).

## 4.5.3 CONTROL ROD DRIVES

Oconee 3 uses the Type C control rod drive mechanism in contrast to Oconee 1 and 2 which use Type A mechanisms. Both types are sealed, reluctance motor-driven screw units and the design requirements are identical. Section 4.5.3.1, "Type A Mechanisms" describes Type A mechanisms, and Section 4.5.3.2, "Type C Mechanisms" on page 4-63 describes Type C mechanisms.

### 4.5.3.1 Type A Mechanisms

The control rod drive mechanism (CRDM) positions the control rod within the reactor core, provides for controlled withdrawal or insertion of the control rod assemblies, is capable of rapid insertion or trip, and indicates the location of the control rod with respect to the reactor core. The speed at which the control rod 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 shim safety drive mechanism releases the CRA and supporting CRDM components permitting the CRA to move by gravity into the core. The reactivity is reduced during such a rod insertion at a rate sufficient to control the core under any operating transient or accident condition. The control rod is decelerated at the end of the rod trip insertion by a buffer assembly in the CRDM upper housing. The buffer assembly supports the control rod in the fully inserted position. The CRDM data is listed in Table 4-21, and criteria applicable to drive mechanisms for both control shim rod assemblies and axial power shaping rod assemblies are given below. Additional requirements for the mechanisms which actuate only control shim rod assemblies are also given below.

#### 4.5.3.1.1 General Design Criteria

1. Single Failure

   No single failure shall inhibit the protective action of the control rod drive system. The effect of a single failure shall be limited to one CRDM.

2. Uncontrolled Withdrawal

   No single failure or sequence of dependent failures shall cause uncontrolled withdrawal of any control rod assembly (CRA).

3. Equipment Removal

   The disconnection of plug-in connectors, modules, and subassemblies from the protective circuits shall be annunciated or shall cause a reactor trip.

4. Position Indication

### 4.5.2.3 Axial Power Shaping Rod Assembly (APSRA)

Gray APSR's are provided for additional control of axial power distribution. Each axial power shaping rod assembly (Figure 4-32) has 16 axial power shaping rods, a stainless steel spider, and a female coupling. The 16 rods are attached to the spider by means of a nut threaded to the upper shank of each rod. After assembly all nuts are lock welded. The axial power shaping rod drive is coupled to the APSRA by a bayonet connection. The female couplings of the APSRA and CRA have slight dimensional differences to ensure that each type of rod can only be coupled to the correct type of drive mechanism.

When the APSRA is inserted into the fuel assembly it is guided by the guide tubes of the fuel assembly. Full length guidance of the APSRA is provided by the control rod guide tube of the upper plenum assembly. At the full out position of the control rod drive stroke, the lower end of the APSRA remains within the fuel assembly guide tube to maintain the continuity of guidance throughout the rod travel length. The APSRAs are designed to permit maximum conformity with the fuel assembly guide tube throughout travel.

Each axial power shaping rod has a section of neutron absorber material. For these gray APSRs, this absorber material is Inconel 600, and the clad is coldworked, Type 304 stainless steel tubing. The tubing provides the structural strength of the axial power shaping rods and prevents corrosion of the absorber material.

Gray APSRs are designed with improved creep life. Cladding thickness and rod ovality control, which are the primary factors controlling the creep life of a stainless steel material, have been improved to extend the creep life of the gray APSR. Minimum design cladding thickness is 25 mils.

The gray APSRs are prepressurized to extend their lifespan.

Pertinent data on gray APSRs is shown in Table 4-19.

These axial power shaping rods are designed to withstand all operating loads including those resulting from hydraulic forces and thermal gradients. The ability of the axial power shaping rod clad to resist collapse due to the system pressure has been established in a test program on cold worked stainless steel tubing. The absorber material does not yield gaseous products under irradiation, therefore, internal pressure is not generated within the clad. Swelling of the absorber material is negligible, and does not cause unacceptable clad strain.

Because of their great length and unavoidable lack of straightness, some slight mechanical interference between axial power shaping rods and the fuel assembly guide tubes must be expected. However, the parts involved are flexible and result in very small friction drag loads. Similarly, thermal distortions of the rods are small because of the low generation and adequate cooling. Consequently, the APSRAs do not encounter significant frictional resistance to their motion in the guide tubes.

### 4.5.2.4 Burnable Poison Rod Assembly (BPRA)

Each BPRA (Figure 4-1) has 16 burnable poison rods, a stainless steel spider, and a coupling mechanism. The coupling mechanism and the 16 rods are attached to the spider. The BPRA is inserted into the fuel assembly guide tubes through the upper end fitting. Retention is provided by the feet on the BPRA spider, which rest upon the fuel assembly holddown spring retainer ring. Thus the BPRA is pinned between this retainer ring and the reactor's upper grid pads. All Oconee fuel which is of the Mk B5 (or later) design, uses this BPRA design.

## 4.5.2 CORE COMPONENTS

This section addresses core components that are not an integral part of the fuel assembly itself. Specifically addressed are the following: control rod assembly, axial power shaping rod assembly, and burnable poison rod assembly.

### 4.5.2.1 Fuel Assemblies

The fuel system (fuel assembly and its components) is addressed in Section 4.2, "Fuel System Design" on page 4-5.

### 4.5.2.2 Control Rod Assembly (CRA)

Each control rod assembly (Figure 4-31) has 16 control rods, a stainless steel spider, and a female coupling. The 16 control rods are attached to the spider by means of a nut threaded to the upper shank of each rod. After assembly, all nuts are lock welded. The control rod drive is coupled to the CRA by a bayonet type connection. Full length guidance for the CRA is provided by the control rod guide tube of the upper plenum assembly and by the fuel assembly guide tubes. The CRAs and guide tubes are designed with adequate flexibility and clearances to permit freedom of motion within the fuel assembly guide tubes throughout the stroke.

Oconee 3, Cycle 8 introduced a new long life control rod assembly design. Future replacement CRAs for all units will be of this type. The extended life control rod assembly (CRA) is nearly identical to B&W's standard design. The present designed spider/coupling arrangement is retained as are all other envelope dimensions. Reference to Table 4-18, demonstrates the differences between the standard and the plant-life CRA design. The major differences are found in the slight reduction in the absorber OD and the use of Inconel 625 clad (as compared to the standard SS 304 material). Inconel 625 CRA cladding was selected because of its added creep and corrosion resistance. In addition, the rodlets are prepressurized with helium, and the cladding is slightly thicker to retard creepdown and ovalization.

Each control rod has a section of neutron absorber material. The absorber material is an alloy of silver-indium-cadmium. End pieces are welded to the tubing to form a water-tight and pressure-tight container for the absorber material.

Both the inconel and the stainless steel tubing provides the structural strength of the control rods and prevents corrosion of the absorber material. A tube spacer similar to that in the fuel assembly is used to prevent absorber motion within the cladding during shipping and handling, and to permit differential expansion in service.

These control rods are designed to withstand all operating loads including those resulting from hydraulic force, thermal gradients, and reactor trip deceleration. The ability of the control rod clad to resist collapse has been established in a test program on cold-worked stainless steel tubing. Because the Ag-In-Cd alloy poison does not yield a gaseous product under irradiation, internal pressure and swelling of the absorber material does not cause excessive stressing or stretching of the clad.

Because of their length and the possible lack of straightness over the entire length of the rod, some interference between control rods and the fuel assembly guide tubes is expected. However, the parts involved, especially the control rods, are flexible and only small friction drag loads result. Similarly, thermal distortions of the control rods are small because of the low heat generation and adequate cooling. Consequently, control rod assemblies do not encounter significant frictional resistance to their motion in the guide tubes.

2. The total deformation of lug from contact with the vessel wall until disc assembly motion is arrested is predicted to be 0.483 inches.
3. The total angular deformation at the plastic hinge is predicted to be 0.016 radians.
4. An analysis was performed on the reactor vessel wall for disc assembly impact and the results indicate that while the stainless steel cladding is deformed locally, the reactor maintains its structural and pressure boundary integrity.

Because of conservative assumptions used in the plastic analysis, actual deformations will be considerably less than the above predicted values. Although plastic deformation may occur as predicted above on impact, the disc will retain its structural integrity. Plastic deformation of the disc dissipates the stored kinetic energy stored in the disc effectively; thus the energy available for rebound is less than 1 percent of the initial impact energy and is too low to overcome the pressure differential and cause impact on the valve body. Disc and body hinge components were analyzed for worst case disc impact loadings and the resulting stresses were found to be less than the allowable limits; therefore, the valve disc free-motion (venting) function will be unaffected.

From the above, it is concluded that vent valve performance will not be impaired during the course of an accident because disc free-motion part stresses remain within allowable limits, disc structural integrity is maintained, vessel pressure boundary integrity is maintained, and plastic deformation of the disc seating surface improves the venting function.

With reference to Figure 4-30, each jackscrew assembly consists of a jackscrew, internally splined mating nut ring, nut ring spring, capture cover and cover attachment fasteners (socket head cap screws). In the figure, the splined nut ring and its spring are hidden from view by the capture cover. The potential for loss of jackscrew assembly parts during the plant lifetime is considered remote on the basis that the jackscrews and capture parts are accessible for visual inspection during scheduled refueling outages. A jackscrew loss is considered remote because a failure in service is highly improbable with the low compressive load (1000 psi) involved and the jack screw is retained in the valve body by a central shoulder and the ends are threaded into the retaining rings. An in-service failure of the splined nut ring and its spring is remote because these parts are subjected to little or no load and even if they did fail all parts would be retained within the capture cover. Capture cover failure and loss is highly improbable on the same basis that is it not loaded in service. The capture cover is attached to the upper retaining ring by socket head cap screws which are lock welded to the cover at installation. By design, these screws are retention rather structural devices and are not loaded in service. These screws do not require a pre-load to hold the formed cover in place; therefore, a loss of pre-load by lock welding would not jeopardize the cover or screw installation or structural integrity. Two fillet welds 180° apart are used to lock weld each screw head to the capture cover and in the absence of loads on both the cover and screws, the likelihood of lock weld failure and loss of screw heads is considered remote. With the capability to inventory these cap screw heads visually at scheduled refuelings, any problem related to the loss of these screws would be apparent early in the plant life and the valve assemblies could be removed for corrective action.

The internals vent valves are described, including materials and hinge part loose clearances in Table 4-17.

The internals vent valves have been tested for ability to withstand the effects of vibratory excitations and for other functional characteristics as described in Section 4.5.4, "Internals Tests and Inspections" on page 4-64.

No further jackscrew problems have occurred or are anticipated on the basis that the surfaces are separated by the low friction "Electrolyze," different materials of different hardnesses are used, loose fits are employed, and thread contact stresses are low (3775 psi).

The final design of this valve is shown in Figure 4-29. The valve disc hangs closed in its natural position to seal against a flat, stainless steel seat inclined 5 degrees from vertical to prevent flow from the inlet coolant annulus to the plenum assembly above the core. In the event of LOCA, the reverse pressure differential will open the valve. At all times during normal reactor operation, the pressure in the coolant annulus on the outside of the core support shield is greater than the pressure in the plenum assembly on the inside of the core support shield. Accordingly, the vent valve will be held closed during normal operation. With four reactor coolant pumps operating, the pressure differential is 42 psi resulting in a several-thousand pound closing force on the vent valve.

Under accident conditions, the valve will begin to open when a pressure differential of less than 0.15 psi develops in a direction opposite to the normal pressure differential. At this point, the opening force on
4 the valve counteracts the natural closing force of the valve. With an opening pressure differential of no greater than 0.3 psi, the valve would be fully open. With this pressure differential, the water level in the core would be above the top of the core. In order for the core to be half uncovered, assuming solid water in the bottom half of the core, a pressure differential of 3.7 psi would have to be developed. This would provide an opening force of about 10 times that required to open the valve completely. This is a conservative limit since it assumes equal density in the core and the annulus surrounding the core. The hot, steam-water mixture in the core will have a density much less than that of the cold water in the annulus, and somewhat greater pressure differentials could be tolerated before the core is more than half uncovered.

An analog computer simulation was developed to evaluate the performance of the vent valves in the core support shield. This analysis demonstrated that adequate steam relief exists so that core cooling will be accomplished.

The behavior of the valve disc during LOCA conditions was investigated and the rather complex dynamic behavior of the disc during LOCA was analyzed as a series of simpler models which provide conservative predictions of peak stresses and deflections.

The valve disc remains closed initially for the LOCA hot leg (36 in. pipe) case and the disc opening on subsequent differential pulses is less than one-half of the initial disc to vessel wall impact velocity for the LOCA cold leg (28 in. pipe) case. Therefore, the disc motion and initial impact with the vessel inside wall was chosen as the worst case and the only one requiring consideration. The cold-leg LOCA pressure time history acting on the disc was approximated by a piecewise linear time function. The momemt due to pressure was equated to the rotary inertia of the disc to determine the velocity of impact with the vessel inside wall.

The model chosen for the initial impact consisted of three effective springs and two masses to represent the disc with its lug, the compliance of the disc, and the vessel inside wall.

Loads generated on impact were based on the conservation of energy. The stresses obtained for these loads indicated that the elastic model assuming conservation of energy was not valid and that the impact must assume plastic deformation. The locations and modes of plastic deformation are illustrated in BAW-10005 (Reference 4 on page 4-69).

The plastic analysis provided the following information:

1. Crush deformation of lug after disc corner contacts the vessel wall is predicted to be 0.165 inches.

service, seven loose rotational clearances would remain to allow unhampered disc free motion. In the worst case, at least four clearances must bind or seize solidly to adversely affect the valve disc free motion.

In addition, the valve disc hinge loose clearances permit disc self-alignment so that the external differential pressure adjusts the disc seal face to the valve body seal face. This feature minimizes the possibility of increased leakage and pressure-induced deflection loadings on the hinge parts in service.

The external side of the disc is contoured to absorb the impact load of the disc on the reactor vessel inside wall without transmitting excessive impact loads to the hinge parts as a result of a loss-of-coolant accident.

### 4.5.1.4 Evaluation of Internals Vent Valve

A vapor lock problem could arise if water is trapped in the steam generator blocking the flow of steam from the top of the reactor vessel to a cold leg leak. Under this condition, the steam pressure at the top of the reactor would rise and force the steam bubbles through the water leg in the bottom of the steam generator. This same differential pressure that develops a water leg in the steam generator will develop a water leg in the reactor vessel which could lead to uncovering of the core.

The most direct solution to this problem is to equalize the pressure across the core support shield, thus eliminating the depression of the water level in the core. This was accomplished by installing vent valves in the core support shield to provide direct communication between the top of the core and the coolant inlet annulus. These vent valves open on a very low-pressure differential to allow steam generated in the core to flow directly to the leak from the reactor vessel. Although the flow path in the steam generator is blocked, this is of no consequence since there is an adequate flow path to remove the steam being generated in the core.

During the vent valve conceptual design phase, criteria were established for valves for this service. The design criteria were (1) functional integrity, (2) structural integrity, (3) remote handling capability, (4) individual part capture capability, (5) functional reliability, (6) structural reliability, and (7) leak integrity throughout the design life. The design criteria resulted in the selection of the hinged-disc (swing-disc) check valve, which was considered suitable for further development.

Because of the unique purpose and application of this valve, B&W recognized the need for a complete detailed design and development program to determine valve performance under nuclear service conditions. This program included both analytical and experimental methods of developing data. It was performed primarily by B&W and the selected valve vendor or his subcontractors.

Vent valve preliminary design drawings were prepared and analyzed both by B&W and the vendor/subcontractor. Specifications and drawings were prepared, and orders were placed with the vendor for the design, development, fabrication, and test of a full-size prototype vent valve. The prototype valve was completed and subjected to the tests described in Section 4.5.4, "Internals Tests and Inspections" on page 4-64. All testing was successfully completed and minor problems encountered during valve assembly handling or use were corrected to arrive at the final design for the production valve (Reference 4 on page 4-69).

The only significant problem encountered during test was seizing of one jack screw. This was attributable to an excessive thickness of "Electrolyze" which spalled off the screw threads. This problem was corrected by reducing the specified "Electrolyze" thickness from 0.0015 in. to 0.0004 in. max. and no further galling was encountered. To further enhance resistance to galling, the final design jackscrew has a 1-1/8 in.-8 Acme thread form instead of a 1 in.-12 UNF and the material is an age hardened corrosion resistant alloy instead of 410 SS.

A perforated flat plate located midway between the two lattice structures aids in distributing coolant flow prior to entrance into the core. Alignment between fuel assemblies and incore instruments is provided by pads bolted to the upper perforated plate.

4. Flow Distributor

The flow distributor is a perforated dished head with an external flange which is bolted to the bottom flange of the lower grid. The flow distributor supports the incore instrument guide tubes and distributes the inlet coolant entering the bottom of the core.

5. Thermal Shield

A cylindrical stainless steel thermal shield is installed in the annulus between the core barrel cylinder and reactor vessel inner wall. The thermal shield reduces the incident gamma absorption internal heat generation in the reactor vessel wall and thereby reduces the resulting thermal stresses. The thermal shield upper end is restrained against inward and outward vibratory motion by restraints bolted to the core barrel cylinder. The lower end of the thermal shield is shrunk fit on the lower grid flange and secured by 96 high strength bolts.

6. Incore Instrument Guide Tube Assembly

The incore instrument guide tube assemblies guide the incore instrument assemblies from the instrument penetrations in the reactor vessel bottom head to the instrument tubes in the fuel assemblies. Horizontal clearances are provided between the reactor vessel instrument penetrations and the instrument guide tubes in the flow distributor to accommodate misalignment. Fifty-two incore instrument guide tubes are provided and are designed so they will not be affected by the core drop described in Section 4.5.1.3, "Description - Reactor Internals" on page 4-50.

7. Internals Vent Valves

Internals vent valves are installed in the core support shield to prevent a pressure imbalance which might interfere with core cooling following a postulated inlet pipe rupture. Under all normal operating conditions, the vent valve will be closed. In the event of the pipe rupture in the cold leg of the reactor loop, the valve will open to permit steam generated in the core to flow directly to the leak, and will permit the core to be rapidly recovered and adequately cooled after emergency core coolant has been supplied to the reactor vessel. The design of the internals vent valve is shown in Figure 4-29 and Figure 4-30.

Each valve assembly consists of a hinged disc, valve body with sealing surfaces, split-retaining ring, and fasteners. Each valve assembly is installed into a machined mounting ring integrally welded in the core support shield wall. The mounting ring contains the necessary features to retain and seal the perimeter of the valve assembly. Also, the mounting ring includes an alignment device to maintain the correct orientation of the valve assembly for hinged-disc operation. Each valve assembly will be remotely handled as a unit for removal or installation. Valve component parts, including the disc, are of captured design to minimize the possibility of loss of parts to the coolant system, and all operating fasteners include a positive locking device. The hinged-disc includes a device for remote inspection of disc function. Vent valve materials are listed in Table 4-16.

The vent valve materials were selected on the basis of their corrosion resistance, surface hardness, antigalling characteristics, and compatibility with mating materials in the reactor coolant environment.

The arrangement consists of eight 14-in. inside diameter vent valve assemblies installed in the cylindrical wall of the internals core support shield (refer to Figure 4-26). The valve centers are coplanar and are 42 in. above the plane of the reactor vessel coolant nozzle centers. In cross section, the valves are spaced around the circumference of the core support shield wall.

The hinge assembly provides eight loose rotational clearances to minimize any possibility of impairment of disc-free motion in service. In the event that one rotational clearance should bind in

The plenum cylinder consists of a large cylindrical section with flanges on both ends to connect the cylinder to the plenum cover and the upper grid. Holes in the plenum cylinder provide a flow path for the coolant water. The upper grid consists of a perforated plate which locates the lower end of the individual CRA guide tube assembly relative to the upper end of a corresponding fuel assembly. The grid is bolted to the plenum cylinder lower flange. Locating keyways in the plenum assembly cover flange engage the reactor vessel flange locating keys to align the plenum assembly with the reactor vessel, the reactor closure head control rod drive penetrations, and the core support assembly. The bottom of the plenum assembly is guided by the inside surface of the lower flange of the core support shield.

#### 4.5.1.3.2 Core Support Assembly

The core support assembly consists of the core support shield, core barrel, lower grid assembly, flow distributor, thermal shield, incore instrument guide tubes, and internals vent valves. Static loads from the assembled components and fuel assemblies, and dynamic loads from CRA trip, hydraulic flow, thermal expansion, seismic disturbances, and loss-of-coolant accident loads are all carried by the core support assembly.

The core support assembly components are described as follows:

1. Core Support Shield

   The core support shield is a flanged cylinder which mates with the reactor vessel opening. The forged top flange rests on a circumferential ledge in the reactor vessel closure flange. The core support shield lower flange is bolted to the core barrel. The inside surface of the lower flange guides and aligns the plenum assembly relative to the core support shield. The cylinder wall has two nozzle openings for coolant flow. These openings are formed by two forged rings, which seal to the reactor vessel outlet nozzles by the differential thermal expansion between the stainless steel core support shield and the carbon steel reactor vessel. The nozzle seal surfaces are finished and fitted to a predetermined cold gap providing clearance for core support assembly installation and removal. At reactor operating temperature, the mating metal surfaces are in contact to make a seal without exceeding allowable stresses in either the reactor vessel or internals. Eight vent valve mounting rings are welded in the cylinder wall for internals vent valves.

2. Core Barrel

   The core barrel supports the fuel assemblies, lower grid, flow distributor, and incore instrument guide tubes. The core barrel consists of a flanged cylinder, a series of internal horizontal former plates bolted to the cylinder, and a series of vertical baffle plates bolted to the inner surfaces of the horizontal formers to produce an inner wall enclosing the fuel assemblies. The core barrel cylinder is flanged on both ends. The upper flange of the core barrel cylinder is bolted to the mating lower flange of the core support shield assembly and the lower flange is bolted to the lower grid assembly. All bolts are lock welded after final assembly. Coolant flow is downward along the outside of the core barrel cylinder and upward through the fuel assemblies contained in the core barrel. A small portion of the coolant flows upward through the space between the core barrel outer cylinder and the inner baffle plate wall. Coolant pressure in this space is maintained lower than the core coolant pressure to avoid tension loads on the bolts attaching the plates to the horizontal formers.

3. Lower Grid Assembly

   The lower grid assembly provides alignment and support for the fuel assemblies, supports the thermal shield and flow distributor, and aligns the incore instrument guide tubes with the fuel assembly instrument tubes. The lower grid consists of two lattice type grid structures, separated by short tubular columns, and surrounded by a forged flanged cylinder. The upper structure is a perforated plate, while the lower structure consists of intersecting plates welded to form a grid. The top flange of the forged cylinder is bolted to the lower flange of the core barrel.

A shop fitup and checkout of the internal components for Oconee 1 in an as-built reactor vessel mockup insured proper alignment of mating parts before shipment. Dummy fuel assemblies and control rod assemblies were used to check fuel assembly clearances and CRA free movement.

To minimize lateral deflection of the lower end of the core support assembly as a result of horizontal seismic loading, integral weld-attached, deflection-limiting guide lugs are welded on the reactor vessel inside wall. These blocks also limit the rotation of the lower end of the core support assembly which could result from flow-induced torsional loadings. The lugs allow free vertical movement of the lower end of the internals for thermal expansion throughout all ranges of reactor operating conditions. In the unlikely event that a flange, circumferential weld, or bolted joint might fail, the lugs limit the possible core drop to 1/2 in. or less. The elevation plane of these lugs was established near the elevation of the vessel support skirt attachment to minimize dynamic loading effects on the vessel shell or bottom head. A 1/2 in. core drop does not allow the lower end of the CRA rods to disengage from their respective fuel assembly guide tubes, even if the CRAs are in the full-out position. In this rod position, approximately 6-1/2 in. of rod length remains in the fuel assembly guide tubes. A core drop of 1/2 in. does not result in a significant reactivity change. The core cannot rotate and bind the drive lines, because rotation of the core support assembly is prevented by the guide lugs.

The core internals are designed to meet the stress requirements of the ASME Code, Section III, during normal operation and transients. Additional criteria and analysis are given in Reference 1 on page 4-69. A detailed stress analysis of the internals under accident conditions has been completed and is reported in B&W Topical Report No. 10008, Part 1 (Reference 2 on page 4-69). This report analyzes the internals in the event of a major loss-of-coolant accident (LOCA) and for the combination of LOCA and seismic loadings. It is shown that although there is some internals deflection, failure of the internals does not occur because the stresses are within established limits. These deflections would not prevent CRA insertion because the control rods are guided throughout their travel, and the guide-to-fuel assembly alignment cannot change because positive alignment features are provided between them and the deflections do not exceed allowable values. All core support circumferential weld joints in the internals shells are inspected to the requirements of the ASME Code, Section III.

#### 4.5.1.3.1 Plenum Assembly

The plenum assembly is located directly above the reactor core and is removed as a single component before refueling. It consists of a plenum cover, upper grid, CRA guide tube assemblies, and a flanged plenum cylinder with openings for reactor coolant outlet flow. The plenum cover is constructed of a series of parallel flat plates intersecting to form square lattices and has a perforated top plate and an integral flange at its periphery. The cover assembly is attached to the plenum cylinder top flange. The perforated top plate has matching holes to position the upper end of the CRA guide tubes. The plenum cover is attached to the top flange of the plenum cylinder by a flange. Lifting lugs are provided for remote handling of the plenum assembly. These lifting lugs are welded to the cover grid. The CRA guide tubes are welded to the plenum cover top plate and bolted to the upper grid. CRA guide assemblies provide CRA guidance, protect the CRA from the effects of coolant cross-flow, and provide structural attachment of the grid assembly to the plenum cover.

Each CRA guide assembly consists of an outer tube housing, a mounting flange, 12 perforated slotted tubes and four sets of tube segments which are oriented and attached to a series of castings so as to provide continuous guidance for the CRA full stroke travel. The outer tube housing is welded to a mounting flange, which is bolted to the upper grid. Design clearances in the guide tube accommodate misalignment between the CRA guide tubes and the fuel assemblies. Final design clearances are established by tolerance studies and Control Rod Drive Line Facility (CRDL) prototype test results. The test results are described in Section 4.2.4.4, "Control Rod Drive Tests and Inspection" on page 4-16.

(Reference 3 on page 4-69) is analyzed in similar fashion to Part 1 except the steady state portion of LOCA is not a significant contributor.

Where it is indicated that substantial coupling, i.e., interrelationship, exists between major components of the Nuclear Steam System (NSS), such as the steam generator, the piping, and the vessel, the dynamic analysis includes the response of the entire coupled system. However, where coupling is found to be small, the component or groups of components are treated independently of the overall system.

The dynamic analysis for LOCA uses predicted pressure-time histories as input to a lumped-mass model. For earthquakes, actual earthquake records normalized to appropriate ground motion, are used as input to the model. The output from the analysis is in the form of internal motions (displacements, velocities, and accelerations), motions of individual fuel assemblies, impact loads between adjacent fuel assemblies, and impact loads between peripheral fuel assemblies and the core shroud. Motions of the reactor vessel, internals and core are confirmed using a time history excited lumped mass solution.

In addition, seismic analysis is also performed using a modal superposition and response spectra approach.

For the simultaneous occurrence of LOCA and the maximum hypothetical earthquake, both time-history excitations are input to the system simultaneously such that maximum structural motions, indicating maximum stresses, are obtained. Outputs are those mentioned above.

The output from the lumped-mass model and additional information such as pressure-time histories on separate internals and core components (including control rods) are used to calculate stresses and deflections. These stresses and deflections are compared to the allowable limits for the various loading combinations as established in Section 3.9.5, "Reactor Pressure Vessel Internals" on page 3-174 to insure that they are less than these allowables.

### 4.5.1.3 Description - Reactor Internals

Reactor internal components include the plenum assembly and the core support assembly. The core support assembly consists of the core support shield, vent valves, core barrel, lower grid, flow distributor, incore instrument guide tubes, and thermal shield. The plenum assembly consists of the upper grid plate, the control rod guide assemblies, and a plenum cylinder. Figure 4-26 shows the reactor vessel, reactor vessel internals arrangement, and the reactor coolant flow path. Figure 4-27 shows a cross section through the reactor vessel, and Figure 4-28 shows the core flooding arrangement.

Reactor internal components do not include fuel assemblies, control rod assemblies (CRAs), or incore instrumentation. Fuel assemblies and control rod assemblies are described in Section 4.2.2, "Description - Fuel System Design" on page 4-8, control rod drives in Section 4.5.3, "Control Rod Drives" on page 4-59, and core instrumentation in Section 7.7.3, "Summary of Alarms" on page 7-90.

The reactor internals are designed to support the core, maintain fuel assembly alignment, limit fuel assembly movement, and maintain CRA guide tube alignment between fuel assemblies and control rod drives. They also direct the flow of reactor coolant, provide gamma and neutron shielding, provide guides for in core instrumentation between the reactor vessel lower head and the fuel assemblies, and support the internals vent valves. The vent valves are designed to vent the stream generated within the core, thereby permitting the rapid re-covering of the core by coolant following a reactor coolant inlet pipe rupture. All reactor internal components can be removed from the reactor vessel to allow inspection of the reactor internals and the reactor vessel internal surface.

# 4.5 REACTOR MATERIALS

## 4.5.1 REACTOR INTERNALS

### 4.5.1.1 Reactor Internal Materials

Reactor internals are fabricated primarily from SA-240 (Type 304) material and designed within the allowable stress levels permitted by the ASME Code, Section III, for normal reactor operation and transients. Structural integrity of all core support assembly circumferential welds is assured by compliance with ASME Code Sections III and IX, radiographic inspection acceptance standards, and welding qualification.

### 4.5.1.2 Design Bases

The reactor internal components are designed to withstand the stresses resulting from startup; steady state operation with one or more reactor coolant pumps running; and shutdown conditions. No damage to the reactor internals will occur as a result of loss of pumping power.

The core support structure is designed as Class I structure, as defined in Section 3.9.5, "Reactor Pressure Vessel Internals" on page 3-174 to resist the effects of seismic disturbances. The basic design guide for the seismic analysis is AEC publication TID-7024, "Nuclear Reactors and Earthquakes."

Lateral deflection and torsional rotation of the lower end of the core support assembly is limited in order to prevent excessive deformation resulting from seismic disturbance thereby assuring insertion of control rod assemblies (CRAs). Core drop in the event of failure of the normal supports is limited by guide lugs so that CRAs do not disengage from the fuel assembly guide tubes (Section 4.5.1.3, "Description - Reactor Internals" on page 4-50).

The structural internals are designed to maintain their functional integrity in the event of any major loss-of-coolant accident. The dynamic loading resulting from the pressure oscillations because of a loss-of-coolant accident will not prevent CRA insertion.

Internals vent valves are provided to relieve pressure resulting from steam generation in the core following a postulated reactor coolant inlet pipe rupture, so that the core will be rapidly re-covered by coolant.

Allowable Stresses

The loading combinations and corresponding stress criteria, including the analytically predicted values of internals deflection for the combined maximum seismic and LOCA loadings, are given in B&W Topical Report "BAW-10008; Part 1; Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident (LOCA) and Maximum Hypothetical Earthquake." Additional criteria for stresses due to flow-induced vibratory loads are given in B&W Topical Report "Design of Reactor Internals and Incore Instrument Nozzles for Flow Induced Vibrations," (Reference 1 on page 4-69).

Methods of Load Analysis to be Employed for Reactor Internals and Fuel Assembly.

In Part 1 of BAW-10008 (Reference 2 on page 4-69), static or dynamic analysis is used as appropriate. In general, dynamic analysis is used for earthquakes and the subcooled portion of the loss-of-coolant accident (LOCA). For the relatively steady-state portion of the LOCA, a static analysis is used. BAW-10035

## 4.4.5 REFERENCES


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3. Stewart, C. W., et al. VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores. 5 vols. Battelle, Pacific Northwest Laboratories, *EPRI NP-2511-CCM-A, Rev. 3*, Richland, Washington, August 1989.

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10. Fuel Rod Bowing in Babcock and Wilcox Fuel Designs, Babcock & Wilcox, *BAW-10147P-A, Rev. 1*, Lynchburg, Va., May 1983.

11. C. E. Barksdale (B&W) to R. Powers (SMUD), Letter, Rancho Seco Unit 1 Response to NRC Questions on Mark-BZ Fuel, March 5, 1984.

12. LYNX2: Subchannel Thermal-Hydraulic Analysis Program, Babcock and Wilcox, *BAW-10130-A*, Lynchburg, VA, July 1985.

respect to mixing turbulence and pressure drop. Additional pressure drop testing has been conducted using 4-rod (5X), 4-rod (1X), 1-rod (1X), and 9-rod (1X) models.

Testing to determine the extent of interchannel mixing and flow distribution has also been conducted. Flow distribution in a square 4-rod test assembly has been measured. A salt solution injection technique was used to determine the average flow rates in the simulated reactor assembly corner cells, wall cells, and unit cells. Interchannel mixing data were obtained for the same assembly. These data have been used to confirm the flow distribution and mixing relationships employed in the core thermal and hydraulic design. Flow tests on a mockup of two adjacent fuel assemblies have been conducted. Additional mixing, flow distribution, and pressure drop data will be obtained to improve future core power capability. The following fuel assembly geometries have been tested to provide additional data:

1. A 9-rod (3 x 3 array) mixing test assembly, to determine flow pressure drop, flow distribution, and degree of mixing.
2. A 64-rod assembly simulating larger regions and various mechanical arrangements within a 15 x 15 fuel assembly and between adjacent fuel assemblies to determine flow distribution in the assembly and between adjacent assemblies.

This instrumentation provided the data necessary to accomplish the objectives set forth for the tests. The tests are summarized in BAW-10037 (Reference 9 on page 4-47).

### 4.4.4.2 Fuel Assembly Heat Transfer and Fluid Flow Tests

0 Although the original design of the reactor is based on the W-3 heat transfer correlation, B&W has conducted a continuous research and development program for fuel assembly heat transfer and fluid flow applicable to the design of the reactor. Single-channel tubular and annular test sections and multiple rod assemblies have been tested at the B&W Research Center. This test work substantiates the thermal design
0 of the reactor core. The multiple rod CHF tests are briefly discussed below.

0 **4.4.4.2.1 Deleted Per 1990 Update**

0

#### 4.4.4.2.2 Multiple-Rod Fuel Assembly Heat Transfer Tests

0 As a part of the development of the 15 x 15 Zircaloy grid Mark-BZ fuel assembly design, a series of CHF
0 tests were run at B&W's Alliance Research Center heat transfer facility. The tests were performed for 15
0 x 15 geometry with Zircaloy grids and full length non-uniform axial flux shapes. A total of 211 data
0 points were obtained covering the following conditions:

0 **Note:** The following conditions were revised in 1990 update.

$$1{,}000 < P < 2{,}400$$

$$0.2 \times 10^6 < G < 3.5 \times 10^6$$

$$-20 < X < +26$$

0 where

0 P = pressure, psia

0 G = mass velocity, lbm/hr-ft$^2$

0 X = local quality, %

0 The BWC correlation was developed from 17 x 17 Mark-C CHF data. The BWC correlation was shown
0 to conservatively represent the Mark-BZ CHF data with a 95/95 DNBR limit of 1.18 (Reference 1 on
0 page 4-47).

0 The BWC correlation was developed by B&W using the LYNX2 computer code (Reference 12 on
0 page 4-47). To verify use of the BWC correlation with the VIPRE-01 code, the Mark-BZ CHF data was
0 predicted and compared with B&W's LYNX2 results. As discussed in Reference 2 on page 4-47, the
0 VIPRE-01 BWC results show that a DNBR limit of 1.18 will provide 95% probability of precluding
0 DNB at a 95% confidence level.

#### 4.4.4.2.3 Fuel Assembly Flow Distribution, Mixing and Pressure Drop Tests

Flow visualization and pressure drop data have been obtained from a ten-times-full-scale (10X) model of a single rod in a square flow channel. These data have been used to refine the spacer grid designs with

A conservative mixing coefficient of 0.01, based on predictions of mixing tests, is used for all DNB analyses.

#### 4.4.3.3.7 Deleted Per 1990 Update.

#### 4.4.3.3.8 Hot Channel Factors

Hot channel factors are applied to the hot channel in the core to conservatively compensate for possible deviations of several parameters from their design values. The power hot channel factor, $F_q$, accounts for variations in average pin power caused by differences in the absolute number of grams of $U_{235}$ per rod. $F_q$ is applied to the heat generation rate of the hot pin of the hot subchannel. The value of $F_q$ used is given in Reference 2 on page 4-47. $F_q$ is increased when calculating Maximum Allowable Peaking (MAP) limits to account for the power spikes occurring as a result of the flux depressions at the spacer grids. The hot channel flow area is also reduced as discussed in Reference 2 on page 4-47 to account for manufacturing tolerances.

#### 4.4.3.3.9 Rod Bow Effects and Penalty

The mechanisms and resulting effects of fuel rod bow are discussed in B&W topical report BAW-10147P-A (Reference 10 on page 4-47). The topical report concludes that the DNB penalty due to rod bow is insignificant and unnecessary because the power production capability of the fuel decreases with irradiation. The rod bow correlation developed in Reference 10 on page 4-47 also conservatively predicts the rod bow behavior of Mark-BZ fuel discussed in Reference 11 on page 4-47.

## 4.4.4 THERMAL AND HYDRAULIC TESTS AND INSPECTION

### 4.4.4.1 Reactor Vessel Flow Distribution and Pressure Drop Test

A 1/6-scale model of the reactor vessel and internals has been tested to evaluate:

1. The flow distribution to each fuel assembly of the reactor core and to develop any necessary modifications to produce the desired flow distribution.
2. Fluid mixing between the vessel inlet nozzle and the core inlet, and between the inlet and outlet of the core.
3. The overall pressure drop between the vessel inlet and outlet nozzles, and the pressure drop between various points in the reactor vessel flow circuit.
4. The internals vent valves for closing behavior and for the effect on core flow with valves in the open position.

The reactor vessel, flow baffle, and core barrel were made of clear plastic to allow use of visual flow study techniques. All parts of the model except the core are geometrically similar to those in the production reactor. The simulated core was designed to maintain dynamic similarity between the model and production reactor.

Each of the 177 simulated fuel assemblies contained a calibrated flow nozzle. The test loop is capable of supplying cold water (80°F) to three inlet nozzles and hot water (180°F) to the fourth. Temperature was measured in the inlet and outlet nozzles of the reactor model and at the inlet and outlet of each of the fuel assemblies. Static pressure taps were located at suitable points along the flow path through the vessel.

experimental data points have been plotted to obtain the maps for the four different types of cells in the reactor core. These are shown in Figure 4-21, Figure 4-22, Figure 4-23, and Figure 4-24. The experimental data points represent the exit conditions in the various types of channels just previous to the burnout for a representative sample of the data points obtained at design operating conditions in the nine rod burnout test assemblies. In all of the bundle tests, the pressure drop, flow rate, and rod temperature traces were repeatable and steady, and did not exhibit any of the characteristics associated with flow instability.

Values of hot channel mass velocity and quality at 114 percent and 130 percent power for both nominal and design conditions are shown on the maps. The potential operating points are within the bounds suggested by Baker. Experimental data points for the reactor geometry with much higher qualities then the operating conditions have not exhibited unstable characteristics (Reference 9 on page 4-47).

0 **4.4.3.3.3 Reactor Coolant Flow System**

0 Another significant variable to be considered in evaluating the design is the total reactor coolant system
0 (RCS) flow. Conservative values for system and reactor pressure drop have been determined to insure
0 that the required system flow is obtained in the as-built plant. Measured RCS flow is considerably above
0 the design flow used in the core reload thermal hydraulic analyses.

0 The difference between the RCS flow and the reactor core flow is the core bypass flow. The core bypass flow is defined as that part of the flow that does not contact the active heat transfer surface area. This part of the flow exists primarily through three different paths. These paths are (1) through the core shroud, (2) through the control rod guide tubes and instrument tubes, and (3) between all interfaces
0 separating the inlet and outlet regions. The core bypass flow is generally less than 9%; however, the
0 bypass flow rate is dependent on the number of assemblies not containing control rods, burnable poison
0 rods, or source rods in each cycle as explained in Reference 2 on page 4-47.

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0 **4.4.3.3.4 Deleted Per 1990 Update**

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0 **4.4.3.3.5 Core Flow Distribution**

0 Inlet plenum effects have been determined from a 1/6 scale model flow test. The isothermal flow test data
0 has shown that the hot bundle receives average or better flow. It is conservatively assumed in all DNB
0 analysis (assuming 4 operating RC pumps) that the flow in the hot bundle is 5 percent less than the
4 average bundle flow (Reference 2 on page 4-47). A more restrictive flow maldistribution factor is
4 assumed for 3 pump operation analyses.

0 Flow redistribution accounts for the reduction in flow in the hot channel resulting from the high flow
0 resistance due to the local or bulk boiling in the hot channel. The effect on flow of the non-uniform
0 design power distribution is inherently considered in the VIPRE-01 code for all of the conditions
0 analyzed.

0 **4.4.3.3.6 Mixing Coefficient**

0 The flow distribution within the hot assembly is calculated using the VIPRE-01 code which allows for the
0 interchange of momentum and heat between channels. The turbulent mixing model incorporated in the
0 VIPRE-01 code and used for all core thermal-hydraulic analyses is discussed in Reference 2 on page 4-47.

0 2. Nuclear peaking factors.

3. Engineering hot channel factors.

4. Core flow distribution hot channel factors.

5. Design reactor power.

0 6. Thermal hydraulic analysis computer codes.

These inputs have been derived from test data, physical measurements and calculations.

0 Critical heat flux (CHF) calculations are performed with the Babcock and Wilcox BWC correlation.
0 Items 2 through 6 on the above list are explained in Chapters 5 and 6 of DPC-NE-2003P-A (Reference 2
0 on page 4-47). VIPRE-01 (Reference 3 on page 4-47) is the computer code used in these analyses.

The design overpower is the highest credible reactor operating power permitted by the Reactor Protective System including maximum instrumentation errors. Normally, trip on overpower will occur at a
0 significantly lower power than the design overpower.

## 4.4.3 THERMAL AND HYDRAULIC EVALUATION

### 4.4.3.1 Introduction

0 A summary of the characteristics of the reactor core design is given in Section 4.1, "Summary
0 Description" on page 4-3. The methodology of the thermal and hydraulic design analysis is presented in
5 DPC-NE-2003P-A (Reference 2 on page 4-47).

0 **4.4.3.2 Deleted Per 1990 Update**

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### 4.4.3.3 Evaluation of the Thermal and Hydraulic Design

#### 4.4.3.3.1 Hot Channel Coolant Conditions

0 The NRC approved VIPRE-01 code is used to calculate the reactor coolant enthalpy, mass flow, vapor
0 void, and DNBR distributions within the core for all expected operating conditions. The VIPRE-01 code
0 is described in detail in (Reference 3 on page 4-47), and the models and empirical correlations that are
0 used are discussed in (Reference 2 on page 4-47).

0 Steady-state analyses yield the MDNBR and quality in the hot channel at nominal and maximum design
0 overpower conditions. Table 4-14 contains a typical hot channel MDNBR value at nominal reactor
5 conditions.

0

#### 4.4.3.3.2 Coolant Channel Hydraulic Stability

Flow regime maps of mass flow rate and quality were constructed in order to evaluate channel hydraulic stability. The confidence in the design is based on a review of both analytical evaluations (References 4 on page 4-47 through 8 on page 4-47) and experimental results obtained in multiple rod bundle burnout tests. Bubble-to-annular and bubble-to-slug flow limits proposed by Baker (Reference 4 on page 4-47) are consistent with the B&W experimental data in the range of interest. The analytical limits and

# 4.4 THERMAL AND HYDRAULIC DESIGN

## 4.4.1 DESIGN BASES

The bases for the thermal and hydraulic design have been established to enable the reactor to operate at 2,568 MWt rated power with sufficient design margins to accommodate both steady-state and transient operation without damage to the core and without exceeding the design pressure limits for the reactor coolant system. The thermal-hydraulic design bases also help to ensure that the fuel rod cladding will maintain its integrity during steady-state operation, design overpower, and anticipated operational transients occurring throughout core life.

Fuel cladding integrity is ensured by limiting the core to the following thermal-hydraulic boundaries during steady-state operation at power levels up to and including the design overpower, and during anticipated transient operation.

1. The fuel pin cladding, fuel pellets, and fuel pin internals must be designed so that the fuel-to-clad gap characteristics ensure that the maximum fuel temperature does not exceed the fuel melting limit at the 112 percent design overpower at any time during core life. See Section 4.2.3.1.3, "Fuel Thermal Analysis" on page 4-13 for a discussion of fuel melting temperature.
0 2. The minimum allowable DNBR during steady-state operation and anticipated transients for Mark-BZ fuel is established as 1.18 with the BWC correlation (Reference 1 on page 4-47). This limit on MDNBR ensures on a 95 percent confidence level that there is a 95 percent probability DNB will not occur.
3. Although generation of net steam is allowed in the hottest core channels, flow stability is required during all steady-state and operational transient conditions.

By preventing a departure from nucleate boiling (DNB), neither the cladding nor the fuel is subjected to excessively high temperatures.

The core flow distribution and coolant velocities have been set to provide adequate cooling capability to the hottest core channels and to maintain minimum DNB ratios greater than the design limit. Fuel assembly design and cladding integrity criteria are discussed in Section 4.2.1.2.4, "Mechanical Limits" on page 4-7.

## 4.4.2 DESCRIPTION OF THERMAL AND HYDRAULIC DESIGN OF THE REACTOR CORE

5 Table 4-14 depicts typical thermal-hydraulic design conditions.

### 4.4.2.1 CORE DESIGN ANALYSIS DESCRIPTION

The methodology of the analysis used with the design bases criterion is fully described in
0 DPC-NE-2003P-A (Reference 2 on page 4-47).

The input information and analytical tools for the thermal hydraulic design and for the evaluation of individual hot channels is as follows:

1. Heat transfer, critical heat flux equations, and data correlations.

## 4.3.8 REFERENCES


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7. Oconee Nuclear Station Unit 2 Startup Report DPR-47, Docket No. 50-270, July 12, 1974.
8. Oconee Nuclear Station Unit 3 Startup Report DPR-55, Docket No. 50-278, March 14, 1975.
9. Letter W. O. Parker, Jr. to H. R. Denton, Oconee Nuclear Station Generic Startup Physics Test Program, Docket Nos. 50-269, -270, -287, July 11, 1980.
10. 3 Letter A. C. Thies to H. R. Denton, Oconee Nuclear Station 1, 2, and 3, Docket Nos. 50-269, -270, -287, May 29, 1981.
11. R. A. Turner, Fuel Densification Report, *BAW-10054, Rev. 2*, Babcock & Wilcox, Lynchburg, Virginia, May 1973.
12. Oconee Unit 1, Cycle 7, Reload Report, *BAW-1660*, Babcock & Wilcox, Lynchburg, Virginia, March 1981.
13. 3 3 Letter W. O. Parker, Jr. to H. R. Denton, Oconee Nuclear Station 1, 2, and 3, Docket Nos. 50-269, -270, -287, August 15, 1980.
14. 3 3 Letter J. F. Stolz to W. O. Parker, Jr., Oconee Nuclear Station, Generic Startup Physics Test Program, March 23, 1981.
15. 3 3 Letter P. C. Wagner to W. O. Parker, Jr., Oconee Nuclear Station 1, 2 and 3, Docket Nos. 50-269, -270, -287, November 30, 1981.
16. 3 3 Letter H. B. Tucker to H. R. Denton, Oconee Nuclear Station 1, 2 and 3, Docket Nos. 50-269, -270, -287, September 2, 1986.
17. 3 3 Letter J. F. Stolz to H. B. Tucker, Oconee Nuclear Station 1, 2 and 3, Revisions to the Startup Physics Testing Program, October 7, 1986.
18. 5 5 Nuclear Design Methodology using CASMO-3/SIMULATE-3P, DPC-NE-1004A, Duke Power, Charlotte, North Carolina, November 1992

| TEST | PLANT CONDITIONS | ACCEPTANCE CRITERIA |
|---|---|---|
| 4. Control Rod Worth | HZP | Individual Groups<br>± 15%<br><br>Sum of Groups<br>± 10% |
| 5. Low Power Testing | 5 - 30 %FP | Relative Core<br>Power Distribution<br>± 10% |
| 6. Intermediate Power Testing | 40 - 75%FP | Total/Radial Peaking<br>±7.5%/±5.0%<br><br>RMS (Radial) < 7.5% |
| 7. Full Power Testing | 90 - 100 %FP | Total/Radial Peaking<br>±7.5%/±5.0%<br><br>RMS (Radial) < 7.5% |
| 8. Reactivity Anomaly | HFP | All Rods Out Critical<br>Boron<br>± 50 ppmB |

3 ### 4.3.7.4 Reactivity Anomaly

3 #### 4.3.7.4.1 Plant Conditions

3 Hot Full Power, ~579°F, ~2155 psig, full RCS flow.

3 #### 4.3.7.4.2 Procedure

3 As a part of the periodic testing program and separate from the startup testing program, the ARO critical
3 boron concentration at power is checked against normalized predicted values approximately each 10
3 EEPD of steady-state operation. With the reactor at steady-state conditions, as near a practical to full
3 power ARO conditions, a sample of the RCS is taken an analyzed for boron concentration. This value of
3 boron concentration is then adjusted to account for the reactivity worth of regulating control rod
3 assemblies in the core at the time of the measurement, and any other minor variations from designed
3 conditions.

3 The results are reviewed by the Site Nuclear Engineering Supervisor and are compared with the
3 normalized predicted ARO boron concentration for the time in the cycle at which the measurement was
3 taken. If the difference between the measured and predicted ARO boron concentration values does not
3 exceed 50 ppm Boron, then the results are acceptable.

3 #### 4.3.7.4.3 Follow-Up Actions

3 If the acceptance criterion (±50 ppmb) is not met and the difference between measured and predicted
3 ARO boron concentration is less than 100 ppm Boron, the results will be reviewed by cognizant engineers
3 to determine the appropriate corrective action required to resolve the discrepancy. This review will be
3 completed with the results and recommended corrective actions approved by representatives from Oconee
3 Nuclear Station Nuclear Engineering and Regulatory Compliance, and General Office Nuclear
3 Engineering within 14 days.

3 If the acceptance criterion (±50 ppmb) is not met and the difference between measured and predicted
3 ARO boron concentration is greater than 100 ppm Boron, then the results will be reviewed by cognizant
3 engineers to determine the appropriate corrective actions required to resolve the discrepancy pursuant to
3 Technical Specification 4.10.

3 Oconee Startup Physics Test Program (OSPTP) Summary

| TEST | PLANT CONDITIONS | ACCEPTANCE CRITERIA |
|---|---|---|
| 1. Control Rod Trip Time Test | RCS Full Flow | 1.66 seconds |
| 2. All Rods Out Critical Boron | HZP | ± 50 ppmB |
| 3. Moderator Temperature Coefficient | HZP | ± $0.3 \times 10^{-4}$ $\Delta\rho/°F$ |

3 **4.3.7.2.3 Follow-Up Actions**

3 If any observed parameter exceeds its specified values in the Technical Specifications, actions will be taken
3 as required by the Technical Specifications.

3 Also, the observed parameter will be reviewed by cognizant engineers to determine the appropriate
3 corrective actions required to resolve the discrepancy. This review will be completed with the results and
3 recommended corrective actions approved by representatives from Oconee Nuclear Station Nuclear
3 Engineering and Regulatory Compliance, and General Office Nuclear Engineering prior to any further
3 escalation of power.

3 If any acceptance criteria are exceeded, the results will be reviewed by cognizant engineers to determine
3 the appropriate corrective actions required to resolve the discrepancy. This review will be completed with
3 the results and recommended corrective actions approved by representatives from Oconee Nuclear Station
3 Nuclear Engineering and Regulatory Compliance, and General Office Nuclear Engineering prior to
3 escalation to 100% FP.

3 **4.3.7.3 Full Power Testing**

3 **4.3.7.3.1 Plant conditions**

3 90 to 100% FP, ~579°F, ~2155 psig, full RCS flow (4 pumps).

3 **4.3.7.3.2 Procedure**

3 Once the unit is between 90 and 100% FP with Xenon equilibrium, the output of the plant OAC reactor
3 calculations program is analyzed. This program processes the signals from fixed incore detectors and
3 provides a relative core power distribution as output. The incore detector outputs are checked, in order to
3 identify malfunctioning detectors. After these have been eliminated, the radial and total peaking factors
3 obtained from the OAC are compared with the values calculated as part of the reload design process on
3 an eighth-core basis. The results are reviewed by the Test Coordinator. If the highest measured radial
3 peaking factor does not exceed the highest predicted radial peaking factor by more than 5.0% of the
3 highest measured radial peaking factor, and if the highest measured total peaking factor does not exceed
3 the highest predicted total peaking factor by more than 7.5% of the highest measured total peaking factor,
3 and if the RMS difference between predicted and measured radial peaking factors is less than 7.5%, then
3 the results are acceptable.

3 **4.3.7.3.3 Follow-Up Actions**

3 If any observed parameter exceeds its specified values in the Technical Specifications, actions will be taken
3 as required by the Technical Specifications.

3 Also, the observed parameter will be reviewed by cognizant engineers to determine the appropriate
3 corrective actions required to resolve the discrepancy. This review will be completed with the results and
3 the recommended corrective actions approved by representatives from Oconee Nuclear Station Nuclear
3 Engineering and Regulatory Compliance, and General Office Nuclear Engineering prior to any escalation
3 of power.

3 If any acceptance criteria are exceeded, the results will be reviewed by cognizant engineers to determine
3 the appropriate corrective actions required to resolve the discrepancy. This review will be completed with
3 the results and recommended corrective actions approved by representatives from Oconee Nuclear Station
3 Nuclear Engineering and Regulatory Compliance, and General Office Nuclear Engineering prior to any
3 escalation of power.

3 **4.3.7.1.2 Procedure**

3 Once the unit is between 5 and 30% FP, the output of the plant OAC reactor calculations program is
3 analyzed. This program processes the signals from fixed incore detectors and provides a relative core
3 power distribution as output. The incore detector outputs are checked in order to identify malfunctioning
3 detectors. After these have been eliminated, the results for corrected assembly power in functioning
3 instrumented symmetric core locations are compared.

3 The results are reviewed by the Test Coordinator. If the reactor calculations outputs appear normal, and
3 the deviation between the highest and lowest corrected assembly power for symmetric core locations is
3 less than ±10%, then the results are acceptable.

3 **4.3.7.1.3 Follow-Up Actions**

3 If the reactor calculations outputs appear abnormal, the raw detector signals are evaluated to determine if
3 a significant core asymmetry exists. If no significant asymmetry exists, power escalation is continued. If
3 an asymmetry exists, the Site Nuclear Engineering Supervisor is contacted to initiate a program of testing
3 and evaluation before further power increase. The problem with the reactor calculations program is
3 investigated and corrected, but this is not a prerequisite for power increase if no significant asymmetry
3 exists.

3 If the reactor calculations outputs appear normal and the deviation between corrected assembly powers
3 for symmetric core locations is greater than ±10%, the cause of the indicated deviation is investigated. If
3 the deviation is due to identifiable reactor calculations program problems, it is corrected per normal
3 procedures and power escalation testing may continue. If the cause of the deviation cannot be identified,
3 the Site Nuclear Engineering Supervisor is contacted to initiate a program of testing and evaluation.

3 The results will be reviewed by cognizant engineers to determine the appropriate corrective actions
3 required to resolve the deviation. This review will be completed with the results and the recommended
3 corrective actions approved by representatives from Oconee Nuclear Station Nuclear Engineering and
3 Regulatory Compliance, and General Office Nuclear Engineering prior to any further escalation of power.

3 **4.3.7.2 Intermediate Power Testing**

3 **4.3.7.2.1 Plant Conditions**

3 40 to 75% FP, ~579°F, ~2155 psig, full RCS flow (4 pumps).

3 **4.3.7.2.2 Procedure**

3 Once the unit is between 40 and 75% FP, the output of the plant OAC reactor calculations program is
3 analyzed. This program processes the signals from fixed incore detectors and provides a relative core
3 power distribution as output. The incore detector outputs are checked, in order to identify
3 malfunctioning detectors. After these have been eliminated, the radial and total peaking factors obtained
3 from the plant OAC are compared with the values calculated using the computer codes utilized during the
3 reload design process on an eighth-core basis.

3 The results are reviewed by the Test Coordinator. If the highest measured radial peaking factor does not
3 exceed the highest predicted radial peaking factor by more than 5.0% of the highest measured radial
3 peaking factor, and if the highest measured total peaking factor does not exceed the highest predicted total
3 peaking factor by more than 7.5% of the highest measured total peaking factor, and if the RMS difference
3 between predicted and measured radial peaking factors is less than 7.5%, then the results are acceptable.

3 recommended corrective actions approved by representatives from Oconee Nuclear Station Nuclear
3 Engineering and Regulatory Compliance, and General Office Nuclear Engineering prior to 100% FP.

### 3 4.3.6.3 Control Rod Worth

#### 3 4.3.6.3.1 Plant Conditions

3 Hot Zero Power, ~532°F, ~2155 psig, steady RCS flow (3 or 4 pumps).

#### 3 4.3.6.3.2 Procedure

3 The measurement of regulating rod group worths begin from a critical steady state condition with all
3 regulating rod groups withdrawn as far as possible (i.e., within 0.12% Δk/k of ARO). From this point a
3 boron concentration necessary to deborate control rod Groups 7, 6 and 5 to fully inserted is calculated.
3 The resulting reactivity change during deboration is compensated for by discrete insertion of control rods
3 with both signals being recorded by a reactivity calculation or Reactimeter. Integral rod worths are
3 calculated by summing the differential rod worths for each control rod group.

3 The results are reviewed by the Test Coordinator and compared with the predicted control rod group
3 worths. If the difference between the measured and predicted individual rod group worths does not
3 exceed 15%, and the difference between the measured and predicted total worth of control rod Groups 5,
3 6 and 7 does not exceed 10%, then the results are acceptable.

#### 3 4.3.6.3.3 Follow-Up Actions

3 If the difference between the measured and predicted total worth of control rod Groups 5, 6, and 7
3 exceeds 10%, then, following calculation of the minimum control rod position for which the worth of the
3 control rods withdrawn would equal 1% Δk/k, additional control rod group worths will be measured.
3 The worths of safety control rod groups will be measured in sequence from Group 4 to Group 2, until
3 either the difference between the measured and predicted total worth of all control rod groups measured
3 does not exceed 10%, or the calculated minimum control rod position is reached. In the latter case,
3 control rod worth testing will halt. The results will be reviewed by cognizant engineers to determine the
3 appropriate additional corrective actions required to resolve the discrepancy. This review will be
3 completed with the results and the recommended actions approved by representatives from Oconee
3 Nuclear Station Nuclear Engineering and Regulatory Compliance, and General Office Nuclear
3 Engineering prior to exceeding 15% FP.

3 If the difference between the measured and predicted control rod worths of any of the individual control
3 rod groups exceeds 15%, the results will be reviewed by cognizant engineers to determine the appropriate
3 corrective actions required to resolve the discrepancy. This review will be completed prior to reaching
3 100% FP.

## 3 4.3.7 POWER ESCALATION TEST PHASE

### 3 4.3.7.1 Low Power Testing

#### 3 4.3.7.1.1 Plant Conditions

3 5 to 30% FP, ~579°F, ~2155 psig, full RCS flow (4 pumps).

3 The results are reviewed by the Test Coordinator and compared with the predicted boron concentration.
3 If the difference between the measured and predicted values does not exceed 50 ppm Boron, the results are
3 acceptable.

3 **4.3.6.1.3 Follow-Up Actions**

3 If the acceptance criterion (± 50 ppmb) between measured and predicted ARO critical boron
3 concentration is not met, the results will be reviewed by cognizant engineers to determine the appropriate
3 corrective actions required to resolve the discrepancy. This review will be completed with the results and
3 recommended corrective actions approved by representatives from Oconee Nuclear Station Nuclear
3 Engineering and Regulatory Compliance, and General Office Nuclear Engineering prior to 100% FP.

3 If the difference between measured and predicted ARO critical boron concentration is greater than 100
3 ppm Boron, the results will be reviewed by cognizant engineers to determine the appropriate corrective
3 actions required to resolve the discrepancy. This review will be completed with the results and
3 recommended corrective actions approved by representatives from Oconee Nuclear Station Nuclear
3 Engineering and Regulatory Compliance, and General Office Nuclear Engineering prior to exceeding 15%
3 FP.

3 ## 4.3.6.2 Moderator Temperature Coefficient

3 **4.3.6.2.1 Plant Conditions**

3 Hot Zero Power, ~532°F, ~2155 psig, steady RCS flow (3 or 4 pumps).

3 **4.3.6.2.2 Procedure**

3 The moderator temperature coefficient (MTC) test begins with the reactor at critical equilibrium
3 conditions. This test is performed by executing a change in RCS average temperature of approximately
3 ± 5°F while data are taken. Stability in RCS temperature is necessary at this first plateau. The hold
3 time at each RCS temperature plateau during the test is approximately five minutes. After data are taken
3 at the first RCS temperature plateau, the RCS average temperature is changed approximately 10°F in the
3 opposite direction and allowed to stabilize. Changes in reactivity associated with the induced RCS
3 temperature transient are measured by a reactivity calculation or Reactimeter. This overall temperature
3 coefficient is corrected for the contribution of the isothermal doppler coefficient or reactivity to give the
3 moderator coefficient of reactivity. The measurement is also corrected to an average temperature of
3 532°F.

3 The results are reviewed by the Test Coordinator and compared with the predicted MTC. If the
3 difference between the measured and predicted values does not exceed $0.3 \times 10^{-4}$ $\Delta k/k/°F$, then the results
3 are acceptable.

3 **4.3.6.2.3 Follow-Up Actions**

3 If the measured maximum positive MTC exceeds $0.5 \times 10^{-4}$ $\Delta k/k/°F$, the results will be reviewed by
3 cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy.
3 This review will be completed with the results and recommended actions approved by representatives
3 from Oconee Nuclear Station Nuclear Engineering and Regulatory Compliance, and General Office
3 Nuclear Engineering prior to exceeding 15% FP.

3 If the $0.3 \times 10^{-4}$ $\Delta k/k/°F$ acceptance criterion is exceeded and the maxium positive MTC is less than $0.5 \times$
3 $10^{-4}$ $\Delta k/k/°F$, the results will be reviewed by cognizant engineers to determine the appropriate corrective
3 actions required to resolve the discrepancy. This review will be completed with the results and

3 ## 4.3.5 PRE-CRITICAL TEST PHASE

3 ### 4.3.5.1 Control Rod Drop Time

3 #### 4.3.5.1.1 Plant Conditions

3 Full reactor coolant system (RCS) flow (4 pumps).

3 #### 4.3.5.1.2 Procedure

3 The control rod drop time for each full-length control rod assembly (CRA) to fall from the fully
4 withdrawn position to the 25% withdrawn position is measured. The sequence of events recorder is
4 normally used to record the time interval between initiation and termination of the event. The test may
3 be performed either by dropping all full-length CRAs simultaneously from the fully withdrawn position,
3 or by dropping one full length CRA group at a time and measuring the drop times for each individual
4 CRA group. In either case, the sequence of events recorder records the drop time of each CRA
3 individually.

3 The results are reviewed by the Test Coordinator and compared with the acceptance criterion, 1.66
4 seconds. The accuracy of the measurement of control rod drop time as performed by the sequence of
4 events recorder is approximately $\pm$ 0.005 seconds.

3 #### 4.3.5.1.3 Follow-Up Actions

3 If any measured control rod drop time is greater than 1.40 seconds but less than 1.66 seconds, then the
3 results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to
3 resolve the discrepancy. This review will be completed prior to 100% FP.

3 If any control rod drop time exceeds 1.66 sec., then the results will be reviewed by cognizant engineers to
3 determine the appropriate corrective actions required to resolve the discrepancy. Also, the actions
3 specified by Technical Specifications 3.5 and 4.7 will be taken.

3 ## 4.3.6 ZERO POWER PHYSICS TEST PHASE

3 ### 4.3.6.1 Critical Boron Concentration

3 #### 4.3.6.1.1 Plant Conditions

3 Hot Zero Power, ~532°F, ~2155 psig, steady RCS flow (3 or 4 pumps).

3 #### 4.3.6.1.2 Procedure

3 The ARO critical boron concentration is measured by establishing an equilibrium RCS boron
3 concentration near the predicted ARO critical boron concentration. Control Rod Groups 1 through 7 are
3 fully withdrawn. Control Rod Group 8 is maintained at the nominal designed position. A sample of the
3 equilibrium boron concentration is then taken and analyzed to determine the critical boron concentration.
3 Since it may not be practical to establish critical equilibrium conditions with Group 7 fully withdrawn, the
3 small amount of inserted worth of Group 7 or worth of Group 8 (from its nominal designed position) is
3 measured by a reactivity calculation or Reactimeter. This reactivity is then used to adjust the boron
3 concentration to obtain the measured ARO boron concentration.

3 c. Full Power Testing (90-100% FP)

3 In addition to the above tests, which comprise the basic Startup Physics Test Program, a separate test, the
3 Reactivity Anomaly at Full Power is performed approximately each 10 EFPD (Effective Full Power
3 Days), during steady-state operation pursuant to Technical Specification 4.10. This procedure is used to
3 verify that the measured "all-rods-out" (ARO) hot full power (FP) critical boron concentration is in
3 agreement with the predicted value. The test conditions, procedure descriptions, acceptance criteria, and
3 review requirements for each of the above are provided in this document.

3 For all these tests, specific acceptance criteria are provided (see OSPTP Summary). Upon completion of
3 each test, the results are reviewed by a designated individual. If the results meet the specific acceptance
3 criteria, then the test is considered to be satisfactorily completed. However, if the results exceed the
3 specific acceptance criteria, an extensive review is performed by cognizant engineers from within Duke
3 Power Company or from outside organizations, as appropriate, to identify and correct the cause of the
3 discrepancy. Continuation of the test program, including any power escalations, will be dependent upon
3 satisfactory resolution of any unacceptable test result. Representatives from Oconee Nuclear Station
3 Nuclear Engineering and Regulatory Compliance, and General Office Nuclear Engineering will approve
3 actions under the conditions stated for each test.

3 The current Startup Physics Test Program for Oconee Nuclear Station was submitted by References 9 on
3 page 4-39 and 13 on page 4-39, approved by Reference 14 on page 4-39, and subsequently modified by
3 References 10 on page 4-39 and 16 on page 4-39, and approved by References 15 on page 4-39 and 17
3 on page 4-39.

When compounded errors are considered as in Figure 4-17 the threshold moderator coefficient is approximately $+2.5 \times 10^{-4}\ \Delta\rho/°F_m$.

5 This analysis is considered to be valid and bounding for the current core designs for the following reasons:

5 1. The minimum moderator temperature coefficient (MTC) threshold value, as listed within Table 4-7,
5 is $+1.0 \times 10^{-4}\ \Delta\rho$/degF. The most positive moderator temperature coefficient assumed within
5 Chapter 15 safety/accident analysis is less than the threshold value.

5 2. There is considerable margin for a BPRA core (i.e., Oconee 2, Cycle 1 within Reference 5 on
5 page 4-39) between the Table 4-7 threshold MTC and the calculated threshold MTC, even when
5 compounded errors are taken into account.

5 3. Current nuclear design bases require that the MTC and the power Doppler coefficient be negative at
5 power. As such, any azimuthal oscillations within current cores are self-damping by virtue of
5 reactivity feedback effects.

Operating procedures are in effect which allow the reactor operator to damp out any axial xenon oscillation if it should occur.

## 4.3.4 NUCLEAR TESTS AND INSPECTIONS

Nuclear Testing and Inspection can be divided into two areas:

1. Initial Core
2. Startup Testing for Reload Cores.

### 4.3.4.1 Initial Core Testing

The startup testing performed on Oconee 1, 2, and 3 initial cores was an extensive program to verify both calculational methods and proper behavior of the core. The results of this testing was reported in References 6 on page 4-39, 7 on page 4-39, and 8 on page 4-39.

### 4.3.4.2 Zero Power, Power Escalation, and Power Testing For Reload Cores

3 The Startup Physics Test Program for Oconee Nuclear Station, or OSPTP, is structured to provide
3 assurance that the installed reactor core following each reload conforms to the design core. This
3 document provides the minimum test program which will be conducted on each Oconee unit. Additional
3 tests may be performed during a specific startup test program as conditions warrant. However, in all
3 cases, the following tests will be performed:

3 1. Pre-critical Test Phase

3 a. Control Rod Drop Time

3 2. Zero Power Physics Test Phase

3 a. Critical Boron Concentration

3 b. Moderator Temperature Coefficient

3 c. Control Rod Worth

3 3. Power Escalation Test Phase

3 a. Low Power Testing (5-30% FP)

3 b. Intermediate Power Testing (40-75% FP)

Fuel Misloading

Misloading the fuel pins in an assembly is prevented by loading controls and procedures. Each fuel rod is identified by an enrichment code, and the design of the reactor is such that only one enrichment is used per assembly. The manufacturing process relies on administrative procedures and quality control checks to assure that fuel rods are placed in the proper assembly. One such administrative procedure which will be practiced to the extent practical is the "campaigning" of enrichments so that only a single enrichment is handled at a given time in fuel fabrication.

Gross fuel assembly misplacement in the core is prevented by administrative core loading procedures and the prominent display of fuel assembly identification markings on the upper end fitting of each assembly. After the core is loaded, an independent check is performed to verify the core loading.

During startup physics testing, misloaded fuel may be discovered by unexpected quadrant power tilt or differences between predicted and measured power distributions.

### 4.3.3.2 Xenon Stability Analysis and Control

Modal and digital analysis of the Oconee 1, Cycle 1 core indicated that a tendency toward xenon instability in the axial mode would exist for a given combination of events (BOL, rodded core). Therefore, eight part-length Axial Power Shaping Rod Assemblies (APSRA) have been included in the design. They will be positioned during operation to maintain an acceptable distribution of power for any particular operating condition in the core, thereby reducing the tendency for axial oscillations. Similar analysis which was performed on the Oconee 2, Cycle 1 core indicated that it would be stable with regard to axial oscillations. Oconee 3, Cycle 1 was assumed to have characteristics similar to those of Oconee 1.

The azimuthal stability of the cores are dependent upon core loadings, power densities, and moderator temperature coefficients. In any event, the cores will not be susceptible to diverging azimuthal oscillations. If the loadings and power densities are low enough, the core will be inherently stable (Oconee 1, Cycle 1). If not, then burnable poison is added in the amount necessary to provide a moderator temperature coefficient that will result in azimuthal stability (Oconee 2&3, Cycle 1). A detailed description of the xenon analyses performed on Unit 1 and 2 cores may be found in Reference 5 on page 4-39.

The first two parts of Reference 5 on page 4-39, which considered modal and one-dimensional digital analyses, pointed out the need for multi-dimensional calculations regarding xenon stability. The reactor core designs for Oconee Units 1 and 2, Cycle 1, have been analyzed in three dimensions with thermal feedback. For the Unit 1 operating core at beginning of life, the predicted azimuthal stability index is -0.07 $hr^{-1}$. Using modal analysis with the three-dimensional results shows that the shape factor must be approximately 50 percent flat for the power coefficient of $-5.05 \times 10^{-6}$ as calculated by previously described methods. Since the curves in Part 1 of Reference 5 on page 4-39 were generated for a power coefficient of $-3.92 \times 10^{-6}$ $\Delta\rho$/MWt, it was necessary to generate two new curves for azimuthal stability. These curves are shown in Figure 4-14 and Figure 4-15. From Figure 4-14 the threshold (i.e., stability index = 0) moderator coefficient for the nominal case is approximately $+3 \times 10^{-4}$ $\Delta\rho/°F_m$. Including compounded errors from Figure 4-15, the threshold moderator coefficient is approximately $+1 \times 10^{-4}$ $\Delta\rho/°F$. Using the least favorable predictions of the Doppler and moderator coefficients, a stability index of -0.067 $hr^{-1}$ is obtained. This corresponds to a power coefficient of $-4.73 \times 10^{-6}$ $\Delta\rho$/MWt. For the Unit 2 operating core at beginning of life (96 FPH), the predicted azimuthal stability index is -.085 $hr^{-1}$. Again, using modal analysis combined with three-dimensional results shows the shape factor to be approximately 40 percent flat for the calculated power coefficient of $-4.67 \times 10^{-6}$ $\Delta\rho$/MWt. Azimuthal stability curves for the nominal and compounded error cases are shown in Figure 4-16 and Figure 4-17 respectively. From Figure 4-16 the nominal threshold moderator coefficient extrapolates to approximately $+5 \times 10^{-4}$ $\Delta\rho/°F_m$.

#### 4.3.3.1.3 Nuclear Design Uncertainty (Reliability) Factors

In various calculations additional conservatism is applied to the calculated parameters. The factors sometimes are analysis dependent and are tabulated in Reference 18 on page 4-39.

#### 4.3.3.1.4 Power Maldistributions

Misaligned Control Rods

The reactor has a control function to protect against a rod out of step with its group. The position of each rod is compared to the average of the group. If an assymetric fault is detected at power levels greater than 60% of rated power, a rod withdrawal inhibit is activated and the Integrated Control System (ICS) runs the plant back to 60% of rated power. If a rod is dropped, the Integrated Control System (ICS) cannot maintain core power to match demand by withdrawal of other rods, and the plant is run back to 60 percent of rated power. Several cases were also analyzed for BOL for Oconee 1, Cycle 1, with single dropped rods. The calculations were performed with half-core X-Y geometry in PDQ07 at rated power without thermal feedback. The results are given in Figure 4-13.

The maximum radial-local power peak is 1.92. The original FSAR design limit is a 2.1 radial-local at rated power with a 1.5 cosine yielding a 1.3 DNBR based on the W-3 correlation. At a 114 percent overpower condition the design limit can also be expressed as a 1.9 radial-local with a 1.5 cosine yielding a 1.3 DNBR. The dropped rods illustrated in Figure 4-13 do not represent violations of the thermal limits of the design.

Several dropped rod cases run with SIMULATE-3P and current core design models indicate less severe radial-local power peaks; primarily because the current cores operate in a feed and bleed mode and the Oconee 1, Cycle 1 core was a rodded core. It should also be noted that dropped rod accidents are analyzed within Chapter 15 (Control Rod Misalignment Accident), and that this analysis showed that the consequence of a dropped rod is minimal such that the core and RCS pressure boundary are preserved, even when the worst assumed safety parameters are used and no credit is taken for ICS action.

Radial power tilts can be detected with the out-of-core and in-core instrumentation, and the operator has the flexibility to monitor the upper or the lower out-of-core detectors to determine the X-Y power symmetry condition at any time.

For the assumed case where one CRA is left out of the core while the remainder of the group is fully inserted, this condition would not occur except with regard to rod "swaps." Since rod swaps are performed at reduced power, and since the operator can monitor the out-of-core detectors, an X-Y tilt resulting from such a condition could be detected and appropriate action taken before the approach to thermal limits could be realized.

The APSR drives are also equipped with the position monitors and the alarm function for a rod out of step with the group average. These drives, however, do not permit rod drops. With the power removed from the rod drive windings of the APSR, the roller nut will not disengage and the rod remains in its position. Since the APSR's are made of low-absorbing (gray) material, it is not likely that thermal limits will be exceeded if one of the rods were stuck and the rest of the group were moved.

Azimuthal Xenon Oscillations

The Oconee reactors are predicted to have a substantial margin to threshold for azimuthal xenon oscillations. Therefore, this mode is not considered to be likely to produce a power peaking problem.

The analytical models and their applications are discussed in this section as well as core instabilities associated with xenon oscillations.

### 4.3.3.1 Analytical Models

Reactor design calculations are made using a large number of computer codes. The following section describes the major analytical models employed by DUKE in the design of Oconee reload cores. Table 4-13 specifies the cycle of each unit when these methodologies were first applied. The methodology used in a particular reload design is stated in the bases behind appropriate reload design change report.

#### 4.3.3.1.1 CASMO-3/SIMULATE-3P-Based Methodology

The CASMO-3/SIMULATE-3P-based calculational methods for nuclear design have been reviewed and approved by the NRC in Reference18 on page 4-39 . This methodology was first applied during the reload design analysis of Unit 1 Cycle 16.

Verification to Measured Data

The verification of the CASMO-3/SIMULATE-3P-based methods for nuclear design is documented in Reference 18 on page 4-39.

#### 4.3.3.1.2 Control of Power Distributions

The reactors are designed to permit power maneuvering on control rods. Various calculations are performed during the maneuvering analysis to develop operational power-imbalance, RPS power imbalance, and rod insertion limits. These three-dimensional calculations account for effects of rod insertion, xenon distribution, and power level on the power distribution. A more detailed discussion of these calculations can be found in Reference 2 on page 4-39.

During startup testing an out-of-core detector correlation test is performed to calibrate the imbalance as measured by the out-of-core detectors to that measured by in-core detectors. Uncertainties in the measurement of imbalance and power level are accounted for to assure that the reactor trips before any DNBR or fuel melt limit is reached.

The out-of-core neutron flux detectors each consist functionally of two six-foot sections of uncompensated ion chambers placed opposite the top and bottom halves of the core. Comparison of the signals from the two detectors gives an indication of the core axial offset or imbalance. This imbalance signal (top core power minus bottom core power) is monitored in the control room. When an imbalance is indicated, the operator will move the APSR's in the direction of the imbalance to reduce the axial offset, i.e.,

positive offset - move APSR's toward top;

negative offset - move APSR's toward bottom.

The integrated control system will automatically compensate for reactivity changes and consequent power swings caused by the part length control rod movement.

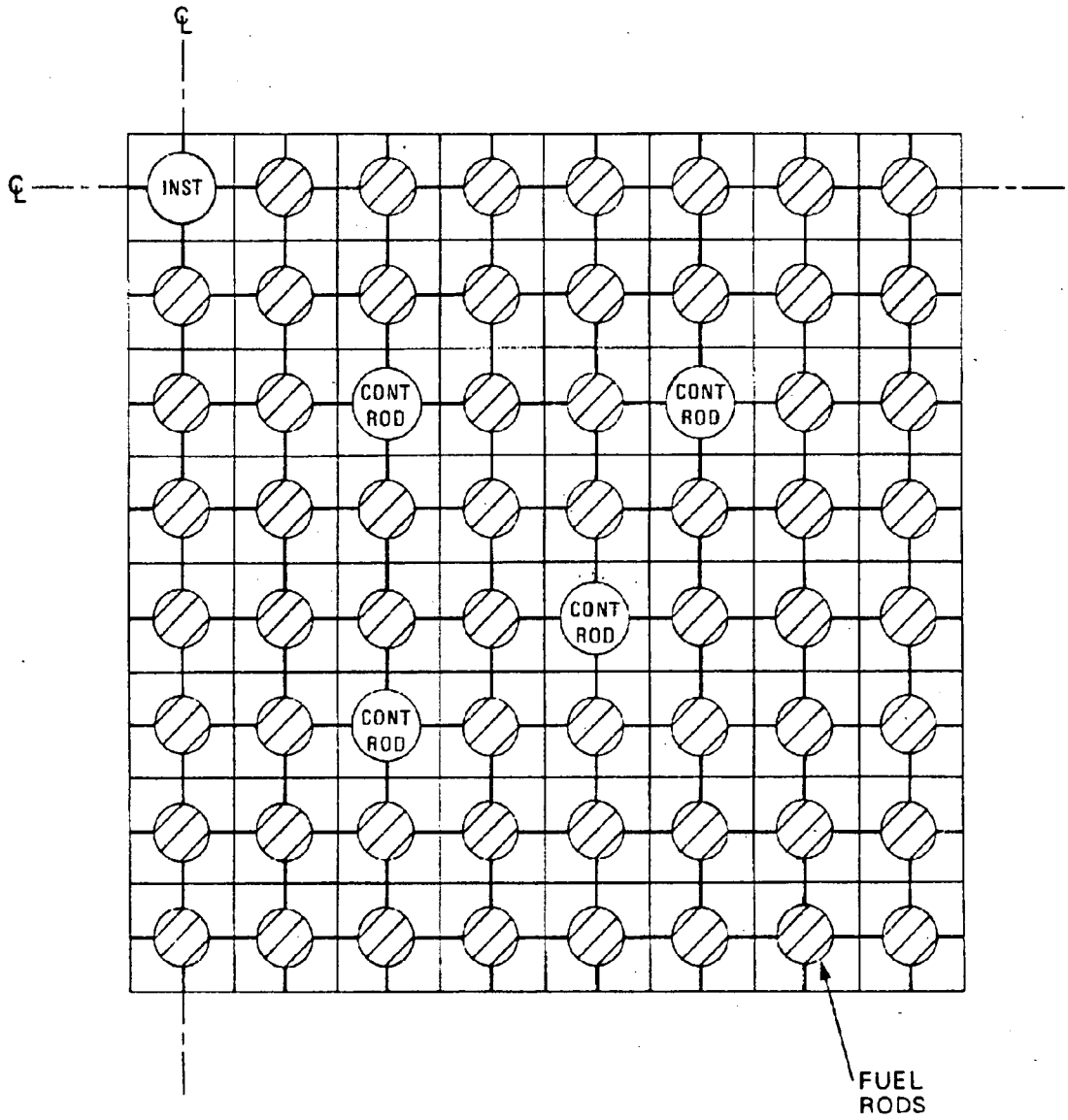
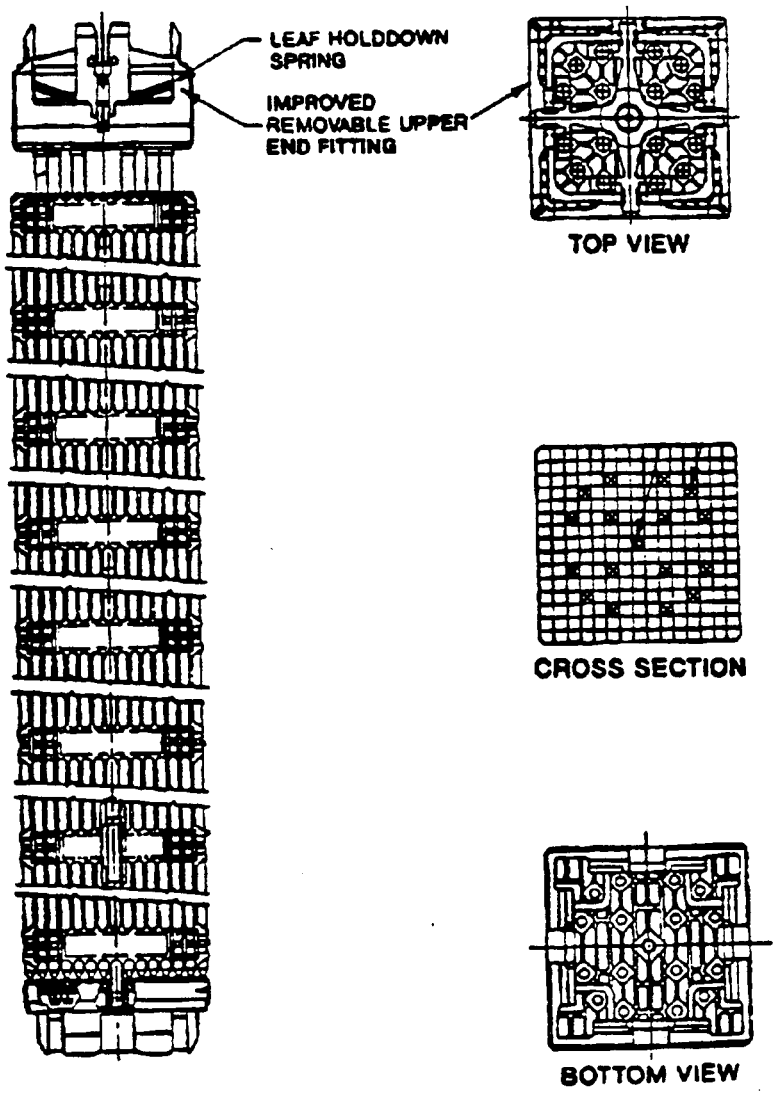



Figure 4-25.
Subchannel Geometry

MARK B-10 AND B-10F FUEL ASSEMBLY




5 Figure 4-37.
5 Mark B-10 and B-10F Fuel Assembly

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# CHAPTER 5. REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

# 5.1 SUMMARY DESCRIPTION

## 5.1.1 GENERAL

### 5.1.1.1 System

The Reactor Coolant System consists of the reactor vessel, two vertical once-through steam generators, four shaft-sealed reactor coolant pumps, an electrically heated pressurizer and interconnecting piping. The system is arranged in two heat transport loops, each with two reactor coolant pumps and one steam generator. The reactor coolant is transported through piping connecting the reactor vessel to the steam generators and flows downward through the steam generator tubes transferring heat to the steam and water on the shell side of the steam generator. In each loop, the reactor coolant is returned to the reactor through two lines, each containing a reactor coolant pump, to the reactor vessel. In addition to serving as a heat transport medium, the coolant also serves as a neutron moderator and reflector, and a solvent for the soluble poison (boron in the form of boric acid). The system pressure settings are listed in Table 5-1; the integrity of the reactor coolant pressure boundary is described in Section 5.2, "Integrity of Reactor Coolant Pressure Boundary" on page 5-9; the reactor vessel design is described in Section 5.3, "Reactor Vessel" on page 5-43; and other major components and subsystems in the reactor coolant pressure boundary (RCPB) are described in Section 5.4, "Component and Subsystem Design" on page 5-49.

The Reactor Coolant System piping diagrams are Figure 5-1 (Oconee 1) and Figure 5-2 (Oconee 2 & 3).

In 1970, the Oconee 1 reactor coolant pumps were replaced with Westinghouse Model 93A pumps. The reactor coolant piping was modified slightly to accommodate the replacement pumps. Both the original pumps and the replacement pumps were bottom suction and side discharge allowing installation of the replacement pumps on the same centerlines as the original pumps. The original motors were utilized with the replacement pumps.

Figure 5-3 and Figure 5-4 show the revised arrangement of the reactor coolant piping for Oconee 1.

### 5.1.1.2 System Protection

Engineered safety features and associated systems are protected from missiles which might result from a loss of coolant accident. Protection is provided by concrete shielding and/or segregation of redundant components.

The reactor vessel is surrounded by a concrete primary shield wall and the heat transport loops are surrounded by a concrete secondary shield wall. These shielding walls provide missile protection for the Reactor Building liner plate and equipment located outside the secondary shielding.

Removable concrete slabs over the reactor vessel area and the concrete deck over the area outside of the secondary shield wall also provide shielding and missile protection.

The Reactor Coolant System is analyzed for maximum hypothetical earthquake to determine that resultant stresses do not jeopardize the safe shutdown of the Reactor Coolant System and removal of decay heat.

### 5.1.1.3 System Arrangement

The system arrangement in relation to shielding walls, the Reactor Building and other equipment in the building are described in Chapter 1, "Introduction and General Description of Plant" on page 1-1. Plan and elevation drawings showing principal dimensions of the Reactor Coolant System in relation to the supporting or surrounding concrete structures are provided in Figure 5-3, Figure 5-4 (Oconee 1), Figure 5-5, Figure 5-6 (Oconee 2) and Figure 5-7, Figure 5-8 (Oconee 3).

### 5.1.1.4 System Parameters

#### 5.1.1.4.1 Flow

The Reactor Coolant System is designed on the basis of 176,000 gpm flow rate in each heat transport loop.

#### 5.1.1.4.2 Temperatures

Reactor Coolant System temperatures as a function of power are shown in Figure 5-9. The system is controlled to a constant average temperature throughout the power range from 15 percent to 100 percent full power. The average system temperature is decreased between 15 percent and 0 percent of full power to the saturation temperature at 900 psia.

#### 5.1.1.4.3 Heatup

All Reactor Coolant System components are designed for a continuous heatup rate of 100°F/hr.

#### 5.1.1.4.4 Cooldown

All Reactor Coolant System components are structurally designed for a continuous cooldown rate of 100°F/hr. System cooldown to 250°F is accomplished by use of the steam generators and by bypassing steam to the condenser with the Turbine Bypass System. The Low Pressure Injection System provides the heat removal for system cooldown below 250°F.

#### 5.1.1.4.5 Volume Control

The only coolant removed from the Reactor Coolant System is that which is letdown to the High Pressure Injection System. The letdown flow rate is set at the desired rate by the operator positioning the letdown control valve and/or opening the stop valve for the letdown orifice.

To maintain a constant pressurizer water level, total makeup to the Reactor Coolant System must equal that which is letdown from the system. Total makeup consists of the seal injection water through the reactor coolant pump shaft seals and makeup returned to the system through the reactor coolant volume control valve (High Pressure Injection System). The pressurizer level controller provides automatic control of the valve to maintain the desired pressurizer water level. Reactor coolant volume changes during plant load changes exceed the capability of the reactor coolant volume control valve, and thus result in variations in pressurizer level. The level is returned to normal as the system returns to steady state conditions.

#### 5.1.1.4.6 Chemical Control

Control of the Reactor Coolant Chemistry is a function of the Chemical Addition and Sampling System. Sampling lines from the letdown line of the High Pressure Injection System provide samples of the reactor coolant for chemical analysis. All chemical addition is made from the Chemical Addition and Sampling

System to the High Pressure Injection System. See Chapter 9, "Auxiliary Systems" on page 9-1 for detailed information concerning the Chemical Addition and Sampling System and Chapter 6, "Engineered Safeguards" on page 6-1 for the High Pressure Injection System.

#### 5.1.1.4.7 Boron

Boron in the form of boric acid is used as a soluble poison in the reactor coolant. Concentrated boric acid is stored in the Chemical Addition and Sampling System and is transported to the Reactor Coolant
2 System in the same manner as described above for chemical addition. The concentrated boric acid may
2 be stored in the concentrated boric acid storage tank (CBAST) or directly in the boric acid mix tank. The
2 CBAST receives concentrated boric acid from the boric acid mix tank. The CBAST is required to
2 contain a specified concentration of boric acid based on the volume in the tank in order to supply a
2 source of concentrated soluble boric acid to the Reactor Coolant System in addition to the borated water
2 storage tank. The concentrated boric acid is pumped to the High Pressure Injection System which
2 transports it to the Reactor Coolant System. Boron concentrations are reduced by running letdown flow
2 through the deborating demineralizers and/or diluting the reactor coolant with demineralized water. All bleed and feed operations for changing the boric acid concentrations of the reactor coolant are made between the High Pressure Injection System and the Coolant Storage System.

#### 5.1.1.4.8 pH

The pH of the reactor coolant is controlled to minimize corrosion of the Reactor Coolant System surfaces which minimizes coolant activity and radiation levels of the components.

#### 5.1.1.4.9 Water Quality

The reactor coolant water chemistry specifications have been selected to provide the necessary boron content for reactivity control and to minimize corrosion of Reactor Coolant System surfaces. The solids content of the reactor coolant is maintained below the design level by minimizing corrosion through chemistry control and by continuous purification by the demineralizer of the High Pressure Injection System. Excess hydrogen is maintained in the reactor coolant to chemically combine with the oxygen produced by radiolysis of the water.

#### 5.1.1.4.10 Vents and Drains

Vent and drain lines are located at the high and low points of the system and provide the means for draining, filling, and venting the heat transport loops and pressurizer. The reactor vessel cannot be drained below the top of the reactor outlet nozzle using these drain lines. Each vent and drain line contains two manual valves in series. Vent lines are routed to a header connected to the quench tank gas space and drain lines are routed to a header connected to the suction of the component drain pump.

## 5.1.2 PERFORMANCE OBJECTIVES

### 5.1.2.1 Steam Output

The Reactor Coolant System is designed to operate at a core power level of 2,568 MWt and transfer a total of 2,584 MWt (including 16 MWt input from reactor coolant pumps) to the steam generators. The system will produce a total steam flow of 11.2 million lb/hr.

### 5.1.2.2 Transient Performance

The Reactor Coolant System will follow step or ramp load changes under automatic control without relief valve or turbine bypass valve action as follows:

#### 5.1.2.2.1 Step Load Changes

Increasing or decreasing load steps of 10 percent of full power in the range between 20 percent and 90 percent full power.

#### 5.1.2.2.2 Ramp Load Changes

Increasing load ramps of 10 percent per minute between 20 percent and 90 percent full power or decreasing load ramps of 10 percent per minute between 100 percent and 15 percent full power are acceptable. Between 15 percent and 20 percent, and between 90 percent and 100 percent full power, increasing ramp load changes of 5 percent per minute are acceptable.

The combined actions of the Control System and the Turbine Bypass System permit a 40 percent load rejection or a turbine trip from 40 percent full power without safety valve action. The combined actions of the Control System, the turbine bypass valves, and the main steam safety valves are designed to accept separation of the generator from the Transmission System without reactor trip.

### 5.1.2.3 Partial Loop Operation

The Reactor Coolant System will permit operation with less than four reactor coolant pumps in operation. The nominal steady-state operating power levels for combinations of reactor coolant pumps operating are as follows:

| Reactor Coolant Pumps Operating | Rated Power, % |
|---|---|
| 4 | 100 |
| 3 | 75 |

2

### 5.1.2.4 Natural Circulation

Natural circulation provides an acceptable method of energy removal from the core with transfer of energy to the Secondary System through the steam generators. The controlling parameters which determine the magnitude of the natural circulation flow rates, i.e., steam generator liquid level and source of feedwater (emergency or main), produce more than adequate circulation rates under steady conditions. The margins to the limits for acceptable operation are more than adequate for steady-state and expected transients.

Natural circulation cooldown mode of operation is not expected to be undertaken at Oconee Nuclear Station except for SBLOCA events which do not allow continued operation of or restart of reactor coolant pumps. In all other situations, procedures recommend that hot shutdown be maintained until those systems required for forced circulation are put back into service.

In response to Generic Letter 81-21, Duke has developed a procedure to continuously vent the reactor vessel head to containment during a natural circulation cooldown to Decay Heat Removal System conditions, as well as prevent upper head voiding. NRC Safety Evaluation Report (Reference 1 on

page 5-8) concurs with Duke that natural circulation cooldown is not a safety concern due to operator training and procedures.

## 5.1.3 REFERENCES

1. Letter from J. F. Stolz (NRC) to H. B. Tucker (Duke) dated June 5, 1985. Subject: NRC Safety Evaluation Report on Duke Response to Generic Letter 81-21 Natural Circulation Cooldown.

# 5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

## 5.2.1 DESIGN CONDITIONS

### 5.2.1.1 Pressure

The Reactor Coolant System components are designed structurally for an internal pressure of 2,500 psig.

### 5.2.1.2 Temperature

The Reactor Coolant System components are designed for a temperature of 650°F with the exception of the pressurizer, surge line, and a portion of the spray line piping which are designed for 670°F.

### 5.2.1.3 Reactor Loads

Reactor Coolant System components are supported and interconnected so that stresses resulting from combined mechanical and thermal forces are within established code limits. Equipment supports are designed to transmit piping rupture reaction loads to the foundation structures.

The Reactor Coolant System supports are on an eight foot six inch thick, heavily reinforced concrete slab which rests on a solid rock subgrade. The minimum ultimate crushing strength of rock cores tested was 720 kips per square foot and the maximum applied dynamic gross load is 30 kips per square foot. Based on the subgrade, the ratio of applied load to bearing capacity of the subgrade, and the monolithic nature of the base slab, differential settlement of the foundation is not anticipated.

### 5.2.1.4 Cyclic Loads

All Reactor Coolant System components are designed to withstand the effects of cyclic loads due to
1 system temperature and pressure changes. Design transient cycles are shown in Table 5-2 and
1 Table 5-23.

Flow-induced vibration analyses have been performed for the fuel assembly, including fuel rods, and for the reactor internals components. The analyses and design criteria for the thermal shield, flow distributor assembly, surveillance holder tubes and shroud tubes, and the "U" baffles are given in B&W Topical Report BAW-10051, Reference 1 on page 5-41.

Components subjected to cross flow are checked for response during design, so that the fundamental frequencies associated with cross flow are above the vortex shedding frequencies. It has also been conservatively determined that the flow induced pressure fluctuations acting on the disc of the vent valve are such that for normal operation there is always a positive net closing force acting on the disc. Emergency operational modes are covered in B&W Topical Report BAW-10008, Part 1, and BAW-10035, References 2 on page 5-41 and 3 on page 5-41.

### 5.2.1.5 Seismic Loads and Loss-of-Coolant Loads

Reactor Coolant System components are designated as Class I equipment and are designed to maintain their functional integrity during earthquake. Design is in accordance with the seismic design bases shown below. The loading combinations and corresponding design stress criteria for internals and pressure

boundaries of vessels and piping are given in the section. A discussion of each of the cases of loading combinations follows:

#### 5.2.1.5.1 Seismic Loads

Case I - Design Loads Plus Design Basis Earthquake (DBE) Loads - For this combination, the reactor must be capable of continued operation; therefore, all components excluding piping are designed to Section III of the ASME Code for Reactor Vessels. The primary piping is designed according to the requirements of USAS B31.1 and B31.7. The $S_m$ values for all components, excluding bolting, are those specified in Table N-421 of the ASME code. The $S_m$ value for bolts are those specified in Table N-422 of the ASME Code.

CASE II - Design Loads Plus Maximum Hypothetical Earthquake (MHE) - In establishing stress levels for this case, a "no-loss-of-function" criterion applies, and higher stress values than in Case I can be allowed. The multiplying factor of 1.2 has been selected in order to increase the code-based stress limits and still insure that for the primary structural materials, i.e., 304 SST, 316 SST, SA302B, SA2102B, and SA106C, an acceptable margin of safety will always exist. A more detailed discussion of the adequacy of these margins of safety is given in BAW-10008, Part 1, "Reactor Internals Stress & Deflection Due to LOCA and Maximum Hypothetical Earthquake." The $S_m$ value for all components are those specified in Table N-421 of the ASME Code.

#### 5.2.1.5.2 Loss-of-Coolant Loads

A loss-of-coolant accident coincident with a seismic disturbance has been analyzed to assure that no loss of function occurs. In this case, primary attention is focused on the ability to initiate and maintain reactor shutdown and emergency core cooling. Two additional cases are considered as follows:

Case III - Design Loads Plus Pipe Rupture Loads - For this combination of loads, the stress limits for Case II are imposed for those components, systems, and equipment necessary for reactor shutdown and emergency core cooling.

Case IV - Design loads plus Maximum Hypothetical Earthquake (MHE) Loads Plus Pipe Rupture Loads - Two thirds of the ultimate strength has been selected as the stress limit for the simultaneous occurrence of MHE and reactor coolant pipe rupture. As in Case III, the primary concern is to maintain the ability to shut the reactor down and to cool the reactor core. This limit assures that a materials strength margin of safety of 50 percent will always exist.

The design allowable stress of Case IV loads is given in BAW-10008 for 304 stainless steel. This curve is used for all reactor vessel internals including bolts. It is based on adjusting the ultimate strength curves published by U.S. Steel to minimum ultimate strength values by using the ratio of ultimate strength given by Table N-421 of Section III of the ASME code at room temperature to the room temperature strength given by U.S. Steel.

In Cases II, III, and IV, secondary stresses were neglected, since they are self-limiting. Design stress limits in most cases are in the plastic region, and local yielding would occur. Thus, the conditions that caused the stresses are assumed to have been satisfied. BAW-10008, Part 1, contains a more extensive discussion of the margin of safety, the effects of using elastic equations, and the use of limit design curves for reactor internals. Table 5-3 provides the stress limits for seismic, pipe rupture, and combined loads.

### 5.2.1.6 Service Lifetime

The design service lifetime for the major Reactor Coolant System components is 40 years. The number
1 of cyclic system temperature and pressure changes (Table 5-2 and Table 5-23), is based on operation for this design lifetime.

### 5.2.1.7 Water Chemistry

The water chemistry is selected to provide the necessary boron content for reactivity control and to minimize corrosion of the Reactor Coolant System surfaces. To ensure the best protection is provided, reactor coolant water quality specifications are based upon the most current revision of the EPRI PWR Primary Water Chemistry Guidelines and vendor recommendations. These are addressed in the Chemistry Section Manual.

### 5.2.1.8 Vessel Radiation Exposure

The reactor vessel is the only Reactor Coolant System component exposed to a significant level of
1 neutron irradiation and is therefore the only component subject to material radiation damage. The
1 predicted exposure from fast neutrons (E > 1.0 MeV) at the inside vessel surface over a 40-year life with
2 an 80 percent load factor has been computed to be as follows (per BAW-2108, Rev. 1, Reference 16 on
2 page 5-41 and BAW-2143P, Reference 23 on page 5-41):

2 Oconee Unit 1 $9.04 \times 10^{18}$ neutrons/cm$^2$
2 Oconee Unit 2 $9.57 \times 10^{18}$ neutrons/cm$^2$
2 Oconee Unit 3 $9.39 \times 10^{18}$ neutrons/cm$^2$

## 5.2.2 CODES AND CLASSIFICATIONS

The codes listed in this section and Table 5-4 include the code addenda and case interpretations issued through Summer 1967 unless noted otherwise. Quality control and quality assurance programs relating to the fabrication and erection of system components are summarized in Section 5.2.3.11, "Quality Assurance" on page 5-32.

5 For inservice inspection of all three units, the applicable ASME Boiler and Pressure Vessel Code is: 1989
5 Edition, no Addenda.

### 5.2.2.1 Vessels

The design, fabrication, inspection and testing of the reactor vessel and closure head, steam generator (both reactor coolant side and secondary side), pressurizer and attachment nozzles on the vessels is in accordance with the ASME Boiler and Pressure Vessel Code, Section III, for Class A vessels.

### 5.2.2.2 Piping

The design, fabrication, inspection and testing of the reactor coolant piping excluding the pressurizer surge line and the spray line is in accordance with USAS B31.7, Code for Pressure Piping, Nuclear Power Piping, dated February, 1968, and as corrected for Errata under date of June, 1968. The pressurizer surge and spray lines were fabricated and initially inspected in accordance with USAS B31.7, February 1968 with June, 1968, Errata. However, the surge line has been analyzed in accordance with the ASME Code, 1977 edition, Summer 1979 Addenda; while the spray line has been analyzed to the 1980 Edition of the ASME Code. The feedwater header and the auxiliary feedwater header for the steam generator meet the requirements of the Code for Pressure Piping, Power Piping USAS B31.1.0 - 1967.

#### 5.2.2.3 Reactor Coolant Pumps

The reactor coolant pump casings are designed, fabricated, inspected and tested to meet the intent of the ASME Boiler and Pressure Vessel Code, Section III, for Class A vessels, but are not code stamped.

#### 5.2.2.4 Relief Valves

The pressurizer code safety valves and the electromatic relief valve comply with Article 9, Section III, of the ASME Boiler and the Pressure Vessel Code.

#### 5.2.2.5 Welding

Welding qualifications are in accordance with the ASME Boiler and Pressure Vessel Code, Section III and Section IX and Section XI, as applicable.

### 5.2.3 SYSTEM DESIGN EVALUATION

#### 5.2.3.1 Design Margin

The Reactor Coolant System is designed structurally for 2,500 psig and 650°F. The system will normally operate at 2,155 psig and 604°F.

In the event of a complete loss of power to all reactor coolant pumps, reactor coolant flow, coastdown, and subsequent natural circulation flow is more than adequate for core cooling and decay heat removal as shown by the analysis in Chapter 15, "Accident Analyses" on page 15-1.

1 The number of transient cycles specified in Table 5-2 and Table 5-23 for the fatigue analysis is conservative.

#### 5.2.3.2 Material Selection

Each of the materials used in the Reactor Coolant System has been selected for the expected environment and service conditions. The major component materials are listed in Table 5-5. All Reactor Coolant System materials normally exposed to the coolant are corrosion-resistant materials consisting of 304 or 316 stainless steel, Inconel, 17-4PH (H1100), Zircaloy, or weld deposits with corrosion-resistant properties equivalent to or better than those of 304 SS. These materials were chosen for specific uses at various locations within the system because of their compatibility with the reactor coolant. There are no novel material applications in the Reactor Coolant System.

To assure long steam generator tube lifetime, feedwater quality entering the steam generator is maintained as high as practical. The current revision of the SGOG EPRI PWR Secondary Chemistry Guidelines and vendor recommendations are used to prepare operating specifications which are addressed in the Chemistry Section Manual.

The selection of materials and the manufacturing sequence for the Reactor Coolant System components, is arranged to insure that no pressure boundary material is furnace-sensitized stainless steel. Safe ends are provided on those carbon steel nozzles of the system vessels which connect to stainless steel piping. All dissimilar metal welds, with the exception of Inconel to Stainless Steel pipe welds, will be made in the manufacturer's shops.

Piping systems designed to resist seismic forces have been restrained by steel supports capable of withstanding these seismic forces. The restraints also act as pipe stops restraining the lines against

whipping. In systems, where it was necessary to use hydraulic snubbers to resist seismic forces, the mechanical action associated with the snubbers makes it possible to consider them as restraints against pipe whipping. When a seismic acceleration equal to or greater than two feet/sec/sec acts on the system, a differential pressure is generated on the ends of the snubber valve piston which is spring centered. This differential pressure is sufficient to cause the piston to shift and close the by-pass ports. With the by-pass ports closed, the snubber acts as a rigid structural member, thus limiting any further movement of the pipe at the point of attachment.

The basic design criteria for pipe whip protection is as follows:

1. All penetrations are designed to maintain containment integrity for any loss of coolant accident combination of containment pressures and temperatures.
2. All penetrations are designed to withstand line rupture forces and moments generated by their own rupture as based on their respective design pressures and temperatures.
3. All primary penetrations, and all secondary penetrations that would be damaged by a primary break, are designed to maintain containment integrity.
4. All secondary lines whose break could damage a primary line and also breach containment are designed to maintain containment integrity.

The pressure boundary of the RCS is fabricated primarily from ferritic materials, while that of the attached systems is fabricated primarily from austenitic material.

Consequently, the RCS components are the only ones that require special protection against nonductile failure and that must comply with the fracture toughness requirements of Appendix G to 10 CFR 50. This protection is ensured by establishing pressure-temperature limitations on the RCS. The margin of safety is controlled by not exceeding the calculated allowable pressure at any given temperature. The following loading conditions require pressure-temperature limits:

1. Normal operations including bolt preloading, heatup, and cooldown.
2. Preservice system hydrostatic test.
3. Inservice system leak and hydrostatic tests.
4. Reactor core operation.

For a better understanding of the required protection against non-ductile failure, typical operational parameters of the RCS are described in the following sections for each of the loading conditions.

#### 5.2.3.2.1 Normal Operation

During bolt preload, the reactor vessel closure studs are tensioned to the specified load. Bolt preloading is not allowed until the reactor coolant temperature or the volumetric average temperature of the closure head region (including the studs) is higher than the specified minimum preload temperature. After the studs are tensioned, system pressure can be increased by the pressurizer until it is above the net positive suction head (NPSH) required for reactor coolant pump (RCP) operation. The heatup transient begins when the RCP is started.

During heatup, the RCS is brought from a cold shutdown condition to a hot shutdown condition. The heat sources used to increase the temperature of the system are the RCP and any residual (decay) heat from the core. Normally, when the pumps are started, the temperature of the water in the pressurizer is about 400°F; this corresponds to the pressure in the RCS, which is about 300 psig. The coolant temperature is at or above the minimum specified bolt preload temperature.

Initially, the reactor coolant temperature may be as low as room temperature for initial core loading or as high as 130°F for subsequent refueling. The system pressure is maintained below the maximum allowable pressure of approximately 625 psig (20 percent of preoperational system hydrostatic test pressure) until the reactor coolant temperature is approximately 270°F.

At any given time throughout the heatup transient, the temperature of the reactor coolant is essentially the same throughout the system except, of course, in the pressurizer. The system pressure, as controlled by the pressurizer heaters, is maintained between the minimum required for RCP NPSH and the maximum established to meet the fracture toughness requirements. The heatup rate is maintained below the maximum rate used to establish the maximum allowable pressure-temperature limit curve.

RCS cooldown brings the system from a hot to a cold shutdown condition. The cooldown is normally accomplished in two phases: The first phase reduces the fluid temperature from approximately 550°F to below the design temperature of the decay heat removal system (approximately 300°F). This temperature reduction is accomplished using the steam generators but bypassing the turbine and dumping the steam directly to the condenser. Once below its design temperature (and pressure), the Decay Heat Removal System (DHRS) is activated in the second phase to further reduce the reactor coolant temperature to that desired.

Before cooldown, the RCS temperature is maintained constant by balancing the heat removal rate from the steam dump with the heat contributed by the RCP and core decay heat. The system pressure is maintained by the pressurizer. The cooldown is normally initiated by stopping one RCP in each loop. The two remaining pumps provide coolant circulation through both steam generators, and the turbine steam bypass flow controls the cooldown rate. The primary pressure during cooldown is controlled with the pressurizer heaters and spray. After cooling down below the DHRS design temperature and pressure, the cooling mode is changed from the steam generators to the DHRS. Before the switch, the RCS pressure is below 625 psig (20 percent of preoperational system hydrostatic test pressure) and below the DHRS pressure but above the pressure required for the RCP to operate.

To minimize the thermal shock on the RCPB, the two RCP remain in operation as the water flow of the DHRS is initiated. The DHRS flow rapidly mixes with the reactor coolant; but during this period, the indicated RCS temperature may fluctuate until mixing is complete. After the switch is completed, the RCP are stopped. During this phase, the cooldown rate is controlled by the temperature and flow of the DHRS.

#### 5.2.3.2.2 Preservice System Hydrostatic Test

Prior to initial operation, the RCS is hydrostatically tested in accordance with ASME Code requirements. During this test, the system is brought up to an internal pressure not less than 1.25 times the system design pressure. This minimum test pressure is in accordance with Article NB-6000 of ASME Section III. Since the system design pressure is 2500 psig, the preservice system hydrostatic test pressure is 3125 psig. Initially, the RCS is heated to a temperature above the calculated minimum test temperature required for adequate fracture toughness. This heatup is accomplished by running the RCP. The pressurizer heaters are used to heat the pressurizer to the required temperature. Before the test temperature is reached, the pressure is maintained above NPSH required for the RCP but below the maximum allowable pressure for adequate fracture toughness. When the test temperature is reached, the RCP are stopped and RCS makeup water is added to fill the pressurizer. The test pressure is then reached using either the pressurizer heaters or the hydrostatic pumps connected to the RCS. The test pressure is held for the minimum specified time, and the examination for leakage follows in accordance with the ASME Code.

#### 5.2.3.2.3 Inservice System Leak and Hydrostatic Tests

When the inservice system leak and hydrostatic tests are required, the system is brought from a cold to a hot shutdown condition. The means of heating the system and increasing the pressure are the same as those used during normal heatup. If it is necessary to cool the system down after either test, normal cooldown procedures are used. These two tests are conducted in accordance with the requirements of ASME Section XI, Article IWB-5000. The test pressure for the inservice leak tests is the pressure that, for the component located at the highest elevation in the system, is no less than the system nominal operating pressure at 100 percent rated reactor power. For the inservice hydrostatic test, ASME Section XI gives a table of the minimum test pressure versus the test temperature at which the system must be tested. However, the test temperature for both the inservice leak and hydrostatic tests is determined by the requirements for fracture toughness.

#### 5.2.3.2.4 Reactor Core Operation

The reactor core is not allowed to become critical until the RCS fluid temperature is above 525°F except for brief periods of low-power physics testing. This temperature is much higher than the minimum permissible temperature for the inservice system hydrostatic pressure test, and it is also at least 40°F above the calculated minimum temperature required at normal pressure for operation throughout the service life of the plant.

### 5.2.3.3 Reactor Vessel

The ability of the reactor pressure vessel to resist fracture is the primary factor in ensuring the safety of the primary system in light water cooled reactors. The beltline region of the reactor vessel is the most critical region of the vessel because it is exposed to neutron irradiation. The general effects of fast neutron irradiation on the mechanical properties of such low-alloy ferritic steels as SA302B, Code Case 1339, used
1 in the fabrication of the Oconee 1 reactor vessel, and SA508, Class 2, used in the fabrication of Oconee 2
1 and 3 reactor vessels, are well characterized and documented in the literature. The low-alloy ferritic steels used in the beltline region of reactor vessels exhibit an increase in ultimate and yield strength properties with a corresponding decrease in ductility after irradiation. In reactor pressure vessel steels, the most serious mechanical property change is the increase in temperature for the transition from brittle to ductile
1 fracture accompanied by a reduction in the Charpy upper-shelf impact strength.

1 10 CFR 50, Appendix G, "Fracture Toughness Requirements," specifies minimum fracture toughness requirements for the ferritic materials of the pressure-retaining components of the reactor coolant pressure boundary (RCPB) of water-cooled power reactors and provides specific guidelines for determining the pressure-temperature limitations on operation of the RCPB. The toughness and operational requirements are specified to provide adequate safety margins during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be
1 subjected over its service lifetime. Although the requirements of 10 CFR 50, Appendix G, became effective on August 13, 1973, the requirements are applicable to all boiling and pressurized water-cooled nuclear power reactors, including those under construction or in operation on the effective date.

1 10 CFR 50, Appendix H, "Reactor Vessel Materials Surveillance Program Requirements," defines the material surveillance program required to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region of water-cooled reactors resulting from exposure to neutron irradiation and the thermal environment. Fracture toughness test data are obtained from material specimens withdrawn periodically from the reactor vessel. These data will permit determination of the condition under which the vessel can be operated with adequate safety margins against fracture throughout its service life.

A method for guarding against brittle fracture in reactor pressure vessels is described in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G. This method utilizes fracture mechanics concepts and the reference nil-ductility temperature, $RT_{NDT}$, which is defined in ASME Section III, Paragraph NB 2331. The $RT_{NDT}$ of a given material is used to index that material to a reference stress intensity factor curve ($K_{IR}$ curve), which appears in Appendix G of ASME Section III. The $K_{IR}$ curve is a lower bound of dynamic, static, and crack arrest fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the $K_{IR}$ curve, allowable stress intensity factors can be obtained for the material as a function of temperature. Allowable operating limits can then be determined using the allowable stress intensity factors.

The $RT_{NDT}$ and, in turn the operating limits of a nuclear power plant, can be adjusted to account for the effects of radiation on the properties of the reactor vessel materials. The radiation embrittlement and the resultant changes in mechanical properties of a given pressure vessel steel can be monitored by a surveillance program in which a surveillance capsule containing prepared specimens of the reactor vessel materials is periodically removed from the operating nuclear reactor and the specimens tested. The increase in the Charpy V-notch 30-ft-lb temperature, or the increase in the 35 mils of lateral expansion temperature, whichever results in the larger temperature shift due to irradiation, is added to the original $RT_{NDT}$ along with a margin value to adjust the $RT_{NDT}$ for radiation embrittlement. This adjusted $RT_{NDT}$ is used to index the material to the $K_{IR}$ curve, which, in turn is used to set operating limits for the nuclear power plant. These new limits take into account the effects of irradiation on the reactor vessel materials.

#### 5.2.3.3.1 Stress Analysis

A stress evaluation of the reactor vessel was initially performed in accordance with Section III of the ASME Code. The evaluation showed that stress levels are within the Code limits.

Table 5-6 lists the reactor vessel steady-state stresses at various load points. The results of the transient analysis and the determination of the fatigue usage factor at the same load points are listed in Table 5-7. As specified in the ASME Code, Section III, Paragraph 415.2(d)(6), the cumulative fatigue usage factor is less than 1.0 for the design cycles listed in Table 5-2. Figure 5-10 illustrates the points of stress analysis for the stresses listed in Table 5-6 and the fatigue usage factors listed in Table 5-7.

These stress summaries demonstrate that all of the requirements for stress limits and fatigue required by ASME Section III for all of the operational requirements imposed by the design specifications have been met. The values tabulated in these summaries are the maximum value obtained in each region. The imposed transients are based on description of the realistic behavior that might be expected for this plant. Transients such as loss of flow and load that cause temperature and pressure variations are included in the reactor vessel specification and Table 5-2. Their effect on accumulated usage factor is included in the stress analysis and in the values reported in included in the stress analysis and in the values reported in summary Table 5-7. These transients are not the major contributors to the largest usage factor of 0.38 for the stud bolts as given in Table 5-7.

#### 5.2.3.3.2 Reference Nil-Ductility Temperature ($RT_{NDT}$)

Throughout the lifetime of a reactor vessel, the impact and tensile properties of the ferritic beltline region materials will change because of neutron irradiation. These changes require periodic adjustment of pressure-temperature relationships for heatup and cooldown during normal, upset, and testing conditions.

To determine the pressure-temperature operating limitations for the RCPB the reference nil-ductility temperature ($RT_{NDT}$) of the ferritic materials must be established. The $RT_{NDT}$ is needed to calculate the critical stress intensity factor ($K_{IR}$). In ASME Section III, Appendix G, $K_{IR}$ is related to temperature, T, and to $RT_{NDT}$ by the following equation:

$$K_{IR} = 26.777 + 1.223 \exp[0.0145(T - RT_{NDT} + 160)] ksi\sqrt{in.}$$

This relationship is applicable only to ferritic materials that have a specified minimum yield strength of 50,000 psi or less at room temperature.

Since the impact properties of the beltline region materials of a reactor vessel will change throughout its lifetime, periodic adjustments are required on the pressure-temperature limit curves of the RCPB. The magnitude of these adjustments is proportional to the shift in $RT_{NDT}$ caused by neutron fluence. Therefore, it is essential to determine the radiation-induced $\Delta RT_{NDT}$ of the beltline region materials.

1

The $RT_{NDT}$ of the ferritic materials, which were specified and tested in accordance with the fracture toughness requirements of the ASME Section III Summer 1972 Addenda (to 1971 Edition) or subsequent addenda, are determined as required by that Code. When enough material is available, the $RT_{NDT}$ of those beltline region materials, which were specified and tested in accordance with an edition or addenda of ASME Section III prior to the Summer 1972 Addenda, are obtained by testing specimens oriented normal to the principal working direction. The test procedure is in accordance with ASME Section III, paragraph NB 2300 (Summer 1972 Addenda).

The Oconee pressure boundaries were designed and constructed in accordance with the requirements of an edition or addenda of ASME Section III issued before the Summer 1972 Addenda. Except for the beltline region materials for which sufficient test material is available, the $RT_{NDT}$ of the ferritic materials must be estimated. This is necessary because the test data required for the exact determination of $RT_{NDT}$ were not required by the applicable ASME Code.

Generally, drop weight tests were not performed, and the Charpy V-notch tests were limited to "fixed" energy level requirements for specimens oriented in the longitudinal (principal working) direction at a temperature of 40°F or lower.

To obtain an $RT_{NDT}$ estimate that is appropriately conservative, B&W has collected and evaluated the data from tests conducted on pressure-retaining ferritic materials to which the new fracture toughness requirements were applied. Based on these evaluations, techniques were developed to estimate $RT_{NDT}$. These techniques as well as the results are described in B&W Topical Report BAW-10046P, Reference 4 on page 5-41.

1 10 CFR 50, Appendix G, requires complete characterization of the unirradiated impact properties of all the beltline region materials of the reactor vessel. The complete characterization includes the determination of $RT_{NDT}$ and Charpy ($C_v$) test curves for the directions normal to and parallel to the principal working direction (other than the thickness direction). Appendix G also requires a minimum $C_v$
1 USE of 75 ft-lb for all beltline region materials unless it is demonstrated that lower values of upper-shelf fracture energy provide an adequate margin for deterioration from irradiation.

For the beltline region materials of reactor vessels that were specified in accordance with the requirements of an edition or addenda of ASME Section III issued before the Summer 1972 Addenda, the complete $C_v$ test curves, including $C_v$ USE, is determined when the material forms part of the reactor vessel surveillance program. For the beltline region materials that do not form part of the surveillance program, and when enough material is available, the $C_v$ test curve and USE are determined only in the direction normal to the principal working direction. No minimum Charpy V-notch USE are required, other than the 50 ft-lbs/35 mils of lateral expansion for the beltline region materials of these reactor vessels. When the unirradiated USE of these materials is below 75 ft-lb/, the procedures described in BAW-10046P are applied to predict the end-of-service USE.

The $C_v$ USE must be estimated for reactor vessel beltline region materials that were specified in accordance with the requirements of an edition or addenda of ASME Section III issued before the Summer 1972 Addenda and for which insufficient material is available for testing. All available data from tests conducted on reactor vessel beltline region materials were collected and evaluated in order to obtain an appropriately conservative estimate. Not all the data were obtained in accordance with the methods specified in ASME Section III, Appendix G, since in some cases the absorbed energy was obtained only at one temperature. Based on these evaluations, estimates of $C_v$ USE were developed. The techniques and results are described in BAW-10046P.

#### 5.2.3.3.3 Neutron Flux at Reactor Vessel Wall

The design value for the fast neutron flux greater than 1.0 MeV at the inner surface of the reactor vessel is $3.0 \times 10^{10}$ n/cm$^2$-sec at a rated power of 2,568 MWt. The most recent corresponding calculated maximum fast neutron flux at the vessel wall is approximately a factor of 3 lower. For 40 years at 80 percent load this corresponds to a fluence of approximately $1 \times 10^{19}$ n/cm$^2$ for the vessel wall.

A semiempirical method is used to calculate the surveillance capsule and reactor vessel flux. The method employs explicit modeling of the surveillance capsule, reactor vessel, and internals and uses a time-weighted average pin-by-pin core power distribution in the two-dimensional DOT IV, version 4.3, computer code. DOT IV is a two-dimensional code which is used to calculate the energy- and space-dependent neutron flux at all points of interest in the specific reactor system configuration. DOT IV employs the discrete ordinates method of solution of the Boltzmann transport equation and has multigroup and asymmetric scattering capability.

The calculational model is an R-theta geometric representation of a plan view through the reactor core midplane using one-eighth core symmetry. The model includes the core with a time-averaged radial power distribution core liner, coolant regions, core barrel, thermal shield, pressure vessel, and concrete. The DOT calculation is carried out with an $S_8$ order of angular quadrature, a $P_3$ expansion of the scattering matrix, and the CASK23E cross-section set. The $P_3$ order of scattering indicates a third order LeGendre polynomial scattering approximation which adequately describes the predominately forward scattering of neutrons observed in the deep penetration of steel and water media. This calculation provides the neutron flux as a function of energy at the detector position and, in addition to the flux, the DOT IV code calculates the saturated specific activity of the various neutron dosimeters located in the surveillance capsule using the ENDF/B5 dosimeter reaction cross-sections. The saturated activity of each dosimeter is then adjusted by a factor which corrects for fraction of saturation attained during the dosimeter's actual detailed irradiation history. Additional corrections are normally made to account for the effects of the following:

1. photon-induced fissions in the U and Np dosimeters,
2. short half-life of isotopes produced in Fe and Ni dosimeters, and
3. Pu-239 generated in the U-238 dosimter.

These calculated activities are used for comparison with the measured dosimeter activity values. The basic equation for the calculated activity ($\mu$Ci/g) is

$$D_i = \frac{N}{A_n\, 3.7\text{x}10^4} f_i \sum_E \sigma_n(E)\, \phi(E) \sum_{j=1} F_j(1 - e^{-\lambda_1 t_j}) e^{-\lambda_i (T - \tau_j)}$$

where:

1

| | | |
|---|---|---|
| $N$ | = | Avagadro's number, |
| $A_n$ | = | atomic weight of target material n, |
| $f_i$ | = | either weight fraction of target isotope in nth material or fission yield of desired isotope, |
| $\sigma_n(E)$ | = | group-averaged cross sections for material n |
| $\phi(E)$ | = | group-averaged fluxes calculated by DOT analysis, |
| $F_j$ | = | fraction of full power during jth time interval $t_j$, |
| $\lambda_i$ | = | decay constant of ith material, |
| 1 $t_j$ | = | length of the jth time period, |
| $T$ | = | sum of total irradiation time, i.e., residual time in reactor and wait time between reactor shutdown and counting, |
| $\tau_j$ | = | cumulative time from reactor startup to end of jth time period, i.e., |

$$\tau_j = \sum_{k=1}^{j} t_k.$$

1 The flux normalization factor $C_i$ is then obtained by the following equation:

$$C_i = \frac{D_i\,(\text{measured})}{D_i\,(\text{calculated})}$$

1 With C specified, the neutron fluence greater than 1 MeV can be calculated from

$$\phi t(E > 1.0\ \text{MeV}) = C \sum_{E=1}^{E=15\text{MeV}} \phi(E) \sum_{j=1}^{j=M} F_j t_j$$

where M is the number of irradiation time intervals; the other values are defined above.

The specific results of these calculations are included in the specific capsule evaluation reports prepared as part of the Reactor Vessel Materials Surveillance Program (FSAR Section 5.2.3.12, "Tests and Inspections" on page 5-33), and are referenced in Technical Specification 3.1.2.

#### 5.2.3.3.4 Radiation Effects

1 The adjusted reference temperatures are calculated by adding the predicted radiation-induced $\Delta RT_{NDT}$, the
1 unirradiated $RT_{NDT}$, and a margin value. The predicted $\Delta RT_{NDT}$ is calculated using the respective
1 neutron fluence and copper and nickel contents. The design curves of Regulatory Guide 1.99 were used to predict the radiation-induced $\Delta RT_{NDT}$ values as a function of the material's copper and phosphorous
1 content and neutron fluence. With the issuance of Rev. 2 of Regulatory Guide 1.99 in May, 1988,
1 $\Delta RT_{NDT}$ values are obtained on the basis of copper and nickel contents.

The effects of radiation on the Charpy USE level of the beltline region material is estimated using the
1 curves shown in Regulatory Guide 1.99, Rev. 2, Figure 2.

1

Several operating plant reactor vessels were manufactured with "high-copper MnMoNi/Linde 80"
1 submerged-arc weld metal. This class of weld metal is susceptible to relatively large changes in impact
1 properties when exposed to fast neutron irradiation. The Charpy V-notch upper-shelf energy ($C_v$USE) of
1 some of these welds may drop below the 50 ft-lb threshold required by federal regulatory requirements (10
1 CFR 50, Appendix G) during the 40-year reactor design life. Should the $C_v$USE drop below 50 ft-lb, certain corrective actions would be required that could severely impact plant availability.

One of the major goals of the B&W Owners Group Program has been to determine the period of time each 177-fuel assembly (FA) reactor vessel can operate without violating the 50 ft-lb $C_v$USE threshold. The work that has been completed in this program includes reports entitled "Prediction of Charpy Upper Shelf Energy Drop in Irradiated Weld Metals," "Pressure Vessel Fluence Analysis for 177-FA Reactors," "Chemistry of B&W 177-FA Owners Group Reactor Vessel Beltline Welds."

1 BAW-1803, Rev. 1, Reference 6 on page 5-41, describes the implementation of predictive methodology developed in this program to determine the service life to reach the 50 ft-lb $C_v$USE threshold for each of the Owners Group reactor vessels. It was also necessary to establish a means of predicting the pre-service $C_v$USE of each of the beltline region reactor vessel welds. The available $C_v$USE data obtained from B&W manufactured, early vintage welds (high-Cu MnMoNi/Linde 80 submerged-arc) were analyzed collectively for this purpose.

1 Based on the developed methods, the limiting Oconee reactor welds are predicted to exhibit a $C_v$USE of
1 more than 50 ft-lb as follows:

1 Oconee Unit 1 > 32 EFPY
1 Oconee Unit 2 23 EFPY
1 Oconee Unit 3 > 32 EFPY.

Completion of work in the other phases of the B&W Owners Group program will provide the justification for plant operation for the 40-year design life.

#### 5.2.3.3.5 Fracture Mode Evaluation

An analysis has been made to demonstrate that the reactor vessel can accommodate without failure the rapid temperature change associated with the postulated operation of the Emergency Core Cooling System (ECCS) at end of vessel design life. A summary of the evaluation follows:

The state of stress in the reactor vessel during the loss-of-coolant accident was evaluated for an initial vessel temperature of 603°F. The inside of the vessel wall is rapidly subjected to 90°F injection water of the maximum flow rate obtainable. The results of this analysis show that the integrity of the vessel is not violated.

The assumed modes of failure are ductile yielding and brittle fracture, which includes the nil-ductility approach and the fracture mechanics approach. The modes of failure are considered separately in the following paragraphs.

Ductile Yielding

The criterion for this mode of failure is that there shall be no gross yielding across the vessel wall using the minimum specified yield strength in the ASME Code, Section III. The analysis considered the maximum combined thermal and pressure stresses through the vessel wall thickness as a function of time

during the safety injection. Comparison of calculated stresses to the material yield stress indicated that local yielding may occur in the inner 8.0 percent of the vessel wall thickness.

Brittle Fracture

Because the reactor vessel wall in the core region is subjected to neutron flux resulting in embrittlement of the steel, this area was analyzed from both a nil-ductility approach and a fracture mechanics approach. The results of the two methods of analysis compare favorably and show that pressure vessel integrity is maintained.

The criterion used in the nil-ductility approach is that a crack cannot propagate beyond any point where the applied stress is below the threshold stress for crack initiation (5-8 ksi), or when the stress is compressive. This approach involves making the very conservative assumption that all of the vessel
1 material could propagate a crack by a low-energy absorption or cleavage mode. End-of-life vessel conditions were assumed. The crack arrest temperature through the thickness of the wall was developed on a stress-temperature coordinate system. The actual quench-induced, stress-temperature condition through the thickness of the wall at several times during the quench was developed and plotted. The maximum depth at which the material in the vessel wall would be in tension or at which the stress in the
1 material would be in excess of the threshold stress for crack initiation (5-8 ksi) was determined by comparison of the plots. The comparison showed that a crack could propagate only through the inner 35 percent of the wall thickness if a crack initiation threshold of 5-8 ksi is applicable.

The foregoing method of analysis is essentially a stress analysis approach which assumes the worst conceivable material properties and a flaw size large enough to initiate a crack. Actually, the outer 83 percent of the vessel wall is at a temperature above the Ductility Transition Temperature (DTT) (NDTT + 60°F) when credit is taken for the neutron shielding, and for the original DTT profile through the wall thickness. The analysis is conservative in that it does not deny that cracks can be initiated, and in that it assumed a crack from 1 to 2 ft long to exist in the vessel wall at the time of the accident. Therefore, it can be concluded that, if a crack were present in the worst location and orientation (such as a circumferentially oriented crack on the inside of the vessel wall), it could not propagate through the vessel wall.

A fracture mechanics analysis was conducted which assumed a continuous surface flaw to exist on the inside surface of the vessel wall. The criterion used for the analysis is that a crack cannot propagate when the stress intensity at the tip of the crack is below the critical crack stress intensity factor ($K_{IC}$). Topical Report BAW-10018, Reference 7 on page 5-41, provides the details of the analysis. This report includes an evaluation considering the Irwin fracture mechanics method and performs a sensitivity analysis of the effect of varying the conservatism of several major parameters on the result.

#### 5.2.3.3.6 Pressurized Thermal Shock

In response to the TMI Action Plan (Item II.K.2.13 "Thermal-Mechanical Report") the effect of cold high pressure injection water entering the reactor vessel during a small break loss of coolant accident or an overcooling transient was considered. The concern was that the cold injection water could rapidly cool the reactor vessel welds and that the resulting thermal stresses, coupled with the relatively high pressure stress on the vessel, would lead to a loss of vessel integrity. This type of event is a particular concern later in life as the vessel neutron fluence increases and the metal becomes more brittle. Various vendor, utility, and EPRI research performed in response to this action item showed that good mixing of the injection water with the warmer Reactor Coolant System fluid would occur, even under near zero loop flow conditions. In particular, the vent valves in the Oconee plant would provide a source of heated water flowing directly from the vessel upper plenum to the downcomer, thus mitigating the cooling effect of the injection flow. The NRC Staff concluded that there is reasonable assurance that vessel integrity would be maintained during a II.K.2.13 event (Reference 12 on page 5-41).

1 The NRC amended its regulations for light water nuclear power plants, effective July 23, 1985, to
1 establish a screening criterion related to the fracture resistance of PWR vessels during PTS events. Only
1 those plants that exceed the screening criterion are required to perform further analysis using Regulatory
1 Guide 1.154. All Oconee units passed the screening criterion (BAW-1895, Reference 20 on page 5-41)
1 and, therefore, met the regulations regarding the PTS concern. This rule was further amended on June
1 14, 1989, to make the definition of $RT_{PTS}$ equal to $RT_{NDT}$ in Regulatory Guide 1.99, Rev. 2.
1 Assessment in accordance with the amended rule is under preparation (BAW-2143, Reference 21 on
1 page 5-41). All Oconee units are anticipated to satisfy this revised screening criterion.

#### 5.2.3.3.7 Closure

The reactor closure is bolted to a ring flange on the reactor vessel. The vessel closure seal is formed by two concentric metal O-ring seals with provisions for leak-off between the O-rings. Reactor closure head leakage will be negligible from the annulus between the metallic O-ring seals during vessel steady-state and virtually all transient operating conditions. Only in the event of a rapid transient operation, such as an emergency cooldown, would there be some leakage past the inner-most O-ring seal. A stress analysis on a similar vessel design indicates this leak rate would be approximately 10 cc/min and no leakage would occur past the outer O-ring seal.

The reactor closure head is attached to the reactor vessel with sixty 6-1/2 in. diameter studs. The studs have a minimum yield strength of 130,000 psi. The studs, when tightened for operating conditions, will have a tensile stress of approximately 30,000 psi. An evaluation of stud failures shows that:

1. 10 adjacent studs can fail before leak occurs.
2. 25 adjacent studs can fail before the remaining studs reach yield strength.
3. 26 adjacent studs can fail before the remaining studs reach the ultimate tensile strength.
4. 43 symmetrically located studs can fail before the remaining studs reach yield strength.

The fatigue evaluation results of the studs is included in Table 5-7.

#### 5.2.3.3.8 Control Rod Drive Service Structure

The control rod drive service structure is designed to support the control rod drives to assure no loss of function in the event of a combined loss of coolant accident and maximum hypothetical earthquake. Requirements for rigidity, imposed on the structure to avoid adversely affecting the natural frequency of vibration of the vessel and internals, as well as space requirements for service routing, result in stress levels considerably lower than design limits. The structure is more than adequate to perform its required function.

#### 5.2.3.3.9 Control Rod Guide Mechanism

Appendix G to 10 CFR 50 requires that the adequacy of the fracture toughness properties of ferritic materials such as type 403 modified stainless steel be demonstrated to the Commission on a case-by-case basis. The type 403 modified steel is used as an RCPB material in the motor tube of the control rod drive mechanism. This section demonstrates that, for this application, the material has adequate fracture toughness for protection against non-ductile failure.

The nominal wall thickness of the motor tube section of interest is more than 1/2 inch and less than 5/8 inch. In the early editions of ASME Section III up to the Winter 1971 Addenda to the 1971 Edition, materials with a nominal section thickness of 1/2 inch or less did not require impact testing. Starting with the Summer 1972 Addenda, the nominal section thickness increased to 5/8 inch or less. Thus, in the early editions of ASME Section III, the Type 403 modified steel required impact testing, but in the new

editions it does not. However, since this material was selected for use, B&W has ordered it to meet the impact toughness requirements for ASME Section III, Summer 1972 and later Addenda, the imposed acceptance standard for nominal wall thicknesses from 5/8 to 3/4 inch, inclusive is presented in paragraph NB-2332. The material has also been specified to meet the requirements of SA 182 grade F6 (forgings) or ASTM A276 (bars) as modified by ASME Code Case 1337.

When ordered according to the early revisions of Code Case 1337 (including Revision 6) and to the early editions of ASME Section III, the type 403 modified forgings or bars were required to be impact-tested at 20°F. The minimum average energy of a set of three Charpy V-notch specimens was 35 ft-lb, with one specimen allowed to be less than 35 but not less than 30 ft-lb. For both forgings and bars, the Charpy specimens were oriented in the axial (longitudinal) direction.

In the Summer 1972 Addenda to the 1971 Edition of ASME Section III, the fracture toughness requirements of all pressure boundary ferritic materials changed; however, no acceptance criterion was given for the martensitic high-alloy chromium steels, such as type 403 modified steel. A year later, the Summer 1973 Addenda re-established the acceptance criteria for the type 4XX steels. Beginning with this addenda, the fracture toughness requirements and acceptance criteria for the type 4XX steels are described in paragraph NB-2332 of ASME Section III. This paragraph requires that three Charpy V-notch specimens be tested at temperatures lower than or equal to the lowest service temperature. The lateral expansion of each specimen must be equal to or greater than 20 mils. The test temperature has been specified as equal to or less than 40°F. The orientations of the specimens are transverse (normal to principal working direction) for the forgings and axial for the steel bars.

The fracture toughness requirements of Code Case 1337, starting with Revision 7, are the same as those of ASME Section III, Summer 1973 Addenda to the 1971 Edition.

It is considered that the fracture toughness requirements of the new edition. of ASME Section III provide adequate protection against nonductile failure. The proof of adequate toughness is based on demonstrating that the type 403 modified steels used in the construction of components designed to an edition or addenda of ASME Section III prior to the Summer 1973 Addenda meet or exceed the toughness requirements of that addenda.

Based on actual test data, the lowest service temperature of the control rod drive mechanism can be as low as 40°F; however, for additional protection against non-ductile failure, B&W has defined the component's lowest service temperature at 100°F. This specified lowest service temperature is 60°F above the temperature at which the fracture toughness requirements are specified and met. The additional 60°F provides margins of safety beyond that required by the ASME code and by Appendix G to 10 CFR 50.

### 5.2.3.4 Steam Generators

Research and Development

In August of 1964, B&W began design and construction of facilities to test full scale sections of the Once Through Steam Generator. Since that time, three different test models of the Once Through Steam Generator have been tested. The design criteria for the test steam generators were as follows:

1. To provide a test steam generator for investigation of the operational characteristics of the steam generator such as heat transfer, pressure drop, control characteristics (including measurements necessary for control), and stability.
2. To provide a test steam generator for investigation of manufacturing procedures, fouling characteristics, and cleaning procedures.

3. To provide a test steam generator which could be non-destructively examined and analyzed with respect to vibration, corrosion, and unit integrity.

The design bases for the test steam generators were:

1. To duplicate tube length, tube thickness, and tube diameters of the full size steam generator.
2. To duplicate important dynamic characteristics such as secondary flow area per tube, downcomer annulus area, and feedwater spray velocity.
3. To operate the test units under temperature, pressure, and control conditions of the full size units.

The general objectives of the model tests include:

1. Heat transfer tests
2. Pressure drop tests
3. Stability tests
4. Fouling and cleaning tests
5. Mechanical design tests including vibration, and structural tests.

In April, 1971, B&W submitted a topical report, BAW-10027, Reference 8 on page 5-41 General results and evaluation of the model tests including the following were reported in BAW-10027:

1. The steady state and transient operation tests have confirmed the analytically predicted performance characteristics of the steam generator, and have provided the data for the control system.
2. Feedwater spray nozzle tests have demonstrated that the design will satisfactorily heat the feedwater.
3. Tube leak simulation tests have demonstrated that a leak in one tube will not propagate by causing a failure in adjacent tubes.
4. Mechanical tests have demonstrated that the tubes can withstand, without failure, the mechanical loads they may experience either during normal operation or accident conditions.
5. Vibration testing demonstrated that the unit contained no undesirable resonance characteristics.
6. Tests to simulate a steam line failure or reactor coolant system failure have demonstrated the integrity of the steam generator under conditions of rapid depressurization and large temperature differentials between the tubes and the shell of the unit.
7. Secondary side fouling tests demonstrated that fouling will be detected by increased pressure drop in the downcomer. Feedwater nozzle flooding causes the downcomer water temperature to fall below saturation temperature. Feedwater nozzle flooding is prevented in high downcomer level limits which restrict and/or limit feedwater flow. Cleaning of the secondary side of the steam generator is required when the high downcomer level limit is activated at full power. If the operator chooses, cleaning may be postponed indefinitely by reducing the power level to the point at which the high downcomer level limit is not actuated.
8. Additional information concerning steam generator research and development, design programs, and evaluations are contained in BAW-10027 as follows:
   a. Objectives and evaluations of all model steam generator tests.
   b. Extrapolation of model tests to full size performance.
   c. Verification test program to be conducted at Oconee 1.
   d. Cleaning processes to be used.

e. Computer programs used in the design of the steam generator and transient analysis.

Because the steam generator is of a, straight tube-straight shell design and because of a minor difference in the coefficient of thermal expansion between Inconel and carbon steel, there exists structural limitation on the mean temperature difference between the tubes and the shell. During normal operation of the steam generator, the tube mean temperature should not be more than 32°F higher than the shell mean temperature. The maximum calculated mean tube to shell $\Delta T$ at normal operating conditions poses no problems to the structural integrity of the reactor coolant boundary. The effect of loss of reactor coolant would impose tensile stresses on the tubes and cause slight yielding across the tubes. Such a condition would introduce a small permanent deformation in the tubes but would in no way violate the boundary integrity. The rupture of a secondary pipe would cause the tubes to become warmer than the shell and may cause tube deformation. Blowdown tests simulating secondary side blowdown on a 37-tube model boiler, show that although a slight buckling in the tubes occurred, there was no loss of reactor coolant.

Calculations confirm that the steam generator tube sheet will withstand the loading resulting from a loss-of-coolant accident. The basis for this analysis is a hypothetical rupture of a reactor coolant pipe resulting in a maximum design pressure differential from the secondary side of 1050 psi. Under these conditions there is no rupture of the primary to secondary boundary (tubes and tube sheet).

The maximum primary membrane plus primary bending stress in the tube sheet under these conditions is 15,900 psi across the center ligaments which is well below the ASME Section III allowable limit of 40,000 psi at 650°F. Under the condition postulated, the stresses in the primary head show only the effect of its role as a structural restraint on the tube sheet. The stress intensity at the juncture of the spherical head with the tube sheet is 14,970 psi which is well below the allowable stress limit. It can therefore be concluded that no damage will occur to the tube sheet or the primary head as a result of this postulated accident.

In regard to tube integrity under loss of reactor coolant, actual pressure tests of 5/8 in. o.d./0.034 inch wall Inconel Tubing show collapse under an external pressure of 4,950 psig. This is a factor of safety of 4.7 against collapse under the 1,050 psig accidental application of external pressure to the tubes.

The rupture of a secondary pipe has been assumed to impose a maximum design pressure differential of 2,500 psi across the tubes and tube sheet from the primary side. The criterion for this accident permits no violation of the reactor coolant boundary (primary head, tube sheet, and tubes).

To meet this criterion, the stress limits delineated in the ASME Pressure Vessel Code, Section III, Paragraph N-714.2 for hydrotest limitations are applicable for the aforementioned abnormal operating circumstance. The referenced section states that the primary membrane stresses in the tube sheet ligaments, averaged across the ligament and through the tube sheet thickness, do not exceed 90 percent of the material yield stress at the operating temperature; in addition, the primary membrane plus primary bending stress in the tube sheet ligaments, averaged across the ligament width at the tube sheet surface location giving a maximum stress, does not exceed 135 percent of the material yield stress at the operating temperature.

An examination of stresses under these conditions show that for the case of a 2,500 psi design pressure differential, the stresses are within acceptable limits. These stresses together with the corresponding stress limits are given in Table 5-8.

The basic design criterion for the tubes assumes a pressure differential of 2,500 psi in accordance with Section III. Therefore, the secondary pressure loss accident condition imposes no extraordinary stress on the tubes beyond that normally expected and considered in Section III requirements.

The superimposed effect of secondary side pressure loss and maximum hypothetical earthquake has been considered. For this condition, the criterion is that there be no violation of the primary to secondary boundary (tube and tube sheet). For the case of the tube sheet, the maximum hypothetical earthquake loading will contribute an equivalent static pressure loading over the tube sheet of less than 5 psi (for vertical shock).

The effect of fluid dynamic forces on the steam generator internals under secondary steam break accident conditions has been simulated in a 37-tube laboratory boiler. Results of the tests show that reactor coolant boundary integrity is maintained under the most severe mode of secondary blowdown.

The ratio of allowable stresses (based on an allowable membrane stress of 0.9 of the nominal yield stress of the material) to the computed stresses for a design pressure differential of 2,500 psi are summarized in Table 5-9.

Additional information discussed in BAW-10027 includes:

1. Discussion of thermal fatigue due to fluctuation and shifting of the liquid-vapor interface on the tubes,
2. Stress distributions and effective elastic constants obtained under thermal inplane and transverse loadings, and analysis of tube to tube sheet complex,
3. Vibration Analysis.

Electroslag welding is utilized on longitudinal seams of the 7-inch shell courses of the steam generator as shown in Figure 5-11. The techniques used in the electroslag welding for the Oconee steam generators are identical to those used in the electroslag welding program reported as Appendix F of Dockets No. 50-237 and 50-249 (Dresden Units 2 and 3). The procedures used were appropriately modified to reflect the difference in materials of the components being welded.

Each weld is subjected to radiographic inspection, ultrasonic inspection, and the finished surfaces of the weld are magnafluxed. In addition, each plate is ordered with excess width so that test specimens may be removed after heat treatment. Physical property test specimens including tensile and impact specimens of the base material heat affected zone and weld metal is obtained from this excess material in accordance with Section III of the ASME Code. Radiographic, ultrasonic, and magnetic particle inspection is performed in accordance with Section III of the ASME Code and as required by Code Case 1355 which permits such welds for Class A vessels.

Physical tests are performed per Section N-511 of Section III of the ASME Code. For example:

1. All weld metal tensile specimens from each heat of weld wire, batch of flux, and for each combination of heat of wire and batch of flux used is obtained and tested after heat treatment.
2. Charpy impact test specimens representing weld metal and heat affected base material for every heat of wire, batch of flux, and combination of heat of wire and batch of flux used is tested.
3. Charpy V-notch impact specimens and tensile specimens are tested for 15 percent of all production welds. Included in this 15 percent are the tests required by 1 and 2 above.

All electroslag welds are made in the vertical position. Two men, one on the inside and one on the outside of the vessel, are used to check the progress of the weld, and to insure that the prescribed welding procedure is being followed. The weld is started in a U-shaped starting fixture about six inches deep attached to the bottom of the joint. The weld stabilizes in this starting tab which is later cut off and discarded. The weld once started is not stopped until the total seam is completed.

The weld receives a heat treatment which consists of a water quench from 1625°F, and a temper of 1150°F, followed by an air cool. This post-weld heat treatment refines the grain of the weld and the base material heat affected zone such that it is virtually indistinguishable from the unaffected base material. The microstructure is the same through the weld.

In 1972, an audit revealed that documentation was incomplete for certain lots of weld filler metal used in the fabrication of the Oconee steam generators. Consequently, an investigation was conducted, the results of which are documented in B&W Topical Report, BAW-1402, Reference 9 on page 5-41, which showed that all weld filler metal having incomplete documentation is satisfactory and acceptable.

### 5.2.3.5 Reliance on Interconnected Systems

The principal heat removal system interconnected with the Reactor Coolant System is the Steam and Power Conversion System. This system provides capability to remove reactor decay heat for the hypothetical case where all station power is lost. Under these conditions decay heat removal from the reactor core is provided by the natural circulation characteristics of the Reactor Coolant System. The turbine driven emergency feedwater pump supplies feedwater to the steam generators. Cooling water flow to the condenser is provided by the emergency discharge line which discharges to the tailrace of the Keowee Dam. The analysis for this unlikely condition of total loss of station electric power is presented
3 in Section 8.3.2.2.4, "Station Blackout Analysis" on page 8-25. Should the condenser not be available to receive the steam generated by decay heat, which is unlikely in view of emergency discharge line flow, the water stored in the feedwater system can be pumped to the steam generators and the resultant steam vented to atmosphere to provide required cooling.

### 5.2.3.6 System Integrity

The Reactor Protective System (Chapter 7, "Instrumentation and Control" on page 7-1) monitors parameters relate to safe operation and trips the reactor to protect against Reactor Coolant System damage caused by high system pressure. The pressurizer code safety valves prevent Reactor Coolant System overpressure after a reactor trip as a result of reactor decay heat and/or any power mismatch between the Reactor Coolant System and the Secondary System.

As a pump-motor shaft is designed to have a natural frequency at least 20 percent above the critical speed, the shaft is too stiff to respond to any of the lower seismic frequencies. The pump and motor bearings are designed to be capable of meeting the seismic design criteria.

The design specification for the control rod drives requires that the drives be capable of withstanding the seismic loadings within the stress limits for Class I equipment.

The purchase specifications for the Emergency Core Cooling System (ECCS) pumps and valves require that the units be capable of operating under the seismic loads predicted to exist at the building elevations where the units will be located. The equipment supplier has certified that the units, based on tests which exceeded the specification requirements on similar units, do adequately meet the purchase specification requirements for operation under seismic loads. The instrumentation transmitters are tested to demonstrate their suitability for the specified seismic conditions.

The center of gravity for this type of equipment is low and both the pump and the driver are rigidly connected to a structural baseplate which in turn is bolted to the building. This type of equipment is structurally quite rigid and in most instances will accommodate very high "g" loadings.

### 5.2.3.7 Overpressure Protection

The Reactor Coolant System is protected against overpressure by the pressurizer code safety valves mounted on top of the pressurizer. The capacity of these valves is determined from considerations of: (1) the Reactor Protective System; (2) pressure drop (static and dynamic) between the points of highest pressure in the Reactor Coolant System and the pressurizer; and (3) accident or transient overpressure conditions.

The combined capacity of the pressurizer code safety valves is based on the hypothetical case of withdrawal of a regulating control rod assembly bank from a relatively low initial power. The accident is terminated by high pressure reactor trip with resulting turbine trip. This accident condition produces a power mismatch between the Reactor Coolant System and Secondary System larger than that caused by a turbine trip without immediate reactor trip, or by a partial load rejection from full load.

The Low Temperature Overpressure Protection (LTOP) System protects the reactor vessel from excessive pressures at low temperature conditions. As a result of Generic Letter 88-11 and a review of operating practices at Oconee, the supporting analyses for the LTOP System have been revised.

The following low temperature overpressure events have been evaluated:

1. Erroneous actuation of the High Pressure Injection System.
2. Erroneous opening of the core flood tank discharge valve.
3. Erroneous addition of nitrogen to the pressurizer.
4. Makeup control valve (makeup to the RCS) fails full open.
5. All pressurizer heaters erroneously energized.
6. Temporary loss of the Decay Heat Removal System's capability to remove decay heat from the RCS.
7. Thermal expansion of the RCS after starting a reactor coolant pump, as a result of the stored energy in the steam generators.

The reactor vessel is protected from damage during these events by the LTOP System. The LTOP System consists of two diverse trains. One train consists of the pressurizer power operated relief valve (PORV) with a lift setpoint based on the low temperature pressure limits. The pressure limits for low temperature operation are the Technical Specification inservice leak and hydrostatic test curves for heatup and cooldown. The second train consists of operator action, assisted by administrative controls, alarms, and an operating philosophy that maintains a steam or gas bubble in the pressurizer during all modes of operation (except for inservice hydrostatic testing).

The pressurizer PORV has a dual setpoint. During normal operation, the lift setpoint is 2450 psig. A lower PORV lift setpoint is used during startup and shutdown conditions. The lower setpoint is enabled by actuation of a switch in the control room whenever the RCS temperature is below 325°F. In order to
2 prevent the LTOP pressure limits from being exceeded, a low pressure setpoint is specified within
2 Technical Specifications.

The second LTOP train relies on operator action to mitigate a low temperature overpressure event. In order to assure that adequate time is available for operator action, administrative controls exist for:

1. RCS pressure;
2. Pressurizer level;
2 3. Nitrogen addition system;

4. Number of operating reactor coolant pumps;
5. Deactivation of the A and B injection trains of the HPI System;
6. Deactivation of both core flood tanks.
7. A dedicated operator provided with approved procedures monitors RCS pressure and pressurizer level during operations at RCS temperatures below 325°F. The sole duty of the operator is to detect and mitigate LTOP transients before the RCS pressure exceeds the low temperature pressure limits.
8. In addition, alarms are provided to alert the operator that an overpressure event is occurring. For the loss of Decay Heat Removal System scenario, various LPI pressure, temperature, and flow alarms are present. For the remaining events, pressurizer level alarms would alert the operator. These alarms help ensure that a time is available for the operator to mitigate an overpressure event prior to exceeding the low temperature pressure limits.

The low temperature overpressure scenarios have been analyzed using conservative assumptions (Reference 13 on page 5-41). Assuming a single failure of either of the two diverse methods of overpressure protection, the analyses demonstrate that the reactor vessel is protected from damage during events which cause increasing pressure.

### 5.2.3.8 System Incident Potential

Potential accidents and their effects and consequences as a result of component or control failures are analyzed and discussed in Chapter 15, "Accident Analyses" on page 15-1.

The pressurizer spray line contains an electric motor-operated backup valve which can be closed should the pressurizer spray valve malfunction and fail to close; this would prevent depressurization of the system to the saturation pressure of the reactor coolant. An electric motor-operated valve located between the pressurizer and the pressurizer electromatic relief valve can be closed to prevent pressurizer steam blowdown in the unlikely event the electromatic relief valve fails to reclose after being actuated. Because of the other protective features in the plant, it is unlikely that the code valves will ever lift during operation. In addition, it is extremely unlikely these valves would stick open, since there is adequate experience to indicate the reliability of code safety valves. The analysis in Chapter 15, "Accident Analyses" on page 15-1 indicate that one high pressure injection pump is sufficient to protect the core for an opening in the system considerably larger than one pressurizer code safety valve in the open position.

The consequences of crud filling one of the two instrument lines from the flow annulus to the flow transmitters has been evaluated.

No mechanism can be postulated which would completely block one of these lines. The Reactor Coolant System is a very clean system and is continuously filtered to assure that no significant particulate matter is circulated. The boric acid in the coolant is in concentrations about a factor of two below its solubility limit at 70°F and no precipitation would occur. The entire flow monitoring system is essentially stagnant because it is a pressure-sensitive device. There is no flow in the sensing lines to induce material into these lines. Any matter of sufficient size to block the instrument lines would have to penetrate the annulus which is of a smaller size than the instrument lines. Blockage of less than four entry ports to the annulus does not significantly impare the flow reading.

If the assumption is made that the line did become blocked, however, two possible situations would arise. The blockage of the high-pressure line would cause the average flow to appear high as flow decreases. Similarly, if the low pressure line is blocked, the average flow will appear higher than normal as flow is decreased. In both cases, the loss of one pump will not cause trip bases on flux-flow if the power is constant as rated power. The results of a single pump coastdown from rated power was analyzed without

trip or power runback. The minimum Departure from Nucleate Boiling Ratio (DNBR) reached when the flow has settled to the three-pump steady state values is 1.34.

If power runback from the Integrated Control System (ICS) is assumed, the reactivity added by control rod insertion is sufficient to reduce the power to 89 percent by the time the flow has reached its new value. Therefore, the hypothetical blocking of the instrument line would not cause the core thermal design limit to be exceeded as a result of the loss of one pump from rated power.

### 5.2.3.9 Redundancy

Each heat transport loop of the Reactor Coolant System contains one steam generator and two reactor coolant pumps. Operation at reduced reactor power is possible with one or more pumps out of service. For added reliability, power to each pump is normally supplied by one of two electrically separated buses. Each of the two pumps per loop is fed from separate buses.

Two core flooding nozzles are located on opposite sides of the reactor vessel to ensure core reflooding water in the event of a single nozzle failure. Reflooding water is available from either the core flooding tanks or the low pressure injection pumps. The high pressure injection lines are connected to the Reactor Coolant System on each of the four reactor coolant inlet pipes.

### 5.2.3.10 Safety Limits and Conditions

#### 5.2.3.10.1 Maximum Pressure

The Reactor Coolant System serves as a barrier which prevents release of radionuclides contained in the reactor coolant to the Reactor Building atmosphere. In the event of a fuel cladding failure, the Reactor Coolant System is the primary barrier against the release of fission products to the Reactor Building. The safety limit of 2,750 psig (110 percent of design pressure) has been established. This represents the maximum transient pressure allowable in the Reactor Coolant System under the ASME Code, Section III.

#### 5.2.3.10.2 Maximum Reactor Coolant Activity

Release of activity into the reactor coolant in itself does not constitute a hazard. Activity in the reactor coolant constitutes a hazard only if the amount of activity is excessive and it is released to the environment. The plant systems are designed for operation with activity in the Reactor Coolant Systems resulting from 1 percent defective fuel. Activity would be released to the environment if the reactor coolant containing gaseous activity were to leak to the steam side of the steam generator. Gaseous activity could then be released to the environment by the steam jet air ejector on the main condenser. In 10 CFR 20, maximum permissible concentrations (MPC) for continuous exposure to gaseous activity have been established. These MPC will be used as the basis for maximum release of activity to the environment which has unrestricted access.

#### 5.2.3.10.3 Leakage

Reactor Coolant System leakage rate is determined by comparing instrument indications of reactor coolant average temperature, pressurizer water level and letdown storage tank water level over a time interval. All of these indications are recorded. The letdown storage tank capacity is 31 gallons per inch of height, and each graduation on the level recorded represents two inches of tank height.

Reactor Coolant System leak detection is also provided by monitoring the Reactor Building sump level and the letdown storage tank level. Since the pressurizer level controller maintains a constant pressurizer level, any Reactor Coolant System volume change due to a leakage would manifest itself as a Reactor

Building sump level change and/or a corresponding letdown storage tank level change. Considering the most adverse initial conditions of a low level in the Reactor Building sump and a high level in the letdown storage tank, a 1 gpm leak from the Reactor Coolant System would initiate a Reactor Building sump high level alarm indication in the control room within 5 hours and a letdown storage tank low level alarm indication in the control room within 16 hours. A three gpm leak would be detected in 1/3 the time given above for detection of a one gpm leak. Normally, with the Reactor Building sump level and the letdown storage tank level between their high alarm and the low alarm respectively, these detection times would be reduced.

If the leak allows primary coolant into the containment atmosphere, additional leak detection is provided by the Reactor Building Gaseous Monitoring System and the Reactor Building Area Monitoring System. The sensitivity and time for detection of a Reactor Coolant System leak by any of the radioactivity monitoring systems depends upon reactor coolant activity and the location of the leak. Alarm indication for each sample point in these systems is in the control room.

If the leak is in a steam generator, the leak will be detected by a decrease in the level of the letdown storage tank as described above and also by main steam line and condenser air ejector off gas radiation monitors. The sensitivity of the radiation monitors for leak detection depends upon the activity of the Reactor Coolant System.

Class I fluid systems other than the Reactor Coolant System pressure boundary will be monitored for leakage by monitoring the various storage and/or surge tanks for the applicable systems. The Radiation Monitoring System for the station will aid in leak detection of systems containing radioactive fluids. In addition to the above, routine Operator and/or Health Physics radiation surveillance will detect leakage in both radioactive and non-radioactive systems.

Natural circulation can be maintained in the Reactor Coolant System for decay heat removal following a complete loss of station power even if the system has been operating with an equipment leak. The natural circulation path will be maintained solid with water until the pressurizer has emptied, which is 6,000 gallons of coolant. A 30 gpm leakage rate in conjunction with a complete loss of station power and subsequent cooldown of the Reactor Coolant System by the Turbine Bypass System (set at 1,040 psia) and steam driven emergency feedwater pump would require a minimum of 60 minutes to empty the pressurizer from the combined effect of system leakage and contraction. Sixty minutes is ample time to restore electrical power to the plant and makeup flow to the Reactor Coolant System.

#### 5.2.3.10.4 System Minimum Operational Components

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources, i.e., pump energy, pressurizer heaters, and reactor decay heat. Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. One steam generator is required to be operable prior to criticality as the steam generator is the means for normal decay heat removal at temperatures above 250°F.

A reactor coolant pump or low pressure injection pump is required to be in operation prior to reducing boron concentration by dilution with make-up water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor.

#### 5.2.3.10.5 Leak Detection

The entire Reactor Coolant System is located within the secondary shielding and is inaccessible during reactor operation. Any leakage drains to the Reactor Building normal sump. Any coolant leakage to the

atmosphere will be in the form of fluid and vapor. The fluid will drain to the sump and the vapor will be condensed in the Reactor Building coolers and also reach the sump via a drain line from the cooler.

All power operated valves containing reactor coolant, located in the Reactor Building, have two sets of stem packing, with a leakoff connection between the packing sets. For the reactor coolant pump, any leakage past the primary mechanical seal is piped to the quench tank and any leakage past the backup seal is piped to the sump.

Locating the actual point of Reactor Coolant System leakage can most readily be accomplished when the reactor is shutdown, thereby allowing personnel access inside the secondary shielding. Location of leaks can then be accomplished by visual observation of escaping steam or water, or of the presence of boric acid crystals which would be deposited near the leak by evaporation of the leaking coolant.

Leakage of reactor coolant into the Reactor Building during reactor operation will be detected by sump/tank levels radioactivity, or both.

All leakage, both reactor coolant and cooling water is collected in the Reactor Building Sump. The sump water level is indicated and annunciated at high level in the control room. Changes in sump water level are an indication of total leakage. Pursuant to the NUREG 0737, Item II.F.1.5 safety grade redundant level transmitter to the normal and emergency containment sumps have been installed. Both sump levels are indicated and recorded in the control room. Measurement of the letdown storage tank coolant level provides a direct indication of reactor coolant leakage. Since the pressurizer level is maintained constant by the pressurizer level controller, any coolant leakage is replaced by coolant from the letdown storage tank resulting in a decrease in tank level. Both the pressurizer and letdown storage tank coolant levels are recorded in the control room. A comparison of these two recordings over a time period yields the total reactor coolant leakage rate.

Changes in the reactor coolant leakage rate in the Reactor Building may cause changes in the control room indication of the Reactor Building atmosphere particulate and gas radioactivities.

### 5.2.3.11 Quality Assurance

Assurance that the Reactor Coolant System will meet its design bases insofar as the integrity of the pressure boundary is concerned, is obtained by analysis, inspection, and testing.

#### 5.2.3.11.1 Stress Analyses

Detailed stress analyses of the individual Reactor Coolant System components including the vessel, piping, pumps, steam generators, and pressurizer have been performed for the Design Bases.

Dynamic analyses have been performed on the complete system treating each steam generator and associated coolant piping as an independent system to include the effect of the design bases earthquake or the maximum hypothetical earthquake in the piping stresses and nozzle stresses.

Independent thermal and dynamic analyses have been performed to insure that piping connecting to the Reactor Coolant System is of the proper schedule and that it does not impose forces on the nozzles greater than allowable. Small nozzles are conservatively designed and utilize ASA schedule 160. The reactor coolant pump casing has been completely analyzed including a dynamic analysis separately from the loop to insure that the stresses throughout the casing are below the allowable for all design conditions.

Stress analysis reports required by codes for the several components have been prepared by the manufacturer and reviewed for adequacy by a separate organization.

#### 5.2.3.11.2 Shop Inspection

Inspection and non-destructive testing of materials prior to and during manufacturing in accordance with applicable codes and additional requirements imposed by the manufacturer have been carried out for all of the Reactor Coolant System components and piping. The extent of these inspections and testing is listed in Table 5-10 for each of the components in the system. Shop testing culminates with a hydrostatic test of each component followed by magnetic particle inspection of the component external surface. Piping will be hydrostatically tested in the field and will undergo a final field inspection.

Preoperational mapping of the reactor vessel by ultrasonic examination was accomplished to establish acceptability of the vessel for service. To meet the requirements of IS-232 of Section XI of the ASME Code, the acceptance standards contained in N625.4 of the 1965 edition of Section III of the ASME Code with Addenda through Summer 1967 were used.

Components are cleaned, packaged to prevent contamination, and shipped over a pre-selected route to the site. For materials purchased or manufactured outside of B&W, the results of the material inspection and testing program have been observed or audited by B&W, and audited by the applicant. In addition there is an independent audit by B&W's Nuclear Power Generation Department Quality Assurance Section.

#### 5.2.3.11.3 Field Inspection

Field welding of reactor coolant piping and piping connecting to nozzles is performed using procedures which will result in weld quality equal to that obtained in shop welding. Non-destructive testing of the welds is identical to that performed on similar welds in the shop and is shown in Table 5-10. Accessible shop and field welds and weld repairs in the reactor coolant piping are inspected by magnetic particle or liquid penetrant tests following the system hydrostatic test.

#### 5.2.3.11.4 Testing

The Reactor Coolant System including the reactor coolant pump internals, reactor closure head, control rod drives, and associated piping out to the first stop valve undergoes a hydrostatic test following completion of assembly. The hydrostatic test is conducted at a temperature 60°F greater than the highest nil-ductility temperature. During the hydrostatic test, a careful examination is made of all pressure boundary surfaces including gasketed joints.

### 5.2.3.12 Tests and Inspections

This section discusses tests and inspections performed during and after the assembly of the individual components into a completed Reactor Coolant System. These tests and inspections are performed to demonstrate the functional capabilities of the components after assembly into a completed system, to inspect the quality of the system closure weldments, and to monitor system integrity during service.

#### 5.2.3.12.1 Construction Inspection

The coolant piping for each loop is shipped to the field in six subassemblies. The loops are then assembled in the field. In order to accommodate the small fabricating and field installation tolerances, a number of the subassemblies are fabricated with excess length. Thus, the final fitting of the coolant piping is accomplished in the field. The ends with excess length are field machined. All carbon steel-to-carbon steel field welds are back-clad with stainless steel following removal of the backing rings. Consumable inserts are used in stainless-to-stainless welds, such as surge line and some coolant pump welds. All welding is inspected in accordance with requirements of the applicable codes or better.

Welding of the auxiliary piping to Reactor Coolant System nozzles is done to the same standards as the main coolant piping. Consumable inserts are used in all cases.

Cleaning of reactor coolant piping and equipment is accomplished both before and after erection of various equipment. Piping and equipment nozzles will require cleaning in the area of the connecting weldments. Most of the piping and equipment are large enough for personnel entry and are cleaned by locally applying solvents and demineralized water and by wire brush to remove trapped foreign particles. Where surfaces and equipment cannot be reached by personnel entry and have been cleaned in vendor shops to the required cleanliness for operation and appropriately protected to maintain cleanliness during handling, shipping, storage, and installation, further cleaning will not be performed. Appropriate checks to verify maintenance of required cleanliness will be performed prior to operation.

#### 5.2.3.12.2 Installation Testing

The Reactor Coolant System will be hydrostatically tested in accordance with USAS B31.7, Nuclear Power Piping Code. The test pressure will affect all parts of the Reactor Coolant System up to and including means of isolation from auxiliary systems, such as valves and blank flanges. The hydrostatic test will be performed at temperature above Design Transition Temperature.

The Reactor Coolant System relief valves will be inspected and shop-tested in accordance with Section III of the ASME code for Nuclear Vessels. The relief pressure setting will be made during the shop test.

#### 5.2.3.12.3 Functional Testing

Prior to initial fuel loading, the functional capabilities of the Reactor Coolant System components will be demonstrated at operating pressures and temperatures. Measurement of pressures, flows, and temperatures will be recorded for various system conditions. Operation of reactor coolant pumps, pressurizer heaters, Pressurizer Spray System, control rod drive mechanism, and other Reactor Coolant System equipment will be demonstrated. For descriptions of the various functional tests performed, refer to Chapter 14, "Initial Tests and Operation" on page 14-1.

#### 5.2.3.12.4 Inservice Inspection

Inservice examination of ASME Code Class 1, 2 and 3 components are performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR
0 50, Section 50.55a(g)(4), to the extent practical within the limitations of design, geometry and materials of construction of the components, except where specific written relief has been granted by the Commission. Technical Specification 4.2 applies to the surveillance of the Reactor Coolant System components.

Vessels, heat exchangers, pumps, valves, and piping, are classified in accordance with 10CFR50.55a and NRC Regulatory Guide 1.26. For each ASME code class, systems have been identified which will be examined. Appropriate Duke drawings and documents provide the exact boundaries for each system to be examined.

Appropriate examination categories and methods are contained in Table IWB-2500, IWF-2500, and IWC-2500, or Subsection IWD of Section XI. Appropriate tests are contained in Sections IWP-2000 and IWV-2000.

1 In general, volumetric examination is performed by ultrasonic techniques. Main steam and feedwater,
1 however, is radiographed where possible. Surface examination is performed by dye penetrant or magnetic particle.

1 Repair procedures are prepared as necessary and Duke Power Company Nuclear Generation Department.
1 These procedures are reviewed for compliance with Section XI. Reexamination to Section XI is included in the repair process.

### 5.2.3.13 Reactor Vessel Material Surveillance Program

The original Oconee design included three reactor vessel surveillance specimen holder tubes (SSHT) located near the reactor inside vessel wall. Each of these SSHT housed two capsules containing reactor vessel surveillance specimens. When failures of the SSHT occurred at other Babcock & Wilcox (B&W) designed plants, the three Oconee units were shut down in succession, starting in March 1976 to inspect the SSHT. The inspection revealed that all of the SSHTs had suffered some damage. To prevent further damage all surveillance capsules and all parts of the SSHT that had failed or were deemed likely to fail during the remainder of that operating cycle were removed from the vessels.

Since the discovery of the damage to the SSHT, B&W has undertaken the design, manufacture and testing of an improved SSHT. SSHT of this improved design were installed in Davis-Besse 1, Crystal River 3
1 and Three Mile Island 2. (Three Mile Island 2 no longer operating but capsules were salvaged for
1 irradiations at other host plants.) All of these plants have the same basic B&W 177 fuel assembly reactor design as Oconee 1, 2, and 3. The acceptability of the redesigned SSHT has been demonstrated by a test program reviewed and approved by the NRC staff and conducted in conjunction with the hot functional test performed at Davis-Besse 1.

Installation of the redesigned SSHT in the Davis Besse 1, Crystal River 3 and Three Mile Island 2 reactor vessels did not present any unusual radiological difficulties because installation was prior to neutron
1 activation of the reactor internals. Studies of methods of installing the redesigned SSHT in the irradiated B&W reactors indicate that substantial installation difficulties will be experienced--primarily because precision machining, alignment and inspection must be performed remotely and under water. Although such problems do not in themselves justify relief from a requirement to reinstall the SSHT in Oconee 1, 2, and 3, they would be likely to cause significant radiation to personnel. Based on its experience in removing the SSHT at Three Mile Island 1 and Rancho Seco 1, B&W estimated that installing SSHT in irradiated reactors could result in personnel exposures totaling about 100 man-rem per reactor. In the interest of maintaining the radiation exposure of plant personnel as low as reasonably achievable, the licensee, in cooperation with B&W and the owners of other B&W 177 fuel assembly plants, has proposed an alternative program that does not require reinstalling the SSHT in Oconee 1, 2, and 3 and the other irradiated B&W plants.

1 The capsules removed from the Oconee vessels which had damaged SSHT were placed in a host reactor, Crystal River 3, as part of the integrated surveillance program discussed herein. These capsules contain samples of plate or forging material and heat-affected zone material from the vessel beltline as well as weld metal. The weld metal is expected to be controlling because it is more radiation sensitive. However, capsules containing other than weld metal will be irradiated also, since the purpose of the surveillance program is to obtain data on materials which would prove to be important later on.

1 This program includes provisions to provide additional information, if required under 10 CFR 50,
1 Appendix G, Paragraph V.C., in addition to the normal requirements of Appendix H.

The plan involves integrating the interrupted surveillance program at Oconee and other plants with the
1 programs for new plants in a manner generally similar to that covered in 10 CFR 50, Appendix H, Paragraph II.C.4, except that the plants are at different sites. There are three distinct features of this plan.

1. The original surveillance materials from one or more reactors that have been in service will now be irradiated in a new host reactor, that can be fitted with the newly-designed capsule holders on the thermal shield in less time and without significant radiation exposure of the workmen, and

2. There will be more weld metal specimens and some larger fracture mechanics (compact tension or CT) specimens placed in the capsules, and
3. A data-sharing feature in which all available irradiation data for the beltline welds of a given reactor some of which will come from other surveillance programs, will be considered in predicting its adjusted reference temperature and in making any fracture analyses for that reactor. Typically, several of the welds in any one vessel were made with the same weld wire and flux as those used on some other reactors. The data sharing feature is required because the welds in these reactors have high radiation sensitivity due to high copper content, large and random variation of copper from point to point in the weld, and low initial upper shelf energy.

The specific program for Oconee 1, 2, and 3 involves installing the Oconee surveillance capsules in extra locations provided in the Crystal River 3 vessel. This plan will accomplish the original purpose of obtaining information on the effect of radiation on material that is representative of the material in the Oconee reactor vessels on a schedule that provides an appropriate lead time over the vessel irradiation rate. The overall integrated program also will provide information relevant to Oconee 1, 2, and 3 from surveillance programs in Crystal River 3, and Davis Besse 1 on material considered to be essentially identical to the actual welds in the Oconee vessels. It is also important to note that still more information relevant to the Oconee vessel materials will be obtained from the NRC funded High Strength Steel Test (HSST) irradiation programs underway. Details are provided below.

#### 5.2.3.13.1 Oconee 1

1 The limiting weld materials for the Oconee 1 vessel are Procedure Qualification (P.Q.) numbers SA-1426
1 and SA-1430* . (BAW-2050, Reference 17 on page 5-41.) These are longitudinal welds in the lower shell
1 course. The end of life (EOL) fluence for these welds is estimated to be 9.08 x $10^{18}$ nvt, (E > 1 MeV) at
1 the inner surface, and both have compositions that are expected to make them relatively sensitive to
1 radiation damage.

The original surveillance material, WF-112, was made using the same heat of filler wire but a different
1 batch of flux as WF-154, one of the radiation sensitive welds in Oconee 2. Metallurgical considerations suggests that the radiation behavior is affected more by the wire than the flux, thus WF-112 is expected to respond to radiation much like WF-154. This data will be a useful part of the data base for B&W vessels.

1 Reference 14 on page 5-41 documents where samples of the pertinent weld materials will be irradiated in the proposed integrated program, what kinds of specimens will be used, and when information will be available. The irradiation schedule and withdrawal dates will be modified to optimize the information
1 obtained as indicated to be appropriate as test results are obtained and evaluated. Reference 14 on
1 page 5-41 is updated periodically to reflect the most recent capsule reports.

#### 5.2.3.13.2 Oconee 2

1 The limiting weld material for the Oconee 2 vessel is P.Q. number WF-25 which is used in the center
1 circumferential weld. (BAW-2051, Reference 18 on page 5-41.) The end of life (EOL) fluence for this
1 weld is estimated to be 9.57 x $10^{18}$ nvt (E > 1 MeV) at the inner surface, and this weld has a composition
1 that is expected to make it relatively sensitive to radiation damage.

The original surveillance material, WF-209-1, while not identical to any of the beltline welds in B&W
1 reactors, is of the same weld wire heat as WF-70 (but different flux lot) and is predicted to be radiation

* Weld materials are specifically identified by the ASME Code by the procedure Qualification Test number. A procedure qualification test is required on each combination of heat of weld wire and batch of flux.

sensitive, based on its copper and nickel contents. Data from WF-209-1 will be a useful addition to the data base for these reactors.

Reference 14 on page 5-41 documents where samples of the pertinent weld materials will be irradiated in the proposed integrated program, what kinds of specimens will be used, and when information will be available. Reference 14 on page 5-41 is updated periodically to reflect the most recent capsule reports.

#### 5.2.3.13.3 Oconee 3

The limiting weld material for the Oconee 3 vessel is WF-67, except for pressurized thermal shock (PTS) for which the limiting material is WF-200. (BAW-2128, Reference 19 on page 5-41.) WF-67 is used for the center circumferential weld (inner 75%) and WF-200 is used for the upper circumferential weld. The end of life (EOL) fluence for WF-67 is estimated to be 8.08 x $10^{18}$ nvt, (E > 1 MeV) at the inner surface and for WF-200, 6.14 x $10^{18}$ nvt (E > 1 MeV). The two weld materials have compositions that are expected to make them relatively sensitive to radiation damage.

The original surveillance material, WF 209-1, is the same as that used in Oconee 2. This discussion of WF-209-1 in 5.2.3.13.2, "Oconee 2" on page 5-36 applies here.

Reference 14 on page 5-41 documents where samples of the pertinent weld materials will be irradiated in the proposed integrated program, what kinds of specimens will be used, and when information will be available. Reference 14 on page 5-41 is updated periodically to reflect the most recent capsule reports.

#### 5.2.3.13.4 Integrated Surveillance Program

BAW-1543A, Rev. 3, Master Integrated Reactor Vessel Material Surveillance Program, September 1989, specifies the Oconee specimens capsules that are to be irradiated in Crystal River 3. These capsules include the weld material and other materials such as plate or forging material samples and weld heat affected zone material samples from the Oconee vessels.

For those welds where no surveillance specimens exist, guidance for predictions will be based on the known chemical composition of those welds. Predictions are based on statistical analysis of the family of "Linde 80" weld metals.

BAW-1543, Rev. 3, presents a "Master Integrated Reactor Vessel Surveillance Program" that provides for additional surveillance capsules which contain tension test, Charpy V-notch, and larger-sized compact fracture specimens of 12 different "Linde 80" weld metals (different wire/flux combinations). These specimens will provide direct data for those materials represented and will provide a statistical base for those other materials for which archive material was not available. In particular, for Oconee-1, SA-1426 and SA-1430 are both from the same weld wire heat for which no archive material was available. For Oconee-2, WF-25 is irradiated in 7 supplementary capsules. For Oconee-3, WF-67 is irradiated in 6 supplementary capsules, and WF-200 is from the same weld wire heat as WF-182-1. WF-182-1 is irradiated in one supplementary capsule and is the weld material in the Davis-Besse RVSP.

All Oconee RVSP capsules, except for standby capsules, have been tested, essentially completing the requirement for reactor vessel surveillance irradiations. However, the supplementary capsules will provide additional fracture toughness data.

Research programs being funded by the NRC have provided information on the effect of radiation on these specific weld materials and on several additional B&W weld materials expected to respond to radiation in a similar manner. These programs, HSST-2 and HSST-3, consist of many tension test, $C_v$ and CT specimens irradiated in a test reactor. Although information on shift in $RT_{NDT}$ will be obtained, the main emphasis of the HSST programs was to develop methods that can be used to better evaluate upper-shelf toughness as compared to using the rather small specimens used in the power reactor programs.

The information to be developed from this program that is directly and indirectly relevant to the Oconee reactor vessels will be sufficient to provide assurance of safety margins against vessel failure that comply with 10 CFR 50, Appendix G.

There are uncertainties involved in applying radiation effects information obtained in other reactors to the Oconee vessels. The major uncertainties involved include:

1. Accuracy of neutron fluence calculations,
2. Magnitude and effect of variation in neutron spectra between reactors,
3. Magnitude and effect of variations in irradiation temperature between reactors,
4. Magnitude and effect of variations in rate of irradiation on material properties.

The effects of these variables have been studied for many years and are discussed below.

1. Neutron flux calculations for the reactor vessel wall and irradiation capsule locations have been developed over many years. The dosimetry used in irradiation capsules has furnished information that was used to check out and refine the calculational methods. It is generally believed that the fast neutron flux and fluence in these locations can be calculated to an accuracy of ±20 percent, particularly if some dosimetry checks are available. Dosimeters from the original Oconee surveillance program were removed and tested for verification of vessel fluence calculations.

   It should be emphasized that the effect of neutron radiation on reactor vessel steel varies as the square root of the fluence, so uncertainties of 20 to 50 percent influence are not highly significant.

   The design of the Oconee vessels, internals and cores is almost identical to that of the other reactors that will be used to obtain radiation effects information.

   These considerations are the basis for the conclusion that uncertainties in the calculation of neutron fluence will be small, and the effect of such uncertainties on the assessment of the radiation effects on the vessel material will also be small.

2. Although differences in neutron energy spectra can cause uncertainties in the effects of radiation on material when evaluated without considering spectrum effects, only very large differences in spectra are significant. The variations from one B&W 177 fuel assembly reactor to another are relatively minor, because they have almost identical geometry.

   The possible differences in neutron spectra that could occur between the B&W power reactors to be involved in the integrated program has been considered. Such effects can be dealt with, if necessary, through the use of neutron damage functions that are being developed for that purpose. However, the worst expected differences are judged inconsequential based on present knowledge of irradiation effects.

3. The effect of the temperature of irradiation has also been the subject of considerable research. It is well known that radiation damage is less severe at 600°F than at 500°F (the temperature range of concern). The differences in effect on the steel appear to be noticeable and should be taken into

account if the irradiation temperature difference is over about 25°F. Enough information is known to permit conservative evaluations of the effect of temperature differences of at least 50°F, and probably even 100°F or more. The differences in the temperature of the surveillance capsules and vessel walls between the B&W power reactors involved in the proposed integrated program are expected to be less than 25°F, and can be conservatively evaluated.

4. The effect of irradiation has also been evaluated by research programs at NRL and other laboratories. The general consensus of experts on this subject is that there will be no major differences in material property changes by irradiation rates varying over 2 to 3 orders of magnitude. However, the differences in the rates of irradiation of specimens in the integrated program and the limiting material in the walls of the affected vessels will be less than one order of magnitude, therefore, it is concluded that there will be no significant uncertainties in this program associated with differences in rate of irradiation.

The integrated, augmented reactor vessel material irradiation program for Oconee 1, 2, and 3, as an alternative to the original program that was interrupted by failure of the associated hardware, will provide the information required to comply with 10 CFR 50, Appendix G, and that the uncertainties involved in using data obtained from surveillance specimens irradiated in various other B&W power reactors to establish Oconee 1, 2, and 3 vessel operating limitations are small and can be accounted for by imposition of appropriate margins.

Additionally, the integrated, augmented program should provide more useful information than could have been extracted from the original surveillance program. The proposed program will also give results of the kind required to meet 10 CFR 50, Appendix G, Paragraph V.C.

An extension of the exemption for Oconee Units 1, 2 and 3 from the requirements for an in-vessel material surveillance program as set forth in 10 CFR 50, Appendix H, was requested by the Duke Power Company in January 1982 (Reference 10 on page 5-41). In its submittal to the NRC, Duke Power Company stated that at present there were no plans to modify the Surveillance Specimen Holder Tubes (SSHT's) or the Core Support Assembly on any Operating B&W plant which would change the geometrical similarity of the reactors or preclude the continued irradiation of the surveillance capsules in the host plants. Thus, adequate surveillance information will continue to be obtained for the Oconee units. An evaluation of the Surveillance Capsules removed from operating B&W plants and an evaluation of the reactor vessel fluence were included in the Duke Submittal to demonstrated adequacy of the Surveillance Program. Duke Power Company submittal concluded that:

1. Based on the Surveillance capsule data obtained on all the B&W-177FA plants to date, it has been demonstrated that the prediction techniques used in establishing the vessel operation limits (i.e., Reg. Guide 1.99, Rev. 2) are conservative.

2. A high degree of accuracy has been demonstrated by B&W in estimation of the reactor vessel fluence using the power histories of the reactors and the dosimetry measurements from the host plants with SSHT's.

3. The Specimen Capsules being irradiated at Crystal River-3 have received neutron fluence greater than the fluence received by the Oconee Reactor Vessels by 7 to 10 EFPYS. The Specimen Capsules are expected to continue to lead the respective reactor vessels accumulated peak fluence for the life of the plant.

NRC granted an extension to the exemption for the Oconee Nuclear Station, Units 1, 2 and 3 from the requirement for an in-vessel Material Surveillance program as set forth in 10 CFR 50, Appendix H, for a period of five years in June 1982 (Reference 11 on page 5-41). The Commission stated in its safety evaluation that the information derived from the surveillance specimens in the host vessel, relevant to Oconee Nuclear Station Units 1, 2 and 3 reactor vessels would be sufficient to provide assurance of safety

margins and comply with 10 CFR 50, Appendix G. In addition, NRC concurred with the Duke position that the dosimetry results have shown that the fluences can be estimated from the power histories with reasonable accuracy and accepted the methodology contained in BAW 1485, June 1978. In June, 1991, the NRC accepted BAW-1543, Rev. 3, and found the program capable of monitoring the effect of neutron irradiation and the thermal environment on the fracture toughness of ferritic reactor vessel beltline materials in the plants that are participating in the material surveillance program. This includes Oconee-1, Oconee-2, and Oconee-3.

## 5.2.4 REFERENCES


1. BAW-10051, Design of Reactor Internals and Incore Instrument Nozzles for Flow Induced Vibrations.
2. BAW-10008, Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Accident.
3. BAW-10035, Fuel Assembly Stress and Deflection Due to Loss-of-Coolant Accident and Seismic Excitation (Nonproprietary version of BAW-10008, Part 2, Rev. 1).
4. BAW-10046P, Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR50, Appendix G.
5. BAW-10100A, Reactor Vessel Material Surveillance Program.
6. 1 1 BAW-1803, Rev. 1, Correlations for Predicting the Effects of Neutron Radiation on Linde 80 Submerged-Arc Welds.
7. BAW-10018, Analysis of the Structural Integrity of a Reactor Vessel Subjected to Thermal Shock.
8. BAW-10027, Once-Through Steam Generator Research and Development (Nonproprietary version of BAW-10002, and BAW-10002, Sup. 1).
9. BAW-1402, Steam Generator Weld Records
10. Letter from W. O. Parker, Jr. to H. R. Denton (NRC) dated January 14,1982 Subject: Exemption from 1OCFR 50 Appendix H requirements for 5 years.
11. Letter from D. G. Eisenhut (NRC) to W. 0. Parker, Jr. dated June 16, 1982.
12. Safety Evaluation by the Office of Nuclear Reactor Regulation Concerning NUREG 0737 Item II.K.2.13, United States Nuclear Regulatory Commission, June 6, 1984.
13. Letter from H. B. Tucker (Duke) to NRC, Oconee Nuclear Station P/T Limit and LTOP Proposed Technical Specifications, November 15, 1989.
14. 1 BAW-1543, Rev. 3, Master Integrated Reactor Vessel Surveillance Program.
15. 1 1 Letter from J. N. Hannon, Office of Nuclear Reactor Regulation, NRC, to J. H. Taylor, B&W Owners Group, June 11, 1991.
16. 2 BAW-2108, Rev. 1 Fluence Tracking System.
17. 1 1 BAW-2050, Analysis of Capsule OC1-C, Duke Power Company, Oconee Nuclear Station Unit-1, Reactor Vessel Material Surveillance Program.
18. 1 1 BAW-2051, Analysis of Capsule OCII-E, Duke Power Company, Oconee Nuclear Station Unit-2, Reactor Vessel Material Surveillance Program.
19. 1 1 BAW-2128, Rev. 1, Analysis of Capsule OCIII-D, Duke Power Company, Oconee Nuclear Station Unit-3, Reactor Vessel Material Surveillance Program.
20. 1 1 BAW-1895, Pressurized Thermal Shock Evaluations in Accordance with 10 CFR 50.61 for Babcock & Wilcox Owners Group Reactor Pressure Vessels.
21. 1 BAW-2143, Evaluation of Reactor Vessel Operating Limitations. Under Preparation.
22. 1 1 BWNS Document 18-1202139-00, "Functional Specification - Pressurizer Surge Line for B&W Lowered-Loop 177 FA Plant".
23. 2 2 BAW-2143P, Evaluation of Reactor Vessel Material Reference Temperatures and Charpy Upper-Shelf Energies.

# 5.3 REACTOR VESSEL

## 5.3.1 DESCRIPTION

The reactor vessel consists of a cylindrical shell, a spherically dished bottom head, and a ring flange to which a removable reactor closure head is bolted. The reactor vessel is supported by a cylindrical support skirt.

The reactor vessel closure head is a spherically dished head welded to a ring flange which mates with and is bolted to the vessel with large-diameter studs. All internal surfaces of the vessel and closure head are clad with stainless steel or nickel-chromium-iron (Ni-Cr-Fe) weld deposit.

The reactor vessel outlines are shown in Figure 5-14 (Oconee 1), Figure 5-15 (Oconee 2), and Figure 5-16 (Oconee 3). The general arrangement of the reactor vessel with internals is shown in Figure 4-26 and Figure 4-27. Reactor vessel design data is listed in Table 5-11.

All major reactor vessel nozzles are installed with full penetration welds. All control rod drive and incore instrument nozzles are installed with partial penetration welds. The gasket leakage tap is installed in each reactor vessel flange with a partial penetration weld. In addition, the Oconee 1 closure head contains eight nozzles initially for instrumentation but now blanked off.

The reactor closure head flange and the reactor vessel flange are joined by sixty 6-1/2 in. diameter studs. Two metallic O-rings seal the reactor vessel when the reactor closure head is bolted in place. Test taps are provided in the annulus between the two 0-rings to afford a means to leak test the vessel closure seal after refueling. To insure uniform loading of the closure seal, the studs are hydraulically tensioned.

The reactor vessels and closure heads are constructed of a combination of formed plates and forgings. The ring forgings in the reactor vessel shells, other than closure flanges, for Oconee 1, 2, and 3 are identified in Figure 5-14, Figure 5-15, and Figure 5-16.

The core support assembly is supported by a ledge on the inside of the vessel flanges, and its location is maintained on this elevation by the closure head flange. The core support assembly directs coolant flow through the reactor vessel and core, supports the core, and guides the control rods in the withdrawn position.

The coolant enters the reactor through the inlet nozzles, passes down through the annulus between the thermal shield and vessel inside wall, reverses at the bottom head, passes up through the core, turns around through the plenum assembly, and leaves the reactor vessel through the outlet nozzles.

The vessel has two outlet nozzles through which the reactor coolant is transported to the steam generators and four inlet nozzles through which reactor coolant reenters the reactor vessel. Two smaller nozzles located between the reactor coolant inlet nozzles serve as inlets for decay heat cooling and emergency cooling water injection (core flooding and low-pressure injection engineered safety features functions). The reactor coolant and the control rod drive penetrations are located above the top of the core to maintain a flooded core in the event of a rupture in a reactor coolant pipe or a control rod drive pressure housing. The reactor vessel is vented through the control rod drives.

The bottom head of the vessel is penetrated by instrumentation nozzles. The closure head is penetrated by flanged nozzles which provide for attaching the control rod drive mechanisms and for control rod extension shaft movement.

Guide lugs welded inside the reactor vessel's lower head limit a vertical drop of the reactor internals and core to 1/2 inch or less and prevent rotation about the vertical axis in the unlikely event of a major internals component failure.

The reactor vessel shell material is protected from fast neutron flux and gamma heating effects by a series of water annuli and stainless steel barriers located between the core and the vessel's wall.

## 5.3.2 VESSEL MATERIALS

### 5.3.2.1 Materials Specifications

The materials used in the reactor vessel are discussed in Section 5.2.3.2, "Material Selection" on page 5-12 and listed in Table 5-5. The original reactor vessel material properties, as used in licensing Oconee, are presented in Table 5-12 and Table 5-13. Additional material physical properties are presented in Table 5-14. These properties have been updated as new data became available as explained in Section 5.2.3.3, "Reactor Vessel" on page 5-15.

### 5.3.2.2 Special Processes for Manufacturing and Fabrication

The reactor vessel and appurtenances are constructed in accordance with the ASME Code, Section III edition and addenda listed in Table 5-4. Processes and materials, including product form used in fabrication of the reactor vessel, are discussed in Section 5.2.3, "System Design Evaluation" on page 5-12, and were selected to ensure reactor vessel integrity, and to meet regulatory requirements and recommendations. Special or unusual processes not meeting the above requirements are not used in construction of the reactor vessel.

### 5.3.2.3 Special Methods for Nondestructive Examination

The required nondestructive examinations carried out during fabrication are presented in Table 5-10. These inspections are performed in accordance with procedures meeting the requirements of the edition and addenda of the ASME Code, Section III listed in Table 5-4. Nondestructive examination techniques used are selected to provide adequate sensitivity, reliability, and reproducibility to inspect surfaces and detect internal discontinuities. Acceptance standards are in accordance with the requirements of the ASME Code, Section III for the given product and/or fabrication process.

## 5.3.3 DESIGN EVALUATION

The summary description of the reactor vessel, including major considerations in achieving reactor vessel safety and vessels contributing to the vessel's integrity, is contained in Section 5.2, "Integrity of Reactor Coolant Pressure Boundary" on page 5-9. B&W is the reactor vessel designer and fabricator.

### 5.3.3.1 Design

The ASME Code, Section III, is the Primary design criteria for the reactor vessel. Chapter 5, "Reactor Coolant System and Connected Systems" on page 5-1 describes the reactor vessel design, including construction features and arrangement drawing. Materials of construction are listed in Table 5-5. The design code is given in Table 5-4. Table 5-11 gives the design basis values used in the design.

#### 5.3.3.2 Materials of Construction

The materials of construction for the reactor vessel are listed in Table 5-5. Special requirements, reason for selection, and suitability of the materials used are included in Section 5.2.3, "System Design Evaluation" on page 5-12. The materials selected have been used extensively in nuclear vessel construction and exhibit well defined properties and serviceability.

#### 5.3.3.3 Fabrication Methods

Fabrication methods used in constructing the reactor vessel are described in Section 5.2.2, "Codes and Classifications" on page 5-11. The suitability of the fabrication methods is demonstrated by the excellent service history of vessels constructed using these methods.

#### 5.3.3.4 Inspection Requirements

Fabrication inspection requirements imposed on the reactor vessel are summarized in Section 5.2.3.11, "Quality Assurance" on page 5-32 and Table 5-10. Preservice and inservice inspection requirements are summarized in Section 5.2.3.12, "Tests and Inspections" on page 5-33.

#### 5.3.3.5 Shipment and Installation

B&W specifies cleanliness requirements during shipment of the reactor vessel to ensure its arrival at the site in satisfactory condition. B&W also provides appropriate instructions and consultation to the owner for onsite cleaning and vessel protection. Temporary protective coatings and/or covers are applied to the vessel during shipment and storage as appropriate for expected environmental conditions. Water chemistry is controlled during initial fill, testing, and operation of the vessel to prevent an environment that may be conducive to material failure.

#### 5.3.3.6 Operating Conditions

The operational limits specified to ensure reactor vessel safety are described in Section 5.2.1, "Design Conditions" on page 5-9. These are compared with normal intended and upset operating conditions in Section 5.2.1, "Design Conditions" on page 5-9. The design transients for the reactor vessel are specified in Section 5.2.1, "Design Conditions" on page 5-9.

#### 5.3.3.7 Inservice Surveillance

A discussion of the reactor vessel material surveillance program is given in the inservice inspection program is discussed in Chapter 17, "Quality Assurance" on page 17-1.

### 5.3.4 PRESSURE - TEMPERATURE LIMITS

#### 5.3.4.1 Design Bases

B&W Topical Report BAW-10046A, Reference 1 on page 5-47, provides the bases for setting operational limits on pressure and temperature. This topical report provides detailed assurance that, throughout the
1 life of the plant, operations will comply to requirements of 10 CFR 50, Appendix G. Regulatory Guide 1.99 is used to predict the effects of neutron irradiation on the beltline region materials. For assurance of
1 compliance with 10 CFR 50, Appendix H, through out the life of the plant, see Section 5.2.3.12, "Tests and Inspections" on page 5-33.

### 5.3.4.2 Limit Curves

Topical Report BAW-10046A provides the following information:

1. Procedures and criteria used
2. Safety margins
3. Bases used to determine the limits
4. Procedures that will be used to revise the limits

The limits of pressure and temperature for the following conditions are provided in Technical Specification 3.1.2.

1. Inservice leak and hydrostatic tests
2. Normal operation, including heatup and cooldown
3. Reactor core operation

These limit curves are periodically updated in accordance with surveillance requirements contained in Technical Specification 4.2.

## 5.3.5 REFERENCES

1 1. BAW-10046A, Rev. 2, Methods of Compliance with Fracture Toughness and Operational Requirements of 10CFR50, Appendix G.

# 5.4 COMPONENT AND SUBSYSTEM DESIGN

## 5.4.1 REACTOR COOLANT PUMPS

The reactor coolant pumps installed on Oconee 1 are Westinghouse Model 93A, while those installed on Oconee 2 and 3 are Bingham. The following briefly describes the significant changes for Oconee 1. Except where noted, the Oconee 1 design is the same as that of Oconee 2 and 3. The reactor coolant flow distribution with less than four pumps operating is presented in Table 5-15.

### 5.4.1.1 Reactor Coolant Pumps (Oconee 1 Only)

Each reactor coolant loop contains two vertical single stage centrifugal-type pumps which employ a controlled leakage seal assembly. A cutaway view of the pump is shown in Figure 5-17 and the principal design parameters for the pumps are listed in Table 5-16. The estimated reactor coolant pump performance characteristic is shown in Figure 5-18. Connections to the pumps are shown on Figure 5-1.

Reactor coolant is pumped by the impeller attached to the bottom of the rotor shaft. The coolant is drawn up through the bottom of the impeller, discharged through passages in the guide vanes and out through a discharge in the side of the casing. The motor-impeller can be removed from the casing for maintenance or inspection without removing the casing from the piping. All parts of the pumps in contact with the reactor coolant are constructed of austenitic stainless steel or equivalent corrosion resistant materials. A list of pressure containing materials is given in Table 5-5.

The pump employs a controlled leakage seal assembly to restrict leakage along the pump shaft, as well as a secondary seal which directs the controlled leakage out of the pump, and a vapor seal which minimizes the leakage of vapor from the pump into the containment atmosphere.

A portion of the high pressure water flow from the high pressure injection pumps is injected into the reactor coolant pump between the impeller and the controlled leakage seal. Part of the flow enters the Reactor Coolant System through a labyrinth seal in the lower pump shaft to serve as a buffer to keep reactor coolant from entering the upper portion of the pump. The remainder of the injection water flows along the drive shaft, through the controlled leakage seal, and finally out of the pump. A small amount which leaks through the secondary seal is also collected and removed from the pump.

Component cooling water is supplied to the thermal barrier cooling coil.

### 5.4.1.2 Reactor Coolant Pumps (Oconee 2 & 3)

The reactor coolant pumps are single suction, single stage, vertical, radially balanced, constant speed centrifugal pumps. This type of pump employs mechanical seals to prevent reactor coolant fluid leakage to the atmosphere. A view of the pump is shown in Figure 5-19 and the principal design parameters are listed in Table 5-17. The reactor coolant pump performance characteristics are shown in Figure 5-20. Connections to the pumps are shown on Figure 5-21 (Oconee 2) and Figure 5-22 (Oconee 3).

The pump casing design utilizes a quad-volute inner case permanently welded to a pressure containing outer case. The configuration of the pressure containing outer case is kept simple so that the casing quality will meet the required radiographic level and the stresses can be analyzed to meet the requirements of the design specification. The quad-volute inner casing consists of four volute passages spaced 90° apart which receive the discharge from the pump impeller and guide it efficiently into the outer casing where it

flows to the discharge nozzle through a passage having a constantly increasing cross-sectional area. The pump casing is welded into the piping system and the pump internals can be removed for inspection or maintenance without removing the casing from the piping.

The pump cover and stuffing box is a unit containing a thermal barrier, recirculation impellers, shaft, journal bearing, and mechanical face-type seals. The pump shaft is coupled to the motor with a spacer coupling which will permit removal and replacement of the seals without removing the motor. The pump cover has a cooling jacket to remove the heat which passes through the thermal barrier. This jacket has a capacity large enough to remove all heat which is transmitted to the cover. However, additional cooling capacity is provided, in case injection cooling water is lost. A recirculation impeller on the shaft immediately above the journal bearing circulates water in the bearing chamber to a heat exchanger and returns it to the chamber. The pump may be operated with loss of either injection water or cooling water.

The Shaft Seal System consists of face-type mechanical seals operating in tandem. Injection water, at a pressure above the pump suction pressure, is injected into the pump bearing chamber. A small portion of the injection water flows into the pump through a restriction bushing. The major portion flows through cooling slots in the o.d. of the bearing steel. The shaft seal system is made up of two mechanical seals operating in tandem, wherein about one-half of the system pressure is expanded in each seal. Each seal is capable of operation at the full system pressure. The fluid which leaks past the face-type mechanical seal passes into a seal leakage chamber and then out to the quench tank. A low pressure mechanical seal at the top of the seal leakage chamber prevents the escape of fluid to the atmosphere.

Electroslag welding is used to make the seven-inch thick circumferential butt weld which welds together the upper and lower halves of the pump casing. This weld is performed in accordance with ASME Code Case 1355-2 which permits electroslag welding of Class A pressure vessels. The casings are cast and welded by ESCO, who is the leading supplier of RCP casings for the industry.

Electroslag welding is a welding process wherein coalescence is produced by heat generated in a conductive molten slag which melts the filler metal and the surfaces of the work to be welded. The weld pool is shielded by this slag and moves along the full cross section of the joint as the welding progresses. The conductive slag is maintained molten by its resistance to the flow of electric current passing between the electrode and the work. Water cooled, non-fusing metal shoes are used to contain the molten metal on both sides of the weld. The welding is performed in a vertical position with the start and finished performed on run-out tabs affixed to the casting. These run-out tabs are later cut off and discarded. The only variables contained in the method of welding are the wide range of amperage (480-units 720H) and voltage (44-52V) needed to control the molten pool of metal.

The weld is examined 100 percent using liquid penetrant and radiographic examination methods in accordance with Section III of the ASME Code. Ultrasonic inspection is not performed because the pump casing material, austenitic stainless steel, precludes achieving meaningful inspection results.

The pump casing receives two heat treatment cycles. The first is a solution annealing treatment where the pump casing halves are furnace heated to 1900°F, held for a specified time, and water quenched. The second heat treatment is a stabilizing treatment in which the welded pump casing is heated to 725°F and air cooled.

The pump is designed such that the pressure boundary casing is similar in shape to a simple pressure vessel. Internal to the casing is a separate diffuser section which provides the correct flow passages to develop the hydraulic characteristics of the pump. This design will permit the initiation of the analysis by separating the vessel into shell elements of simple geometry (such as rings, cylinders, etc.) of which the structural behavior is known. The pressure, mechanical, and thermal loads acting on the structure are

applied to the shell elements with a system of forces required to maintain the static equilibrium of each element.

The following computer programs have been utilized by Mechanics Research institute in performing the code calculations on this casing for the pump vendor:

1. SEAL SHELL 2

   SEAL SHELL 2 is a digital computer program prepared by Westinghouse Electric Corporation for the U. S. Atomic Energy Commission. The program determines the stresses, loads, and deformations for a shell of revolution which applied axisymmetric end loads (axial, radial, and moment), pressure (internal or external), and thermal gradients.

2. MAX STRESS

   MAX STRESS is a digital computer program prepared by Mechanics Research Institute to maximize the stresses in the nozzle due to pressure, external bending, and axial loads. The first phase of the program calculates the torsional moment, bending moment, bending moment angle, axial load, normal shear load, and normal shear angle for each possible combination of loads.

   Phase 2 of the program calculates the maximum principal stress and twice the maximum shear stress at the inside fiber, midwall fiber, and outside fiber for each of the possible combination of loadings. The maximums are then printed along with the maximum axial stress, maximum hoop stress (VQ/It), and maximum midwall axial stress.

   The program also has the option of maximizing the stress at any section around the circumference of the nozzle. The stress was generally maximized within $\pm 10°$ of the section analyzed.

3. THAN

   THAN or Thermal Analyzer solves three-dimensional transient heat flow problems, producing the temperature history of a physical system of any arbitrary geometry. This is accomplished through the concept of lumped parameters, and the problem is expressed as an electrical analog of the heat transfer problem. The program utilizes finite difference techniques for problem solution. Steady-state problems are also solved by the program and depend only on the boundary conditions and lumped parameter characteristics. From these data supplied by the THAN program, temperature time histories (in the case of the bowl) are determined.

An analysis in accordance with paragraph N-415.1 of the ASME Code was performed to determine if the pump casing required a fatigue analysis for the number of design cycles specified. This analysis showed that the pump casing bowl met all the requirements of paragraph N-415.1. Thus a fatigue analysis was not required. However, a fatigue analysis was performed on the pump casing cover in which the worst possible stress combination was considered at the two most critical points in the cover. It was found from this analysis, with this very conservative approach, that the maximum cumulative usage factor is only 0.125 for the design cycles specified for this plant.

A summary of the code allowables is listed in Table 5-18 and shown pictorially on Figure 5-23 and Figure 5-24. The reinforcement area is as defined in paragraph N-454 of the ASME Code Section III. The stress analysis performed on the bowl and the attached nozzles showed that the stresses are within the allowable limits. Note that a factor of two was applied to the nozzle loading due to seismic reactions and when these were combined with the dead weight and thermal expansion reactions, the stress levels were within the realistic allowable stress intensities shown in Table 5-18. A summary of maximum calculated stresses is given in Table 5-18.

The casing cover analysis indicates that the thermal stresses and pressure stresses on the cover are within the Section III code allowables.

There are no deviations from the applicable ASME Code requirements in the design and fabrication of the pump casings other than code stamping.

## 5.4.2 STEAM GENERATOR

The steam generator general arrangement is shown in Figure 5-25 (Oconee 1 and 2) and Figure 5-26 (Oconee 3). Principal design data are tabulated in Table 5-20.

The once-through steam generator supplies superheated steam and provides a barrier to prevent fission products and activated corrosion products from entering the Steam System.

The steam generator is a vertical, straight tube, tube and shell heat exchanger which produces superheated steam at constant pressure over the power range. Reactor coolant flows downward through the tubes and transfers heat to generate steam on the shell side. The high pressure (reactor coolant pressure) parts of the unit are the hemispherical heads, the tube sheets and the tubes between the tube sheets. Tube support plates maintain the tubes in a uniform pattern along their length. The unit is supported by a skirt attached to the bottom head.

The shell, the outside of the tubes, and the tube sheets form the boundaries of the steam producing section of the vessel. Within the shell, the tube bundle is surrounded by a cylindrical baffle. There are openings in the baffle at the feedwater inlet nozzle elevation to provide a path for steam to afford contact feedwater heating. The upper part of the annulus formed by the baffle plate and the shell is the superheat steam outlet, while the lower part is the feedwater inlet heating zone.

Vent, drain, and instrumentation nozzles, and inspection handholes are provided on the shell side of the unit. The reactor coolant side has manway openings in both the top and bottom heads, and a drain nozzle on the bottom head. Venting of the reactor coolant side of the unit is accomplished by a vent connection on the reactor coolant inlet pipe to each unit.

Feedwater is supplied to the steam generator through an auxiliary feedwater ring located at the top of the steam generator to assure natural circulation of the reactor coolant following the unlikely event of the loss of all reactor coolant pumps.

Four heat transfer regions exist in the steam generator as feedwater is converted to superheated steam. Starting with the feedwater inlet these are:

### 5.4.2.1 Feedwater Heating Region

Feedwater is heated to saturation temperature by direct contact heat exchange. The feedwater entering the unit is sprayed into the downcomer annulus formed by the shell and the cylindrical baffle around the tube bundle. Steam is drawn by aspiration into the downcomer and heats the feedwater to saturation temperature.

The saturated water level in the downcomer provides a static head to balance the static head in the nucleate boiling section, and the required head to overcome pressure drop in the circuit formed by the downcomer, the boiling sections, and the bypass steam flow to the feedwater heating region. The downcomer water level varies with steam flow from 15 - 100 percent load. A constant minimum level is held below 15 percent load.

#### 5.4.2.2 Nucleate Boiling Region

The saturated water enters the tube bundle just above the lower tube sheet and the steam-water mixture flows upward on the outside of the tubes counter current to the reactor coolant flow. The vapor content of the mixture increases almost uniformly until DNB is reached, and then film boiling and super heating occurs.

#### 5.4.2.3 Film Boiling Region

Dry saturated steam is produced in the film boiling region of the tube bundle.

#### 5.4.2.4 Superheated Steam Region

Saturated steam is raised to final temperature in the superheater region. The amount of surface available for superheat varies inversely with load. As load decreases the superheat section gains surface from the nucleate and film boiling regions. Mass inventory in the steam generator increases with load as the length of the heat transfer regions vary. Changes in temperature, pressure, and load conditions cause an adjustment in the length of the individual heat transfer regions and result in a change in the inventory requirements. If the inventory is greater than that required, the pressure increases. Inventory is controlled automatically as a function of load by the feedwater controls in the Integrated Control System.

Typical steam generator feedwater quality values are shown in Table 5-6.

### 5.4.3 REACTOR COOLANT PIPING

The general arrangement of the reactor coolant piping is shown in Figure 5-3, Figure 5-4, Figure 5-5, Figure 5-6, Figure 5-7, and Figure 5-8. Principal design data are tabulated in Table 5-21.

The major piping components in this system are the 28-inch i.d. cold leg piping from the steam generator to the reactor vessel and the 36-inch i.d. hot leg piping from the reactor vessel to the steam generator. Also included in this system are the 10-inch surge line and the 2-1/2-inch spray line to the pressurizer. The system piping also incorporates the auxiliary system connections necessary for operation. In addition to drains, vents, pressure taps, injection, and temperature element connections, there is a flow meter section in each 36-inch line to the steam generators to provide a means of determining the flow in each loop.

The 28-inch and 36-inch piping is carbon steel clad with austenitic stainless steel. Short sections of 28-inch stainless steel transition piping are provided between the pump casing and the 28-inch carbon steel lines.

For Oconee 1 only a 28 in. i.d. x 31 in. i.d. stainless steel transition section is installed between the existing 28 in. i.d. coolant piping and the 31 in. i.d. pump suction.

Also a 28 in. i.d. small angle elbow section between the pump discharge nozzle and the reactor inlet pipe is installed to account for the radial discharge of the replacement pump. The original pump had a tangential discharge nozzle. The elbow section is carbon steel with a section of stainless for welding to the pump casing nozzle.

Stainless steel or Inconel safe-ends are provided for field welding the nozzle connections to smaller piping. The piping safe-ends are designed so that there will not be any furnace sensitized stainless steel in the pressure boundary material. This is accomplished either by installing stainless steel safe-ends after stress relief or using Inconel. Smaller piping, including the pressurizer surge and spray lines, is austenitic

stainless steel. All piping connections in the Reactor Coolant System are butt-welded except for the flanged connections on the pressurizer for the relief valves.

Thermal sleeves are installed where required to limit the thermal stresses developed because of rapid changes in fluid temperatures. They are provided in the following nozzles: the four high pressure injection nozzles on the reactor inlet pipes; the two core flooding low pressure injection nozzles on the reactor vessel; and the surge line nozzle and spray line nozzle on the pressurizer.

## 5.4.4 REACTOR COOLANT PUMP MOTORS

The reactor coolant pump motors are large, vertical, squirrel cage, induction machines. The motors have flywheels to increase the rotational-inertia, thus prolonging pump coastdown and assuring a more gradual loss of main coolant flow to the core in the event pump power is lost. The flywheel is mounted on the upper end of the rotor, below the upper radial bearing and inside the motor frame. An anti-reverse device is included in the flywheel assembly to eliminate reverse rotation when there is back flow. Prevention of back rotation also reduces motor starting time.

The motors are enclosed with water-to-air heat exchangers so as to provide a closed circuit air flow through the motor. Radial bearings are floating pad type, and the thrust bearing is a double-acting Kingsbury type designed to carry the full thrust of the pump. A High Pressure Oil System with separate pumps is provided with each motor to jack and float the rotating assembly before starting. Once started, the motor provides its own oil circulation.

Instrumentation is provided to monitor motor cooling, bearing temperature, winding temperature, winding differential current, and speed.

In evaluating the design of the reactor coolant pump motor as it relates to the safety of the Reactor Coolant System, many items have been considered, namely: the overspeed of the motor; flywheel and shaft integrity; bearing design and system monitoring; seismic effects; and quality control and documentation.

An analysis of these considerations are given as follows as an indication of the safety and reliability that is integral with the motors:

### 5.4.4.1 Overspeed Considerations

The reactor coolant pump motors normally receive their electrical power from the nuclear generating unit through the unit's Auxiliary Electric System. On load rejection, the generating unit is designed to separate from the transmission network and remain in a standby operating condition carrying its own auxiliaries.

Figure 5-27 shows the turbine speed response following load rejection with the steam control valves wide open (VWO). On load rejection with VWO, the speed of the turbine-generator will increase under the control of the Normal Speed Governing Control System. The maximum speed attainable under the Normal Speed Governing Control System is less than 106 percent with the unit auxiliaries connected. This governing system is comprised of three independent control activities, namely: the speed control unit, power unbalance relay and the fast acting intercept valves all of which function to limit overspeed to below 106 percent.

As indicated in Figure 5-27 there are additional safety devices backing up the speed governing system, namely:

1. Mechanical overspeed trip which operates at 110 percent turbine-generator speed.

2. Generator overfrequency relay trip which is an electrical trip that operates at 111 percent turbine-generator speed.
3. Electrical back-up overspeed relay trip which operates at 112 percent turbine-generator speed.

In addition, each individual reactor coolant pump motor control circuit includes an overfrequency relay which trips the motor at 115 percent motor (or turbine-generator) speed. Therefore, it is evident that the reactor coolant pump motors speed will be limited to less than 115 percent.

### 5.4.4.2 Flywheel Design Consideration

For conservatism, the design of the flywheel on the reactor coolant pump motor is based on a design speed of 125 percent. The primary stress at the flywheel bore radius, with a speed of 125 percent, is 20,000 psi which is less than 50 percent of the 50,000 psi minimum yield strength of the flywheel material. This, therefore, yields a centrifugal stress design safety margin of 250 percent at 125 percent speed.

The Duke Power Company specification on the motor calls for 500 motor starts in forty years; the flywheels have been designed for 10,000 starts yielding a safety factor of 20. However, calculation based on the material used in the flywheel results in 400,000 cycles required for crack initiation which results in a flywheel fatigue design safety factor of 800.

### 5.4.4.3 Flywheel Material, Fabrication, Test and Inspection

#### 5.4.4.3.1 Material

The flywheel is manufactured from vacuum degassed ASTM 533 steel.

#### 5.4.4.3.2 Fabrication and Test

1. Flywheel blanks are flame cut from a plate with enough surplus material to allow for the removal of the flame affected metal.
2. At least three charpy tests are made on each plate parallel and normal to the rolling direction to determine that the blank meets specifications.
3. A complete 100 percent volumetric ultrasonic test is made on the blank and tension and bend tests are also made prior to shipment of a blank to Westinghouse Electric Company.
4. Following the machining of the flywheel at the Westinghouse plant, a complete 100 percent volumetric ultrasonic test is conducted on the fly wheel and a liquid penetrant test is conducted on the bore.
5. After the flywheel is installed and the motor is completely assembled, a 125 percent overspeed test for one minute is conducted on the assembled unit.
6. Following the overspeed test, a periphery sonic test is conducted on the flywheel through access holes in the motor frame.
7. To assure the original integrity of each flywheel during operation, the following inservice inspections will be performed.

   At approximately three-year intervals, the bore and keyway of each reactor coolant pump flywheel shall be subject to an in-place, volumetric examination. Whenever maintenance or repair activities necessitate flywheel removal, a surface examination of exposed surfaces and a complete volumetric examination shall be performed, if the interval measured from the previous such inspections is greater than 6 2/3 years. Results of the examination will be evaluated by the original acceptance criteria and compared with the original examination data to assure the absence of unacceptable defects.

### 5.4.4.4 Shaft Design and Integrity

The shear stress on the shaft in the vicinity of the flywheel is 5520 psi with short circuit torque on the motor. The minimum strength of the shaft material is 23,000 psi which results in a safety factor of four under the maximum torque condition. Because of the conservatism used in the design of the shaft, it is concluded that shaft failure is not credible.

### 5.4.4.5 Bearing Design and Failure Analysis

The motor pump assembly is supported by a Kingsbury type thrust bearing which consists of a runner and upper and lower thrust plates. The history of the Kingsbury type bearing design indicates that the device is highly reliable and has a non-locking failure mode.

Provided on the motor are a number of devices to warn the operator of bearing trouble and these devices are each independent in their operation. The thrust bearing monitoring devices are as follows:

1. Two thermocouples located diametrically opposite to each other in the upper thrust plates.
2. Two thermocouples located diametrically opposite to each other in the lower thrust plates.
3. One thermocouple in the upper oil pot.
4. Oil pot level alarm device.
5. Vibration device.

These devices are arranged to provide alarm indications to the control room operator. If a thrust bearing fails and the motor is shut down, the result would be melting of the bearing babbitt and, finally automatic tripout of the motor on overload. Since seizure of the bearing will not result from a bearing failure, it is concluded that missiles will not be produced.

### 5.4.4.6 Seismic Effects

The pump motor units have been analyzed against the combination effects of mechanical and seismic loads including the gyroscopic effects of the flywheel to verify that the stress limits will not be exceeded and the pump motor unit will operate through the maximum hypothetical earthquake.

### 5.4.4.7 Documentation and Quality Assurance

The Duke Power Company and the motor supplier, Westinghouse Electric Corp., have a rigid quality assurance program directed at assuring the integrity of the reactor coolant pump motors.

A quality assurance folder is developed by Duke Power Company on each motor and the folder includes the following:

1. Specifications and addendum
2. Description of the manufacturer's quality control organization and engineering order handling.
3. Copies of all inspection reports relating to the appropriate motor.
4. Samples of quality control drawings.
5. Copies of all test reports including flywheel material vendor test reports; Westinghouse motor test reports; bearing assembly reports; shaft tests; sonic test reports on the machined flywheel prior to assembly on the motor and following the 125 percent speed test; and certification on the motor test report that the overspeed test was conducted on the assembled motor.

6. Copies of the Duke Form QA-2 which is the manufacturer's certification to Duke Power Company Design Engineering that the motors were manufactured per specification and the Duke Power quality assurance program.
7. Copies of Duke Form QA-1 which the indication to the field quality control engineer that the motor described thereon was manufactured to the specification and the Duke quality assurance program.
8. Copies of Duke Form QC-31 which is the field receiving report on the motor.

   A copy of each quality assurance folder is sent to the field quality control engineer and a copy is kept in the Design Engineering Department file.

5 Babcock & Wilcox has analyzed the reactor coolant pump assembly action resulting from postulated Reactor Coolant System breaks. B&W Topical Report, BAW-10040, Reference 1 on page 5-66, describes the homologous pump model used for the speed calculations and presents results for the spectrum of breaks analyzed.

A discussion of the linear elastic fracture analysis to determine the structural failure speed of the reactor coolant pump motor flywheel assembly is also included.

## 5.4.5 REACTOR COOLANT EQUIPMENT INSULATION

The Reactor Coolant System components are insulated with metal reflective type insulation. The insulation is supported by rings welded to weld pads on the components during field installation of the insulation. The weld pads to which the holding rings are attached are added to the components prior to final stress relief of the component.

The insulation units are removable and are designed for ease of removal and installation in such areas as field welds, nozzles, and bolted closures. The insulation units permit free drainage of any condensate or moisture from within the insulation unit.

## 5.4.6 PRESSURIZER

The pressurizer general arrangement is shown in Figure 5-28 and principal design data are tabulated in Table 5-22.

The electrically heated pressurizer establishes and maintains the Reactor Coolant System pressure within prescribed limits, and provides a steam surge chamber and a water reserve to accommodate reactor coolant density changes during operation.

The pressurizer is a vertical cylindrical vessel with a bottom surge line penetration connected to the reactor coolant piping at the reactor outlet. The pressurizer contains removable electric heaters in its lower section and a water spray nozzle in its upper section. Heat is removed or added to maintain Reactor Coolant System pressure within desired limits. The pressurizer vessel is protected from thermal effects by a thermal sleeve in the surge line and by an internal diffuser located above the surge pipe entrance to the vessel.

During outsurges, as Reactor Coolant System pressure decreases, some of the pressurizer water flashes to steam, thus assisting in maintaining the existing pressure. Heaters are then actuated to restore the normal operating pressure. During insurges, as system pressure increases, water from the reactor vessel inlet piping is sprayed into the steam space to condense steam and reduce pressure. Spray flow and heaters are controlled by the pressure controller. The pressurizer water level is controlled by the level controller.

Since all sources of heat in the system, core, pressurizer heaters, and reactor coolant pumps, are interconnected by the reactor coolant piping with no intervening isolation valves, relief protection is provided on the pressurizer. Overpressure protection consists of two code safety valves and one electromatic relief valve.

To eliminate abnormal buildup or dilution of boric acid within the pressurizer, and to minimize cooldown of the coolant in the spray and surge lines, a bypass flow is provided around the pressurizer spray control valve. This continuously circulates approximately one gpm of reactor coolant from the heat transport loop. A sampling connection to the liquid volume of the pressurizer is provided for monitoring boric acid concentration. A steam space sampling line provides capability for monitoring of or venting accumulated gases.

During cooldown and after the decay heat system is placed in service, the pressurizer can be depressurized and cooled by circulating through a connection from the High Pressure Injection System to the pressurizer spray line.

Electroslag welding is utilized in the fabrication of the pressurizer, only in the longitudinal seams of the shell courses. A total of three individual electroslag welds are made in the fabrication of each pressurizer. The electroslag welding process and quality control is the same as described in Section 5.2.3.4, "Steam Generators" on page 5-23.

Normal Reactor Coolant System pressure control is by the pressurizer steam cushion in conjunction with the pressurizer spray, electromatic relief valve, and heaters. The system is protected against overpressure by Reactor Protective System circuits such as the high pressure trip and by pressurizer relief valves located on the top head of the pressurizer. The schematic arrangement of the relief valves shown in Figure 5-1 and Figure 5-2. Since all sources of heat in the system, i.e., core, reactor coolant pumps, and pressurizer heaters, are interconnected by the reactor coolant piping with no intervening isolation valves, all relief valves are located on the pressurizer. Reactor Coolant System pressure settings and relief valve capacities are listed in Table 5-1.

Reduction of pressure during Reactor Coolant System cooldown is accomplished by the pressurizer spray provided by the reactor coolant pump. Below a system temperature of approximately 250°F, the Low Pressure Injection System is used for system heat removal and the steam generators and reactor coolant pumps are removed from service. During this period, spray flow is provided by a branch line from one high pressure injection line to the pressurizer spray line for further pressure reduction or complete depressurization of the Reactor Coolant System.

### 5.4.6.1 Pressurizer Spray

The pressurizer spray line originates at the discharge of a reactor coolant pump in the same heat transport loop that contains the pressurizer. Pressurizer spray flow is controlled by a solenoid valve using an on-off control in response to the opening and closing pressure set points. An electric motor operated valve in series with the spray line is to provide for remote spray line isolation.

### 5.4.6.2 Pressurizer Heaters

The pressurizer heaters replace heat lost during normal steady state operation, raise the pressure to normal operating pressure during Reactor Coolant System heatup from the cooled down condition, and restore system pressure following transients. The heaters are grouped in four banks and are controlled by the pressure controller. The first bank utilizes proportional control and will normally operate at partial capacity to replace heat lost, thus maintaining pressure at the set point. On-off control is used for the

remaining three banks. A low level interlock prevents the heaters from being energized with the heaters uncovered.

Based on B&W calculations and startup testing experience, a conservative value for the total heat loss from the Reactor Coolant System, under normal hot standby conditions, is 107 kilowatts. On this basis, it was determined that a minimum of 126 kilowatts of pressurizer heaters, which corresponds to the smallest single bank of pressurizer heaters, should be available from an assured power source within two hours after loss of offsite power in order to establish and maintain natural circulation at hot standby conditions. This calculated heat loss is similar to heat loss estimates that have been accepted for other pressurized water reactors.

The pressurizer heaters for each unit are supplied from non-safety-related motor control centers (MCC). The MCC are in turn powered via load centers from the 4160-volt engineered safeguard buses. These buses are powered from a hydro station which is the emergency generation source (EGS) in the event of loss of offsite power. This emergency source has ample capacity to provide emergency power to all pressurizer heaters and is capable of doing so promptly following an accident. The pressurizer heaters are divided among the three 4160 volt EGS buses such that the loss of one entire 4160 volt bus will not preclude the capability to supply sufficient pressurizer heaters to maintain natural circulation under hot standby conditions.

Uncovering energized direct immersion heaters does not immediately harm the heaters. Three heaters, one for each bundle assembly, are tested in air to provide an accelerated life test as follows:

1. Tested for 100 hours at sheath temperature of 600°F to 1600°F with a watt density of 85 watt/in.$^2$.
2. Cycled 400 times with a cycle time of 15 minutes on and 15 minutes off with a watt density of 65 watt/in.$^2$.

The heaters successfully completed this test, which simulated a total of 200 hours "on" time for the heaters in an uncovered environment while in an energized condition. Moreover, the heater sheath is designed for 2500 psig and 670°F with the heater terminal also designed for these same conditions. Therefore, the heater sheath could fail and the pressurizer vessel integrity would be maintained. This conclusion has been substantiated in tests conducted by the heater vendor for a similar design.

### 5.4.6.3 Pressurizer Code Safety Valves

Two pressurizer code safety valves are mounted on individual nozzles on the top head of the pressurizer. The valves have a closed bonnet with bellows and supplementary balancing piston. The valve inlet and outlet is flanged to facilitate removal for maintenance or set point testing.

#### 5.4.6.3.1 Safety Valve Testing and Qualification

During the EPRI Safety Valve testing, it was determined that the short inlet Dresser 31739A valve successfully met all the test requirements with the "reference" ring settings. The performance of the valve was determined to be dependent on the ring settings. Duke Power Company evaluated the safety impact of the inadequate safety ring settings and determined that for the limiting RCS overpressure transients the plant safety can be maintained. In October 1982, all the Oconee Nuclear Station safety valves were adjusted to the recommended settings resulting from EPRI tests. Duke Power Company has also initiated a program to determine the optimal ring settings for the Dresser 31739A safety valves. Based on the test data, operability of the Oconee safety valves has been demonstrated for expected and operating conditions as required by NUREG-0737, Item II.D.1.A.

### 5.4.6.4 Pressurizer Electromatic Relief Valve

The pressurizer electromatic relief valve, also called power operated relief valve (PORV), is mounted on a separate nozzle on the top head of the pressurizer. The main valve operation is controlled by the opening or closing of a pilot valve which causes unbalanced forces to exist on the main valve disc. The pilot valve is opened or closed by a solenoid in response to the pressure set points. Flanged inlet and outlet connections provide ease of removal for maintenance purposes.

The Power Operated Relief Valve (PORV) in each Oconee unit is actuated by a DC solenoid-operated pilot valve that is connected to a Class IE DC system. The block valve for the PORV is an AC motor operated valve and is connected to an AC emergency power supply. The power supplies for the PORV and its associated block valve are therefore independent and diverse.

#### 5.4.6.4.1 PORV and Block Valve Testing and Qualification

Under the EPRI Test Program, for all tests applicable to Oconee, the Dresser PORV was opened and closed on demand. The functionability of the Dresser PORV has been shown for all expected operating and accident conditions applicable to Oconee Nuclear Station and the requirements of NUREG-0737, Item II.D.1.A have been met.

Under the EPRI PORV Block Valve Test Program, a Westinghouse motor-operated gate valve was tested on steam to full differential pressure conditions. Oconee Nuclear Station uses the same Westinghouse valve and LimiTorque operator for PORV block valve application. Based upon the successful EPRI tests for the valve-operator combination, the Oconee PORV block valves meet the intent of NUREG-0737 Item 11.D.1.B. The program test results were submitted to NRC in April 1982 and October 1985 (Reference 2 on page 5-66).

### 5.4.6.5 Relief Valve Effluent

Effluent from the pressurizer electromatic-relief and code safety valves discharges into the quench tank which condenses and collects the relief valve effluent. After the quench tank receives relief valve effluent, the tank contents are cooled to normal temperature by the component drain pump and quench tank cooler of the Coolant Storage System. The tank fluid is circulated from the tank through the cooler and returned to the tank by spraying into the tank vapor space. The quench tank is protected against overpressure by a rupture disc sized for the total combined relief capacity of the two pressurizer code safety valves and the pressurizer electromatic relief valve. The quench tank can be remotely vented to the Gaseous Waste Disposal System.

An Acoustical Monitoring System is installed on each unit. It is a reliable, single channel system, powered from a battery backed vital bus. It will provide the operator with positive indication of valve position and an annunciation of an open valve in the control room. The valve position indication components have been seismically and environmentally qualified as appropriate for conditions applicable to their location.

Backup valve position indication is provided by temperature sensors located downstream of the PORV and safety valves and by the quench tank level indicator.

## 5.4.7 INTERCONNECTED SYSTEMS

### 5.4.7.1 Low Pressure Injection

The Low Pressure Injection System provides the capability below about 250°F for cooling the Reactor Coolant System during plant cooldown. During this mode of operation, coolant is drawn from the Reactor Coolant System through a nozzle on the reactor outlet pipe, circulated through the low pressure injection coolers by the low pressure injection pumps and then injected back into the Reactor Coolant System through two nozzles on the reactor vessel into the inlet side of the core. The heat received by this system is rejected to the Low Pressure Service Water System. Components in these two systems are redundant for reliability purposes.

The Low Pressure Injection System also performs an emergency injection function for a loss of coolant accident and provides long term emergency core cooling; this is described in Chapter 6, "Engineered Safeguards" on page 6-1.

### 5.4.7.2 High Pressure Injection

The High Pressure Injection System controls the Reactor Coolant System coolant inventory, provides the seal water for the reactor coolant pumps, and recirculates Reactor Coolant System letdown for water quality maintenance and reactor coolant boric acid concentration control. Letdown of reactor coolant is through a nozzle on the outlet coolant pipe from one steam generator. The discharge of the high pressure injection pumps connects to a nozzle on each of the reactor inlet pipes downstream of the reactor coolant pumps. The reactor coolant which is letdown is returned to the Reactor Coolant System through the nozzles in a different heat transport loop from the heat transport loop containing the letdown line. Components are redundant for reliability purposes (Section 9.1, "Fuel Storage and Handling" on page 9-3).

The High Pressure Injection System utilizes four injection nozzles in carrying out the high pressure emergency injection function after a loss of coolant accident.

The High Pressure Injection/Makeup Nozzle assemblies at Oconee incorporate a thermal sleeve to provide a thermal barrier between the cold HPI/MU Fluid and the HOT HPI Nozzle. In 1982, High Pressure Injection/Makeup Nozzle cracking problems were identified on several operating B&W plants. A task force formed by B&W owners group identified the root cause of the failures and undertook modifications, in consultations with NRC to eliminate such future failures.

Site inspections of Oconee 1, 2 and 3 were conducted. Oconee 2 and Oconee 3 were found to have nozzle cracking and thermal sleeve displacement problems. The radiographic and ultrasonic testing of the Oconee 1 indicated that no abnormal conditions were present in any of the nozzles; this is attributable to the unique double thermal sleeve design of the Oconee 1 nozzles.

The B&W owners task force studied the safe end nozzle cracking problems on a generic basis and reported its findings to the NRC (Reference 3 on page 5-66). The task force concluded that all cracked safe ends of the HPI/MU nozzles were associated with loose thermal sleeves; the cracked safe ends were associated with the makeup nozzles only, and the cracks were propagated by thermal fatigue. B&W recommended modifications to the design and inspection of the HPI/MU nozzles. The modified design installs a hard rolled thermal sleeve which prevents thermal shock to the nozzle assembly and helps reduce flow induced vibrations more effectively. An in-service inspection program had been developed to provide early detection of the safe-end cracking problems. The Oconee 1 makeup nozzles did not require modifications but are now subject to an augmented ISI program.

### 5.4.7.3 Core Flooding System

The Core Flooding System floods the core in the event of a loss of coolant accident. Connection to the reactor vessel is through the two nozzles described above for low pressure injection. The low pressure injection and core flooding lines tie together and connect to the same nozzle on the reactor vessel.

2 The core flood nozzles have flow restrictors installed to minimize blowdown due to postulated core flood line break.

### 5.4.7.4 Secondary System

The principal Decay Heat Removal System interconnected with the Reactor Coolant System is the Steam and Power Conversion System. The Reactor Coolant System is dependent upon the Steam and Power Conversion System for decay heat removal at normal operating conditions and for all reactor coolant operating temperatures above 250°F. The system is discussed in detail in Section 10.2, "Turbine-Generator" on page 10-5.

The Turbine Bypass System routes steam to the condensers when the turbine has tripped or is shutdown and also during large plant load reduction transients when steam generation exceeds the demand. Overpressure protection for the secondary side of the steam generators is provided by the turbine bypass system and by safety valves mounted on the main steam lines outside of the Reactor Building. The Auxiliary Feedwater System will supply water to the steam generators in the event that the Main Feedwater System is inoperative. The physical layout of the Reactor Coolant System provides natural circulation of the reactor coolant to ensure adequate core cooling following a loss of all reactor coolant pumps.

### 5.4.7.5 Sampling

A sample line from the pressurizer steam space to the Chemical Addition and Sampling System permits detection of non-condensible gases in the steam space. This sample line also permits a bleeding operation from the vapor space to the letdown line of the High Pressure Injection System to transport accumulated noncondensible gases in the pressurizer to the letdown storage tank.

### 5.4.7.6 Remote RCS Vent System

The Oconee design has the capability for venting post-accident non-condensible gases that, in sufficient quantities, could accumulate at high points in the RCS and impair natural circulation. Although such an event is highly unlikely, the remote RCS vents on the RCS hot legs and reactor vessels will enable venting of these gases.

The design of the RCS High Point Vent System consists of two valves installed in series in each of the following existing vent connections: steam generator piping high points, and reactor vessel head high point. The redundant valve in each vent line assures that venting operations can be terminated under postulated single failure. The three pairs of valves each receive electrical power from a different safety related power source. Vent valve position indication is provided by limit switches within each solenoid valve. The valves require power to open and fail close on loss of power. The existing power operated relief valve can be used to vent the pressurizer.

The reactor vessel head vent is attached to an existing Axial Power Shaping Rod motor tube and closure assembly. Two normally deenergized solenoid valves are installed in the vent line and controlled from the control room. The vent ties into a hot leg vent and discharges into the air stream from the Reactor Building Cooling Units when operated.

#### 6.3.2.3.7 Heat Exchanger Characteristics

The decay heat removal coolers are designed to remove the decay heat generated during a normal shutdown. In addition, each cooler is capable of cooling the injection water during the recirculation mode following a loss-of-coolant accident to provide for removal of decay heat which provides adequate core cooling. The heat transfer capability of a decay heat removal cooler as a function of recirculated water
0 temperature is illustrated in Figure 6-18. Note that this figure is representative in nature and is provided
3 for information only. It is not intended to constitute design commitments or performance requirements
3 for the coolers.

#### 6.3.2.3.8 Relief Valve Settings

0 Relief valves are provided to protect the low pressure injection piping and components from overpressure.
3 On Units 1 and 2 these relief valves will be set at 370 psig, the system design pressure at 250°F for the "B"
3 LPI Coolers and at 515 psig, the system design pressure at 250°F for the "A" LPI Coolers. On Unit 3 the
3 relief valves will be set at 505 psig, the system design pressure at 250°F.

#### 6.3.2.3.9 Component Data

Component data for each ECCS System is given in the following tables:

1. High Pressure Injection System - Table 6-8
2. Low Pressure Injection System - Table 6-9
3. Core Flooding System - Table 6-10

#### 6.3.2.3.10 Quality Control

Quality Standards for the Emergency Core Cooling System components are given in Table 6-3.

### 6.3.2.4 Applicable Codes and Classifications

The High Pressure Injection, Low Pressure Injection, and Core Flooding Systems are designed and
5 manufactured to the Codes and Standards in Table 6-3 or FSAR Section 3.2.2.2, "System Piping
5 Classifications" on page 3-41 which allows use of substitute codes.

### 6.3.2.5 Material Specifications and Compatibility

All components with surfaces in contact with water containing boric acid are protected from corrosion and deterioration. The High Pressure Injection System, which operates continuously with borated reactor coolant, is constructed entirely of stainless steel. With the exception of the borated water storage tank, the major components in low pressure injection are constructed of stainless steel. The borated water storage tank is carbon steel with an interior phenolic coating. The core flooding piping and valves are stainless steel and the tanks are constructed of stainless clad carbon steel.

### 6.3.2.6 System Reliability

System reliability is assured by the system functional design including the use of normally operating equipment for safety functions, testability provisions, and equipment redundancy; by proper component selection; by physical protection and arrangement of the system; and by compliance with the intent of the AEC General Design Criteria. There is sufficient redundancy in the Emergency Core Cooling System to assure that no credible single failure can lead to significant physical disarrangement of the core. This is demonstrated by the single failure analysis presented in Table 6-11. This analysis was based on the

have been tested satisfactorily under the conditions which would be encountered during the loss-of-coolant accident. Both the high pressure and low pressure injection pump casings are liquid penetrant tested by methods described in the ASME Boiler and Pressure Vessel Code, Section VIII, and have been hydrotested and qualified to be able to withstand pressures as great or greater than 1.5 times the system design pressure. The pumps are designed so that periodic testing may be performed to assure operability and ready availability. The operating characteristics of each engineered safeguard pump are verified by shop testing before installation of the pumps.

#### 6.3.2.3.3 Heat Exchangers

The low pressure injection system heat exchangers (decay heat removal coolers) are designed and manufactured to the requirements of the ASME VIII and the TEMA-R (Rigorous) Standards. In addition to these requirements, uniformity of the tubes is assured by eddy current testing, and the tubes are seal welded to the tube sheet to decrease the possibility of leakage. All tube welded ends are liquid penetrant tested to assure the absence of welding flaws. The heat exchangers have been fabricated with surface areas greater than those dictated by the most severe heat transfer conditions.

#### 6.3.2.3.4 Valves

All remotely operated valves in the Emergency Core Cooling Systems are manufactured and inspected in
5 accordance with the intent of the ASME Nuclear Power Piping Code B31.7 or FSAR Section 3.2.2.2,
5 "System Piping Classifications" on page 3-41 which provides allowances for substitute codes. Liquid penetrant, radiography, ultrasonic, and hydrotesting are performed as the Code classification requires.

The seats and discs of these valves are manufactured from materials which will be free from galling and seizing. All valve material is certified to be in accordance with ASTM specifications. All remotely
3 operated valves in these systems are of the backseating type.

#### 6.3.2.3.5 Coolant Storage

The letdown storage tank has a total coolant volume of 600 $ft^3$ and normally contains approximately 2,600 gallons of water. This tank provides water to the high pressure injection pumps until the borated water storage tank outlet valves are opened. The letdown storage tank is designed and inspected in accordance with the requirements of ASME III-C.

Each unit is provided with a borated water storage tank as described in Chapter 9, "Auxiliary Systems" on page 9-1.

Provisions are made for sampling the water and adding concentrated boric acid solution or demineralized water.

3 Each core flooding tank contains approximately 7,000 gallons of borated water with a boron
3 concentration maintained in accordance with the Core Operating Limits Report.

#### 6.3.2.3.6 Pump Characteristics

Curves of total dynamic head and NPSH versus flow are shown in Figure 6-16 for the high pressure
3 injection pumps and in Figure 6-17 for the low pressure injection pumps. These curves are representative
3 in nature and are provided for information only. They are not intended to constitute design commitments
3 or performance requirements for the pumps. Refer to the Inservice Test Program for actual performance
3 requirements for HPI and LPI pumps.

4 Reactor Building Emergency Sump (RBES) to maintain continuous liquid flow through the core and
5 assure post-LOCA boric acid solubility. Additionally, the design of the reactor vessel and vessel internals
5 around the hot leg nozzles provides a third path that can assure post-LOCA boric acid solubility. At least
5 two of the three paths will always be available.

In the event of an accident where the Reactor Coolant System piping remains intact, then the Low Pressure Injection System will operate in the recirculation mode with suction being taken from the normal decay heat line. If in this mode of operation a decay heat isolation valve should fail closed, then a bypass line to the emergency sump would be opened. Recirculation would then take place with suction being taken from the emergency sump.

#### 6.3.2.2.3 Core Flooding System

The Core Flooding System provides core protection continuity for intermediate large Reactor Coolant System pipe failures. It automatically floods the core when the Reactor Coolant System pressure drops below 600 psig. The Core Flooding System is self-contained, self-actuating, and passive in nature. The combined coolant volume in the two tanks is sufficient to re-cover the core assuming no liquid remains in the reactor vessel following the loss-of-coolant accident.

The discharge pipe from each core flooding tank (CFT) is attached directly to a reactor vessel core flooding nozzle. Each core flooding line at the outlet of the CFT's contains an electric motor operated stop valve adjacent to the tank and two in-line check valves in series. The stop valves at the core flooding tank outlet are fully open during reactor power operation. Valve position indication is shown in the control room. During power operation when the Reactor Coolant System pressure is higher than the Core Flooding System pressure, two series check valves between the flooding nozzles and the CFT's prevent high pressure reactor coolant from entering the core flooding tanks.

The driving force to inject the stored borated water into the reactor vessel is supplied by pressurized nitrogen which occupies approximately one-third of the core flooding tank volume. Connections are provided for adding both borated water and nitrogen during power operation so that the proper level and pressure can be maintained. Each core flooding tank is protected from overpressurization by a relief valve installed directly on the tank. The size of these relief valves is based upon maximum water makeup rate to the tank. Redundant level and pressure indicators and alarms are provided in the control room for each tank.

### 6.3.2.3 Equipment and Component Descriptions

#### 6.3.2.3.1 Piping

The high pressure injection and low pressure injection lines are designed for the normal operating conditions. The system temperature and pressure requirements are greater than those encountered during emergency operation. The Low Pressure Injection System piping and valves are subjected to more severe conditions during decay heat removal operation than during emergency operation and, therefore, operate well within the design conditions. Table 6-4 gives the design pressure and temperatures of these systems. To assure system integrity, major piping has welded connections except where flanges are dictated for maintenance reasons.

#### 6.3.2.3.2 Pumps

The pumps used in the Emergency Core Cooling Systems are of proven design and have been used in many other applications. Pumps similar to the high pressure injection pumps have been used in boiler feed pump service and in high pressure makeup pump nuclear reactor service. Pumps similar to the low pressure injection pumps are used extensively in refinery service. The low pressure injection pump seals

reactor vessel through the reactor coolant inlet lines. The following automatic actions accomplish this change:

a. The isolation valves in the purification letdown line and in the seal return lines close.

b. The high pressure injection pumps start.

3 c. The throttle valve in each high pressure injection line opens.

d. The valves in the lines connecting to the borated water storage tank outlet header open.

3 In addition to the automatic action described, the pumps and valves may be remote manually operated from the control room. If any of the valve operators necessary to accomplish the above functions are 3 inoperable, the valves may be left in their emergency position during operation provided control of normal 3 parameters is not inhibited.

Operation of the High Pressure Injection System in the emergency mode will continue until the system 3 action is manually terminated. The HPI system is not designed to withstand a single passive failure since 3 the duration of system usage during an accident is not considered to be long term.

#### 6.3.2.2.2 Low Pressure Injection System

The Low Pressure Injection System is designed to 1) maintain core cooling for larger break sizes and 2) control the boron concentration in the core while operating in the recirculation mode. The Low Pressure Injection System operates independently of and in addition to the High Pressure Injection System. A description of the normal reactor operation mode for the system is given in Chapter 9, "Auxiliary Systems" on page 9-1.

Automatic actuation of the Low Pressure Injection System is initiated at: (a) Reactor Coolant System pressure of 550 psig (500 psig Technical Specification value) or (b) a Reactor Building pressure of 3 psig (4 psig Technical Specification value). Initiation of operation provides the following actions:

a. The valves in the lines connecting to the borated water storage tank outlet header open.

b. The low pressure injection pumps start on receipt of an engineered safeguards signal.

c. The inlet valves in the low pressure injection lines open.

d. Low pressure service water pumps start.

e. Service water valves from decay heat removal coolers open.

Low pressure injection is accomplished through two separate flow paths, each including one pump and one heat exchanger and terminating directly in the reactor vessel through core flooding nozzles located on opposite sides of the vessel.

The initial operation of the Low Pressure Injection System involves pumping water from the borated water storage tank into the reactor vessel. With all pumps operating and assuming the maximum break size, this mode of operation lasts for a minimum of about 30 minutes. When most of the borated water storage tank inventory is exhausted, a low water level alarm is annunciated in the control room. At this time the operator will take action to open the suction valve from the Reactor Building emergency sump, permitting recirculation of the spilled reactor coolant and injection water from the Reactor Building emergency sump.

Following a large break LOCA located in the reactor inlet piping, the boric acid concentration within the core region will increase. Recrystallized boric acid could deposit on fuel assemblies and hinder heat 4 transfer. The LPI system provides two redundant gravity flow paths from the reactor outlet piping to the

a. Injection of borated water from the borated water storage tank by the High Pressure Injection System.

b. Rapid injection of borated water by the Core Flooding System.

c. Injection of borated water from the borated water storage tank by the Low Pressure Injection System.

d. Long term core cooling by recirculation of injection water from the Reactor Building sump to the core by the Low Pressure Injection pumps.

e. Gravity drain from the reactor outlet piping to the Reactor Building emergency sump by the Low Pressure Injection System.

Although the high and low pressure emergency injection systems operate to provide full protection across the entire spectrum of break sizes, each system may operate individually and each is initiated independently. High pressure injection prevents uncovering of the core for small coolant piping leaks where high system pressure is maintained, and to delay uncovering of the core for intermediate-sized leaks. The core flooding and low pressure injection systems are designed to re-cover the core at intermediate-to-low pressures, and to assure adequate core cooling for break sizes ranging from intermediate breaks to the double-ended rupture of the largest pipe. The Low Pressure Injection System is also designed to permit boron concentration control and long-term core cooling in the recirculation mode after a LOCA. The injection and core flooding functions are subdivided so that there are two separate and independent strings, each including one high pressure pump, one low pressure pump, and
1 one core flooding tank.

Much of the equipment in these systems serves a function during normal reactor operation. In those cases where equipment is used for emergency functions only, such as the Reactor Building Spray System, systems have been designed to permit meaningful periodic tests. Operational reliability is achieved by using proven component designs, and by conducting tests where either the component or its application was considered unique. Quality control procedures are imposed on the components of the engineered safeguards systems. These procedures include use of accepted codes and standards as well as supplementary test and inspection requirements to assure that all components will perform their intended function under the design conditions following a LOCA.

## 6.3.2 SYSTEM DESIGN

### 6.3.2.1 Schematic Piping and Instrumentation Diagrams

The schematic diagrams for the Emergency Core Cooling System are shown in Figure 6-1. Instrumentation is shown schematically in Chapter 7, "Instrumentation and Control" on page 7-1.

### 6.3.2.2 ECCS Operation

#### 6.3.2.2.1 High Pressure Injection System

During normal reactor operation, the High Pressure Injection System recirculates reactor coolant for purification and for supply of seal water to the reactor coolant circulating pumps. This normal operation mode is described in Chapter 9, "Auxiliary Systems" on page 9-1. The High Pressure Injection System is initiated at: (a) a low Reactor Coolant System pressure of 1,600 psig (1500 psig Technical Specification value) or (b) a Reactor Building pressure of 3 psig (4 psig Technical Specification value). Automatic actuation of the valves and pumps by the actuation signals switches the system from its normal operating mode to the emergency operating mode to deliver water from the borated water storage tank into the

# 6.3 EMERGENCY CORE COOLING SYSTEM

0

**Note**

0 This section of the FSAR contains information on the design bases and design criteria of this
0 system/structure. Additional information that may assist the reader in understanding the system is
0 contained in the design basis document (DBD) for this system/structure.

## 6.3.1 DESIGN BASES

The Emergency Core Cooling System (ECCS) is designed to cool the reactor core and provide shutdown capability following initiation of the following accident conditions:

1. Loss-of-coolant accident (LOCA) including a pipe break or a spurious relief or safety valve opening in the RCS which would result in a discharge larger than that which could be made up by the normal make-up system.
2. Rupture of a control rod drive mechanism causing a rod cluster control assembly ejection accident.
3. Steam or feedwater system break accident including a pipe break or a spurious relief or safety valve opening in the secondary steam system which would result in an uncontrolled steam release or a loss of feedwater.
4. A steam generator tube rupture.

The primary function of the ECCS is to remove the stored and fission product decay heat from the reactor core during accident conditions.

The ECCS provides shutdown capability for the accident above by means of boron injection. It is designed to tolerate a single active failure (short term) or single active or passive failure (long term). It can meet its minimum required performance level with onsite or offsite electrical power and under simultaneous Safe Shutdown Earthquake loading.

The Emergency Core Cooling System for one reactor unit is shown in Figure 6-1. The overall Emergency Core Cooling System is comprised of the following independent subsystems:

a. High Pressure Injection System

b. Low Pressure Injection System

c. Core Flooding System

The principal design basis for the Emergency Core Cooling System as described in the proposed AEC General Design Criterion 44 has been met. Protection for the entire spectrum of break sizes is provided. Two separate and independent flow paths containing redundant active components are provided in the ECCS. Redundancy in active components assures performing the required functions should a single failure occur in any of the active components. Separate power sources are provided to the redundant active component. Separate instrument channels are used to actuate the systems. The adequacy of the installed ECCS to prevent fuel and clad damage is discussed in Chapter 15, "Accident Analyses" on page 15-1.

The ECCS is designed to operate in the following modes:

## 6.2.5 REFERENCES


5 1. DPC-NE-3003-P-A, "Duke Power Company Oconee Nuclear Station Mass and Energy Release and
5 Containment Response Methodology", Revision 0, March, 1995.

5 2. BAW-1016-P-A, "RELAP5/MOD2-B&W--An Advanced Computer Program for Light Water
5 Reactor LOCA and Non-LOCA Transient Analysis", Revision 1, April, 1990.

5 3. "CAP--Containment Analysis Package (FATHOMS 2.4)", Numerical Applications, Inc., October 10,
5 1989.

5 4. BAW-10030, "CRAFT - Description of Model for Equilbrium LOCA Analysis Program", Revision 0,
5 October 1971.

5 5. BAW-10034, "Multinode Analysis of B&W's 2568 MWt Nuclear Plants During a LOCA", Babcock
5 & Wilcox, Revision 3, May 1972.

5 6. Letter from A. C. Thies (Duke) to A. Schwencer (NRC) dated June 28, 1973.

5 7. ANSI/ANS-5.1-1979, "Decay Heat Power in Light Water Reactors", American Nuclear Society.

5 8. ANSI/ANS-56.4-1983, "Pressure and Temperature Transient Analysis for Light Water Reactor
5 Containments", American Nuclear Society, December 1983.

5 9. NP-1850-CCM-A, "RETRAN-02--A Program for Transient Analysis of Complex Fluid Flow
5 Systems", Electric Power Research Institute, Revision 4, November 1988.

5 10. DPC-NE-3000-P-A, "Duke Power Company Nuclear Station Thermal-Hydraulic Transient Analysis
5 Methodology", Revision 1, December 1995.

5 11. BAW-10104-P-A, "B&W's ECCS Evaluation Model", Babcock & Wilcox, Revision 5, April 1986.

5 12. BAW-10103-P-A, "ECCS Analysis of B&W's 177-FA Lowered Loop NSSS", Babcock & Wilcox,
5 Revision 3, July 1997.

5 13. Letter from W. O. Parker, Jr. (Duke) to B. C. Rusche (NRC) dated October 10, 1975.

protect public health and safety. Leak testing will be continued until a satisfactory leak rate has again been demonstrated.

A considerable background of operating experience is being accumulated on containments and penetrations. Full advantage of this knowledge has been taken in all phases of design, fabrication, installation, inspection and testing. Practical improvements in design and details have been incorporated as they are developed, where applicable.

The steel-lined Reactor Building is self-sufficient, and other than valves and hatch doors, there are no operating parts. The containment boundary is extended only by listed penetrations and further described and tabulated in Section 6.2.3, "Containment Isolation System" on page 6-27.

barrier could be violated in any way that would be significant to the public health and safety or to that of the station personnel. Adequate administrative controls are enforced to minimize the possibility of human error. Station operators are trained and licensed in accordance with regulations. Safety analyses are presented in Chapter 15, "Accident Analyses" on page 15-1.

Penetrations such as the personnel access and emergency hatches cannot be opened except by deliberate action and are interlocked and alarmed by fail-safe devices such that the Reactor Building will not be breached unintentionally. The liner plate over the foundation slab is protected by cover concrete. Wherever access to the liner plate is blocked by interior concrete, means are provided so that weld seams
0 can be tested for leakage. The liner plate is protected against corrosion by suitable coatings. Walls and floors for biological and missile shielding, and for access and operating purposes, also provide compartmentation which constitutes protection for the liner during operating as well as accident conditions.

Once the adequacy of the liner has been established initially, there is no reason to anticipate progressive deterioration during the life of the station which would reduce the effectiveness of the liner as a vapor barrier. Inside the Reactor Building, the atmosphere is subject to a high degree of temperature control. The outside of the liner is protected by 3 3/4 feet of prestressed concrete which is exceptionally resistant to all weather conditions.

Inspection on a periodic basis, as necessary, will be conducted in all spaces accessible under full power operation. Biological shielding is provided to reduce radiation to limits which make occupancy of spaces adjacent to the liner permissible.

All penetrations except the following are grouped within or vented to the penetration room. Any leakage that might occur from these penetrations will be collected and discharged through high efficiency particulate air (HEPA) filters and charcoal filters to the unit vent as described in Section 6.5, "Fission Product Removal and Control Systems" on page 6-55. In this manner, leakage which might occur from these penetrations will be isolated from leakage which might occur through the Reactor Building itself.

3 1. (1) Main Steam Lines
3 2. Normal Sump Drain Line
3 3. Emergency Sump Drain Line
3 4. (2) RB Emergency Sump Recirc. Lines
3 5. (2) Fuel Transfer Tubes
3 6. Quench Tank Drain Line
3 7. RC Post Accident Sample Line

The above lines are not considered a source of significant leakage because they are welded to the liner plate.

Individual major penetrations or groups of penetrations are tested by means of permanently installed pressure connections or temporarily installed pressure or vacuum boxes. If necessary, liner plate weld seams will be tested by the vacuum box soap bubble method where accessible, or by means of the permanently installed backup channels and angles where inaccessible.

In any event, sources of excessive leakage will be located and such corrective action as necessary will be taken. This will consist of repair or replacement. Appropriate action will also be taken to minimize the possibility of recurrence of excessive leakage, including such redesign as might prove to be necessary to

3 #### 6.2.3.3 Periodic Operability Tests

3 Each containment isolation valve will be tested periodically during normal operation or during shutdown
3 conditions to assue its operability when needed. A description of periodic testing programs for
3 containment isolation valves and other penetrations is provided in Section 3.8.1.7.4, "Leakage Monitoring"
3 on page 3-118.3.8, "Design of Class 1 Structures" on page 3-75 .

### 6.2.4 CONTAINMENT LEAKAGE TESTING

#### 6.2.4.1 Periodic Leakage Testing

Tests and surveillance are performed periodically to verify that leakage from the containment is maintained within acceptable limits. These tests include:

Integrated Leak Rate Tests

3 Local Leak Detection

2 These tests are discussed in detail in Section 3.8.1.7.4, "Leakage Monitoring" on page 3-122.

#### 6.2.4.2 Continuous Leakage Monitoring

No continuous Reactor Building leakage monitoring system is provided.

The barrier to leakage in the Reactor Building is the one-quarter inch steel liner plate. All penetrations are continuously welded to the liner plate before the concrete in which they are embedded is placed. The penetrations, shown on Figure 6-13 and Figure 6-14, become an integral part of the liner and are so designed, installed, and tested.

The steel liner plate is securely attached to the prestressed concrete Reactor Building and is an integral part of this structure. This Reactor Building is conservatively designed and rigorously analyzed for the extreme loading conditions of a highly improbable hypothetical accident, as well as for all other types of loading conditions which could be experienced. Thorough control is maintained over the quality of all materials and workmanship during all stages of fabrication and erection of the liner plate and penetrations and during construction of the entire Reactor Building.

The comprehensive program for preoperational testing, inspection, and postoperational surveillance is described in detail in Section 3.8, "Design of Class 1 Structures" on page 3-75 and is summarized in the following paragraphs.

During construction, the entire length of every seam weld in the liner plate was leak tested. Individual penetration assemblies are shop tested. Welded connections between penetration assemblies and the liner plate were individually leak tested after installation. Following completion of construction, the entire Reactor Building, the liner, and all its penetrations were tested at 115 percent of the design pressure to establish structural integrity. The initial leak rate tests of the entire Reactor Building were conducted at the maximum calculated peak accident pressure and at one-half this pressure to demonstrate vapor tightness and to establish a reference for periodic leak testing for the life of the station. Multiple and redundant systems based on different engineering principles are provided as described in this chapter to provide a very high degree of assurance that the accident conditions will never be exceeded and that the vapor barrier of the containment will never be jeopardized.

Under all normal operating conditions and under accidental conditions short of the worst loss-of-coolant accident, virtually no possibility exists that any leakage could occur or that the integrity of the vapor

### 6.2.3.2 System Design

The fluid penetrations which require isolation after an accident may be classed as follows:

Type A. Each line connecting directly to the Reactor Coolant System has two Reactor Building isolation valves. One valve is inside and the other is outside the Reactor Building. These
3 valves may be either a check valve and an automatic remotely operated valve, two
3 automatic remotely operated valves, or two check valves, depending upon the direction of
3 normal flow (referred to as Type I penetrations in the DBD).

Type B. Each line connecting directly to the Reactor Building atmosphere has two isolation valves. At least one valve is outside and the other may be inside or outside the Reactor
3 Building. These valves may be either a check valve and an automatic remotely operated
5 valve, or one check valve and one, normally closed manual valve, or two automatic
3 remotely operated valves, or two check valves, depending upon the direction of normal
5 flow (referred to as Type II penetrations in the DBD). For piping not part of the
5 process flow, double isolation will be used. One or more of the isolations will be a
5 normally closed manual valve located on the vent, drain, or test connection. The other
5 isolation valve may be located on the process piping.

3 Type C. Each line not directly connected to the Reactor Coolant System or not open to the
3 Reactor Building atmosphere has at least one valve, either a check valve or an automatic
3 remotely operated valve. This valve is located outside the Reactor Building. A seismic
3 closed loop forms the inside barrier for most Type C penetrations. Since the Component
3 Cooling System has a non-seismic closed loop, penetrations for this system have an
3 additional automatic remotely operated valve or check valve located inside the Reactor
3 Building (referred to as Type III penetrations in the DBD).

3 Type D. Each line connected to either the Reactor Building atmosphere or the Reactor Coolant
3 System, but which is not normally open during reactor operation, has two isolation
3 valves. They may be manual valve(s) with provisions for locking in a closed position,
3 check valve(s), and/or remotely operated valve(s), depending upon the direction of the
3 normal flow (referred to as Type IV penetrations in the DBD).

There are additional subdivisions in each of these major groups. The individual system flow diagrams show the manner in which each Reactor Building isolation valve arrangement fits into its respective
3 system. For convenience, each different valve arrangement is shown in Table 6-7 and Figure 6-9 of this
3 section. The symbols on Figure 6-9 are described at the end of Table 6-7. This table lists the mode of actuation, the type of valve, its normal position and its position under Reactor Building isolation conditions. The specific system penetrations to which each of the arrangements is applied is also
3 presented. It may be noted that only electric motor-operated, manual normally closed, or check valves
3 are used inside the Reactor Building. Each valve will be tested periodically during normal operation or during shutdown conditions to assure its operability when needed.

3 Fluid penetrations which do not require isolation after an accident are also classified as Type A through
3 D, however the redundant containment isolation provisions described above are not applicable. Such
3 penetrations are identified on Figure 6-9 as "PA" for Post Accident.

There is sufficient redundancy in the instrumentation circuits of the engineered safeguards protective system to minimize the possibility of inadvertent tripping of the isolation system. Further discussion of this redundancy and the instrumentation signals which trip the isolation system is presented in Chapter 7, "Instrumentation and Control" on page 7-1.

Reactor Building Spray Nozzles

With the Reactor Building Spray inlet valves closed, low pressure air or fog can be blown through the test connections. Visual observation will indicate flow paths are open.

During these tests, the equipment can be visually inspected for leaks. Valves and pumps will be operated and inspected following maintenance on the system to assure proper operation.

The RBCU equipment, piping, valves, and instrumentation are arranged so that they can be visually inspected. The cooling units and associated piping are located outside the secondary concrete shield. Personnel can enter the Reactor Building during power operations to inspect and maintain this equipment. The service water piping and valves outside the Reactor Building are inspectable at all times. Operational tests and inspections will be performed prior to initial startup.

The cooling units will be tested periodically as follows:

a. The fans can be started and inspected for proper operation.

b. The return line service water valves will be opened, and the lines checked for flow.

Additional discussion of tests of the containment heat removal systems is provided in Section 3.8, "Design of Class 1 Structures" on page 3-75.

## 6.2.3 CONTAINMENT ISOLATION SYSTEM

### 6.2.3.1 Design Bases

3 The general design basis governing isolation requirements is:

> Leakage through all fluid penetrations not serving accident-consequence limiting systems is to be minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss-of-isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the Reactor Building, and various types of isolation valves.

3 Reactor Building Essential and Non-essential Isolation occurs on an Engineered Safeguards signal of 3
3 psig (4 psig Technical Specification value) in the Reactor Building. Reactor Building Non- essential
3 Isolation occurs on an Engineered Safeguards signal of 1600 psig(1500 psig Technical Specification value)
3 in the Reactor Coolant System. For details on Reactor Building Essential and Non- essential Isolation,
3 refer to Section 7.3, "Engineered Safeguards Protective System" on page 7-19 and Table 7-2 and Table 7-3.
3 Valves which isolate penetrations that are normally directly open to the Reactor Building (the Reactor
3 Building purge valves and sump drain valves) will also be closed on a high radiation signal. (The
3 radiation monitor signal is not an Engineered Safeguards signal). Although normally open to the Reactor
3 Building, the Reactor Building Gaseous Radiation Monitor penetrations are not closed on a high
3 radiation signal; they remain open (except during ES isolation) to provide continuous monitoring.

The isolation system closes all fluid penetrations, not required for operation of the engineered safeguards systems, to prevent the leakage of radioactive materials to the environment.

All remotely operated Reactor Building isolation valves are provided with position limit indicators in the control room. All solenoid valves used in actuating pneumatic RB isolation valves are environmentally qualified to the requirements of the IE Bulletin 79-01B.

The Reactor Building Spray System will deliver 3,000 gal/min through the spray nozzles within 92 seconds after the Reactor Building reaches a setpoint of less than or equal to 30 psig (typical value is 10 psig), although some flow would be expected at least 18 seconds earlier.

The Reactor Building Cooling System provides the design heat removal capacity following a loss-of-coolant accident with all three coolers operating by continuously circulating the steam-air mixture past the cooling tubes to transfer heat from the containment atmosphere to the low pressure service water.

Building pressure is limited below the design pressure. The design heat load at these conditions is 240 x $10^6$ Btu/hr. The design inlet cooling water is 75°F, although the expected cooling water range is 45 - 85°F. RBCU performance has been evaluated at elevated LPSW temperatures (up to 90°F), and show adequate capacity. The design heat removal capacity for these units is shown in Figure 6-6. The safety analysis given in Chapter 15, "Accident Analyses" on page 15-1 demonstrates system effectiveness.

### 6.2.2.4 Tests and Inspection

The active components of the Reactor Building Spray System can be tested as follows:

Reactor Building Spray Pumps

The delivery capability of one pump at a time can be tested by opening the valve in the line from the borated water storage tank, opening the corresponding valve in the test line, and starting the corresponding pump. Pump discharge pressure and flow indication demonstrate performance.

Borated Water Storage Tank Outlet Valves

These valves will be tested in performing the pump test above.

Reactor Building Spray Injection Valves

With the pumps shut down and the borated water storage tank outlet valves closed, these valves can each be opened and closed by operator action. In the event the valve operators are inoperable, the valves are tested manually.

The RBCU's and associated piping are located outside the secondary concrete shielding. The ductwork required to operate during an accident is located outside of the secondary shielding.

#### 6.2.2.2.7 System Actuation

The Reactor Building Spray System will be activated when Reactor Building pressure reaches a setpoint of less than or equal to 30 psig (typical value is 10 psig). The system components may also be actuated by operator action from the control room for performance testing.

In the event of a loss-of-coolant accident, the RBCU's are initiated at a Reactor Building pressure of 3 psig (4 psig Technical Specification Limit). The cooling units are placed in operation as follows:

a. Valve LPSW-565 is closed, stopping water flow in the Reactor Building Auxiliary Cooling Units, and valve LPSW-566 is opened, establishing flow to RBCU "B".

b. The Low Pressure Service Water valves at the discharge of the coolers go to the full open
4 position. Normally, these valves are operating with an intermediate setting (1200-1400 gal/min per loop). The loop flow under accident conditions is 1,400 gal/min.

c. The idle cooling unit fan is started; and the speed of all fans is switched to half speed to change the horsepower capacity, required by the denser building atmosphere.

d. The links that hold the fusible dropout plates in the duct work melt and drop off, assuring that a positive path for recirculation of the Reactor Building atmosphere is available.

e. Depending upon the severity of the accident, the blowout plates at the bottom of the downcomer are designed to be forced out by any shock wave, allowing attenuation of the wave before it reaches the cooling coils. Analysis has shown this to be a highly unlikely scenario due to duct deformation, and therefore the blowout plates are not needed for this function.

#### 6.2.2.2.8 Environmental Considerations

None of the active components of the Reactor Building Spray System are located within the Reactor Building, so none are required to operate in the steam-air environment produced by the accident.

Figure 6-8 depicts the Reactor Building post-accident steam-air conditions. The RBCU fans and motors are designed for operation in the post-accident conditions. Cooling capability of the coolers has been satisfactorily tested in this environment.

#### 6.2.2.2.9 Quality Control

Quality standards for the Reactor Building Spray System components are given in Table 6-3.

### 6.2.2.3 Design Evaluation

The Reactor Building Spray System, acting independently of the Reactor Building Cooling System is
2 capable of reducing the containment pressure after a loss-of-coolant accident. The Reactor Building Spray System is at least equivalent in heat removal capacity to the cooling system. The Reactor Building Spray System is designed for long term postaccident operation. In combination with cooling units, it
2 affords redundant alternative methods to remove heat from containment. Any of the following
2 combinations of equipment will provide sufficient heat removal capability:

a. The Reactor Building Spray System alone.

b. Three cooling units alone.

c. Two cooling units and the Reactor Building Spray System at one-half capacity.

RBCU Coolers

The cooling surface of the cooling units has been designed for and satisfactorily tested under simulated
4 post-accident conditions. A conservative design has resulted in a heat exchanger which has a design heat transfer capability in excess of the expected heat transfer requirements.

The Reactor Building cooler is located in the discharge ducting for the fan. The air-steam mixture flows across the tube bank, resulting in condensation of a portion of the steam and removal of sensible heat
4 from the air. Figure 6-6 shows the design heat transfer capability of each unit at various Reactor Building temperature conditions. Figure 6-6 is based on a Low Pressure Service Water temperature of 75°F. Actually, the cooling water is drawn from a point near the bottom of the lake and the anticipated service water temperature would be in the range of 45 to 85°F. Therefore, the curve shown in Figure 6-6 is conservative for most of the year. RBCU heat transfer capacity has been evaluated for LPSW entering temperatures up to 90°F and show acceptable results. Figure 6-7 shows how the Reactor Building cooling rate varies with the air-steam mixture flow rate. It can be seen that even if the mixture flow rate decreases by 40 percent, the cooling capability decreases by less than 7 percent.

2

RBCU Fans

Circulation of the Reactor Building atmosphere under accident conditions is by the same fans used for normal ventilation. Upon actuation by an engineered safeguards signal, the fan motors switch from full speed to half speed and the idle unit is started at half speed (Section 6.2.2.2, "System Design" on page 6-22). Prototype fan motors combination testing has demonstrated the capability to supply design flow of steam-air mixture through the coolers. The control circuitry of the RBCU fans has been modified to remain in the ES state after reset of the ES channels. This modification ensures that deliberate separate action is required to shutdown the RBCU's. This modification is made pursuant to the requirements of IE Bulletin 80-06.

#### 6.2.2.2.5 Reliability Considerations

A failure analysis has been made on all active components of the BS System to show that the failure of any single active component will not prevent fulfilling the design function. This analysis is shown in Table 6-5.

Inside the Reactor Building, the RBCU's are located outside the secondary shield at an elevation above the water level in the bottom of the Reactor Building during post-accident conditions. In this location the units are protected from being flooded.

The major equipment of the Reactor Building Cooling Units is arranged in three independent strings with three duplicate service water supply lines. In the unlikely event of a failure in one of the three cooling units, the Reactor Building Spray System independently, or half of the Reactor Building Spray System capacity combined with the remaining two cooling units, will provide cooling capacity in excess of that required. Fan-motor operation under design LOCA condition has been demonstrated by prototype test.

A failure analysis of the cooling units is presented in Table 6-6.

#### 6.2.2.2.6 Missile Protection

BS System protection against missile damage is provided by direct shielding or by physical separation of duplicate equipment. The spray headers are located outside and above the primary and secondary concrete shield.

high speed, serve to cool the Reactor Building atmosphere. The Engineered Safeguards System (Channels 5 and 6) is actuated when the Reactor Building pressure reaches 3 psig (4 psig Technical Specification Limit). Upon ES actuation, the fan motors associated with the RBCU's operating at high speed (A and C units) change to low speed, and the idle unit (B unit) is energized at low speed.

Performance of the cooling system is monitored by flow instrumentation in the service water return line from each cooler and by the Reactor Building temperature and pressure instrumentation.

#### 6.2.2.2.2 Codes and Standards

BS System equipment is designed to the applicable codes and standards given in Chapter 9, "Auxiliary Systems" on page 9-1.

4 The cooling coils for the RBCU's are constructed in accordance with ASME Section III, Class 3
4 guidelines. The Low Pressure Service Water System is designed to USAS B31.1.

#### 6.2.2.2.3 Materials Compatibility

All materials in the BS System are compatible with the reactor coolant. The major components of the system are constructed of stainless steel. Minor parts such as pump seals utilize other corrosion resistant materials.

The materials for the RBCU's have been selected to be compatible with the use of untreated service water
4 to minimize corrosion in accordance with ASME guidelines.

#### 6.2.2.2.4 Component Design

BS Pumps

The Reactor Building Spray pumps are similar to those used in refinery service. These pumps are liquid-penetrant tested by methods described in the ASME Boiler and Pressure Vessel Code Section VIII and are hydrotested and qualified to be able to withstand pressures greater than 1.5 times the design pressure. The pumps are designed so that periodic testing may be performed to assure operability at all times.

Curves of total dynamic head and NPSH versus flow are shown in Figure 6-5.

BS Valves

The remotely operated valves of the Reactor Building Spray System are designed and manufactured to the same requirements as the valves in the Emergency Core Cooling Systems. Refer to Section 6.3, "Emergency Core Cooling System" on page 6-33.

RB Spray Headers and Nozzles

120 full core spray nozzles are arranged on each of the two Reactor Building Spray headers. The spray nozzles are spaced in the headers to give uniform spray coverage of the Reactor Building volume above the operating floor.

BS Piping

Except for the sections of lines requiring flanged connections for maintenance, the entire system is welded construction. Table 6-4 lists the design conditions for this system.

5 temperature. The assumptions used in these analyses result in a conservatively calculated minimum
5 containment pressure response. This method has been shown to result in the maximum peak clad
5 temperature.

5 The analysis of the minimum containment pressure response for the reflood analysis is performed on a
5 generic basis as detailed in References 11 on page 6-32 and 12 on page 6-32. The mass and energy
5 release to the Reactor Building and the resulting pressure response for the worst case LOCA, 8.55 $ft^2$ cold
5 leg break at the pump suction, is shown in Figure 6-50 (mass releases), Figure 6-51 (energy releases), and
5 Figure 6-52 (pressure response). The CRAFT2 model which is used for this analysis is shown in
5 Figure 6-53. Since a generic Reactor Building is utilized in the analysis, a comparison of the generic
5 building heat sink and containment systems data with Oconee specific data was compiled to verify the
5 conservatism of the generic analysis (Reference 13 on page 6-32).

## 6.2.2 CONTAINMENT HEAT REMOVAL SYSTEMS

### 6.2.2.1 Design Bases

Two independent engineered safeguards systems, the Reactor Building Spray System and the Reactor
2 Building Cooling System, are provided to remove heat from the containment following an accident.

### 6.2.2.2 System Design

#### 6.2.2.2.1 Piping and Instrumentation Diagrams

A schematic diagram of the Reactor Building Spray (BS) System is shown in Figure 6-2. The system serves no function during normal operation.

Removal of post-accident energy is accomplished by directing borated water spray into the Reactor Building atmosphere. The system consists of two pumps, two Reactor Building Spray headers, isolation valves, and the necessary piping, instrumentation and controls. The pumps and remotely operated valves for each unit can be operated from the control room. The Reactor Building Spray System is sized to furnish 100 percent of the design cooling capacity (240 x $10^6$ Btu/hr) with both of the spray paths in operation. Both paths operate independently, and the Reactor Building Spray System also operates
2 separately from the Reactor Building Cooling Units, which independently possess post-accident cooling capability.

A high Reactor Building pressure signal of less than or equal to 30 psig (typical value is 10 psig) from the Engineered Safeguards System (Channels 7 and 8) initiates operation of the BS system. The two pumps start, taking suction initially from the borated water storage tank through the intertie with the Low Pressure Injection System, and initiate building spray through the spray headers and nozzles. After the water in the borated water storage tank reaches a low level, the spray pump suction is transferred to the Reactor Building sump automatically when the operator places the Low Pressure Injection System in the recirculation mode. The Reactor Building emergency sump water is cooled by the Low Pressure Injection System as described in Section 6.3, "Emergency Core Cooling System" on page 6-33.

This system shares borated water storage tank capacity with the Low Pressure Injection System and the High Pressure Injection System.

Figure 6-3 illustrates the Reactor Building Cooling Units (RBCU's). Each cooling unit consists of a fan, cooling coils, and the required distribution duct work. The Reactor Building atmosphere is circulated past cooling coils by fans and returned to the building. Cooling water for the cooling units is supplied by the Low Pressure Service Water System. During normal operation these units, with two fans operating at

- It is assumed that failure of a feedwater control valve to close on a feedwater isolation signal is beyond the licensing basis.
- Credit is taken for the trip of the main feedwater pumps in the mass and energy release analyses for steam line breaks with automatic feedwater isolation available.

#### 6.2.1.4.3 Initial Conditions

The criteria presented in Reference 8 on page 6-32 are used as the bases for the choices of initial conditions in the steam line break mass and energy release analyses. The specific conservatisms are:

1. End of core life conditions are chosen to maximize the energy addition to the primary system. The initial fuel temperature used is 1072°F.
2. 102% power is assumed, corresponding to the licensed core thermal power plus a 2% measurement uncertainty allowance. This maximizes the available generated energy and stored core energy for release to the secondary side.
3. The assumed reactor vessel average temperature is the nominal at power value, 579°F, plus a 2°F uncertainty allowance. This maximizes the stored energy in the Reactor Coolant System.
4. The assumed Reactor Coolant System pressure is the nominal value, 2155 psig, plus a 30 psi uncertainty allowance. This maximizes time to reactor trip and thus the energy transferred to the secondary system.
5. Steam line pressure is left at the nominal value rather than being increased to delay the generation of a feedwater isolation signal. This is required so that RETRAN-02 model calculated steam generator tube heat transfer areas correspond to the physical tube areas.
6. A conservatively large steam generator initial fluid mass is assumed to maximize the inventory available for release through the break.
7. End of core life fuel and moderator temperature feedback is assumed to maximize positive reactivity insertion from the cooldown. A low effective delayed neutron fraction and prompt neutron lifetime are also chosen to maximize the positive reactivity added by the cooldown.
8. The control rods are assumed to be positioned such that a reactor trip inserts only the amount of negative reactivity which produces and maintains the minimum shutdown margin required by the Technical Specifications.
9. The core boron concentration is assumed to be zero, which is consistent with end of core life conditions.

#### 6.2.1.4.4 Description of Blowdown Model

The RETRAN-02/MOD005 computer code, described in Reference 9 on page 6-32 is used to generate the mass and energy releases for steam line breaks inside containment. The models used for this calculation are generally described in Reference 10 on page 6-32 with modifications for the containment mass and energy release calculations as described in Reference 1 on page 6-32. The calculational methods for applying this code and model to calculate mass and energy releases for steam line breaks are also described in Reference 1 on page 6-32. Reference 1 on page 6-32 also discusses and justifies the conservatisms in this calculational method. Reference 9 on page 6-32 presents the heat transfer correlations used to calculate the heat transferred from the steam generator tubes and shell and justifies their application. No liquid entrainment is assumed in the break flow. The analysis methodology used credits the Feedwater Isolation System in conjunction with operator action to manually isolate motor-driven emergency feedwater at ten minutes after a postulated steam line break occurs.

### 6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies on Emergency Core Cooling System

The pressure response of the containment to a LOCA is analyzed to determine the backpressure in the containment as a boundary condition for the reflood analysis and the calculation of the peak clad

#### 6.2.1.3.5 Metal-Water Reaction

The energy released by steam/cladding metal-water reaction is considered in the short-term LOCA mass and energy release calculation. Reference 1 on page 6-32 provides the methodology for modeling this energy source. The energy from the metal-water reaction is not considered in the long-term LOCA mass and energy release analysis. Reference 1 on page 6-32 gives a quantitative discussion of the significance of this energy source and the justification for omitting it.

### 6.2.1.4 Mass and Energy Release Analyses for Postulated Secondary System Pipe Ruptures Inside Containment

The limiting secondary system pipe rupture from a containment response point of view is the steam line break. This is because the feedwater exiting a steam line break will have been heated to a higher temperature inside the steam generator via heat transfer across the steam generator tubes. In contrast, the feedwater exiting a feedwater line break will only be as hot as the outlet of the last feedwater heater upstream of the break location. Therefore, only steam line breaks are evaluated in this section. The model used is adjusted as described in Reference 1 on page 6-32 to prevent any predicted liquid entrainment from decreasing the break enthalpy below the enthalpy of dry steam. Only the results of double-ended guillotine breaks of the maximum steam line pipe area are presented. This break size is 6.305 $ft^2$. For peak containment pressure this is the worst case because the rate of mass and energy release to containment is maximized. For peak containment temperature the response depends on both steam mass flow rate and on steam enthalpy. The steam enthalpy depends on hot leg temperature, steam flow and steam generator pressure. A double-ended steam line break results in higher steam enthalpies than smaller breaks. The larger break flow increases the overcooling of the primary side, which increases the hot leg temperature through an increase in reactor power. The higher feedwater flow enhances the heat transfer between the primary and the secondary sides, and the low steam generator pressure occurring during a large break tends to increase the steam enthalpy. Sensitivity studies were done in which the break flow was reduced by factors of 0.9, 0.8 and 0.5. In all three cases a conservatively high exit enthalpy was used. The results show that the peak containment temperature response is most severe for the double-ended break.

#### 6.2.1.4.1 Mass and Energy Release Data

The mass and energy release data for the limiting secondary line break analyzed, a 6.3 $ft^2$ double-ended guillotine break of a main steam line at the steam generator outlet, is presented in Table 6-32.

#### 6.2.1.4.2 Single Failure Analysis

The failure of an emergency feedwater control valve is chosen as the single failure for the steam line break mass and energy release analysis. Other potential single failures were considered:

- Failure of a 4160V switchgear was also analyzed. The failure of one of the three available switchgear results in the loss of one train of LPI and one train of HPI. This failure also results in the loss of one Reactor Building Cooling Unit and one Reactor Building Spray train as discussed in Section 6.2.1.1.3.2, "LOCA Long-Term Containment Temperature Response" on page 6-11. Note that this single failure continues to be limiting for and conservatively assumed in the containment response analysis.
- There are no steam line isolation valves at Oconee.
- Although the feedwater isolation valves receive a feedwater isolation signal, this is used only to provide a redundant means of accomplishing the feedwater isolation function. The steam line break mass and energy release analyses credit the faster closing feedwater control valves to provide the feedwater isolation function. Therefore the failure of a feedwater isolation valve has no effect on these analyses.

10. The initial structural metal temperature is assumed to be constant across the entire volume of the metal and in equilibrium with the adjacent primary or secondary coolant. This is conservative since there will be some temperature gradient across the metal with the hottest temperature at the coolant surface.
11. Some of the fuel rods are assumed to be at a higher than average temperature as described in Reference 1 on page 6-32. No credit is taken for offsetting lower temperatures in other fuel rods, thereby effectively increasing the fuel rod stored energy above expected average for the assumed power level and coolant temperature.
12. The turbine stop valves are assumed to close coincident with the opening of the break. This maximizes the rate of secondary side heatup and therefore the rate of secondary to primary heat transfer and energy release to containment.
13. No credit is taken for opening of the turbine bypass valves. This maximizes the amount of hot secondary side fluid remaining in the steam generators and able to transfer its energy via the primary side to containment.
14. Except for the offsite power lost sensitivity study case in the short-term LOCA analysis, the main feedwater flow boundary condition assumed allows up to the nominal hot full power flow to be delivered to the steam generators. This is conservative since it maximizes the amount of higher energy secondary side inventory available to transfer heat to the containment via the primary side.
15. The BWST liquid level is the lower Technical Specification limit less a level uncertainty of 20.2 inches. Minimizing the amount of BWST inventory minimizes the available sensible heat capacity of the BWST liquid and therefore maximizes the break steaming rate. This is because the BWST water is colder than the sump liquid inventory.
16. The BWST temperature is assumed to be a conservatively high value of 115°F, which minimizes available sensible heat capacity of the BWST liquid and therefore maximizes the break steaming rate.
17. Main feedwater temperature is at the nominal at power value, 453°F, plus a 7°F uncertainty allowance. This maximizes the potential energy release to containment. For further conservatism this assumed main feedwater temperature is maintained during the analysis although actual temperature would decrease as bleed steam was lost due to the break.
18. Main feedwater flow is maximized by controlling flow to the higher natural circulation setpoint even before the reactor coolant pumps are tripped. The nominal level setpoint is increased by a 10.5% operating range allowance for instrument uncertainty. This is conservative since it maximizes the amount of higher energy secondary side inventory available to transfer heat to the containment via the primary side.
19. Emergency feedwater temperature is at a conservatively high value of 120°F. This maximizes the potential energy release to containment. This is only relevant for the peak pressure large break analyses since the long-term large break analyses conservatively use hotter main feedwater.

#### 6.2.1.3.3 Description of Analytical Models

The mass and energy releases during the blowdown and core reflood periods of a postulated LOCA are calculated by the RELAP5/MOD2-B&W computer code (Reference 2 on page 6-32). The methodology for applying this code is given in Reference 1 on page 6-32. After the first 30 minutes of releases quasi-steady state conditions are reached. Beyond this point the BFLOW and FATHOMS (Reference 3 on page 6-32) codes are used to calculate mass and energy releases for the remainder of the accident as detailed in Reference 1 on page 6-32.

#### 6.2.1.3.4 Single Failure Analysis

The assumed single failure is the same as discussed above for the containment response analysis, the failure of a 4160 V switchgear, resulting in the loss of one HPI pump and one LPI pump.

#### 6.2.1.3.1 Mass and Energy Release Data

The short-term LOCA peak pressure mass and energy releases are given in Table 6-29. The long-term LOCA mass and energy releases calculated by RELAP5/MOD2-B&W for the first 1800 seconds are given in Table 6-30. The mass and energy releases used by the FATHOMS computer code for periods beyond 1800 seconds are given in Table 6-31.

#### 6.2.1.3.2 Energy Sources

The generated energy sources considered in the LOCA mass and energy release calculations are fission power, fission product and actinide decay energy, and metal-water reaction. The analyzed (actual) core power level is conservatively assumed to be 2% above the licensed (indicated) power. The assumed core axial power distribution is chosen to maximize the amount of steam exiting the break. The short-term LOCA mass and energy release calculation conservatively determines fission power by modeling moderator density, fuel, temperature, and boron feedbacks as described in Reference 1 on page 6-32. The long-term LOCA mass and energy release calculation uses the fission power given in Figure 6-49. Fission product and actinide decay power is calculated as a function of time based on the methodology from Reference 7 on page 6-32. For conservatism an upper bound of two standard deviations above the mean value is used. The modeling of metal-water reaction energy is discussed in Section 6.2.1.3.5, "Metal-Water Reaction" on page 6-20.

The stored energy sources considered in the LOCA mass and energy release calculations are fluid stored energy in the initial primary and secondary system inventories, stored energy in the primary and secondary structural metal components, stored energy in the fuel rods, and the energy content of the fluid added to the primary and secondary systems during the accident. The specific conservatisms used in modeling these stored energy sources are:

1. The nominal volume of the Reactor Coolant System calculated based on cold dimensions is increased by 1% to account for the increase in volume due to thermal expansion to operating temperatures.
2. Initial pressurizer liquid mass is increased by assuming the initial level is at the high-high level alarm setpoint of 315 inches.
3. The assumed reactor vessel average temperature is the nominal at power value, 579°F, plus a 2°F uncertainty allowance. This maximizes the stored energy in the Reactor Coolant System.
4. Reactor Coolant System pressure is at the nominal at power value, 2155 psig, plus a 30 psi uncertainty allowance. This maximizes the saturation temperature and therefore the energy content of the pressurizer fluid.
5. The value of steam generator total fluid mass used is the maximum which can be obtained with the RELAP5/MOD2-B&W computer code at the chosen operating conditions.
6. The core flood tank (CFT) temperature is assumed to be a conservatively high value of 120°F. This minimizes available sensible heat capacity of the CFT liquid and therefore maximizes the break steaming rate.
7. The CFT initial pressure used is the upper Technical Specification limit plus a 30 psi instrument uncertainty. This maximizes the amount of noncondensible gas released to containment and therefore the containment pressure.
8. The CFT liquid volume is the lower Technical Specification limit less an instrument uncertainty of 38 $ft^3$. This minimizes available sensible heat capacity of the CFT liquid and maximizes the amount of noncondensible gas released to containment, both of which, as explained above, tend to increase containment pressure and temperature.
9. The RCS flow rate assumed is 366,080 gpm. This conservatively low flow rate maximizes the temperature difference across the reactor vessel, lowering the cold leg temperature and raising the hot leg temperature. Since slightly more of the RCS volume is at the hot leg temperature, this increases the initial RCS liquid stored energy.

5 *6.2.1.1.3.4 Functional Capability of Normal Containment Ventilation Systems*

5 Normal containment ventilation is provided by four Reactor Building auxiliary cooling units (RBACUs)
5 and two of the three RBCUs. The function of these units during normal operation is described in Section
5 9.4.6, "Reactor Building Cooling System" on page 9-59. Upper and lower limits on containment
5 pressure during normal operation are maintained by complying with the Technical Specifications.

5 *6.2.1.1.3.5 Post-Accident Monitoring of Containment Conditions*

5 Post-accident monitoring instrumentation is provided for the following containment parameters:

5 Reactor Building pressure
5 Reactor Building air temperature
5 Reactor Building normal sump level
5 Reactor Building emergency sump level
5 Reactor Building wide range sump level

5 Section 7.5, "Display Instrumentation" on page 7-43 discusses the range, accuracy, and response of this
5 instrumentation and the tests conducted to qualify the instruments for use in the post-accident
5 containment environment.

5 **6.2.1.2 Containment Subcompartments**

5 The pressure response of the Reactor Building subcompartments following the design basis LOCA has
5 been evaluated using mass and energy release rates calculated by the CRAFT code (Reference 4 on
5 page 6-32) using the system model in Reference 5 on page 6-32, with the pressure response calculated by
5 the COPRA code (Reference 6 on page 6-32). The Reactor Building subcompartments include the
5 reactor compartment and the east and west steam generator compartments. For each compartment the
5 worst case LOCA break size and location is identified, including the effect of piping restraints on the
5 maximum break size. The flow through the subcompartment vents is calculated using a sonic choking
5 model for a homogeneous steam-water-air mixture, with a vent discharge coefficient of 0.6. A discharge
5 coefficient of 1.0 is used for the system blowdown calculation.

5 The reactor compartment has a volume of 5520 $ft^3$, one 6 $ft^2$ vent flowpath, and concrete shield plugs
5 with a total flow area of 69 $ft^2$. Only the vent flowpath is assumed to be available for pressure relief.
5 Although the maximum break area within the compartment has been determined to be 3.0 $ft^2$, hot leg
5 breaks of 8.0, 5.0, and 3.0 $ft^2$ were analyzed, as well as the maximum cold leg break of 8.55 $ft^2$. The
5 CRAFT mass and energy release rates are given in Figure 6-44 and Figure 6-45. The resulting pressure
5 differential across the compartment walls are shown in Figure 6-46. The peak pressure of 160 psi, which
5 occurs for the 8.0 $ft^2$ hot leg break, is only 78 percent of the design differential pressure of 205 psi.

5 The west steam generator compartment has a volume of 61,750 $ft^3$ and a total vent flow area of 1333 $ft^2$.
5 The east compartment has a volume of 60,400 $ft^3$ and a flow area of 1222 $ft^2$. The discharge coefficients
5 for each of the flowpaths and the effective discharge coefficient calculated to result in the correct choked
5 flow are given in Table 6-28 and Figure 6-47. The maximum hot leg break of 14.1 $ft^2$ was analyzed using
5 the CRAFT mass and energy release rates in Figure 6-44 and Figure 6-45. The resulting pressure
5 differentials across the compartment walls are shown in Figure 6-48. The structural integrity of the
5 compartments is sufficient to withstand 130 percent of the peak differential pressure of 15 psi.

5 **6.2.1.3 Mass and Energy Release Analyses for Postulated Loss-of-Coolant**
5 **Accidents**

5 *Analysis Method and Computer Codes*

5 The analysis method used in this section is described in Reference 1 on page 6-32 The computer codes
5 used in this section are RETRAN-02/MOD5 (Reference 9 on page 6-32) for calculating the steam line
5 break mass and energy releases and FATHOMS (Reference 3 on page 6-32) for calculating the
5 containment pressure and temperature response.

5 *Mass and Energy Release Rate Data*

5 The mass and energy release rate data used for the steam line break analyses described in this section are
5 given in Section 6.2.1.4, "Mass and Energy Release Analyses for Postulated Secondary System Pipe
5 Ruptures Inside Containment" on page 6-20.

5 *Initial Condition Assumption Conservatisms*

5 Initial condition assumptions in the containment response analyses are adjusted to give a conservative
5 answer:

5 1. The initial pressure assumption is adjusted by 0.3 psi above the proposed upper Technical
5 Specification limit for cases in which high initial pressure is conservative.
5 2. The initial temperature assumption is conservatively high for full power operation. It is known from
5 the LOCA analyses described in the previous section that a lower initial temperature maximizes the
5 containment peak pressures due to a higher initial air mass. However, a higher initial temperature
5 reduces the cooling capacity of the structural heat sinks, outweighing the impact of a higher initial air
5 mass.
5 3. The nominal containment free volume is reduced by 2%
5 4. A low initial relative humidity is used to maximize the initial air mass.

5 The initial conditions used are tabulated in Table 6-22.

5 *Containment Heat Removal Systems*

5 The Reactor Building cooling units (RBCUs) are modeled as described in Section 6.2.1.1.3.2, "LOCA
5 Long-Term Containment Temperature Response" on page 6-11 The steam line break peak pressure is
5 reached long before the borated water storage tank empties. Therefore the recirculation phase is not
5 simulated, and no credit is taken for heat removal by the LPI coolers during the injection phase. The
5 RBS is modeled as described in Section 6.2.1.1.3.2, "LOCA Long-Term Containment Temperature
5 Response" on page 6-11 for the injection phase.

5 *Single Failure*

5 The assumed single failure is the same as discussed above for LOCA, the failure of a 4160 V switchgear,
5 resulting in the loss of one HPI pump, one LPI pump, and one RBCU.

5 *Structural Heat Sinks*

5 The structural heat sinks within containment are those described in Section 6.2.1.1.3.1, "LOCA Short
5 Term Containment Pressure Response" on page 6-10 and tabulated in Table 6-23. For the steam line
5 break containment response calculation the surface areas of these heat structures are reduced by 1% for
5 conservatism.

5 Liquid injection from the core flood tanks is not explicitly modeled in FATHOMS but is considered in
5 the mass and energy releases discussed in Section 6.2.1.3, "Mass and Energy Release Analyses for
5 Postulated Loss-of-Coolant Accidents" on page 6-17. The nitrogen cover gas from these tanks is
5 assumed to be injected to increase the containment pressure calculated by FATHOMS. The amount of
5 injected nitrogen is based on the mass which would be present at the pressure and temperature initial
5 conditions of the mass and energy release calculation.

5 Assumed values for ECCS equipment performance parameters are given in Table 6-26.

5 ***Single Failure***

5 While a component single failure generally has little impact on the peak pressure analysis described in the
5 previous section, it has a much greater impact on the long-term containment response. The most limiting
5 single failure is therefore chosen to yield a conservative long-term containment response. The most
5 restrictive single failure is chosen as the one which disables the greatest number of containment heat
5 removal components.

5 An evaluation was performed to determine the most limiting single failure with respect to containment
5 cooling. This evaluation indicated that the failure of a 4160V switchgear represents the most limiting
5 single failure. Electrical switchgear power a myriad of safety related equipment including injection systems
5 and containment cooling systems. The failure of one of the three available switchgear will result in the
5 loss of the following components:

5 • One HPI pump
5 • One LPI pump
5 • One RBS pump
5 • One RBCU

5 All other ECCS equipment is available following a nominal, transient-specific actuation delay.

5 The switchgear failure is more limiting than a loss of offsite power (LOOP) and failure of one Keowee
5 hydroelectric unit because the second hydroelectric unit is available to power the standby busses through
5 CT-4 (underground) or through the switchyard (overhead). Therefore, all ECCS equipment would be
5 available after a small time delay.

5 ***Structural Heat Sinks***

5 The structural heat sinks within containment are those described in Section 6.2.1.1.3.1, "LOCA Short
5 Term Containment Pressure Response" on page 6-10 and tabulated in Table 6-23. For the LOCA
5 long-term containment response calculation the surface areas of these heat structures are reduced by 1%
5 for conservatism.

5 *6.2.1.1.3.3 Steam Line Break Containment Pressure and Temperature Response*

5 This section provides analyses of the pressure and temperature response of the containment to postulated
5 secondary system pipe ruptures. The results of the limiting case are given in Table 6-27. The pressure
5 and temperature response of this limiting case are given in Figure 6-42 (containment pressure) and
5 Figure 6-43 (containment temperature). The period of time during which the calculated temperature
5 exceeds the equipment qualification limit is very short compared to the time that the equipment is
5 exposed to high temperatures during its qualification testing. This short duration of calculated
5 temperatures above the equipment qualification limit is not long enough to cause the equipment internal
5 temperatures to reach values as high as those reached during the qualification testing.

used to enhance performance due to a higher ΔT across the RBCU. For example, one of the cases assumes two RBCUs at 28 million Btu/hour each and an LPSW temperature of 55°F.

Low Pressure Injection (LPI) cooler test data at a heat removal rate of 102 million Btu/hour is used to determine the relationships between cooler degradation (number of plugged tubes and amount of tube surface fouling) and thermal performance parameters such as fluid flow rates and temperatures. These relationships are then modeled with specified heat removal rates of 93, 97, and 102 million Btu/hour to determine an LPI cooler overall heat transfer coefficient as a function of LPI temperature. Since assumed cooler degradation, fluid flow rates, and LPSW temperature are constant during a simulation, and since LPI temperature changes during the accident, this change determines LPI cooler heat removal rate for a particular case. For example, the case referred to above also assumes a 93 million Btu/hr heat removal rate at the following conditions:

- LPI temperature of 250°F
- LPI flow rate of 3000 gpm
- LPSW temperature of 90°F
- LPSW flow rate of 5000 gpm

The actual heat removal rate calculated by FATHOMS for this case is reduced by accounting for a 2685 gpm LPI flow rate (minimum actual flow for 3000 gpm indicated flow rate), increased by accounting for an assumed colder LPSW temperature of 55°F, and then varied as calculated LPI temperature varies. The preceding discussion applies to the recirculation phase. No credit is taken for heat removal by the LPI coolers during the injection phase.

The single operating (refer to single failure discussion below) Reactor Building Spray (RBS) pump is conservatively modeled with respect to flow rate and temperature. The injection phase flow rate used is the nominal flow rate of 1500 gpm per pump less an allowance of 143 gpm for flow indication uncertainty. The recirculation phase flow rate used is the nominal flow rate of 1000 gpm per pump with BS-15 and BS-20 closed, less an allowance of 72 gpm for operating in the recirculation phase with these valves open, less an additional allowance of 128 gpm for flow indication uncertainty. Some cases were also run with an additional 200 gpm reduction in recirculation phase RBS flow rate. The injection phase temperature used is a conservatively high for the borated water storage tank, the source of RBS water during the injection phase. The recirculation phase RBS temperature is the sump temperature calculated by FATHOMS. No credit is taken for aligning the RBS pumps to take suction from the outlet of the LPI coolers.

Assumed values for containment heat removal equipment performance parameters are given in Table 6-26.

*Emergency Core Cooling Systems*

The single operating Low Pressure Injection (LPI) pump (refer to single failure discussion below) is assumed in the FATHOMS computer code to be supplying a conservatively low flow rate to the reactor vessel. The flow rate assumed is the nominal value less an allowance for flow indication uncertainty. The injection phase temperature used is a conservatively high for the borated water storage tank, the source of LPI water during the injection phase. The recirculation phase temperature is calculated by FATHOMS based on the heat removal from the LPI coolers.

The two operating High Pressure Injection (HPI) pumps (refer to single failure discussion below) are assumed to be supplying a conservatively low flow rate to the cold legs. The HPI water injected into the broken cold leg is added directly to the containment sump. The injection phase temperature used is a conservatively high for the borated water storage tank, the source of HPI water during the injection phase. No credit is taken for HPI flow during the recirculation phase.

5 The cases plotted present the four combinations of minimum and maximum LPSW temperature and
5 RBCU heat removal rate for each assumed Reactor Building Spray flow rate. All results plotted are for
5 the minimum tabulated LPI cooler heat removal rate, 93 million Btu/hour. The equipment qualification
5 criteria plotted on these figures are potentially challenged only at two points, the 100,000 second transition
5 from 240°F to 200°F and the 20.683 day transition from 200°F to 125°F. All the cases tabulated meet
5 these two criteria. All the previous criteria are met regardless of the performance of the active
5 containment heat removal equipment. This is because the amount of mass and energy releases generated
5 by a double-ended guillotine pump discharge break in a 2568 MWt B&W designed NSSS will not exceed
5 the equipment qualification criteria curve as long as the analyzed amount of passive heat structures remain
5 available for energy absorption.

5 ***Analysis Method and Computer Codes***

5 The analysis method used in this section is described in Reference 1 on page 6-32. The computer codes
5 used in this section are RELAP5/MOD2-B&W (Reference 2 on page 6-32) for calculating the mass and
5 energy releases during the first 30 minutes, BFLOW for calculating the longer term LOCA mass and
5 energy releases, and FATHOMS (Reference 3 on page 6-32) for calculating the containment pressure and
5 temperature response.

5 ***Mass and Energy Release Rate Data***

5 The mass and energy release rate data used for the LOCA analyses described in this section are given in
5 Section 6.2.1.3, "Mass and Energy Release Analyses for Postulated Loss-of-Coolant Accidents" on
5 page 6-17.

5 ***Initial Condition Assumption Conservatisms***

5 Initial condition assumptions in the containment response analyses are adjusted to give a conservative
5 answer:

5 1. A nominal initial pressure is used, although this parameter has very little effect due to the long
5 duration of this analysis.
5 2. The initial temperature assumption is conservatively high for full power operation.
5 3. The nominal containment free volume is reduced by 2%
5 4. A high initial relative humidity is used, although this parameter has very little effect due to the long
5 duration of this analysis.

5 The initial conditions used are tabulated in Table 6-22

5 ***Containment Heat Removal Systems***

5 The Reactor Building cooling units (RBCUs) are modeled based on performance relative to a reference
5 value of 80 million Btu/hour. This reference performance is based on the heat removal rate associated
5 with:

5 • Low Pressure Service Water (LPSW) temperature of 90°F
5 • Containment air temperature of 286°F
5 • Containment air mass characteristic of 110°F and 100% relative humidity

5 The performance of the two operating coolers (refer to single failure discussion below) is parameterized in
5 terms of the percentage of this reference heat removal rate and the LPSW temperature. A reduction
5 below 80 million Btu/hour reflects degradation in RBCU performance. A colder LPSW temperature is

5 analysis considers only a single break size and location: a double-ended guillotine break located at the A1
5 cold leg pump discharge. There is no need to analyze a spectrum of large break locations since a suitably
5 bounding site can be chosen by inspection. The qualitative bases for this position is explained in the
5 following paragraphs. Reference 1 on page 6-32 also extensively analyzed the mass and energy releases
5 from and containment response to small break LOCAs. The conclusion of these analyses is that small
5 break LOCAs are not more limiting than large break LOCAs with respect to challenging the containment
5 equipment qualification acceptance criteria.

5 The basis for choosing a cold leg break as opposed to a hot leg break is obvious once the characteristics
5 of each break are considered. Although it is true that an identical quantity of decay heat will be generated
5 regardless of the break location, the manner in which this energy is partitioned between the vapor and
5 liquid break flow streams is the dominant consideration.

5 Because the long-term containment response is concerned with temperature in containment as a function
5 of time, it is expected that an energy release profile which is dominated by steam relief will generate a
5 more severe containment response. This is because steam relief to the atmosphere will have a greater
5 impact on containment temperature than if the energy is released primarily in the liquid phase, which has
5 only a slight interaction with the containment atmosphere (convection at the pool surface). Indeed, this
5 observation has been validated with the FATHOMS computer code in numerous analyses. It might
5 appear that the Reactor Building Spray System acts to homogenize the containment atmosphere such that
5 the phase in which the energy is released is insignificant. However, when the complicated interactions
5 between the equipment used to cool the containment atmosphere (Reactor Building coolers and sprays)
5 and the equipment used to cool the containment sump (LPI coolers) are examined by analysis, it is
5 apparent that containment will never become completely homogenized. Therefore, the partitioning of
5 energy released to containment between the vapor and liquid phases is the dominant factor in the
5 long-term containment response.

5 Due to the geometry of a B&W reactor system, steaming from a cold leg break location will never become
5 completely suppressed. This means that steam will always exit the break no matter how much the decay
5 heat power drops. In contrast, it is possible to completely suppress steaming from a hot leg break site as
5 decay power decreases. Decay heat will eventually be absorbed as sensible heat by the injection fluid and
5 thus steaming from the break will cease. Naturally, when this occurs, decay heat will be transferred to
5 containment in the liquid phase, resulting in a less severe containment response.

5 The cold leg pump discharge break location is selected rather than the pump suction location. For a
5 pump suction break the cold HPI fluid injected into the broken cold leg pump discharge piping will
5 interact with steam exiting the core through the vent valves and condense a large portion of this steam
5 before it reaches the break. Thus, the steam release will be less for a pump suction break. Consequently,
5 the pump discharge break location is limiting.

5 An accident chronology is presented in Table 6-24 for the most limiting LOCA, an 8.55 $ft^2$ double-ended
5 guillotine cold leg break at the reactor coolant pump outlet. Table 6-25 presents results for various
5 combinations of LPI cooler heat removal rate, LPSW temperature, RBCU heat removal rate, and
5 recirculation phase Reactor Building Spray flow rate. The results of four of these cases are presented in
5 the following figures:

5 Figure 6-36 Containment pressure (600 gpm RBS flow rate)
5 Figure 6-37 Containment atmosphere temperature (600 gpm RBS flow rate)
5 Figure 6-38 Containment sump water temperature (600 gpm RBS flow rate)
5 Figure 6-39 Containment pressure (800 gpm RBS flow rate)
5 Figure 6-40 Containment atmosphere temperature (800 gpm RBS flow rate)
5 Figure 6-41 Containment sump water temperature (800 gpm RBS flow rate)

energy releases and FATHOMS (Reference 3 on page 6-32) for calculating the containment pressure and temperature response.

***Mass and Energy Release Rate Data***

The mass and energy release rate data used for the LOCA analyses described in this section are given in Section 6.2.1.3, "Mass and Energy Release Analyses for Postulated Loss-of-Coolant Accidents" on page 6-17.

***Initial Condition Assumption Conservatisms***

Initial condition assumptions in the LOCA containment peak pressure response analyses are adjusted to give a conservative answer:

1. The initial pressure assumption is adjusted by 0.3 psi above the proposed upper Technical Specification limit.
2. The initial temperature assumption is conservatively low for full power operation. This maximizes the initial containment air mass, which maximizes the air partial pressure contribution to the pressure peak.
3. The nominal containment free volume is reduced by 2%
4. A low initial relative humidity is used to maximize the initial air mass.

The initial conditions used are tabulated in Table 6-22.

***Containment Heat Removal Systems***

No credit is taken in the LOCA peak pressure analysis for either the Reactor Building cooling units or the Reactor Building Spray System. The peak pressure occurs within the first 20 seconds after the postulated break, prior to the assumed actuation of either of these heat removal systems.

***Emergency Core Cooling Systems***

The emergency core cooling systems are not explicitly modeled in FATHOMS for the LOCA peak pressure analysis, but are considered in the mass and energy releases discussed in Section 6.2.1.3, "Mass and Energy Release Analyses for Postulated Loss-of-Coolant Accidents" on page 6-17.

***Single Failure***

A component single failure generally has little impact on the peak pressure analysis. This is because peak pressures usually occur before the engineered safeguards equipment has time to activate and become effective.

***Structural Heat Sinks***

The structural heat sinks within containment are divided into nine groups for the purposes of containment pressure and temperature response modeling. These nine structures are tabulated in Table 6-23. The concrete and steel portions of the building cylinder, the building dome, and the building base are combined in three structures of two materials each.

*6.2.1.1.3.2 LOCA Long-Term Containment Temperature Response*

This section provides analyses of the long-term (~1 day) pressure response of the containment to a spectrum of postulated Reactor Coolant System pipe ruptures. The long-term large break containment

5 **6.2.1.1.2 Design Features**

5 Since the design of the Engineered Safeguards Systems and their operation is discussed more fully in
5 Section 6.3, "Emergency Core Cooling System" on page 6-33, only their relation to the basis of Reactor Building design is discussed below. The Engineered Safeguards Systems are provided to limit the consequences of an accident. Their energy removal capabilities limit the internal pressure after the initial peak so that Reactor Building design limits are not exceeded and the potential for release of fission products is minimized.

5 Following a LOCA, the Emergency Core Cooling Systems inject borated water into the Reactor Coolant System to remove core decay heat and to minimize metal-water reactions and the associated release of heat and fission products. Flashed primary coolant, Reactor Coolant System sensible heat, and core decay heat transferred to Reactor Building are removed by two engineered safeguards systems: the Reactor Building Spray and/or the Reactor Building Cooling Systems.

5 Following a secondary line break at power, main feedwater and turbine- driven emergency feedwater flow
5 to the faulted steam generator are isolated by the Feedwater Isolation System. Motor-driven emergency
5 feedwater flow to the faulted steam generator is isolated manually by the operator.

The Reactor Building Spray System removes heat directly from the Reactor Building atmosphere by cold water quenching of the Reactor Building steam.

The air recirculation and cooling systems remove heat directly from the Reactor Building atmosphere to the Service Water System with recirculating fans and cooling coils.

5 The low pressure injection coolers remove heat from the containment sump liquid to the Service Water
5 System with heat exchange through tubes.

Section 3.8, "Design of Class 1 Structures" on page 3-75 provides a detailed description of the Reactor Building design.

5 **6.2.1.1.3 Design Evaluation**

5 *6.2.1.1.3.1 LOCA Short Term Containment Pressure Response*

5 This section provides analyses of the short-term (3 minutes) pressure response of the containment to a
5 spectrum of postulated Reactor Coolant System pipe ruptures. The break size and location of each
5 postulated loss-of-coolant accident is given in Table 6-21. The pressure and temperature response of the
5 four break location sensitivity studies, Cases 1A through 1D, are given in the following figures:

5 Figure 6-28 Containment pressure for a 14.1 $ft^2$ break at the reactor vessel outlet (1A)
5 Figure 6-29 Containment temperature for a 14.1 $ft^2$ break at the reactor vessel outlet (1A)
5 Figure 6-30 Containment pressure for a 14.1 $ft^2$ break at the steam generator inlet (1B)
5 Figure 6-31 Containment temperature for a 14.1 $ft^2$ break at the steam generator inlet (1B)
5 Figure 6-32 Containment pressure for a 8.55 $ft^2$ break at the RCP discharge (1C)
5 Figure 6-33 Containment temperature for a 8.55 $ft^2$ break at the RCP discharge (1C)
5 Figure 6-34 Containment pressure for a 8.55 $ft^2$ break at the RCP suction (1D)
5 Figure 6-35 Containment temperature for a 8.55 $ft^2$ break at the RCP suction (1D)

5 ***Analysis Method and Computer Codes***

5 The analysis method used in this section is described in Reference 1 on page 6-32. The computer codes
5 used in this section are RELAP5/MOD2-B&W (Reference 2 on page 6-32) for calculating the mass and

# 6.2 CONTAINMENT SYSTEMS

## 6.2.1 CONTAINMENT FUNCTIONAL DESIGN

5 ### 6.2.1.1 Containment Structure

5 #### 6.2.1.1.1 Design Bases

The Reactor Building completely encloses the Reactor Coolant System to minimize release of radioactive material to the environment should a serious failure of the Reactor Coolant System occur. The structure provides adequate biological shielding for both normal operation and accident situations. The Reactor Building is designed for an internal pressure of 59 psig. The leakage rate will not exceed 0.25 percent by volume in 24 hours under the conditions of the maximum hypothetical accident as described below.

5 The Reactor Building is designed for an external pressure 3.0 psi greater than the internal pressure. The
5 design external pressure of 3.0 psi corresponds to a margin of 0.5 psi above the differential pressure that
5 could be developed if the building is sealed with an internal temperature of 120°F with a barometric
5 pressure of 29.0 inches of Hg and the building is subsequently cooled to an internal temperature of 80°F
5 with a concurrent rise in barometric pressure to 31.0 inches of Hg. The weather conditions assumed here
5 are conservative since an evaluation of National Weather Service records for this area indicates that from
5 1918 to 1970 the lowest barometric pressure recorded is 29.05 inches of Hg and the highest of 30.85 inches
5 of Hg.

The principal design basis for the structure is that it be capable of withstanding the internal pressure
5 resulting from a loss-of-coolant accident or a secondary line rupture with no loss of integrity. In a
5 LOCA, the total energy contained in the water of the Reactor Coolant System is assumed to be released
5 into the Reactor Building through a break in the reactor coolant piping. In a secondary line break event
5 the energy contained in the water in the secondary coolant system, as well as energy transferred across the
5 steam generator tubes from the Reactor Coolant System is assumed to be released into the Reactor
5 Building through a break in the steam line piping. However, in the case of a secondary line break, the
5 release of energy essentially stops when the faulted steam generator empties and is no longer being
5 supplied with feedwater. In either case, subsequent pressure behavior is determined by the building volume, engineered safeguards, and the combined influence of energy source and heat sinks.

Energy is available for release into the containment structure from the following sources:

| | LOCA | Secondary Line Break |
|---|---|---|
| 5 | Reactor Coolant Stored Energy | Secondary Coolant Stored Energy |
| 5 | Reactor Stored Energy | Secondary System Stored Energy |
| 5 | Reactor Decay Heat | Reactor Coolant Stored Energy |
| 5 | Metal-Water Reactions | Reactor Stored Energy |
| 5 | Secondary Coolant Stored Energy | Reactor Decay Heat |
| 5 | Secondary System Stored Energy | |

## 2 6.1.5 PIPING DESIGN CONDITIONS

Piping Design Conditions for the Engineered Safeguards Systems are listed on Table 6-4.

## 2 6.1.6 ENGINEERED SAFEGUARDS MATERIALS

Materials used in Engineered Safeguards components are addressed in applicable sections where appropriate.

3) Bonnet flange leakage.

b. Flanges

c. Pump shaft seals

While leakage rates have been assumed for these sources, maintenance and periodic testing of these systems will preclude all but a small percentage of the assumed amounts. With the exception of the boundary valve discs, all of the potential leakage paths may be examined during periodic tests or normal
3 operation. Boundary valves which have been identified to have leakage paths are tested periodically. All
3 other valve disc leakage is retained in the other closed systems and, therefore, will not be released to the Auxiliary Building.

While valve stem leakage has been assumed for all valves, the manual valves in the recirculation complex are backseating and do not rely on packing alone to prevent stem leakage.

*Leakage Assumptions*

| Source | | Quantities |
|---|---|---|
| a. | Valves - Process | |
| 1. | Disc Leakage | 10 cc/hr./in. of nominal disc diameter |
| 2. | Stem leakage | 1 drop/min. |
| 3. | Bonnet flange | 10 drops/min. |
| b. | Valves - Instrumentation | |
| | Bonnet flange and stem | 1 drop/min. |
| c. | Flanges | 10 drops/min. |
| d. | Pump seals | 50 drops/min. |

For the analysis, it was assumed that the water leaving the Reactor Building was at 252 °F. This assumption is conservative as this peak temperature would only exist for a short period during the post-accident condition. Water downstream of the coolers was assumed to be 115 °F. The Auxiliary Building was assumed to be at 70 °F and 30 percent relative humidity. Under these conditions, approximately 22 percent of the leakage upstream of the coolers and 4 percent of the leakage downstream of the coolers would flash into vapor. For the analysis, however, it was assumed that 50 percent of the leakage upstream of the coolers would become vapor because of additional heat transfer from the hot metal.

*Design Basis Leakage:* The design basis leakage quantities are tabulated in Table 6-2.

*Leakage Analysis Conclusions:* It was concluded from this analysis (in conjunction with the discussion and analysis in Section 15.15.4, "Effects of Engineered Safeguards Systems Leakage" on page 15-70) that leakage from Engineered Safeguards Systems outside the Reactor Building does not pose a public safety problem.

## 2 6.1.4 QUALITY CONTROL STANDARDS

Quality Control Standards for the Engineered Safeguards Systems are listed in Table 6-3.

The NPSH available to the Low Pressure Injection and Reactor Building Spray pumps during the post-LOCA recirculation phase has been calculated based on:

a. "As Built" piping drawings.

b. Pipe and fitting losses calculated using the information in Crane Technical Paper No. 410.

c. Total indicated flow in a single string (i.e., consisting of one Low Pressure Injection Pump and one Reactor Building Spray pump served by a single sump suction line) is 4,000 gal/min. This consists of 3,000 gal/min to the Low Pressure Injection pump and 1,000 gal/min to the Reactor Building Spray pump. Instrument uncertainties have been applied to these values to provide conservatism in the NPSH analysis.

d. Sump water temperatures and Reactor Building pressures were determined from analysis of hot leg break with conservative building cooling assumptions (LPSW temperature, RBCU capacity, etc.).

e. Reactor Building Spray pump shaft center line are elevation 760 ft. 1 in.

f. Low Pressure Injection pump shaft center line at elevation 761 ft. 1 in.

g. Water level in the Reactor Building sump is 782 ft. 6 in. based on the following assumptions: (height above Reactor Building basement level is 5.0 ft.)

   1) The Technical Specification minimum levels were used for the BWST and the CFT's, with six feet of level remaining in the BWST at time of switchover.

   2) Some water is maintained in the Reactor Building atmosphere as vapor. The quantity was determined using the results of a CONTEMPT Computer Run for a 5.0 $ft^2$ break with 2 fan coolers and one Reactor Building Spray pump operating.

   3) The break is conservatively assumed to occur at the top of the hot leg, thereby keeping the Reactor Coolant System full.

h. Unit 3 NPSH available to each pump in the "B" string, LP-P3B and BS-P3B. Calculations show this to be the worst case flowpath of all possible units and trains.

The required NPSH's indicated above reflect the manufacturer's certified test.

Available NPSH has been determined to meet or exceed the required NPSH for worst case accident conditions with conservative inputs as identified above. Curves of total dynamic head and NPSH versus flow are shown in Figure 6-5 for the Reactor Building Spray Pumps and in Figure 6-17 for the Low Pressure Injection Pumps. These curves are representative in nature and are provided for information only. They are not intended to constitute design commitments or performance requirements for the pumps. Refer to the Inservice Test Program for actual performance requirements for BS and LPI pumps.

*Bases of Leakage Estimates:* While the reactor auxiliary systems involved in the recirculation complex are closed to the Auxiliary Building atmosphere, leakage is possible through component flanges, seals, instrumentation, and valves.

The leakage sources considered are:

a. Valves

   1) Disc leakage when valve is on recirculation system boundary.

   2) Stem leakage.

The Reactor Building Penetration Room Ventilation System shown on Figure 6-4 collects and filters air leakage to control and minimize the release of radioactive material from Reactor Building penetrations following an accident. Two full capacity filtering paths are provided.

## 2 6.1.2 EQUIPMENT OPERABILITY

Operability of engineered safeguards equipment is assured in several ways. Much of the equipment in these systems serves a function during normal reactor operation. In those cases where equipment is used for emergency functions only such as the Reactor Building Spray System, systems have been designed to permit meaningful periodic tests. Operational reliability is achieved by using proven component design, 5 and by conducting tests where either the component or its application was considered unique. In-house quality control procedures are imposed on the components of the Engineered Safeguards Systems. These procedures include use of accepted codes and standards as well as supplementary test and inspection requirements to assure that all components will perform their intended function under the design conditions following a loss-of-coolant accident.

The purpose of this section is to describe the physical arrangement, design, and operation of the Engineered Safeguards Systems as related to their safety function.

Reactor Building isolation is described in Section 6.2, "Containment Systems" on page 6-9. Other sections of the report contain information which is pertinent to the Engineered Safeguards Systems. Chapter 7, "Instrumentation and Control" on page 7-1 describes the actuation instrumentation of these systems. Chapter 15, "Accident Analyses" on page 15-1 describes the analysis of the Engineered Safeguards Systems' capability to provide adequate protection during accident conditions. Chapter 9, "Auxiliary Systems" on page 9-1 discusses functions performed by these systems during normal operation and gives further design details and descriptive information concerning those systems.

## 2 6.1.3 LEAKAGE AND RADIATION CONSIDERATIONS

The use of normally operating equipment for engineered safeguards functions and location of some of this equipment outside the Reactor Building require that consideration be given to direct radiation levels after fission products have accumulated in these systems and leakage from these systems.

The shielding for components of the engineered safeguards is designed to meet the following objectives in the event of a maximum hypothetical accident:

a. To provide protection for personnel to perform all operations necessary for mitigation of the accident.

b. To provide sufficient accessibility in all areas around the station to permit safe continued operation of the unaffected nuclear units.

*Summary of Post-Accident Recirculation:* Following a loss-of-coolant accident, flow is initiated in the Low Pressure Injection System from the borated water storage tank to the reactor vessel. Flow is also initiated by the Reactor Building Spray Systems to building spray headers. When the borated water storage tank level reaches the low level alarm, recirculation from the Reactor Building emergency sump is initiated by the operator for both the reactor core cooling flow and the Reactor Building sprays. The operator will maintain the 3,000 gal/min design flow rates of the Low Pressure Injection pumps, but will throttle the Reactor Building Spray pumps from the 1,500 gal/min design flow rate to 1,000 gal/min in order to ensure adequate NPSH. The post-accident recirculation system includes all piping and equipment both internal and external to the Reactor Building as shown on Figure 6-1, up to the stop and test line valves leading to the borated water storage tank.

2 # 6.1 ENGINEERED SAFEGUARDS

Engineered safeguards are those systems and components designed to function under accident conditions to prevent or minimize the severity of an accident or to mitigate the consequences of an accident. During accident conditions when reactor coolant is lost, the engineered safeguards act to provide emergency cooling to assure structural integrity of the core, to maintain the integrity of the Reactor Building, and to collect and filter Potential Reactor Building penetration leakage. Separate and independent engineered safeguards are provided for each of the three reactor units at Oconee. Special precautions are taken to assure high quality in the system design and components.

The engineered safeguards include provisions for:

a. High pressure injection.

b. Low pressure injection.

c. Core flooding.

d. Two types of Reactor Building cooling.

e. The collection and control of Reactor Building penetration leakage.

f. Reactor Building isolation.

Figure 6-1 and Figure 6-4 depict the portion of the Engineered Safeguards System related to core and building protection (see (a) through (d) above). A general description of the engineered safeguards provisions is presented below and a more detailed description is presented in the latter portion of this section. Since each reactor unit has the same arrangement of Emergency Safeguard Systems, the performance of the systems is described on a unit basis.

2 ## 6.1.1 GENERAL SYSTEMS DESCRIPTION

The High and Low Pressure Injection Systems and the Core Flooding Tanks are designed to form collectively an overall Emergency Core Cooling System (ECCS), which is designed to prevent melting or physical disarrangement of the core over the entire spectrum of Reactor Coolant System break sizes. Figure 6-1 shows the Emergency Core Cooling Systems for one reactor unit. The High Pressure Injection System is arranged so that three pumps are available for emergency use. The Low Pressure Injection System is arranged to assure that two pumps are normally available and a third pump is installed but normally valved off. The Core Flooding System for each unit is composed of two separate pressurized tanks containing borated water at Reactor Building ambient temperature. These tanks automatically discharge their contents into the reactor vessel at a preset Reactor Coolant System pressure without reliance on any actuating signal, any motive power or any external actuated component.

Reactor Building integrity is assured by two full-capacity, independent, pressure reducing systems operating on different principles; the Reactor Building Spray System and the Reactor Building Emergency Cooling System. (Refer to Figure 6-2 and Figure 6-3). These systems have the redundancy required to
2 meet the single failure criterion. These systems operate to lower Reactor Building pressure over the spectrum of Reactor Coolant System break sizes and to reduce the driving force for leakage of radioactive materials from the Reactor Building.

# CHAPTER 6. ENGINEERED SAFEGUARDS

# LIST OF FIGURES

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## 5.4.9 REFERENCES

1. BAW-10040, Reactor Coolant Pump Assembly Overspeed Analysis
2. Letter from H. B. Tucker (Duke) to H. R. Denton (NRC) dated October 1, 1985. Subject: Performance Testing of Relief and Safety Valves.
3. Babcock & Wilcox Owners Group Safe End Task Force Report on Generic HPI/MU Nozzle Component Cracking. B&W Document #77-1140611-00, Submitted to NRC on February 15, 1983.
4. B&W - 1543, Rev. 2, Addendum 1, July 1987. "Integrated Reactor Vessel Material Surveillance Program (Addendum)."

**THIS IS THE LAST PAGE OF THE CHAPTER 5 TEXT PORTION.**

coolant pressure and A the flow-sectional area of the pipe. The thrust was applied as an equivalent static force using a dynamic load factor of 2.0. Assuming the force to be a point load acting at the midpoint of the span between supports, the piping stresses were calculated using beam models. The supports are located so as to prevent the formation of plastic hinges in the piping, which would lead to an unstable linkage-type structure and possible impacting against the generator.

To evaluate the effect of fluid jet impingement on the generator, an equivalent static pressure load on the shell was calculated. A break of 14 $ft^2$ for the hot leg or 8.5 $ft^2$ for the cold leg was assumed. The maximum initial mass velocity was computed using the methods outlined in the report "Maximum Two-Phase Vessel Blowdown From Pipes, APED-4827," by F. J. Moody. It was assumed that the fluid leaves the break in a direction normal to the pipe and that its velocity undergoes a 90° change in direction upon impinging on the steam generator. The resulting shell pressure loading was calculated to be 1300 psi.

A shell analysis was performed on the steam generator to determine the stress intensity due to the above loading. A B&W proprietary digital computer code, which considers two-dimensional shells with asymmetric loading, was utilized. The loading distribution and stress model are shown in Figure 5-31 and Figure 5-32.

The maximum stress intensity was computed to be 38,600 psi. This is less than the allowable stress of 46,670 psi. Based on these results for the 36- inch i.d. pipe break, it was concluded that the steam generator shell could also withstand the reduced loading which would be generated by a 28-inch i.d. break.

reactor coolant piping also filled with water, and steam lines under the MHE. The foundation is also designed to restrain the steam generator under the combined forces due to a circumferential rupture of a 28-inch coolant line and a simultaneous MHE.

Forces imposed on the generator by the rupture of a 36-inch coolant line are transferred to the shielding walls by a support structure located near the top of the generator.

### 5.4.8.4 Piping

The reactor coolant piping, inlet and outlet lines, are supported by the reactor vessel and steam generator nozzles. The piping will withstand the forces imposed on it by the MHE.

### 5.4.8.5 Pump and Motor

The reactor coolant pump casing, internals, and motor weight are supported by the 28 inch coolant lines and constant load hangers attached to the motor. In the cold condition, the coolant piping will support the coolant pump and motor without the hangers. The hangers are designed to withstand the forces imposed on them by the MHE.

### 5.4.8.6 LOCA Restraints

Each steam generator has a support located opposite the upper tube sheet and transfers forces from the generator into the shield walls in the event of a circumferential rupture of the 36-inch line.

Each 28-inch reactor coolant inlet line and 36-inch reactor coolant outlet line has a restraint located outside of and bolted to the primary shield to limit pipe motion in the event of a circumferential rupture of the piping inside the primary shield.

A detailed study of the primary loop was performed to determine potential pipe break locations which could possibly cause either fluid impingement or pipe impact forces on the Secondary System. The results of this evaluation indicated the most credible break locations which could cause either of these effects are:

1. A guillotine break at the pump discharge in the cold leg piping;
2. A longitudinal split in the vertical pump suction segment of the cold leg piping; or,
3. A longitudinal split in the vertical segment of the hot leg piping.

All of the above breaks could potentially affect the generator because of their proximity to it. The main steam lines, however, are shielded from the effects of pipe breaks by the generator.

The primary piping and steam generator were analyzed for each of the above breaks and supports provided to restrain the pipe from whipping into the generator. In addition, the stresses in the generator shell due to the fluid impingement forces were calculated and found to be within acceptable limits.

The restraints on the primary loop are shown in Figure 5-29 and Figure 5-30. The coolant pump is restrained by steel supports from the primary shield wall. The hot leg piping is restrained by the concrete support at the primary cavity penetration, an intermediate steel support from the primary wall, and another steel support near the generator upper tube sheet. The vertical segment of the cold leg piping is restrained by a steel support midway along its length, which would spread any rupture load over a larger area of the generator shell.

To verify the location and size of the piping supports, the piping was analyzed for rupture loads occurring at the worst point along its length. The rupture thrust force was assumed equal to P x A, where P is the

One independent remotely operated vent is provided at the high point of each 36-inch RCS hot leg line. Each vent makes use of the existing manual vent line. A tee has been added after the first manual valve and a new manual valve has been added after the tee. The first manual valve (1RC-19, 1RC-38, 2RC-19,
4 2RC-196, 2RC-38, 3RC-19, 3RC-38) is in the open position. The function of the first valve has been transferred to the second valve (1RC-168, 1RC-169, 2RC-168, 2RC-169, 3RC-168, 3RC-169). The new vent runs from the tee through two solenoid valves and discharges into the air stream from the Reactor Building Coolant Units. The solenoid valves are remotely controlled from the Control Room. The function of the manual vent is unaffected.

The reactor coolant vent system is acceptable to the NRC and in conformance with the requirements of 10CFR 50.44 paragraph (c)(3)(iii) and the guidelines of NUREG 0737 Item II.B.1, and NUREG-0800 Section 5.4.12.

## 5.4.8 COMPONENT FOUNDATIONS AND SUPPORTS

The supports for all major components listed in this section are analyzed in detail to insure adequate structural integrity for their intended function during normal operating, seismic, and accident conditions. Following calculation of sources of loading, stresses and motions at significant locations are computed and compared to applicable criteria. Details of this analysis are given in Chapter 3, "Design of Structures, Components, Equipment, and Systems" on page 3-1.

### 5.4.8.1 Reactor Vessel

The reactor vessel is bolted to a reinforced concrete foundation designed to support and position the vessel and to withstand the forces imposed on it by a combination of loads including the weight of vessel and internals, thermal expansion of the piping, design basis earthquake (DBE), and dynamic load following reactor trip.

The foundation, in addition, restrains the vessel during the combined forces imposed by the circumferential rupture of a 36-inch reactor outlet line and a simultaneous maximum hypothetical earthquake (MHE).

The vessel foundation further is designed to provide accessibility for the installation and later inspection of incore instrumentation, piping, and nozzles; to contain ductwork and vent space for cooling air to remove heat losses from the vessel insulation; and to provide a sump and drainage line for leak detection.

### 5.4.8.2 Pressurizer

The pressurizer is supported on a structural steel foundation by eight lugs welded to the side of the vessel.

The foundation and supports are designed to withstand the loads imposed by thermal expansion of the pressurizer, the weight of the pressurizer including its contents and attached piping, relief valve reaction forces, and forces imposed by the design basis earthquake. In addition, the foundation and supports will restrain the vessel during the combined forces imposed by the circumferential rupture of the 10-inch surge line coupled with the MHE.

The foundation is also designed to permit accessibility to pressurizer surfaces for inspection.

### 5.4.8.3 Steam Generator

The steam generator foundation is designed to support and position the generator. The foundation is designed to accept the loads imposed by the generators and feedwater piping filled with water, the attached

## 7.4.4 REFERENCES

1. *Evaluation of Transient Nuclear Instrumentation Power Range Flux Error* - Duke Power Company - March 1981.
2. Qualification Testing of Protective System Instrumentation Babcock and Wilcox - *BAW - 10003 Rev. 3 - April, 1974 and BAW - 10003A Rev. 4 - January, 1976.*
3. *Evaluation of Reactor Protective System Grounding Concern* Babcock and Wilcox - March, 1978.
4. *177 FA Plants NI/RPS Ground Problem Discussion and Recommended Test Scheme* Babcock and Wilcox - March, 1978.

"Emergency Feedwater Controls" on page 7-37). Each OTSG has two independent level control systems each of which is capable of supplying a signal to the OTSG emergency feedwater level control valve. All automatic initiation logic and control functions are independent from the Integrated Control System (ICS).

#### 7.4.3.2.2 System Design

Each OTSG is provided with two independent level control systems, each of which consists of a level
2 transmitter, controller, power supply, signal isolator and E/P converter to supply a signal to that OTSG's emergency feedwater level control valve. Two transmitters are provided on each OTSG and monitor the
2 0-388 inch range of water in the OTSG at cold shutdown. A computer alarm from the signal isolator is provided to perform a signal deviation check and will alarm if greater than 15 percent deviation is present. The operator has a selector switch on the main control board with which he can select either control
2 channel on each OTSG. Also provided on the main control board is a pushbutton (automatic/manual)
2 and manual loader which may be utilized to override the automatic level control signal provided the control switch which governs this transfer is first engaged.

#### 7.4.3.2.3 System Evaluation

Each level channel is separate and independently powered from its counterpart on each OTSG. Redundancy is provided with two trains/channels monitoring each steam generator. Each level channel per steam generator is capable of performing the necessary control and modulation of the feedwater control valves. In addition, sufficient alarms and indications are provided to alert the operator to a system failure and ensure correct manual operation of a level control valve.

5 discharge pressure is determined by pressure switch actuation. Each pump is also provided with a control switch with which the operator may start the pump manually.

Indication is provided in the control room to allow the operator to monitor Emergency Feedwater System parameters during a cooldown.

Discharge flow from the EFW pumps is normally aligned and controlled by discharge control valves located in the supply line to each steam generator's emergency feedwater connection. These valves are controlled independently of the Integrated Control System and arranged to fail to the automatic control mode upon loss of DC control power to the manual/auto select solenoid. If the selected train of
3 automatic control fails, then the valve would fail open. Also, upon loss of all station air, the valves will
5 maintain their position with $N_2$ backup. If $N_2$ backup fails, then the valve would fail open. In
5 automatic, the control valve's manual/auto select solenoid valves are de-energized, thereby aligning the control valve to automatic control and positioning the valve per the automatic setting. These valves may be automatically controlled, or manually controlled by the operator to limit or increase emergency
2 feedwater as necessary to maintain steam generator inventory and cooldown rate. A pushbutton is provided for each control valve to allow the individual valve to be placed in either an automatic level
5 control mode or in a manual level control mode of operation. In automatic, the valves will be positioned
5 and be controlled by the automatic level control. Independent level transmitters are utilized in the automatic steam generator level control system. Upon loss of all four reactor coolant pumps, such as during blackout conditions, the level control setpoint is automatically raised to promote natural circulation in the Reactor Coolant System.

2 The steam supply for the EFW pump turbine is provided from either main steam line and/or Auxiliary
2 Steam. This common supply to the turbine will fail open upon loss of power to the normally energized solenoid valve. Upon loss of station air, the common supply is maintained by nitrogen bottle back-ups which are used on the essential common supply valves. Should the nitrogen bottle back-ups fail, the common supply would fail to open. Upon receipt of a manual or automatic start signal, the solenoid
5 valve will de-energize and immediately start the turbine. The turbine start signal is sealed-in whenever it is started automatically. Reset of the loss of feedwater logic does not stop the turbine driven pump.
2 Manual operator action is required to stop the turbine and reset the start logic sequence.

Cooling water is initiated automatically, upon manual or automatic start of the motor driven EFW pumps.

Alarms are provided to alert the operator of conditions exceeding normal limits. Essential plant parameters are annunciated or alarmed by the process computer in addition to specific EFW System alarms.

#### 7.4.3.1.3 System Evaluation

Redundancy is provided with separate, full capacity, motor and turbine driven pump subsystems. Failure of either the motor driven pumps or the turbine driven pump will not reduce the EFW System below minimum required capacity. Pump controls, and instrumentation are separate and independent in design.

### 7.4.3.2 Steam Generator Level Control

#### 7.4.3.2.1 Design Basis

The Steam Generator Level Control System (SGLCS) provides automatic Once Through Steam Generator (OTSG) water level control while the Emergency Feedwater System is supplying feedwater to the steam generators. SGLCS is designed to automatically control and modulate emergency feedwater supply to the steam generators during all initiating conditions for the EFW System (Section 7.4.3,

2 Automatic starting of the MDEFWP's is determined by the position of the control room selector switch
2 for each pump. The MDEFWP's are provided with a four position selector switch which allows the
2 operator to select between Off, Auto 1, Auto 2 and Run. When the selector switch is in the Auto 1
3 position, LOW STEAM GENERATOR WATER LEVEL in either steam generator (OTSG) will start
2 the pump after a 30 second time delay to prevent spurious actuations. When the selector switch is in the
2 Auto 2 position, LOW STEAM GENERATOR WATER LEVEL or LOSS OF BOTH MAIN
2 FEEDWATER PUMPS will start the pump. Loss of both main feedwater pumps is sensed by pressure
2 switches which monitor feedwater pump turbine control oil pressure and feedwater pump discharge header
2 pressure. Loss of both Main Feedwater Pumps actuation may be by either the control oil pressure
2 switches sensing loss of feedwater pumps or low discharge header pressure. This design allows the
2 operator to select the Auto 1 switch position during startup and shutdown conditions when the operating
2 main feedwater pump(s) approach low pump discharge actuation pressures and still have automatic
2 initiation of EFW prior to OTSG dry out.

2 Turbine Driven EFW Pump (TDEFWP):

2 Automatic starting of the TDEFWP is determined by the position of the control room selector switch for
2 the pump. The TDEFWP is provided with a three position-pull to lock selector switch which requires
2 that the control room operator manually take the switch to the OFF position through a deliberate action.
2 The operator can select between Off, Auto and Run. When the selector switch is in the Auto position,
5 LOSS OF BOTH MAIN FEEDWATER PUMPS, with exception to loss due to the Main Steam Line
5 Break (MSLB) logic, will start the pump. The MSLB circuitry will trip both Main Feedwater Pumps and
5 inhibit the TDEFWP from starting. The TDEFWP may be manually started by setting the selector
5 switch to RUN. Loss of both main feedwater pumps is sensed by pressure switches which monitor
2 feedwater pump turbine control oil pressure and feedwater pump discharge header pressure.

5 Once automatically started, the MDEFW pumps and the TDEFWP will continue to operate until
5 manually secured by the operator. Each emergency feedwater discharge line to each steam generator is
5 provided with a control valve and check valve. The control valves are normally closed in the automatic
5 mode due to steam generator level > 30″. The control valves are arranged to fail to the automatic
control mode upon loss of DC control power to the manual/auto select solenoid. If the selected train of
automatic control fails, then the valve would fail open. Also, upon loss of station air, the valves will
maintain their position with $N_2$ backup. If $N_2$ backup fails then the valve would fail open. These modes
of operation show that emergency feedwater isolation is not possible with valve control circuitry or motive
2 force failure. Open/Closed valve position indication is provided for each control valve in the main control
2 room at the valve manual loader.

5 In automatic, the manual/auto select solenoid valve on each control valve is de-energized, allowing the
5 valve to be positioned automatically. The control valves will receive an air signal for valve modulation in
response to steam generator level, independent from the ICS. The EFW pumps are normally aligned to
discharge into separate lines feeding a separate steam generator through the emergency feedwater header.

Power for the motor driven pumps is normally provided by the normal station auxiliary power system.
3 During loss of offsite power operation, these pumps are aligned to the Emergency Power System. Motive
steam for the turbine driven pump is provided from either of the two steam generators through the main
steam lines upstream of the main turbine stop valves.

Each of the EFW pumps is supplied with its own independent starting circuit which monitors main
feedwater pump trip status. These independent control circuits are powered by the 125 VDC station
batteries. They are actuated either by low feedwater pump turbine hydraulic control oil pressure or by
low feedwater pump discharge pressure of both main feedwater pumps. Low feedwater pump turbine
5 hydraulic control oil pressure is also determined by pressure switch actuation. Low feedwater pump

Alarms:

1. Low RC flow.
2. Plant computer alarm and printout for low flow.

Indication:

1. Loop flow indication on console falls to zero.
5 2. Loop flow indication in each RPS channel falls to zero on Unit 1. Flow is not displayed in Units 2
5 and 3 RPS channel cabinets unless STAR CTC is connected to channel.
3. Total flow indication on console falls approximately 50 percent.
5 4. Total flow indication in each RPS channel falls approximately 50 percent on Unit 1. Flow is not
5 displayed in Units 2 and 3 RPS channel cabinets unless STAR CTC is connected to channel.

*7.4.2.3.1.3 Conclusion*

The conclusion of this analysis is that the operator has adequate indication and alarm facilities to quickly recognize a common mode failure in the flow instrumentation for the reactor protection system. Corrective action would therefore be positive and prompt.

**7.4.2.3.2 Coincident LOCA and Systematic Failure of Low RCS Pressure Trip Signal.**

Several break sizes and locations for the loss-of-coolant accident have been investigated with an assumed systematic failure of the low Reactor Coolant System pressure trip signal. Although this failure is not considered credible, the analysis has shown that either the void shutdown mechanism or the power/flow comparator should provide backup to shut down the reactor and render the Emergency Core Cooling System (ECCS) effective.

## 7.4.3 EMERGENCY FEEDWATER CONTROLS

### 7.4.3.1 Emergency Feedwater and Pump Controls

#### 7.4.3.1.1 Design Basis

The EFW System is designed to start automatically in the event of loss of both main feedwater pumps or
3 low feedwater pump discharge header pressure or low water level in either steam generator after a 30
2 second delay to prevent spurious actuations. The EFW System will supply sufficient feedwater for
2 approximately five hours of cooldown at a flowrate of as much as 450 gpm to enable the Reactor Coolant System of reach conditions at which the Decay Heat Removal System may be operated.

Three EFW pumps are provided, powered from diverse power sources, two independent motor driven pumps powered from the Emergency Power System, each supplying feedwater to one steam generator; and one turbine driven pump, supplying feedwater to both steam generators, driven from steam contained in either steam generator. All automatic initiation logic and control functions are independent from the Integrated Control System (ICS).

#### 7.4.3.1.2 System Design

2 The EFW pumps will start automatically as outlined below:

2 <u>Motor Driven EFW Pumps (MDEFWP's)</u>:

Indication:

1. Break in a 1-inch HP Instrument Line

   a. Control room indication of the Reactor Building atmosphere particulate and gas radioactivities increases.

   b. Loop flow indication on console falls to zero.

   c. Loop flow indication in each RPS channel falls to zero on Unit 1. Flow is not displayed in Units 2 and 3 RPS channel cabinets unless STAR CTC is connected to channel.

   d. Total flow indication on console falls approximately 50 percent.

   e. Total flow indication in each RPS channel falls approximately 50 percent on Unit 1. Flow is not displayed in Units 2 and 3 RPS channel cabinets unless STAR CTC is connected to channel.

   f. Makeup flow goes to maximum value.

   g. RC pressure falls on console indicators and with each RPS channel.

   h. Reactor Building pressure and temperature indication rises.

2. Break in a 1-inch LP Instrument Line

   Identical indication as listed for HP line break except all loop flow indication goes full scale, total flow indication increases above normal.

*7.4.2.3.1.2.2 Break in a ½-inch Instrument Line*

A break of a ½-inch instrument line will result in a reactor trip due to low RC pressure. If the break occurs in a HP line, the reactor will trip due to a high power/flow ratio if the power/flow limit is exceeded.

The operator will receive the same alarms and indication as described for the 1-inch instrument line break.

*7.4.2.3.1.2.3 Leak in One of the Instrument Lines*

If the leak occurs in a HP line the operator will receive a low flow alarm for a 5 percent change in indication flow and a high flow alarm for a similar leak in the LP line. At this alarm Point, the leakage is in excess of 1 gallon per minute, hence Reactor Building radiation monitors will readily detect such a condition and result in leak evaluation, and subsequent action as required by Technical Specifications.

Depending on the size of the leak, alarms and indication described in Section 7.4.2.3.1.2.1, "Break in 1 Inch Instrument Lines" on page 7-35 may occur. If the leak occurs on either of the ΔP transmitters associated with the RPS-A input or the fifth channel input a mismatch alarm will occur and the SASS will select the valid signal for indication and ICS control as described in Section 7.4.2.2.2, "Non-Nuclear Process Instrumentation in Regulating Systems" on page 7-31.

*7.4.2.3.1.2.4 Rupture of ΔP Transmitter*

If the ΔP transmitter ruptures, such that the high pressure and low pressure sides of the transmitter are no longer isolated, the pressure between the HP and LP headers to which the transmitter is connected will be equalized. Since zero ΔP corresponds to zero flow, the output of all five ΔP transmitters for that affected loop will drop to zero. This will result in an immediate reactor trip if the power/flow limit is exceeded.

The operator will receive the following alarms and indication:

control console and as the selected input to the ICS via the automatic signal selector described in Section 7.4.2.2.2, "Non-Nuclear Process Instrumentation in Regulating Systems" on page 7-31.

Each of the four Reactor Protective System channels receives a ΔP signal form a different one of the four ΔP transmitters. In other words, one transmitter is exclusively assigned to one protective channel. The identical arrangement and assignment of transmitters is used for each of the two primary reactor coolant loops.

Within each Reactor Protective System channel, the square roots of the ΔP signals from each loop are
5 extracted to obtain loop flow signals. The loop flow signals are summed to obtain a total reactor coolant
5 flow signal. The three flow signals are displayed within the channel's cabinet on Unit 1. Units 2 and 3
5 require the connection of the STAR CTC to the channels's STAR module to display the information.
5 The three signals are monitored by the plant computer.

5 The reactor operator can read the individual loop flows at the control console on all units and within each
5 reactor protective channel's cabinets on Unit 1. Units 2 and 3 requires the connection of the STAR CTC
5 to display the flow information within the reactor protective channel's cabinets. The flow information is
5 available to the operator on the plant computer for each unit.

*7.4.2.3.1.2 Failures Considered*

The following failures are considered:

1. Break in one of the 1-inch instrument lines.
2. Break in one of the 1/2-inch instrument lines.
3. A leak in one of the instrument lines.
4. Rupture of ΔP transmitter bellows.

*7.4.2.3.1.2.1 Break in 1 Inch Instrument Lines*

A break of a 1-inch instrument line will result in a reactor trip due to low RC pressure. If the break occurs in a HP line, the reactor will trip due to a high power/flow ratio if the power/flow limit is exceeded.

The operator will receive at least the following alarms and indications:

Alarms:

1. Break in 1-inch HP Instrument Line
   a. Low RC flow.
   b. Plant computer alarm and printout for low flow.
   c. Letdown storage low level.
   d. Pressurizer low level.
   e. Low reactor coolant pressure.
   f. Plant computer alarm and printout for low RC pressure.
2. Break in a 1-inch LP Instrument Line

   Identical alarms as listed for HP line break except RC flow is alarmed on high value.

c. Pressurizer electromatic relief valve.

The heaters are grouped in banks which are energized below preset pressures.

The selected signal also provides input to a pressure controller which automatically modulates the output of one bank of heaters to maintain a preset pressure. The spray and relief valves are opened above preset pressures.

The selected signal is recorded and high and low pressures alarmed.

Reactor coolant pressure is recorded on two single-pen strip chart recorders. One recorder has a range of 1700-2500 PSIG, and its input is selectable from one transmitter measuring "A" loop pressure or one transmitter measuring "B" loop pressure. The other recorder has a range of 0-2500 PSIG, and its input is from a transmitter in the "A" loop.

Reactor coolant temperature is also recorded on two single-pen strip chart recorders. A recorder indicating average temperature receives its input from one of two RTD's in "A" loop and one of two RTD's in "B" loop and has a range of 520°F to 620°F. The other temperature recorder has a range of 50°F to 650°F, and its input is selectable from either of four RTD's two located in "A" loop cold legs and two located in "B" loop cold legs.

4. Coolant Pump Control

   Interlock signals of reactor coolant inlet temperature are provided to each pump switching logic to prevent operation of more than three pumps during startup until a preset temperature is reached.

5. Feed and Bleed Control

   The feed and bleed control instrumentation in the High Pressure Injection System provides control and interlocks to permit adjustment of the reactor coolant boron concentration.

## 7.4.2.3 System Evaluation

The quantity and types of process instrumentation have been selected to provide assurance of safe and orderly operation of all systems and processes over the full operating range of the plant. Some of the criteria for design are:

1. Separate instrumentation has been provided for the protective systems and vital control circuits.
2. Time of response and accuracy of measurements are adequate for protective and control functions to be performed.
3. Where wide process variable ranges are required and precise control is involved, both wide range and narrow range instrumentation are provided.
4. All electrical and electronic instrumentation required for operation is supplied form redundant vital and uninterruptable instrumentation buses.

### 7.4.2.3.1 Failure in RC Flow Tube Instrument Piping

#### *7.4.2.3.1.1 Reactor Coolant Flow Indication*

In each primary loop, reactor coolant flow is detected by measuring the ΔP developed across a flow tube that is an integral part of the outlet piping of the loop. Each flow tube has a high pressure (HP) tap and a low pressure (LP) tap. Connections to the taps are made with 1-inch lines. The 1-inch lines are terminated at root valves located inside the secondary shield wall to HP and LP headers. Five ΔP transmitters are connected between the two headers. Four are used to provide information to the Reactor Protective System. The fifth is used to provide the operator with flow indication and alarms at the

Pressure drop measurement across the valves is provided for input by redundant differential pressure transmitters. The selected input signal is also indicated.

h. Steam Generator Level

Selected "startup" level and "operate" level inputs are provided from each steam generator. Redundant measurements of each level are provided by differential pressure transmitters. Temperature compensation to augment the predetermined compensation for normal operating temperature is provided by two fast response resistance elements and associated transmitters which measure steam generator lower downcomer temperature.

The selected "operate." level input is recorded and "high" level alarmed. The selected "startup" level input is indicated and "low" level alarmed.

A full range level measurement is provided for indication of each steam generator level but does not provide protective or regulating systems input.

i. Steam Generator Outlet Pressure

Selected outlet pressure input is provided from each steam generator. Measurement is made by pressure transmitters in both outlet lines of each steam generator. The selected input is also indicated.

j. Turbine Header Pressure

Turbine header pressure measurement is provided for input by a pressure transmitter in each header line from the steam generators. The selected pressure signal is also recorded, and high and low pressures alarmed. Additional redundant transmitters in each header line provide indication only.

#### 7.4.2.2.3 Other Non-Nuclear Process Instrumentation

The following instrumentation is provided for measurement and control of process variables necessary for proper operation:

1. Pressurizer Temperature

Pressurizer temperature is measured by two fast response resistance elements and their associated transmitters. The selected output signal is indicated and supplies input for pressurizer level temperature compensation.

2. Pressurizer Level Control

Pressurizer level is measured by three differential pressure transmitters. One signal is selected for temperature compensation and output for recording, level control, alarms and interlock to de-energize the pressurizer electric heaters on low level. The level controller output positions the makeup control valve in the High Pressure Injection to maintain a preset level. Pressurizer level is lowered by reactor coolant letdown or by manual control at the control room.

3. Reactor Coolant Pressure Control

The reactor coolant pressure signal for control is provided by a fifth channel measurement. Selective redundant measurement by one of the other four pressure transmitters is also provided from the Reactor Protective System.

The selected signal is used as an input to pressure switches which provide signals for automatic control of:

a. Pressurizer electric heaters.

b. Pressurizer spray control valve.

The SASS can also detect a mismatch between the two input signals and provides indication of the mismatch on the SASS panel. The plant computer also receives the same signals as SASS and provides mismatch alarms to the operator via the plant computer.

The following inputs to the Integrated Control System are provided:

a. Reactor Outlet Temperature

Selected loop or unit average outlet temperature input is provided in each loop by two fast response resistance elements and associated transmitters.

b. Reactor Controlling Average Temperature

Instrumentation separate from the Reactor Protective System supplies input to the Integrated Control System.

Loop or unit average temperature signals are selected for indication and input as controlling average temperature. Automatic selection determined by loop flows is provided for input of the appropriate signals.

Reactor inlet temperature signals required for loop, and unit average or differential temperatures are provided in each loop by two fast response resistance elements and associated transmitters.

c. Reactor Inlet Differential Temperature

Reactor inlet differential temperature is indicated and provided for input to the Integrated Control System.

d. Reactor Coolant Flow

Reactor coolant flow signals are provided for each loop and summed for total flow. Total flow is recorded and "low" total flow is alarmed. Flow measurement in each loop is provided by a transmitter, independent of the Reactor Protective System, which measures flow through the flow tube.

Selective redundant measurement by one of the other four loop flow transmitters is also provided from the Reactor Protective System.

Loop "low" flow signals provide the logic for automatic selection of reactor controlling average temperature.

Contacts from reactor coolant pump motor breakers provide fast indication to the ICS that a pump has tripped.

e. Feedwater Temperature

Feedwater temperature input is provided by two fast response resistance elements and associated transmitters. The selected input also provides signals for indication and feedwater temperature compensation.

f. Feedwater Flow

Feedwater flow input is provided from a full range flow calculator for each loop. The calculator automatically selects and computes startup or main feedwater flow signals to provide the required full range flow input signal. The main feedwater flow measurement in each loop is provided by redundant differential pressure transmitters that measure flow through a flow nozzle. Startup feedwater flow measurement in each loop is provided by a differential pressure transmitter that measures flow through a flow nozzle.

g. Feedwater Control Valves Differential Pressure

Reactor Building pressure inputs to the Engineered Safeguards Protective System are provided by:

1) Three pressure transmitters which are located outside the Reactor Building. These provide inputs for initiation of Reactor Building isolation, high pressure injection, low pressure injection, and Reactor Building cooling.

2) Three groups of two pressure switches each are located outside the Reactor Building. These provide input signals of high Reactor Building pressure for initiation of Reactor Building spray by safeguards actuation.

Table 7-5 provides pertinent information concerning the NNI sensors supplying inputs to the RPS and ESPS, respectively.

#### 7.4.2.2.2 Non-Nuclear Process Instrumentation in Regulating Systems

Selective redundant measurements and input signals are provided for the process variables required for critical control functions. Automatic selection of valid signals is provided by a Smart Automatic Signal Selector (SASS). The SASS manufactured by B&W detects a rapid change in signal and automatically switches the SASS output signal to the remaining valid input signal.

Two SASS racks are provided for automatic signal selection of process instrumentation signals. The first rack of instrumentation is located in the fifth channel cabinet which is the regulating channel of instrumentation as described in Section 7.1.2.5, "Separation" on page 7-3. This SASS rack monitors:

1. Reactor Coolant System Pressure
2. Reactor Coolant Flow Loop A
3. Reactor Coolant Flow Loop B
4. Power Range Neutron Flux

The second rack of SASS instrumentation is located in ICS Cabinet 8. This SASS rack monitors the following process signals and selects the valid signal independent of the control board mounted key switch.

1. Feedwater Flow Loop A
2. Feedwater Flow Loop B
3. OTSG Operate Level Loop A
4. OTSG Operate Level Loop B
5. T-Hot Loop A
6. T-Hot Loop B
7. T-Cold Loop A
8. T-Cold Loop B
9. Turbine Header Pressure
10. OTSG Start-up Level Loop A
11. OTSG Start-up Level Loop B
12. Spare 0-10 VDC Channel

response to those systems and provides instrumentation for startup, operation, and shutdown of the reactor system under normal and emergency conditions.

### 7.4.2.2 System Design

The non-nuclear instrumentation provides measurements used to indicate, record, alarm, interlock, and control process variables such as pressure, temperature, level, and flow in the reactor coolant, steam supply, and auxiliary reactor systems as shown in system drawings in Chapter 5, "Reactor Coolant System and Connected Systems" on page 5-1, Chapter 9, "Auxiliary Systems" on page 9-1, Chapter 10, "Steam and Power Conversion System" on page 10-1 and Chapter 11, "Radioactive Waste Management" on page 11-1. Process variables required on a continuous basis for the startup, operation, and shutdown of the unit are indicated, recorded, and controlled at the control rooms. Alternate essential indicators and controls are provided at other locations to maintain the reactor in a hot shutdown condition if the control rooms have to be evacuated. Other instrumentation is provided at auxiliary panels with alarm at the control rooms.

Response time and accuracy of measurements are adequate for reactor protective and regulating systems and other control functions to be performed.

#### 7.4.2.2.1 Non-Nuclear Process Instrumentation in Protective Systems

Four independent measurement channels are provided for each process parameter for input to the Reactor Protective System.

Three independent measurement channels are provided for each process parameter and input to the Engineered Safeguards Protective System.

a. Reactor Outlet Temperature

   Reactor outlet temperature inputs to the Reactor Protective System are provided by two last-response resistance elements and associated transmitters in each loop.

b. Reactor Coolant Flow

   Reactor coolant flow inputs to the Reactor Protective System are provided by eight high-accuracy differential pressure transmitters which measure flow through calibrated flow tubes welded into the reactor outlet pipe. The power/flow monitor of the reactor protective system utilizes this flow measurement to prevent reactor power from exceeding a permissible level for the measured flow. Operation of each reactor coolant pump breaker is also monitored as an indication of flow.

   An additional differential Pressure transmitter for each loop is utilized as the normal flow measurement providing input to the Integrated Control System via the automatic signal selector (SASS) described in Section 7.4.2.2.2, "Non-Nuclear Process Instrumentation in Regulating Systems" on page 7-31.

c. Reactor Coolant Pressure

   Reactor Protective System inputs of reactor coolant pressure are provided by two pressure transmitters in each loop. An additional pressure transmitter, independent of the Reactor Protective System, is provided for pressurizer pressure control.

   Engineered Safeguards Protective System inputs of reactor coolant pressure in each loop are provided by redundant pressure transmitters. One pressure signal is utilized for recording, low pressure alarm, and interlock to decay heat removal return flow valves.

d. Reactor Building Pressure

and IEEE No. 279. An annunciator alarm exists to indicate a nuclear instrumentation out of calibration condition.

### 7.4.1.3 System Evaluation

4 The nuclear instrumentation will monitor the reactor over a minimum 10+decade range from source
4 range to 200 percent of rated power. The full power neutron flux level at the power range detectors will be approximately 3.2 x $10^9$ nv. The detectors employed will provide a linear response up to approximately 2.5 x $10^{10}$ nv before they are saturated.

4 The wide range channels fully overlap the source range and the power range channels as shown in Figure 7-7, providing the continuity of information needed during startup.

The steady-state radial flux distribution within the reactor core will be measured by the incore neutron detectors (Section 7.6.1, "Regulation Systems" on page 7-67). Both out-of-core and incore detectors will be used to obtain the axial power distribution. The sum of the outputs from the two sections of each (out-of-core) power range detector will be calibrated to a heat balance. The sum will be recalibrated whenever it is determined that the sum disagrees with the heat balance by 2 percent or more. The signals from the two sections of the detector may be individually read and compared independent of the sum of the outputs. The operator, therefore, may correlate the difference signal against the core power distribution obtained from the incore system.

#### 7.4.1.3.1 Primary Power

The nuclear instrumentation draws its primary power from vital buses and uninterruptable buses described in Section 8.3.2.1.4, "120 Volt AC Vital Power Buses" on page 8-21 and Section 8.3.2.1.5, "240/120 Volt AC Uninterruptible Power System" on page 8-22.

#### 7.4.1.3.2 Reliability and Component Failure

The requirements established for the Reactor Protective System apply to the nuclear instrumentation. All channel functions are independent of every other channel, and where signals are used for safety and/or control, electrical isolation is employed to meet the criteria of Section 7.1.2, "Identification of Safety Criteria" on page 7-3.

#### 7.4.1.3.3 Relationship to Reactor Protective System

The relation of the nuclear instrumentation to the RPS is described in Section 7.2, "Reactor Protective System" on page 7-7. Power range channels NI-5, -6, -7, and -8 are associated with the Reactor Protective System. NI-5 also provides information for the Integrated Control System through the automatic signal selector (SASS) described in Section 7.4.2.2.2, "Non-Nuclear Process Instrumentation in Regulating Systems" on page 7-31.

The periodic test requirements of the Reactor Protective System are not dictated by the accuracy of the power range channels. The accuracy of the linear amplifiers is better than $\pm 0.2$ percent including drift.

## 7.4.2 NON-NUCLEAR PROCESS INSTRUMENTATION

### 7.4.2.1 Design Bases

The non-nuclear process instrumentation provides the required input signals of process variables for the reactor protective, regulating, and auxiliary systems. It performs the required process control functions in

The fifth power range channel, NI-9, provides reactor power information to the ICS via SASS (Section 7.4.2.2.2, "Non-Nuclear Process Instrumentation in Regulating Systems" on page 7-31) and to a recorder located on the control console above the dual indicators. The channel is in no way associated with the
4 RPS. While channel NI-9 is the normal source of reactor power information to the ICS, the operator
4 may elect to use a selected channel supplying the RPS as the source. Power range neutron flux is input to the SASS from NI-5 (RPS-A) and NI-9. SASS will automatically switch the output to the ICS to the
4 valid input in the event of a signal failure. Either NI-5 or NI-9 can be switch selected by the operator to
4 feed the power range recorder. Isolation amplifiers are used to provide isolation from the RPS. Isolation amplifiers are used to buffer all signals leaving the system cabinets, preventing the reflection of faults on external signal lines back into the system.

#### 7.4.1.2.1 Neutron Detectors

4 The detectors used in the source range and wide range channels are fission chambers. The same
4 detector/electronics string provides both source range and wide range outputs.

4

Uncompensated ion chambers are used in the power range channels. Each power range detector consists of two 72-inch sections with a single high voltage connection and two separate signal connections. The outputs of the two sections are summed and amplified by the linear amplifiers in the associated power range channel to obtain a signal proportional to total reactor power ($\phi$). A signal proportional to the difference in percent full power between the top and bottom halves of the core, the reactor power imbalance or $\Delta\phi$, is derived from the difference in currents from the top and bottom sections of the detector. The difference signal is displayed on the control console to permit the operator to maintain proper axial power distribution. The manual test and calibration facilities provide a means for reading the output of the individual sections of the detector. Each detector has a combined sensitive volume extending approximately from the bottom to the top of the reactor core.

The physical locations of the neutron detectors are shown in Figure 7-8. The power range detectors for channels NI-5, -6, -7, and -8 are positioned adjacent to each of the four quadrants of the core. The power
4 range detector for channel NI-9 is adjacent to the power range detector for channel NI-5. The
4 source/wide range detectors are located adjacent to each of the four quadrants of the core.

Table 7-4 provides pertinent characteristics of the out-of-core neutron detectors. The flux ranges illustrated in Figure 7-7 are seen to be compatible with these characteristics. Nearly identical Westinghouse out-of-core detectors are presently in use at power reactors as follows:

| Tube Type | Reactors | Utility |
|---|---|---|
| 4 FC | Haddam Neck | Connecticut Yankee Power |
| 4 | San Onofre | Southern California Edison |
| 4 | Three Mile Island | GPU Nuclear |
| 4 | Crystal River 3 | Florida Power Corp. |
| 4 | | |
| UCIC | Haddam Neck | Connecticut Yankee Power |

#### 7.4.1.2.2 Test and Calibration

Test and calibration facilities are built into the system to permit an accurate calibration of the system and the detection of system failures in accordance with the requirements of Reactor Protective System design

# 7.4 SYSTEMS REQUIRED FOR SAFE SHUTDOWN

## 7.4.1 NUCLEAR INSTRUMENTATION

The nuclear instrumentation system is shown in Figure 7-6. The system meets the intent of the Proposed IEEE "Criteria for Nuclear Power Plant Protection Systems," dated August, 1968, (IEEE No. 279), for those elements associated with the Reactor Protective Systems.

### 7.4.1.1 Design Bases

The nuclear instrumentation (NI) system is designed to supply the reactor operator with neutron information over the full operating range of the reactor and to supply reactor power information to the RPS and to the Integrated Control System (ICS).

The system sensors and instrument strings are redundant in each range of measurement. Measurement ranges are designed to overlap to provide complete and continuous information over the full operating range of the reactor.

### 7.4.1.2 System Design

The nuclear instrumentation has nine channels of neutron information divided into three ranges of
4 sensitivity: source range, wide range, and power range. The three ranges combine to give a continuous
4 measurement of reactor power from source level to approximately 200 percent of rated power or ten + decades of information. A minimum of one decade of overlapping information is provided between successive higher ranges of instrumentation. The relationship between instrument ranges is shown in Figure 7-7.

4 The source range instrumentation has four redundant count rate channels originating in four high
4 sensitivity fission chambers. These channels are used over a counting range of 0.1 to $10^5$ counts/sec as displayed on the operator's control console in terms of log counting rate. The channels also measure the
4 rate of change of the neutron level as displayed for the operator in terms of startup rate from -1 to +7
4 decades/min.

4 The wide range instrumentation has four log N channels originating in four electrically identical fission
4 chambers. Each channel provides ten + decades of flux level information in terms of the log of chamber
4 count rate and startup rate. The fission chamber/wide range monitor output range is from $10^{-8}$ to 200%
4 power. The startup rate range is from -1 to +7 decades/min. A high startup rate of +2 decades/min. in
4 any channel will initiate a control rod withdraw inhibit.

The power range channels have five linear level channels originating in five composite uncompensated ion chambers. The channels output is directly proportional to reactor power and covers the range from 0 to 125 percent of rated power. The gain of each channel is adjustable providing a means for calibrating the output against a reactor heat balance.

Power range channels NI-5, -6, -7, and -8 supply reactor power level information continuously to the RPS. Dual indicators on the control console provide the operator with both total reactor power information ($\phi$), and reactor power imbalance information ($\Delta\phi$), from each of the four channels. The method of obtaining $\phi$ and $\Delta\phi$ is described in Section 7.4.1.2.1, "Neutron Detectors" on page 7-28.

### 7.3.3.5 Manual Trip

A manual trip switch is provided in each Engineered Safeguards Protective System channel. There are eight manual trip pushbuttons on the control console, one for each protective channel.

### 7.3.3.6 Bypassing

The trip functions of the High and Low Pressure Injection and Reactor Building Non-Essential Isolation Systems are bypassed whenever the reactor is to be depressurized below the trip point of the systems. Bypassing must be initiated manually within a fixed pressure band above the protective system trip point. The High Pressure Injection and Reactor Building Non-Essential Isolation System may be bypassed only when the reactor pressure is 1,750 psi or less, and the Low Pressure Injection System may be bypassed only when the reactor pressure is 900 psi or less. The bypass is automatically removed when the reactor pressure exceeds the 1,750 and 900 psi values. This is in accordance with IEEE 279, Section 4.12. The removal set points are above the trip points in order to obtain a pressure band in which the trips may be bypassed during a normal cooldown. The bypasses do not prevent actuation of the HP and LP Injection and Reactor Building Non-Essential Isolation Systems on high Reactor Building pressure. Bypassing is under administrative control. Since the ESPS incorporates triple redundancy in its analog input subsystems, there are three HP injection bypass switches and three LP injection bypass switches. Two of the three switches must be operated to initiate a bypass. Once a bypass has been initiated, the condition is indicated by the plant annunciator and by lamps associated with the bypass switches. The switches are backlighted. No provisions are made for manual removal of a bypass before its automatic removal set point is reached.

### 7.3.3.3 Physical Isolation

The arrangement of modules within the system cabinets is designed to reduce the chance of physical events impairing system operation. Control wiring between the UC modules and the final actuating devices is physically separated and protected against damage which could impair system operation.

Separation between redundant channels of equipment, control cables, and power cables provides independence of redundant ES functions. Power and control cables for each group of ES equipment are routed in cable trays that contain no cable for redundant equipment. Cables for Reactor Building cooling units enter each Reactor Building through three separate penetrations located at least 25 feet apart and are routed in three different directions to the cooling units. The only other ES equipment located inside the Reactor Buildings are electric motor operated isolation valves which are all common to the odd numbered actuation group discussed above.

### 7.3.3.4 Periodic Testing and Reliability

The number of elements which can fail in a single instrument string is small as the system coincidence logic is not complex. The redundancy of the logic and the division of protective devices between logics forms a system having two parallel protective channels either of which is capable of performing the required functions. These characteristics are basic to an inherently reliable system. Logic elements are relays which have been selected for reliability and subjected to confirming tests under loads identical to those encountered in the system. The resultant calculated probability of logic failure is several orders of magnitude less than the known or estimated probability of failure of all other system elements.

The built-in test facilities permit an electrical trip test of each analog instrument string by the substitution of signals at the isolation amplifiers.

When an analog instrument string is placed in test, all associated analog subsystem outputs go to the trip state. This assures that protective action cannot be defeated by placing analog instrument strings in test.

To avoid a full protective channel or system trip, the logic is tested in parts, one element at a time. The continuity of the electrical connections from the output of the coincidence logic up to each $R_o$ relay is tested by means of the LT and UC modules. A LT module can neither prevent a trip of the associated protective channel when protective action is called for nor initiate a trip of the associated protective channel.

An individual protective device may be actuated by means of the UC module manual switch. Operating this switch energizes the $R_o$ relay as if the protective channel has tripped actuating the associated final device. The module lamp confirms that the module test relay returned to its normal state upon release of the manual switch.

On-line checks of the system will confirm the normal state of the system, principally by comparative readings of similar analog indications between redundant measurements and by the status lamps on bistables and logic modules.

The set points of the pressure switches used for ESPS channels 7 and 8 may be checked by connecting a source of pressure and a pressure gauge to the pressure transmitter connections provided inside the Reactor Building. This check may be made regardless of reactor power when access to the building is attained. The design provides access for this check at all reactor power levels.

The example just presented shows the independence and redundancy of the system. There is redundancy of sensors, logic, and equipment. The redundancy is preserved and kept effective by independence of sensors, instrument strings, logic, and control elements in the final actuator. These characteristics enable the system to tolerate single failures at all levels.

The system protective devices (pumps, valves, etc.) require electrical power in order to operate and perform their functions. The power for operating the CR relays is taken from the power source of the associated device. Loss of power to a CR relay or device does not impair the system functions since there is a second redundant device for each required function. The power for the $R_o$ relays, logic, and instruments is taken from the plant's system of battery backed vital buses since loss of power at this level could affect the performance capability of the system. The system will tolerate the loss of one vital bus without loss of protective capability.

### 7.3.3.1 Redundancy and Diversity

The system as evaluated above is shown to have sufficient diversity and redundancy to withstand single failures at every level.

### 7.3.3.2 Electrical Isolation

The use of isolation amplifiers will effectively prevent any faults (shorts, grounds, or cross connection of signals) on any analog signal leaving the system from being reflected into or propagating through the system. The direct connection of any analog signal to a source of electrical power can, at worst, negate information from the measured variable involved. The use of individual $R_o$ relays for each controlled device effectively preserves the isolation of each device and of elements of one protective channel from another. Faults in the control wiring between an $R_o$ relay and its CR relay in the controller of a protective device will not affect any other device or protective channel action.

Separation of redundant Engineered Safeguards (ES) functions is accomplished by assigning the eight actuation channels (Table 7-2) to three groups. Isolation for power, control, equipment location, and cable routing is maintained throughout. Channels 1, 3, 5 and 7 are assigned to one group (odd actuation channels). Channels 2, 4, 6 and 8 are assigned to a second group (even actuation channels). Equipment which is actuated by both the even and odd actuation channels is assigned to a third group. All equipment required to perform a specific ES junction is assigned to the same group. For example, a pump motor and all valves required for that pump to perform its function are assigned to the same group.

For Oconee 1, AC power for equipment controlled by the odd numbered actuation channels is supplied from Switchgear Group 1TC (4KV), motor control center 1XS1 (600 and 208 volts), actuation power from Vital Power Panelboard 1KVIA and DC control power from DC Panelboard 1DIA. ES functions which are redundant to those controlled by the odd numbered actuation channels are controlled by the even numbered actuation channels. AC power for this equipment in Oconee 1 is supplied from Switchgear Group 1TD (4KV), Motor Control Center 1XS2 (600 and 208 volts), from Vital Power Panelboard 1KVIB, and DC control power from DC Panelboard 1DIB. Where a third unit of ES equipment is used to provide additional redundancy, it is actuated by both the even and odd actuation channels. AC power for this equipment in Oconee 1 is supplied from Switchgear Groups 1TE or 2TC (4KV), Motor Control Center 1XS3 (600 and 208 volts), actuation power from either Vital Power Panelboard 1KVIA for odd channel actuation or Vital Power Panelboard 1KVIB for even channel actuation, and DC power from DC Panelboard 1DIC. Similar arrangements are employed for ES equipment in Oconee 2 and 3 with different power and control sources for each unit. These are described in Section 8.3, "Onsite Power Systems" on page 8-9.

### 7.3.2.5 Availability of Information

All system analog signals are indicated within the system cabinets and are monitored by the plant computer. All bistable outputs are indicated within the cabinets. Logic outputs are indicated within the cabinets and their state monitored by the plant computer.

Plant annunciators provide the operator with immediate indication of changes in the status of the ESPS. Included are all test switches, except those that are spring loaded to return to the operate position.

### 7.3.2.6 Summary of Protective Action

Actions initiated by the Engineered Safeguards Protection System are tabulated in Table 7-2. The devices actuated by the Engineered Safeguards Protection System are listed in Table 7-3. Channels indicated may be referred to applicable systems as shown in Figure 7-5. All actuated devices remain in their emergency modes after the reset of an engineered safeguards actuation signal until the devices are reset by operator action.

## 7.3.3 SYSTEM EVALUATION

The ESPS is a basic three-channel redundant system employing 2-out-of-3 coincidence between measure variables.

The system will tolerate the failure of one of three variables among either the reactor coolant pressure measurements or Reactor Building pressure measurements without losing its ability to perform its intended functions.

The High and Low Pressure Injection and Reactor Building Non-Essential Isolation Systems are actuated by either reactor coolant pressure or Reactor Building pressure, thus providing diversity in actuation. The system will tolerate single or multiple failures within one protective channel without affecting the operation of other protective channels. This is the result of keeping each of the protective channel logics independent of every other protective channel. The independence is carried through the protective channel logic and up to the final actuating CR control relay. This is best illustrated by considering the actuation arrangement for the high pressure injection pumps (Figure 7-5).

There are three High Pressure Injection System pumps which operate in the event of an accident. HP-P1A is under the control of protective channel 1, HP-P1C is under the control of protective channel 2, while HP-P1B is under the control of both channels. Within the motor controller of HP-P1A and HP-P1C there is a single CR control relay controlled by the $R_o$ relay in the pump's associated Test and Block module. The operation of the protective channel logic, the $R_o$ relay in relation to the CR relay, was described previously. Should any two of the three reactor coolant pressure variables drop below their bistable set point, both protective channel 1 and 2 logics will trip, energizing the appropriate CR relays, and start the pumps.

Within the motor controller of HP-P1B there are two independent CR relays, each controlled by separate $R_o$ relays in separate Test and Block modules, one in channel 1 and one in channel 2. The arrangement is identical to the way a channel would control any device since all elements are independent and duplicated through the CR relay. The only common element is the power source for the CR relays which is common to the motor controller. Loss of this power prevents the motor control from operating as well as the pump. Relays that monitor actuator coils for each motor or valve control detect either an open coil or a loss of control power.

There are three independent reactor coolant pressure sensors. The output of each sensor terminates in an isolation amplifier which provides individually isolated outputs. One output of each pressure measurement goes to the plant computer for monitoring. One output goes to bistables, for initiating high pressure injection and Reactor Building non-essential isolation action and for low pressure injection action. The bistables are identical except for their set point. Bistable action is initiated when the low reactor coolant pressure set points are reached.

The output of the three high pressure injection and Reactor Building Non-Essential Isolation System bistables is combined in series with the trip outputs of three Reactor Building pressure bistables. The combination of reactor coolant pressure and Reactor Building pressure bistables outputs allows either variable to initiate high pressure injection and Reactor Building non-essential isolation.

The series outputs of the bistables are brought together in two identical 2-out-of-3 coincidence logics which form two Engineered Safeguards Protective System channels. Either of the two protective channels is independently capable of initiating the required protective action through redundant high pressure injection and Reactor Building Non-Essential Isolation System equipment.

The outputs of the three Low Pressure Injection System bistables are also combined in series with the independent trip outputs of the three Reactor Building pressure bistables. The combination functions in identically the same way as described for the High Pressure Injection System, with two 2-out-of-3 coincidence logics and protective channels.

### 7.3.2.3 Reactor Building Cooling and Reactor Building Essential Isolation System

There are three Reactor Building pressure sensors. The output of each sensor terminates in an input isolation amplifier, which provides individually isolated outputs. One isolated output of each pressure measurement goes to the plant computer for monitoring. One output of each pressure measurement goes to a bistable which initiates action when its high building pressure trip point is exceeded. Each input isolation amplifier module contains an analog meter for indicating the measured pressure. Each of the three bistables has contact outputs that are combined in series with the output of the High and Low Pressure Injection System bistables as previously described.

The outputs of the three bistables are brought together in two identical 2-out-of-3 coincidence logics which provide two Engineered Safeguards Protective System channels. Either of the two channels is independently capable of initiating the required protective action. Each protective channel uses redundant protective system devices.

### 7.3.2.4 Reactor Building Spray System

The Engineered Safeguards Protective System channels of the Reactor Building Spray System are formed by two identical 2-out-of-3 logic networks with the active elements originating in six Reactor Building pressure sensing pressure switches.

The independent pressure switches containing normally open contacts form one protective channel's 2-out-of-3 logic inputs. Three other identical pressure switches form the 2-out-of-3 logic inputs of the second protective channel. Either of the two protective channels is capable of initiating the required protective action.

### 7.3.2.1 System Logic

The Engineered Safeguards Protective System is a basic 2-out-of-3 coincidence logic system. Each input variable is measured three times, the three redundant signals terminate in three bistables as shown in Figure 7-5.

The Engineered Safeguards Protective System consists of eight 2-out-of-3 coincidence logic networks for actuating the equipment in four safeguards systems, thus each system is actuated by a pair of 2-out-of-3 logic and its outputs are referred to as an Engineered Safeguards Protective System channel. Each safeguards system is therefore actuated by two redundant coincidence logics or protective channels.

The coincidence logic output is normally de-energized. Trip action consists of closing the electrical path through the logic.

The output of the protective channel coincidence logic is connected to the channel's unit control module (UC modules). There is one UC module for every item, (pump, valve, etc.) controlled by the protective channel. A protective channel's UC modules are connected in parallel with the output of the coincidence logic.

The output of the coincidence logic follows a normally closed path in each UC module, finally terminating in an output relay, $R_o$, within each module. The $R_o$ relays of a protective channel's UC modules are driven in parallel with the output of the protective channel coincidence logic.

The contacts of the $R_o$ relay are normally open across a control line terminating in a control relay, CR, in the controller of the equipment assigned to the individual UC module. Power for operating the CR relay is taken from the equipment controller in series with the $R_o$ relay contacts. Trip action involves energizing the $R_o$ relay, closing its contacts which in turn energizes the CR relay actuating the assigned equipment.

Each protective channel is equipped with a logic test module (LT module). The LT module, together with the UC module, provides the necessary circuitry to permit trip testing of an individual protective device without tripping an entire protective system or channel.

The UC module also provides a means whereby following a system trip, an individual protective device may be removed from the control of the Engineered Safeguards Protective System and returned to manual control. This block action cannot be initiated prior to a system trip.

The design of the system's logic can be summarized in terms of the systems operation as follows:

1. Each protective action is initiated by either of two protective channels with 2-out-of-3 coincidence between input signals.
2. Protective action is initiated by applying power from the protective channel logic to the individual $R_o$ relays in the UC modules, which in turn energize the CR relays in each protective device controller.
3. There is a UC module for every safeguards system component (valve, pump, etc.)

### 7.3.2.2 High and Low Pressure Injection and Reactor Building Non-Essential Isolation Systems

The instrumentation, logic, and actuation of the High and Low Pressure Injection Systems are identical in design. The systems differ only in their actuation set point.

# 7.3 ENGINEERED SAFEGUARDS PROTECTIVE SYSTEM

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**Note**

0 This section of the FSAR contains information on the design bases and design criteria of the
0 system/structure. Additional information that may assist the reader in understanding the system is
0 contained in the design basis document (DBD) for this system/structure.

The Engineered Safeguards Protective System (ESPS) monitors parameters to detect the failure of the Reactor Coolant System and initiates operation of the High and Low Pressure Injection Systems, the Building Isolation, the Reactor Building Cooling and the Reactor Building Spray Systems. In addition, the signal is used to start the standby power source and initiate a transfer to the standby power source when required as described in Section 8.3.1.1.3, "4160 Volt Auxiliary System" on page 8-10.

## 7.3.1 DESIGN BASES

The design basis of the system includes the items of Section 7.1.2, "Identification of Safety Criteria" on page 7-3 with the following additions:

### 7.3.1.1 Loss of Power

1. The loss of vital bus power to the instrument strings will, with the exception of Reactor Building Spray, initiate a trip of that portion of the logic associated with the affected instrument string.
2. The loss of vital bus power to the system logic will not initiate system actuation.

### 7.3.1.2 Equipment Removal

1. Removing modules from an instrument string will, with exception of Reactor Building Spray, initiate a trip in that portion of the logic associated with the affected instrument string.
2. Removing logic modules from one protective channel does not affect any other protective channel and does not initiate system action.

### 7.3.1.3 Control Logic of ESF Systems

All systems receiving the ES signal remain in the emergency mode required by the ES actuation after the signal is reset. A separate deliberate action is required to shut off the ES systems and power supplies.

The following systems have been modified to conform to the above requirement of I.E. Bulletin 80-06:

1. HPI Pumps
2. Penetration room exhaust fans
3. Reactor Building Cooling Unit fans
4. Keowee Start

## 7.3.2 SYSTEM DESIGN

## 7.2.4 REFERENCES

1. H. B. Tucker letters to H. R. Denton (NRC), November 4, 1983 and February 27, 1986. Response to Generic Letter 83-28 Item 1.2.

The shutdown bypass switch enables the power/imbalance/flow, power/RC pumps, low pressure, and pressure-temperature trips to be bypassed allowing control rod drive tests to be performed after the reactor has been shut down and depressurized below the low reactor coolant pressure trip point. Before the bypass may be initiated, a high pressure trip bistable, which is incorporated in the shutdown bypass circuitry, must be manually reset. The setpoint of the high pressure bistable (associated with shutdown bypass) is set below the low pressure trip point. If pressure is increased with the bypass initiated, the channel will trip when the high pressure bistable (associated with shutdown bypass) trips. The user of the shutdown bypass key switch is under administrative control.

For maintenance purposes, a bistable may be removed from the system and a dummy bistable inserted in
5 its place, thus bypassing the original function. Installing a dummy bistable forces the protective channel
5 into a trip state upon removal of the bistable. Thus, the removal and substitution cannot be performed
5 without passing through a tripped condition. The Units 2&3 installation of the BWNT STAR hardware
5 in the flux/imbalance/flow (fif) trip string requires the use of jumpers to bypass the trip string. The
5 installation of jumpers to bypass the fif trip does not require the removal of the STAR processor module,
5 therefore, the protective channel is not forced into a tripped condition. The use of dummy bistable
5 modules and jumpers is under administrative control.

### 7.2.3.9 Post Trip Review

Post trip review data and information capabilities are provided by use of sequence of events and time history recording equipment. This equipment provides sufficient information on plant parameters to assure that the course of the reactor trip can be reconstructed as well as provide root cause determination. Specific capabilities are provided in Reference 1 on page 7-17.

the Reactor Buildings thirty feet apart. Located under the control rooms between the outside of the Reactor Buildings and the cable and equipment rooms, four separate trays are provided per unit which carry nothing but nuclear, RPS, ES, and accident monitoring instrumentation cables. Three separate routes are followed by these trays. RPS channel C and RPS channel D follow the same route but are separated vertically by 1-1/2 feet. A detailed review of cable tray and pipe routing in this area indicates that no more than two RPS channels could be damaged by a single pipe failure or missile. Equipment locations in the Auxiliary Building provide the basis for vertical arrangement of trays following the same route from the Reactor Buildings. Switchgear for power equipment is located at lower elevations and instrumentation cabinets are located at higher elevations. Therefore, vertical separation of classes of cables in trays is as follows from top trays down:

1. Instrumentation cable trays
2. Control cable trays
3. Power and control cable trays
4. Power cable trays

Inside the cable rooms, cables from each protective channel are routed in trays separate from those carrying cables from any other protective channel. Included in these trays are instrumentation cables from the Reactor Building, control and interconnecting cables associated with that protective channel, and non-protective instrumentation and control cables. Both protective and non-protective cables are individually armored and are flame retardant.

Reactor trip cables from the four RPS cabinets are routed separately to a reactor trip switch located on the main control board. From the trip switch, the cables follow four separate paths to the reactor trip breakers and the control rod drive cabinets.

### 7.2.3.6 Primary Power

The primary source of 120V ac power for the Reactor Protective System comes from four vital buses, one for each protective channel, as described in Section 8.3.2.1.4, "120 Volt AC Vital Power Buses" on page 8-21.

### 7.2.3.7 Manual Trip

Manual trip may be accomplished from the control console by a trip switch. This trip is independent of the automatic trip system. Power from the control rod drive power breakers' undervoltage coils comes from the RT modules. The manual trip switches are between the reactor trip module output and the breaker undervoltage coils. Opening of the switches opens the lines to the breakers, tripping them. There is a separate switch in series with the output of each reactor trip module. All switches are actuated through a mechanical linkage from a single pushbutton.

### 7.2.3.8 Bypassing

Each protective channel is provided with two key-operated bypass switches, a channel bypass switch and a shutdown bypass switch.

The channel bypass switch enables a protective channel to be bypassed without initiating a trip. Actuation of the switch initiates a visual alarm on the main console which remains in effect during any channel bypass. The key switch will be used to bypass one protective channel during on-line testing. Thus, during on-line testing, the system will operate in 2-out-of-3 coincidence. The use of the channel bypass key switch is under administrative control.

5 1. Prior to startup (following a refueling outage), all Reactor Protective System channels, logic, and
5 control rod drive power breakers are electrically trip tested to prove their operability. Testing is
5 performed on a 45 day staggered test bases, for example:

5 2. 45 days after startup, protective channel A is electrically trip tested for every input up to and including
5 the channel trip relay.

5 3. 90 days after startup, protective channel B is similarly tested.

5 4. 135 days after startup, protective channel C is similarly tested.

5 5. 180 days after startup, protective channel D is similarly tested.

5 6. 225 days after startup, protective channel A is similarly tested.

5 The rotational cycle is repeated so that a different protective channel is electrically trip tested every 45
5 days.

The control rod drive power breaker with a reactor trip module is tested monthly.

Rotational testing has several advantages. It significantly reduces the probability of system failure as compared to testing all protective channels at one time. It also reduces the chance of systematic errors entering the system.

### 7.2.3.5 Physical Isolation

The need for physical isolation has been met in the physical arrangement of the protective channels within separate cabinets and wiring within the cabinets separating power and signal wiring so as to reduce the possibility of some physical event impairing system functions. The systems sensors are separated from each other. There are four pressure taps for the reactor coolant pressure measurements to reduce the likelihood of a single event affecting more than one sensor. Outside the Reactor Protective System cabinets, vital signals and wiring are separated and physically protected to preserve protective channel independence and maintain system redundancy against physical hazards.

Redundant detectors and transmitter applied in the Reactor Protective System are located to provide physical separation. Redundant out of core nuclear detectors are located in separate quadrants around the reactor vessels. Two resistance thermometers assigned to the RPS are located on each reactor coolant outlet header. Cables approach redundant temperature detectors from opposite directions. Redundant pressure transmitters are located outside the secondary shield in four separate quadrants of the Reactor Buildings. Two reactor coolant pressure transmitters for RPS are connected to each of the two loops. Separate flow transmitters for each RPS channel are applied to sense the flow in each loop. This arrangement results in detectors and transmitters associated with one RPS channel being located in essentially (the reactor vessels are not in the center of the Reactor Buildings) the same quadrant of a Reactor Building, and with redundant detectors and transmitters located in another quadrant of the Reactor Building. Since each RPS channel receives a flow signal from both loops, one of the flow transmitters for each channel is not located with the other RPS transmitters for that channel. Location and cable routing for these transmitters is such that separation of at least seven feet is provided between redundant channels inside the Reactor Buildings. Cables for redundant RPS and ES detectors and transmitters are routed in separate directions to four separate Reactor Building penetrations in trays carrying only nuclear instrumentation, RPS, ES, and accident monitoring instrumentation. These penetration assemblies are assigned to nuclear instrumentation, ES instrumentation, accident monitoring instrumentation, and RPS cables exclusively. Two of these penetration assemblies are located sixty feet apart in separate quadrants of each Reactor Building. One is used for RPS and ES channel A instrumentation; the other for RPS and ES channel B instrumentation. A penetration assembly for RPS and ES channel C instrumentation and one for RPS and ES channel D are located on the opposite side of

The isolation amplifier circuits have been prototype tested to assess their effectiveness to isolate the input signal from output circuit faults. They are capable of blocking a direct connection (i.e., a hot short) across their output of 410 vdc (300 v rms) without effecting the input source. The redundancy and coincidence logic of the system permits the system to tolerate failures and thus reduces the chance of an inadvertent reactor trip. For Units 2 and 3, the BWNT STAR hardware for the Flux/imbalance/flow trip string is installed. This hardware uses optical isolators rated at 500 VAC, ± 700 VDC galvanic isolation.

### 7.2.3.4 Periodic Testing and Reliability

The use of 2-out-of-4 logic between protective channels permits a channel to be tested on-line without initiating a reactor trip. Maintenance to the extent of removing and replacing any module within a protective channel may also be accomplished in the on-line state without a reactor trip.

To prevent either the on-line testing or maintenance features from creating a means for unintentionally negating protective action, a system of interlocks initiates a protective channel trip whenever a module is placed in the test mode or is removed from the system. However, provisions are made in each protective channel to supply an input signal which leaves the channel in a non-tripped condition for testing or maintenance.

The test scheme for the Reactor Protective System is based upon the use of comparative measurements between like variables in the four protective channels, and the substitution of externally introduced digital and analog signals as required, together with measurements of actual protective function trip points.

On-line testing may be performed at different intervals and levels within the system consistent with satisfactory system reliability characteristics. The reliability of the system for random failures has been assured by careful selection of components, failure testing of logic elements, environmental testing of the system's modules, and long-term prototype proof-testing with the Babcock and Wilcox Test Reactor (BAWTR).

The reliability of the system logic, primarily the relays and coincidence networks in the RT modules, has been made very high so as to eliminate the need for frequent tests of the logic. The logic relays are of two classes; one class designed for high speed, light electrical loads, and more than $10^7$ operations under load; and the other class for switching electric loads of up to 10 amperes and than $10^6$ operations. Confirmation tests of operational reliability of these two types of relays, operated under load as they are used in the RPS, have been performed with no sign of failure or wear to $5 \times 10^6$ and $1.2 \times 10^6$ operations respectively.

The system test scheme includes frequent visual checks and comparisons within the system on a regular schedule in which all protective channels are checked at one time, together with less frequent electrical tests conducted on a rotational plan in which tests are conducted on different protective channels at different times.

A regular check of all Reactor Protective System indications is required. The check includes such things as comparing the value of the analog variables between protective channels and observing that the equipment status is normal. In addition, power-range protective channel readings are compared with a thermal calculation of reactor power. These checks are designed to detect the majority of failures that might occur in the analog portions of the system as well as the self-annunciating type of failure in the digital portions of the system. The electrical tests are designed to detect more subtle failures that are not self-evident or self-annunciating and are detectable only by testing.

Electrical tests are conducted on a rotational basis in accordance with a preliminary test scheme as follows:

Each of the reactor trip modules (2-out-of-4 logic networks) controls a control rod drive breaker or contactor. Thus, a trip in any 2 of the 4 protective channels initiates a trip of all the breakers and contractors. The breakers and contactors, however, are arranged in what is effectively a 1-out-of-2 logic (Figure 7-4). This system combines the advantages of the 2-out-of-4 and the 1-out-of-2 × 2 system alone. The combination results in a system that is considered superior to either of the basis systems alone.

In evaluating system performance, it is arbitrarily assumed that "failure" can either prevent a trip from occurring or can initiate trip action.

The redundant Reactor Protective System inputs operate in a true 2-out-of-4 logic mode so that the failure of an input leaves the system in either a 2-out-of-3 or a 1-out-of-3 logic mode, with either state providing sufficient redundancy for reliable performance.

The system can tolerate several input function failures without a reduction in performance capability provided the failures occur in unlike variables in different protective channels, or are of a different mode in different protective channels, or all occur within one protective channel. When a single protective channel fails, the system is left in either a 2-out-of-3 or 1-out-of-3 logic mode as explained below.

The protective channel trip relay of each channel is located in the reactor trip module associated with the channel. Within each reactor trip module, there is a logic relay for each protective channel. These combine in each module to form the 2-out-of-4 logic. A Failure Mode and Effects analysis of the reactor trip module has demonstrated that single failures within the module or in its interconnections can produce only the following effects:

1. Trip the breaker associated with the module.
2. Place the system in a 2-out-of-3 mode, as if the associated protective channel had a cannot trip failure.
3. Place the system in a 1-out-of-3 mode, as if the associated protective channel had tripped.

The combination of reactor trip modules and control rod drive breakers and contactors form a 1-out-of-2 × 2 logic. At this level the system will tolerate a "cannot trip" type of failure of one reactor trip module, or of the breaker and/or contactors associated with one reactor trip module without degrading the system's ability to trip all control rods. The failure analysis demonstrates that no single failure involving a reactor trip module will prevent its associated breakers and contactors from opening.

### 7.2.3.2 Redundancy

The design redundancy of the Reactor Protective System would allow physically removing all the components associated with a single protective channel. Doing so would have all the remaining components and protective channels operational in a 1-out-of-3 system.

### 7.2.3.3 Electrical Isolation

All signals leaving the Reactor Protective System are isolated from the system either by the use of
5 isolation amplifiers for analog signals, by relay contacts (in the case of digital signals), or by optical
5 isolators for the BWNT STAR hardware and relay contacts. The effect of this isolation is to prevent faults occurring to signal lines outside of the Reactor Protective System cabinets from being reflected into more than one Protective channel. The isolation thus provided also assures that two or more protective channels cannot interact through the cross-coupling or faulting of related signal lines.

Faults such as short, open, or grounded circuits and cross-coupling of analog output signals from two or more channels have no effect upon the protective channels or their functions.

Reactor Building pressure exceeding the trip point specified in Table 7-1, the contact buffer de-energizes the protective channel's trip relay.

#### 7.2.2.4 Setpoint Adjustments for Single Loop Operation

Following amendments 165/165/162 to the facility operating license, single loop power operation is prohibited.

#### 7.2.2.5 Availability of Information

The modules, logic, and analog equipment associated with a single protective channel are contained
5 wholly within two Reactor Protective System cabinets. Within these cabinets, there is a meter for most of
5 the analog signals employed by the protective channel, and a visual indication of the state of every logic
5 element. The exceptions to having local meters for indication of the analog signals are the Units 2 and 3
5 RC flow and flux imbalance signals. This information may be obtained by connecting the calibration test
5 computer to the cabinet hardware. At the top of one cabinet, and easily visible at all times, is a protective channel status panel. Lamps on this panel give a quick visual indication of the trip status of the particular protective channel and of the RT module associated with it. Additional lamps on the panel give visual indication of a channel bypass or a fan failure.

In addition to the visual indications and readouts within the protective channel cabinets, each trip function, power supply, and analog signal is monitored by the plant computer. Separate from the computer, trip actions are sequence-annunciated in the plant status annunciator. Such sequencing permits the operator to identify readily the protective channel trip actions. Process instrumentation including power, imbalance, flow, temperature, and pressure is indicated on the main control console.

Plant annunciator windows provide the operator with immediate indications of changes in the status of the Reactor Protective System. The following conditions are annunciated for each Reactor Protective System channel:

1. channel trip
2. fan failure in channel
3. channel on test
4. shutdown bypass initiated
5. manual bypass initiated
6. dummy bistable installed

Any time a test switch is in other than the operate position, annunciator (3) will be lit and the associated protection channel will be tripped. Under this condition, annunciator (1) will be lit unless annunciator (5) is lit (i.e., the channel is bypassed).

### 7.2.3 SYSTEM EVALUATION

#### 7.2.3.1 System Logic

The RPS is a four-channel, redundant system in which the four protective channels are brought together in four identical 2-out-of-4 logic networks of the RT modules. A trip in any 2 of the 4 protective channels initiates a trip of all four logic networks. The system to this point has the reliability and advantages of a pure 2-out-of-4 system.

0 Normally, the trip point corresponding to only two pumps in operation is set at 0 percent full power. If
0 two pumps in one loop are lost, a reactor trip will be initiated. Prior to startup with two pumps in one
0 loop, the Power/RC Pumps trip point corresponding to only two pumps in one loop in operation must
0 be increased to 55 percent full power.

#### 7.2.2.3.4 Reactor Outlet Temperature Trip

The reactor outlet temperature is measured by resistance elements. The bridge for each resistance element is considered a part of, and is located within, its associated protective system channel cabinet.

The reactor outlet temperature signal from the temperature bridge passes through a signal converter and then is applied to a bistable trip module. When the temperature exceeds the trip point of the bistable, the
4 bistable trips, de-energizing the protective channel trip relay.

#### 7.2.2.3.5 Pressure-Temperature Trip

Figure 7-3 shows typical operating reactor coolant pressure-temperature boundaries formed by the combined reactor high temperature, high pressure, low pressure, and the pressure-temperature comparator trip settings. The pressure-temperature comparator trips whenever the specified reactor pressure-outlet temperature relationship is exceeded. The comparator forms the boundary line A-B in Figure 7-3.

#### 7.2.2.3.6 Reactor Coolant Pressure Trip

The reactor coolant pressure signal from the pressure transmitter is received by an isolation module in the associated protective channel cabinet. This module acts as a signal conditioner and isolation unit.

Pressure signals go to a high pressure bistable trip module and a low pressure trip module. When the pressure exceeds the trip point of the high pressure bistable, the bistable trips de-energizing the protective channel trip relay.

The low pressure bistable trips when the pressure falls below the trip point, tripping the protective channel trip relay.

#### 7.2.2.3.7 Main Turbine Trip

Pressure switches monitoring the hydraulic fluid pressure in the Turbine Emergency Trip System header will input an open indication to the RPS on turbine trip. Contact buffers located in each RPS channel provide isolation for the RPS System from the field contacts. Upon sensing field contact change state, the contact buffer will initiate an RPS trip when a turbine trip is indicated. This trip is bypassed below a predetermined flux level for unit startup.

#### 7.2.2.3.8 Loss of Main Feedwater Trip

Control oil pressure switches or feedwater header discharge pressure switches for each feedwater pump will input an open indication to the RPS on feedwater pump trip. Isolation contact buffers in the RPS sense the field contact inputs and initiate an RPS trip when both pumps have tripped. This trip is bypassed below a predetermined flux level for unit startup.

#### 7.2.2.3.9 Reactor Building Pressure Trip

Each of the four protective channels receives Reactor Building pressure information from an independent pressure switch. A contact buffer in each protective channel continuously monitors the state of the associated pressure switch. When the state of the pressure switch changes to that corresponding to a

7. When Reactor Building pressure exceeds a fixed maximum limit.

The RPS also automatically trips the reactor to protect the Reactor Coolant System whenever the reactor pressure exceeds a fixed maximum limit.

In addition the RPS automatically trips the reactor upon main turbine trip or trip of both main feedwater pumps.

The abnormal conditions that initiate a reactor trip are keyed to the above listing and tabulated in Table 7-1.

### 7.2.2.3 Description of Protective Channel Functions

The functions of the RPS described below apply to each protective channel.

#### 7.2.2.3.1 Over Power Trip

The nuclear instrumentation provides a linear neutron flux signal in the power range as an indication of reactor power to a protective system bistable trip module.

When the neutron flux signal exceeds the trip point of the bistable, the bistable trips, de-energizing the associated protective channel trip relay.

#### 7.2.2.3.2 Nuclear Over Power Trip Based on Flow and Imbalance

Neutron flux and the reactor coolant flow are continuously monitored. A linear neutron flux signal is received from the nuclear instrumentation and a total reactor coolant flow signal is received from the flow tubes. A power level trip setpoint is established for a bistable trip module (Unit 1) and STAR module (Units 2 and 3) as the percentage reactor coolant flow rate multiplied by the flux to flow ratio. The reactor power imbalance (power in the top half of the core minus the power in the bottom half of the core) reduces the power level trip setpoint such that the four pump power-imbalance boundaries illustrated in Figure 7-2 are not exceeded. Less than four pump power-imbalance protection is provided by the power level trip setpoint decrease due to flow decrease. When the neutron flux signal exceeds the power level trip setpoint established by the total reactor coolant flow and the reactor power imbalance the bistable trips (Unit 1) or the STAR module trips (Units 2 and 3), de-energizing the associated protective channel trip relay.

All flow ΔP cells for a single loop are connected to common 1-inch "low" and "high" lines from the flow tube in that loop. Severance of the "low" line will result in maximum indicated flow for the loop in all four protective channels. All console indicators for the loop will go to 110 percent full flow. Severance of the "high" line will result in zero indicated flow for the loop and possibly a power/flow reactor trip. See Section 7.4.2.3.1, "Failure in RC Flow Tube Instrument Piping" on page 7-34 for more details.

The use of Type C Control rod drive mechanisms on Oconee 3 requires the use of a slightly higher trip delay time for Oconee 3. This difference is accounted for in the affected safety analysis.

#### 7.2.2.3.3 Power/Reactor Coolant Pumps Trip

The reactor coolant (RC) pumps are monitored to determine that they are running. Loss of a single pump initiates four independent signals, one to each protective channel. This information is received by a pump monitor logic which counts the number of RC pumps in operation and identifies the coolant loop in which the pumps are operating. The pump monitor logic output controls the trip point of a power/pump comparator, and initiates a protective channel trip for the conditions in Table 7-1.

logic relay in one reactor trip module. Therefore, each reactor trip module has four logic relays controlled by the four protective channels. The four logic relays combine to form a 2-out-of-4 coincidence network in each reactor trip module. The coincidence logics in all reactor trip modules trip whenever any two of the four protective channels trip.

The reactor trip modules are given the same designation as the protective channel whose trip relay they contain and in whose cabinet they are physically located. Thus, the protective channel A reactor trip module is located in protective channel A cabinet, etc. (Figure 7-1). The coincidence logic in each reactor trip module controls one or more breakers in the control rod drive power system.

The coincidence logic contained in the Reactor Protective System channel A RT module controls breaker A in the Control Rod Drive System as shown in Figure 7-1, channel B RT module controls breaker B, channel C RT module controls breakers C and E, and channel D RT module controls breakers D and F. Breakers A and B control all the 3 phase primary power to the rod drives; breakers C and D control the DC power to rod groups 1 through 4; and breakers E and F control the gating power to rod groups 5 through 8 and the auxiliary power supplies. The control rod drive circuit breaker combinations that initiate a reactor trip can best be stated in logic notation as: AB + ADF + BCE + CDEF. This is a 1-out-of-2 logic used twice and is referred to as a 1-out-of-2 × 2 logic. It should be noted that when any 2-out-of-4 protective channels trip, all reactor trip module logics trip, commanding all control rod drive breakers to trip.

The undervoltage coils of the control rod drive breakers receive their power from the protective channel associated with each breaker. The manual reactor trip switch is interposed in series between each RT module logic and the assigned breakers undervoltage coil.

In response to NRC Generic Letter 83-28 automatic actuation of the AC and DC breaker shunt trip attachments for the Reactor Trip System and Manual Trip Actuation have been installed. This upgrade improves the reactor trip breaker reliability.

For the reactor trip breakers in each channel a relay is installed with its operating coil in parallel with the existing undervoltage device. The output contacts of these relays controls the power to the shunt trip devices. Thus, when power is removed from the breaker undervoltage trip attachment on either a manual or automatic trip signal, the shunt trip attachment is energized to provide an additional means to trip the breaker. Test switches are installed to permit independent testing of the shunt and undervoltage trip devices and silicon controlled rectifiers. Loss of shunt trip control power is annunciated in the control room indicating that the shunt trip device is not operable.

### 7.2.2.2 Summary of Protective Functions

The four Reactor Protective System protective channels are identical in their functions, which combine in the system logic to trip the reactor automatically and protect the reactor core for the following conditions:

1. When the reactor power, as measured by neutron flux, exceeds a fixed maximum limit.
2. When the reactor power, as measured by neutron flux, exceeds the limit set by the reactor coolant flow and power imbalance.
3. When the reactor power exceeds the limit set by the number and combination of reactor coolant pumps in operation.
4. When the reactor outlet temperature exceeds a fixed maximum limit.
5. When a specified reactor pressure-outlet temperature relationship is exceeded.
6. When the reactor pressure falls below a fixed minimum limit.

# 7.2 REACTOR PROTECTIVE SYSTEM

0 Note

0 This section of the FSAR contains information on the design bases and design criteria of this
0 system/structure. Additional information that may assist the reader in understanding the system is
0 contained in the design basis document (DBD) for this system/structure.

The Reactor Protective System (RPS) monitors parameters related to safe operation and trips the reactor to protect the reactor core against fuel rod cladding damage. It also assists in protecting against Reactor Coolant System damage caused by high system pressure by limiting energy input to the system through reactor trip action.

## 7.2.1 DESIGN BASES

The Reactor Protective System includes all design basis features of Section 7.1.2, "Identification of Safety Criteria" on page 7-3 with the following additions:

### 7.2.1.1 Loss of Power

A loss of power to a reactor protective channel will cause that protective channel to trip.

### 7.2.1.2 Equipment Removal

The Reactor Protective System initiates a protective channel trip whenever a module or subassembly is removed from the equipment cabinet. Removing a reactor trip module causes the associated control rod breaker to trip.

### 7.2.1.3 Diverse Means of Reactor Trip

In the unlikely event of a systematic or complete failure of the Reactor Coolant System low pressure signals to trip the reactor following the initiation of emergency core cooling, there is a separate, diverse means of assuring reactor trip. A high pressure in the Reactor Building is independently sensed by four sensors, and independent signals are fed from these sensors to the four Reactor Protective System channels to provide the desired diverse reactor trip signal.

## 7.2.2 SYSTEM DESIGN

### 7.2.2.1 System Logic

The system as shown in Figure 7-1 consists of four identical protective channels, each terminating in a trip relay within a reactor trip (RT) Module. In the normal untripped state, each protective channel functions as an AND gate, passing current to the terminating relay and holding it energized as long as all inputs are in the normal energized (untripped) state. Should any one or more inputs become de-energized (tripped), the terminating relay in that protective channel becomes an OR gate.

Each of the four protective channels terminates in a channel trip relay within a reactor trip module. There are four such modules. Each protective channel trip relay has four contacts, each controlling a

Vital Pwr Pnlbd 1KVIC

RPS Ch C

ESG Channel Even-Odd

DC Pnlbd 1DID

Vital Pwr Pnlbd 1KVID

RPS Ch D

### 7.1.2.6 Manual Trip

Manual trip switches, independent of the automatic trip instrumentation, are provided.

### 7.1.2.7 Testing

Manual testing facilities are built into the protective systems to provide for:

1. Preoperational testing to give assurance that a protective system can fulfill its required protective functions.
2. On-line testing to prove operability and to demonstrate reliability.
3. BWNT STAR module provides both manual and automated test capability, and self diagnostic tests performed during start-up and operation. The front panel of the STAR module has LED indicators which indicate module status.

## 7.1.3 IDENTIFICATION OF PROTECTIVE EQUIPMENT

All safety related sensors, transmitters, transducers, cabinets, etc. located outside the control room are physically identified by placement of a permanent, conspicuous tag on or adjacent to the device. A typical tag bears the wording "Safety Related." The following are examples of equipment that should be tagged:

Swgr 1TC

LD Ctr 1X8

MCC 1XS1

ESG channel 1, 3, 5, & 7

DC Pnlbd 1DIA

Vital Pwr Pnlbd 1KVIA

RPS Ch A

Swgr 1TD

LD Ctr 1X9

MCC 1XS2

ESG channel 2, 4, 6, & 8

DC Pnlbd 1DIB

Vital Pwr Pnlbd 1KVIB

RPS Ch B

Swgr 1TE

MCC 1XS3

DC Pnlbd 1DIC

# 7.1 INTRODUCTION

Instrumentation and control systems include the Reactor Protective System, the Engineered Safeguards Protective Systems, the Rod Drive Control System, the Integrated Control System, the Nuclear Instrumentation System, the Non-Nuclear Instrumentation System, and the Incore Monitoring System.

## 7.1.1 IDENTIFICATION OF SAFETY-RELATED SYSTEMS

The protective systems, which consist of the Reactor Protective System and the Engineered Safeguards Protective Systems, perform important control and safety functions. The protective systems extend from the sensing instruments to the final actuating devices, such as circuit breakers and pump or valve motor contactors.

## 7.1.2 IDENTIFICATION OF SAFETY CRITERIA

### 7.1.2.1 Design Bases

The protective systems are designed to sense plant parameters and actuate emergency actions in the event of abnormal plant parameter values. They meet the intent of the Proposed IEEE "Criteria for Nuclear Power Plant Protection Systems" dated August, 1968. (IEEE No. 279).

### 7.1.2.2 Single Failure

The protective options meet the single failure criterion of IEEE No. 279 to the extent that:

1. No single component failure will prevent a protective system from fulfilling its protective functions when action is required.
2. No single component failure will initiate unnecessary protective system action where implementation does not conflict with the criterion above.

### 7.1.2.3 Redundancy

All Reactor Protective System functions are implemented by redundant sensors, measuring channels, logic, and actuation devices. These elements combine to form the protective channels.

### 7.1.2.4 Independence

Redundant protective channels are electrically independent and are packaged to provide physical separation.

### 7.1.2.5 Separation

Protective channels are physically separate and are electrically isolated from regulating instrumentation. Only one string of instrumentation may be selected at a given time for use in a system control function, and electrical isolation is assured through the use of isolation amplifiers. A fifth channel of regulating instrumentation not associated with protection is normally employed for control purposes.

# CHAPTER 7. INSTRUMENTATION AND CONTROL

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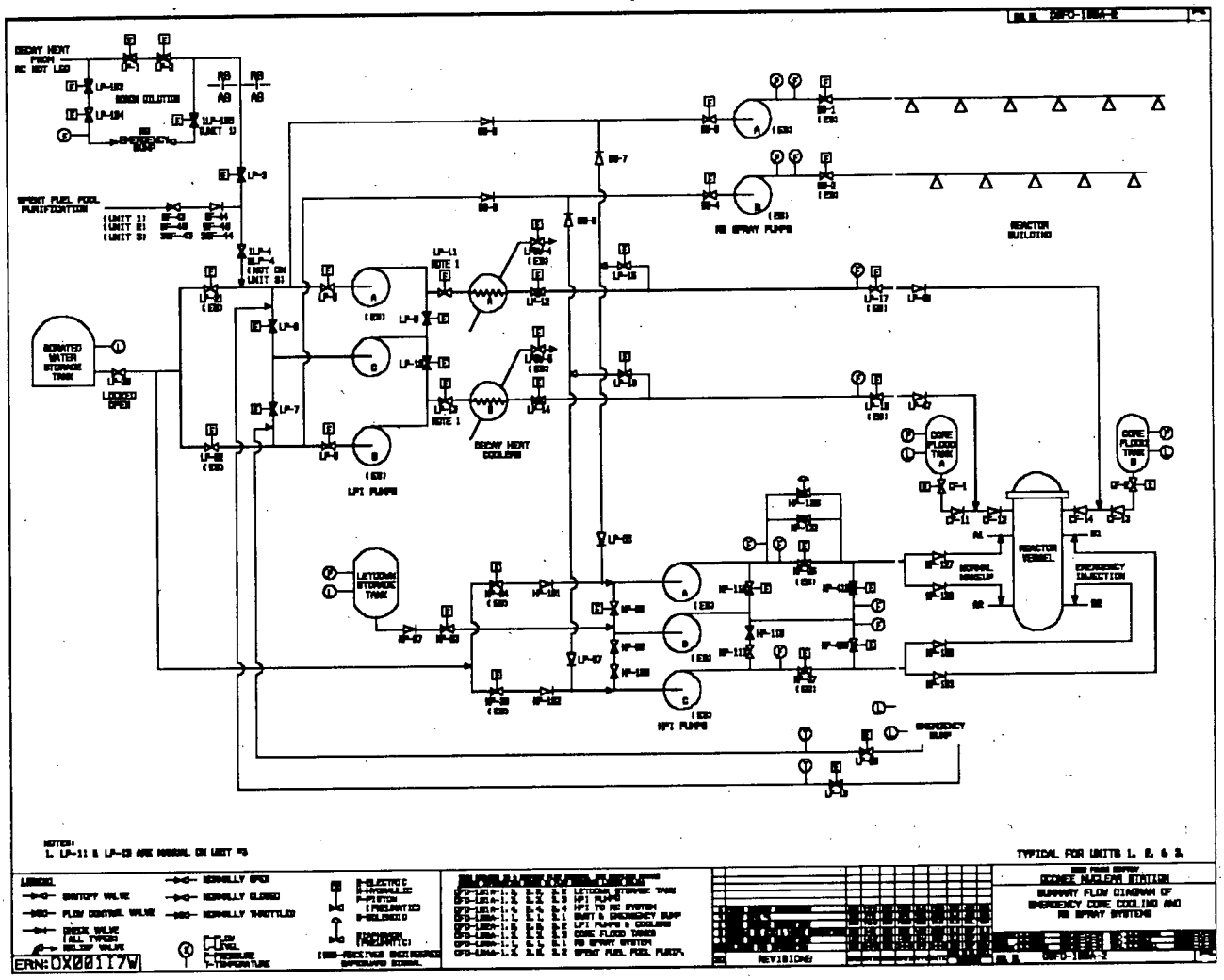


Figure 6-1.
Flow Diagram of Emergency Core Cooling Systems

5 **Table 6-27. Summary of Calculated Containment Pressures and Temperatures for Secondary System Pipe Rupture Cases**

5
5
5

| Break Location | Break Size | Peak Pressure (psig) | Peak Temperature (F) | Time of Peak Pressure (sec) | Energy Released to Containment up to End of Blowdown ($10^6$Btu) |
|---|---|---|---|---|---|
| S/G outlet | 12.6 | 58 | 397 | 200 | 221.2 |

**Table 6-22. Containment Response Analyses Initial Conditions**

| | | | |
|---|---|---|---|
| **General Information** | | | |
| External containment design pressure (psig) | 59 | | |
| Internal containment design pressure (psig) | 3.0 | | |
| Containment design temperature (°F) | | | |
| Containment free volume (ft³) | 179,000,000 | | |
| Containment design leak rate (/1st day at peak accident pressure) | 0.25% | | |
| | **Large LOCA Short Term** | **Large LOCA Long-Term** | **Steam Line Break** |
| **Initial Conditions** | | | |
| Reactor power (MWt @ 102% design overpower) | 2619 | 2619 | 2619 |
| Average reactor coolant temperature (°F) | 581 | 581 | 581 |
| Containment pressure (psig) | 1.5 | 0 | 1.5 |
| Containment atmosphere temperature (°F) | 110 | 125 | 125 |
| Containment atmosphere relative humidity (%) | 0 | 100 | 0 |
| Low pressure service water temperature (°F) | N/A | variable | N/A |
| Borated water temperature (°F) | N/A | 115 | 120 |
| Borated water storage tank level (ft) | N/A | 44.3 | N/A |
| Core flood tank volume (both, ft³) | 1944 | 1944 | 1940 |

**Table 6-18. Inventory of Iodine Isotopes in Reactor Building (at t = o)**

| Isotope | Initial Inventory |
|---|---|
| | Curies/MWt |
| Iodine 131 | $2.51 \times 10^4$ |
| Iodine 132 | $3.81 \times 10^4$ |
| Iodine 133 | $5.63 \times 10^4$ |
| Iodine 134 | $6.58 \times 10^4$ |
| Iodine 135 | $5.10 \times 10^4$ |

**Table 6-19. Single Failure Analysis for Reactor Building Penetration Room Ventilation System**

| Component | Malfunction | Comments |
|---|---|---|
| 1. Fan | Fails | The other fan retains full capacity. |
| 2. Fan Discharge Valve | Fails to open | The other fan retains full capacity. |
| 3. Fan Discharge Valve | Fails to close | Check valve prevents recirculation. |
| 4. Vacuum Relief Valve | Failure to open. | The other vacuum relief valve opens. |
| 1 5. PR-13 and PR-17 | Loss of air to remote manual loaders. | Valves fail as-is. |
| 6. PR-20 | Loss of air to remote manual loaders. | Valve stays shut. |

2 **Table 6-20. Parameters for Boron Precipitation Analysis**

| | | |
|---|---|---|
| 3 | Initial Reactor Core Power | 2568 MWth |
| 2 | LPI Flow Rate | 402 lbm/sec |
| 2 | Core Mixing Mass | 60000 lbm |
| 3 | LPI Injection Enthalpy | 123 BTU/lbm |
| 3 | Containment Pressure | 25 psia |
| 3 | BWST Boron Concentration | 3000 ppm |
| 2 | CFT Boron Concentration | 4000 ppm |

**Table 6-6. Single Failure Analysis For Reactor Building Cooling System**

| Component | | Malfunction | Comments |
|---|---|---|---|
| 1. | Circulating fan | Fails to operate. | The cooling capacity of the cooling units is reduced; however, the Reactor Building Spray System provides separate full capacity cooling. |
| 2. | Cooler service water outlet valve. (LPSW-18, -21, -24) | Fails to open fully | Valve will normally be partially open. If the valve fails to open fully, the unit will operate under reduced heat removal capability. The Reactor Building Spray System provides full heat removal capability. |
| 3. | Cooler service water inlet valve. (LPSW-16, -19, -22) | Inadvertently left closed. | The flow through this string will be unavailable for cooling. It is unlikely that this condition would occur during an accident since the position and flow are monitored during normal operation. The Reactor Building Spray System will provide the required cooling. |
| 4. | Service water pump (1A, 1B, 1C). | Fails to operate. | The two remaining pumps will provide full low pressure service water flow to all components. |
| 5. | Service water pump (3A, 3B). | Fails to operate. | The one remaining pump will provide full low pressure service water flow to all components. |

## 6.6.7 SYSTEM PRESSURE TESTS

5 Classes 2 and 3 system pressure testing complies with Section XI Articles IWC-5000 and IWD-5000 in effect as stated in Section 6.6.1, "Components Subject to Examination."

## 6.6.8 AUGMENTED INSERVICE INSPECTION TO PROTECT AGAINST POSTULATED PIPING FAILURES

Class 2 high energy fluid piping systems will be inspected in accordance with Article IWC-2000 of Section XI up to the isolation valve outside containment. The examination areas, methods, extent, and frequency will be as specified in Article IWC-2000. Those lines requiring augmented inservice inspection will be contained in the Oconee Nuclear Station Inservice Inspection Plan.

**THIS IS THE LAST PAGE OF THE CHAPTER 6 TEXT PORTION**

# 6.6 INSERVICE INSPECTION OF CLASS 2 AND 3 COMPONENTS

## 6.6.1 COMPONENTS SUBJECT TO EXAMINATION

Class 2 and 3 components, indicated in the Oconee Inservice Inspection Plan, are equivalent to Quality Groups B and C respectively of Regulatory Guide 1.26. These components will be examined in accordance with the provisions of the ASME Boiler and Pressure Vessel Code Section XI in effect as specified in 10CFR50.55a(g) to the extent practical. Requests for relief from inservice inspection requirements determined to be impractical will be submitted to the NRC for review in accordance with NRC guidelines for submitting such requests.

## 6.6.2 ACCESSIBILITY

Class 2 and 3 systems at Oconee were installed before any inservice inspection requirements existed for these systems. In most cases adequate clearance is available to perform the inspection required by Section XI. In cases where adequate clearance is not available, the use of alternate inspection techniques will be investigated. If no alternate techniques appear practical, relief will be requested.

## 6.6.3 EXAMINATION AND PROCEDURES

The examination techniques to be used for inservice inspection include radiographic, ultrasonic, magnetic particle, liquid penetrant, eddy current, and visual examination methods. For all examinations, both remote and manual, specific procedures will be prepared describing the equipment, inspection technique, operator qualifications calibration standards, flaw evaluation, and records. These techniques and procedures will meet the requirements of the Section XI edition in effect as stated in Section 6.6.1, "Components Subject to Examination."

## 6.6.4 INSPECTION INTERVALS

The inservice inspection interval for ASME Class 2 and 3 components is 10 years. The inspection schedule will be developed in accordance with IWC-2400 and IWD-2400. Detailed inspection listings and scheduling will be contained in the Oconee Inservice Inspection Plan.

## 6.6.5 EXAMINATION CATEGORIES AND REQUIREMENTS

The examination categories to be used are those listed in Tables IWC-2500-1 and IWD-2500-1 of ASME Section XI. Specific examinations will be identified by an Item Number, composed of the Item Number assigned in Tables IWC-2500-1 and IWD-2500-1 of ASME Section XI, plus an additional number to uniquely identify that examination.

## 6.6.6 EVALUATION OF EXAMINATION RESULTS

Evaluation of examination results shall be in accordance with the Section XI in effect as stated in Section 6.6.1, "Components Subject to Examination" where these evaluation standards are contained in Section XI. For examination where evaluation standards are not contained in Section XI, evaluation shall be performed in accordance with the original construction code.

## 6.5.4 REFERENCES

1. Cottrell, W. B. and Savolainen, A. W., Editors, U. S. Reactor Containment Technology, *ORNL-NSIC-5, Volume II.*

#### 6.5.1.6 Materials

Carbon steel and suitable coatings are used to obtain desired service life.

### 6.5.2 CONTAINMENT SPRAY SYSTEMS

4 No credit is taken for this system for fission product removal or control in LOCA analysis (see 15.14.7,
4 "Environmental Evaluation" on page 15-64). Credit is taken for this system for fission product removal
4 in the MHA off-site dose analyses only. (see 15.15.1, "Identification of Accident" on page 15-69).

### 6.5.3 FISSION PRODUCT CONTROL SYSTEMS

5 Credit is taken for fission product control by the Containment Hydrogen Recombiner System which is
5 addressed in Section 15.16, "Post-Accident Hydrogen Control" on page 15-73; however, this system is
5 not considered an Engineered Safeguards System.

The Reactor Building penetration room is maintained at a negative pressure of greater than 0.06 in. $H_20$
1 with respect to the outside atmosphere when the penetration room fans are in operation.

Even in the event of unfiltered leakage of all the iodine input to the penetration room due to high wind velocity, the improvement in atmospheric dilution more than compensates for bypassing of the penetration room filter by this portion of the iodine. At a wind velocity of greater than 8.1 mph, the improvement in X/Q compensates for the complete loss of the filtering system in the calculation of offsite dose. A wind velocity of 8.1 mph will cause a reduction in pressure of .032 in. $H_20$ along the penetration room wall. (This assumes that wind velocity is exactly parallel to the wall which is the worst case assumption). By maintaining the penetration room at a negative pressure of 0.06 in. $H_20$, a conservative margin of pressure is established.

The equipment in this system is designed and rated in accordance with the following standards:

Pre-Filter- Filter efficiency is determined by the "American Filter Institute Dust Spot Test" utilizing atmospheric dust.

Absolute Filter- The basic design criteria for this filter is set forth in AEC Health and Safety Bulletin 212 (6-25-65) which incorporates U.S. Military Specification MIL-F-51068A captioned "Filter, Particulate, High Efficiency, Fire Resistant".

In addition, the dust holding capacity is determined by utilizing the test procedures of AFI "Code of Testing Air Cleaning Devices Used in General Ventilation", Section I (1952).

Adsorptive (Carbon) Filter- The specified ignition temperature of the carbon is checked using the methodology of ASTM D-3466. This test is conducted on one sample from each lot of carbon.

Fans- Fan performance is determined by prototype test according to procedures set forth by the Air Moving and Conditioning Association (AMCA) 1960 Standard Test Code.

### 6.5.1.4 Tests and Inspections

The Reactor Building PRVS may be actuated during normal operation for testing and inspection. The high efficiency particulate air (HEPA) filters and the charcoal iodine filters are tested to ensure that they are able to remove airborne materials from penetration leakage.

Sight glasses in the PRVS drain lines and humidity sensors are available for monitoring the penetration room humidity. (Procedures are implemented to monitor the humidity, and prompt action is to be taken
3 to reduce the humidity to less than 80 percent when these values are exceeded.) External carbon sample canisters are installed on the filters to facilitate sampling. Provision is made to check penetration room negative pressure relative to either the Auxiliary Building or the outside.

Testing and inspection of the system is as required by the Technical Specifications.

### 6.5.1.5 Instrumentation Requirements

Instrumentation is used only to monitor system performance and has no control function other than to guide the operator in adjusting the final control elements.

Penetration room pressure and humidity and loss of air flow through either filter are monitored.

penetration room are very large in comparison with the minute leakage that might exist due to imperfect seals.

From time to time the system will be activated to purge the filters of any moisture that may accumulate. The air will be taken from the penetration room where it will be sufficiently warm to accomplish this purpose. Dampers are placed in the system inlets to prevent moisture from being carried through by natural circulation.

The only penetrations which do not pass through the penetration rooms are:

5 1. Reactor Building Fuel Transfer Tube and Reactor Coolant SSF Makeup (Penetration No. 11a).
5 2. Reactor Building Fuel Transfer Tube and Reactor Coolant SSF Letdown (Penetration No. 12a).
5 3. One Main Steam Line per unit --1B (Penetration No. 28) and 2A & 3A (Penetration No. 26).
5 4. Normal Personnel Access Lock (Penetration No. 90).
5 5. Permanent Equipment Hatch which contains a double-gasketed closure (Penetration No. 91).
5 6. Emergency Personnel Access Lock (Penetration No. 92).
5 7. Reactor Building Normal Sump Drain (Penetration No. 5a).
5 8. Reactor Coolant Post-Accident Liquid Sample Lines (Penetration No. 5b).
5 9. Reactor Coolant Quench Tank Drain (Penetration No. 29).
5 10. Reactor Building Emergency Sump Recirculation A (Penetration No. 36).
5 11. Reactor Building Emergency Sump Recirculation B (Penetration No. 37).
5 12. Reactor Building Emergency Sump Drain (Penetration No. 40).
5 13. Reactor Coolant Decay Heat Drop Line and Post-Accident Boron Dilution Line (Penetration No. 62
5 -- Units 2 and 3 only).

5 Penetrations 7 through 13 are embedded lines.

The main steam lines are not considered a source of significant leakage because they are welded to the liner plate. The access openings can be tested during normal operation and are not considered sources of significant leakage. There are double seals at each of these access openings, and the space between these double seals is connected to the penetration room. The refueling tube is equipped with a blind flange which is only opened during shutdown for transfer of fuel to the spent fuel pool.

### 6.5.1.3 Design Evaluation

A single failure analysis of the various portions of this system is presented in Table 6-19.

Redundant fans, cross connected piping, and locked open filter inlet valves render incredible a loss of
3 cooling air flow to the filters. However, even if air flow is lost through a filter, Calculation OSC-4024
3 (Performed for PIR #4-090-0057) has shown charcoal ignition temperature will not be reached.

3

Adequate instrumentation is provided to detect loss of air flow through either filter. Reduction in air flow below a preset minimum would result in low Penetration Room vacuum and cause an alarm in the control room. Flow indication with readout outside the penetration filter area is furnished for each filter. Following an accident, filter instrumentation is monitored by an operator every four hours.

# 6.5 FISSION PRODUCT REMOVAL AND CONTROL SYSTEMS

The systems addressed below reduce accidental release of fission products following a design basis accident.

## 6.5.1 ENGINEERED SAFEGUARDS (ES) FILTER SYSTEMS

Included in this section is a discussion of the Reactor Building Penetration Room Ventilation System.

### 6.5.1.1 Design Bases

The Reactor Building Penetration Room Ventilation System (PRVS) is designed to collect and process potential Reactor Building penetration leakage to minimize environmental activity levels resulting from post-accident Reactor Building leaks. Experience (Reference 1 on page 6-59) has shown that Reactor Building leakage is more likely at penetrations than through the liner plates or weld joints.

The design basis for filtration was a requirement to remove 25 percent of the core iodine inventory. The 25 percent was derived using the standard assumption that during a Maximum Hypothetical Accident 50 percent of the halogens are released from the core and that 50 percent of the iodine released plates out within the Reactor Building. The initial inventory of the individual isotopes in terms of Curies/MWt is given in Table 6-18.

### 6.5.1.2 System Design

This section addresses the design only as related to fission product removal. More details of system design and operation are addressed in Section 9.4.7.2, "System Description" on page 9-62.

The system schematic and characteristics are shown on Figure 6-4 and Figure 6-22, respectively. Figure 6-23 and Figure 6-24 show penetration and opening locations in the penetration rooms. Mechanical openings, electrical openings, and construction details are illustrated in Figure 6-25, Figure 6-26, and Figure 6-27, respectively.

Penetration rooms are formed adjacent to the outside surface of each Reactor Building by enclosing the area around the majority of the penetrations.

Each unit's penetration room is provided with two fans and two filter assemblies. Both fans, discharging
1 through a single line to the unit vent, may be controlled from the main control room.

During normal operation, this system is held on standby with each fan aligned with a filter assembly. The engineered safeguards signal from the Reactor Building will actuate the fans. Control room instrumentation monitors operation.

Particulate filtration is achieved by a medium efficiency pre-filter and a high efficiency (HEPA) filter. Adsorption filtration is accomplished by an activated charcoal filter. When the system is in operation, a
0 negative pressure will be maintained in the penetration room to ensure inleakage. Penetration room pressure is displayed in the control room and excessive and insufficient vacuum are annunciated. It can be assumed that no pressure differentials exist in the room, so that an instrument string sensing pressure at a single point can be used. This is because the communicative paths between various parts of the

## 6.4.5 REFERENCES

1. J. F. Stolz (NRC) to H. B. Tucker (Duke) November 24, 1986.

3 outside air filter trains can maintain their respective control room zones at a positive pressure to prevent
3 uncontrolled infiltration into the control room zones.

### 6.4.2.4 Interaction With Other Zones and Pressure-Containing Equipment

The control room envelope is bounded on the north, south, and west by the Auxiliary Building and on the east by the Turbine Building. The Ventilation Systems serving the Auxiliary Building and Turbine
0 Buildings are separate from the Control Room Ventilation System.

3 Interaction with other areas is minimal as air for pressurizing the Control Room Zone is taken from outside and is filtered through charcoal filters to eliminate airborne radioactive contaminants.

Pressure retaining equipment generally is not permitted in the control room zone. Exceptions to this are several hand held fire extinguishers for local fire control and several self-contained breathing apparatus with additional bottles of replenishment air.

### 6.4.2.5 Toxic Gas Protection

Chlorine gas is used in the Water Treatment System. Other gases used on site are Ammonia, Hydrazine, Hydrogen, Liquid Nitrogen, and welding gases. No potential sources of toxic gas releases were identified off site. Protection of control room operators against potential toxic gas release accidents has been found to be adequate by the NRC (Reference 1 on page 6-53).

Self-contained type breathing apparatus are available to operator personnel. The Oconee 1 and 2 Control Room has six apparatus with twelve refill bottles and the Oconee 3 Control Room has three apparatus with six refill bottles.

## 6.4.3 TESTING AND INSPECTION

0 The Control Room Ventilation System is normally operable and is accessible for periodic inspection. The
0 pressurization portion of the system is tested periodically to demonstrate its readiness and operability as required by the Technical Specifications.

## 6.4.4 INSTRUMENTATION REQUIREMENTS

Sufficient indications in the form of status lights and performance readouts are provided in the control
0 room to evaluate system operation and indicate system malfunctions.

0 A radiation monitor is located in the return air side of the Control Room Ventilation System as described in Section 9.4.1.1, "Design Bases" on page 9-53.

# 6.4 HABITABILITY SYSTEMS

## 6.4.1 DESIGN BASES

Oconee Nuclear Station's design pre-dates General Design Criterion 19 (GDC-19) of Appendix A to 10 CFR 50, however control room habitability was a design consideration at Oconee as discussed in Section 3.0.1.

The Oconee Nuclear Station control rooms are located in the Auxiliary Building. Oconee 1 and 2 have a shared control room while Oconee 3 has a separate control room. Figure 6-19 shows the location of the two control rooms with regard to other major structures of the station. Figure 6-20 and Figure 6-21 show the Oconee 1 and 2 and Oconee 3 control room general arrangement, respectively.

The facility is provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection is provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10CFR20 limits. The control room shielding meets the NUREG-0578 requirements. It is possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost.

## 6.4.2 SYSTEM DESIGN

### 6.4.2.1 Definition of Control Room Envelope

The control room envelope includes the control room and all rooms the control room personnel may require access to during emergency plant operation. This envelope is designated as the Control Room
0 Zone and is comprised of the Offices, Computer Rooms, Operator's Break Area, and Operator's Toilet Room.

All controls and displays necessary to bring the plant to a safe shutdown condition are included within the control room envelope.

### 6.4.2.2 Ventilation System

0 The Control Room Ventilation System is described in detail in Section 9.4.1, "Control Room Ventilation" on page 9-53. The ventilation system was designed and installed in accordance with HVAC Industry Standards and practices for commercial and industrial systems.

### 6.4.2.3 Leak Tightness

0 Outside air filter trains are provided as part of the Control Room Ventilation System to provide filtered
3 pressurization air to offset the exfiltration from the control room zone. This minimizes uncontrolled
3 infiltration into the control room zone by creating a positive pressure with respect to adjacent zones.

0

0 The Oconee 1 and 2, and Oconee 3 Control Room Ventilation Systems are designed as independent
3 ventilation systems; ie, independent of the Cable and Electrical Equipment Rooms. Two 50% capacity

## 6.3.6 REFERENCES

1. Qualification test of Limitorque valve operator, motor brake, and other units in a simulated reactor containment post-accident environment, Final Report F-C3327, July, 1972.
2. Qualification test of Limitorque valve operators in a simulated reactor containment post-accident steam environment, Final Report F-C3441, September 1972.
3. Agar, J. D. (B&W), letter to Swindlehurst, G. B. Oct. 19, 1989 DPC 89-124.
3 4. B&W Calculation 86-1176201-01 "187FA-LL 48 Second ECCS delay time," dated 10-13-89.
4 5. Instruction Manual for Rotork Valve Actuators, OM-245-1023.

inspected for leaks. Items for inspection will be pump seals, valve packing, flange gaskets, heat exchangers, and safety valves for leaks to atmosphere.

## 6.3.5 INSTRUMENTATION REQUIREMENTS

The High Pressure Injection System is actuated automatically by a low Reactor Coolant System pressure of 1,600 psig (1500 Technical Specification value) or by a Reactor Building pressure of 3 psig (4 psig Technical Specification value). All of the pumps and valves can also be remotely operated from the control room. In the event valve operators are not functional for ES valves on the HPI pump suction,
3 letdown or seal return, these valves may be left in their ES position during operation provided control of
3 normal plant parameters is not inhibited. Flow instrumentation is available in each HPI train during an
3 accident.

The Low Pressure Injection System is automatically actuated by a low Reactor Coolant System pressure of 550 psig (500 psig Technical Specification value) or Reactor Building pressure of 3 psig (4 psig Technical Specification value). All of the pumps and automatic valves can also be remotely operated from the control room. In the event valve operators are not functional for ES valves on the LPI pump suction, these valves may be left in their ES position during operation.

The Core Flooding System is actuated at a Reactor Coolant System pressure of 600 psig. At this point the differential pressure across the inline check valves allows them to open releasing the contents of the tanks into the reactor vessel.

The Engineered Safeguards Actuation instrumentation for the Emergency Core Cooling System is provided with redundant channels and signals as described in Chapter 7, "Instrumentation and Control" on page 7-1. The control room layout is arranged so that all indicators and alarms are grouped in one sector at a convenient location for viewing. Switches and controls are also located conveniently.

2 The results of the analysis show the maximum allowable boric acid concentration establishd by the NRC,
2 which is the boric acid solubility limit minus 4 weight percent, will not be exceeded in the vessel if a
2 boron dilution flow of 40 gpm (Reference 20 on page 15-66) from the hot leg to the sump is initiated
2 within 9 hours following a LOCA.

2 Since there are redundant methods to establish this dilution flow, no diverse means is required to be
2 provided to prevent the buildup of boron concentration. All components of the ECCS are ANS Safety
2 Class 2 and Seismic Category 1.

#### 6.3.3.3 Loss of Normal Power Source

1 Following a loss-of-coolant accident assuming a simultaneous loss of normal power sources to the LOCA
1 unit, the emergency power source and both the Low Pressure and High Pressure Injection Systems will be
3 in full operation within 48 seconds after actuation, even assuming a single failure. The electrical power
1 system design is based on the assumption that ESG actuation in one unit occurs simultaneously with a
1 loss of offsite power to all three units. However, accident scenarios in FSAR Section Chapter 15,
3 "Accident Analyses" on page 15-1 assume loss of offsite power to the LOCA unit only. All calculations
3 for the Oconee Units have assumed a 48 second delay from receipt of the actuation signal to start of flow for both the HPI and LPI Systems. Upon loss of normal power sources including the startup source and initiation of an engineered safeguards signal, the 4160 volt engineered safeguards power line is connected to the underground feeder from Keowee hydro (Section 8.3.1, "AC Power Systems" on page 8-9). The Keowee hydro unit will start up and accelerate to full speed in 23 seconds or less. An analysis has shown that by energizing the HPI and LPI valves (which have opening times of 14 seconds and 15 seconds
3 respectively at normal bus voltage) and pumps after a 10 second swapover time (required by the single
3 failure), the design injection flow rate (HPI - 450 gal/min, LPI - 3000 gal/min) will be obtained within 48 seconds.

#### 6.3.3.4 Single Failure Assumption

3 Previously, the worst single failure for a LOCA and loss of offsite power was assumed to be the loss of one bus of emergency power (which results in the loss of one train of HPI and one train of LPI). This single failure scenario has been analyzed assuming a delay time of 35 seconds from ES actuation signal to delivery of flow to the RCS. The failure of transformer CT-4 has been identified as a more limiting single failure for the large break LOCA (Reference 3 on page 6-50). With the assumed loss of offsite power, this single failure results in a 48 second delay from ES actuation to delivery of flow to the RCS (Reference 3 on page 6-50). Both trains of ECCS would be available at 48 seconds. Reference 3 on page 6-50 demonstrates that having both ECCS trains injecting at a later time is more limiting than having one ECCS train injecting at an earlier time. The net effect of the changed single failure assumption is less than a 50°F increase in the peak cladding temperature.

### 6.3.4 TESTS AND INSPECTIONS

#### 6.3.4.1 ECCS Performance Tests

Table 6-15 summarizes performance testing for the Emergency Core Cooling System.

#### 6.3.4.2 Reliability Tests and Inspections

All active components, listed in Table 6-15, of the Emergency Injection System will be tested periodically to demonstrate system readiness. The High Pressure Injection System will be inspected periodically during normal operation for leaks from pump seals, valve packing, and flanged joints. During operational testing of the low pressure injection pumps, the portion of the system subjected to pump pressure will be

Injection response of the Core Flooding System is dependent upon the rate of reduction of Reactor Coolant System pressure. For the maximum pipe break (14.1 $ft^2$), the Core Flooding System is capable of reflooding the core to the hot spot in less than 25 seconds after a rupture has occurred.

Special attention has been given to the design of core flooding nozzles to assure that they will take the differential temperature imposed by the accident condition. Special attention has also been given to the ability of the injection lines to absorb the expansion resulting from the recirculating water temperature.

The gravity flow path from the reactor outlet piping to the Reactor Building emergency sump will maintain a minimum core flow in excess of 40 gal/min to assure boric acid solubility. The flow path is
3 open within 9 hours following a large LOCA.

The Low Pressure Injection System is connected with other safeguards systems in three respects, i.e., (1) the High Pressure and Low Pressure Injection Systems and the Reactor Building Spray System take their suction from the borated water storage tank; (2) the low pressure injection pumps and the Reactor Building spray pumps share common suction lines from the Reactor Building sump during the coolant recirculation mode; and (3) the Low Pressure Injection System and the Core Flooding System utilize common injection nozzles on the reactor vessel.

2 **6.3.3.2.1 Boron Precipitation Evaluation**

2 In response to the RCS depressurization associated with a LOCA, the ECCS actuates and begins injecting
2 borated water into the system to reflood the core, keep the reactor subcritical, and provide for long term
2 cooling. The boiloff of the ECCS delivered water along with flow stagnation in the reactor vessel can
2 result in an increase in the boron concentration. If unrealized, this process could result in localized
2 recrystallization of the boric acid and the potential for deposits to build up on the fuel assemblies and
2 internals and hinder effective heat removal. In order to prevent this occurrence, analytically based
2 operating procedures have been developed to assure sufficient circulation and dilution of the coolant.

4 In the initial long term phase of post-LOCA heat removal, a natural circulation flowpath from the core
4 through the vent valves to the downcomer occurs which sufficiently circulates the coolant through the
4 core. At some point in time the flowpath through the vent valves will no longer be available as the decay
4 heat becomes insufficient to drive the flow. In addition, natural circulation flow through the gaps between
4 the reactor vessel hot leg nozzles and the reactor internals has also been evaluated to be available.
5 Operator action must be taken to initiate at least one of the two gravity flow paths to provide further
5 assurance that flow is established and post-LOCA boric acid solubility is maintained. The method for
2 performing this function is by means of a drain line from the hot leg to the Reactor Building sump which
2 draws coolant from the top of the core, thereby inducing core circulation. The system has been designed
2 with redundant drain lines and has been shown to be single failure proof. The boron concentration of the
2 liquid leaving through the drain line is equal to the core boron concentration. Most of the core decay
2 heat is removed by steam flow through the vent valves. ECCS pump flow will continue to be provided to
2 the RCS cold legs and will preclude any boron concentration buildup in the vessel for breaks in the hot
2 leg.

2 An analysis has been performed to determine the maximum boron concentration in the reactor vessel
2 following a hypothetical LOCA. This analysis uses the methodology and assumptions described in
2 Reference 20 on page 15-66 with the principal input parameters given in Table 6-20. The analysis
2 considers the increase in boric acid concentration in the reactor vessel during the long term cooling phase
3 of a LOCA, conservatively assuming no liquid flow will occur through the hot leg nozzle gaps during a
2 cold leg break LOCA (B&W PSC 2-91). The calculation of boric acid concentration in the reactor vessel
2 considers a cold leg break of the RCS in which steam is generated in the core from decay heat while the
2 boron associated with the boric acid solution is completely separated from the steam and remains in the
2 effective vessel volume.

transmitters associated with a LOCA. The non-metallic materials selected for the electric motor valve operators based on irradiation testing are: melamine used in the limit switches (all plastic material used is melamine), viton for all seals, Humble Nebula EP #1 as the lubricant, and Class "H" insulation for the motor.

## 6.3.3 PERFORMANCE EVALUATION

In establishing the required component redundancy for the Emergency Core Cooling System, several factors related to equipment availability were considered:

a. The probability of a major Reactor Coolant System failure is very low; i.e., the probability that the equipment will be needed to serve its emergency function is low.

b. The fractional part of a given component lifetime for which the component is unavailable due to maintenance is estimated to be very small. On this basis, the probability that a major Reactor Coolant System accident would occur while a component from the Emergency Core Cooling System was out of service for maintenance is several orders of magnitude below the low basic accident probability.

c. The maintenance period for important equipment can usually be scheduled for a period of time when the reactor is shut down. Where maintenance of an engineered safety feature component is required during operation, the periodic test frequency of the similar redundant components can be increased to insure availability.

d. Where the systems are designed so that the components serve a normal function in addition to the emergency function or where meaningful periodic tests can be performed, there is also a low probability that the required emergency action would not be performed when needed; i.e., equipment reliability is improved by using the equipment for other than emergency functions.

### 6.3.3.1 High Pressure Injection System (HPI)

One high pressure injection string can deliver 450 gal/min at 585 psig reactor vessel pressure. The safety
0 analysis in Chapter 15, "Accident Analyses" on page 15-1 has shown that two high pressure injection
0 pumps through two injection trains are sufficient to prevent core damage for those smaller leak sizes which do not allow the Reactor Coolant System pressure to decrease rapidly to the point where the Low Pressure Injection System is initiated.

4 After receiving an actuation signal, the HPI system valves for injection will open sufficiently to admit the
4 required flow within 14 seconds and the HPI pumps will reach full speed within 6 seconds. One of the three high pressure injection pumps is normally in operation and a positive static head of water assures that all pipe lines are filled with coolant. The high pressure injection lines contain thermal sleeves at their connections into the reactor coolant piping to prevent over stressing the pipe juncture.

Operation of this system does not depend on any portion of another engineered safety feature. The system can be operated in conjunction with the Low Pressure Injection System if the HPI System must be operated in the recirculation mode.

### 6.3.3.2 Low Pressure Injection and Core Flooding Systems

Two pumps will deliver 6,000 gal/min to the reactor vessel through two separate injection lines. One pump can deliver approximately 3,000 gal/min to the reactor vessel at 100 psig.

After receiving an actuation signal, the low pressure injection valves will reach full open within 15 seconds and the low pressure injection pumps will reach full speed within 8 seconds.

a. Reactor Coolant System pressure transmitters.

b. Reactor Building isolation valves and associated position indications. For all containment isolation valves inside containment, the valves may be left in their closed position if the operator is not functional.

c. Reactor Building air cooling unit fans and cooling coils.

d. Instrument cables for pressure transmitters, level, and valve position indication.

e. Power cables for the Reactor Building fan motors and isolation valves.

f. Isolation valves and flow verification instrumentation in the gravity flow path from the reactor outlet piping to the Reactor Building emergency sump.

Non-nuclear instrumentation (Item a) in the Reactor Protection System and the Engineered Safeguards System located inside the Reactor Building are qualified in accordance with Criteria for Nuclear Power Plant Protection Systems, IEEE No. 279, dated August 30, 1969, to establish the adequacy of equipment performance in the LOCA environment.

A valve operator similar to those being used (Item b) was satisfactorily tested for performance under conditions expected to exist in the Reactor Building after the LOCA. The operator was tested in accordance with Level 4 of the Standard Draft, dated June 7, 1968, prepared by Sub-Committee 2 (Equipment Qualification Testing) of the IEEE/NSG/Technical Committee for Standards.

Table 6-12 provides an analysis for valve motors which may become submerged following a LOCA.

A scaled down Reactor Building cooling coil unit (a 24 x 24 inch section identical in construction with the full-size unit) has been satisfactorily tested under post-accident condition. The maximum test conditions were 70 psig, 286 °F and 100 percent relative humidity.

Other equipment and components located in the primary containment or elsewhere in the plant must be operable during and subsequent to a loss-of-coolant or steam-line-break accident and are as follows:

a. Equipment Outside Containment

Safety related equipment and components which are located outside the containment and which therefore are not subject to the abnormal environmental conditions present within the containment during an accident are given operational performance tests on either the actual equipment or prototype units. A list of the equipment located outside the containment is tabulated in Table 6-13.

b. Equipment Inside Containment

Safety related equipment located within the containment is qualified for the application by tests to demonstrate operability under the accident environment. A list of this equipment along with a brief description of the qualification tests is tabulated in Table 6-14.

Instrument transmitters and electric motor valve operators in the Reactor Protection System and the Engineered Safeguards System located inside the Reactor Building were designed to withstand the potential effects of radiation due to normal and accident conditions. Non-metallic materials and lubricants were selected on the basis of their susceptibility to radiation damage demonstrated by irradiation tests. The instrument transmitters were successfully irradiation tested at the Babcock & Wilcox Nuclear Development Center (NDC). The transmitters with two dosimeters attached to each were placed in a sealed aluminum box and positioned near fuel elements in the NDC storage pool. The test was conducted in two parts; the first part simulated the environmental dose to the transmitters associated with the 40-year design plant lifetime, and the second part simulated the maximum expected dose to the

The technical specification testing requires that these valves be operated quarterly to assure their continued availability. During the life of the facility, these valves will be appropriately operated subsequent to any required maintenance, repair or replacement.

9) Actual Seismic Conditions

In summary, we want to emphasize the results of the dynamic seismic analysis for the specific piping systems in which these valves are located (Table 6-17). The displacement and acceleration values in Table 6-17 are representative of the valves originally installed and their latest replacements. The maximum acceleration of 1.05g indicated is considerably less than the maximum g force in either the horizontal or vertical direction that the seven valves are required to withstand. The entire scope of testing verifies valve operability from conditions of extreme duress to normal operation, and the results of the earliest environmental, vibratory, and load testing have been verified in later independent testing.

## 6.3.2.7 Protection Provisions

### 6.3.2.7.1 Seismic Design

Components in the Emergency Core Cooling System are designated as Class I equipment and are designed to maintain their functional integrity during earthquake (2.6).

### 6.3.2.7.2 Missile Protection

Protection against missile damage is provided by either direct shielding or by physical separation of duplicate equipment. For most of the routing inside the Reactor Building, the ECCS Piping will be outside the primary and secondary shielding, and hence, protected from missiles originating within these areas. The portions of the injection lines located between the primary reactor shield and the reactor vessel wall are not subject to missile damage because there are no credible sources of missiles in this area.

The high pressure injection lines enter the Reactor Building via penetrations on opposite side of the building. Each injection line splits into two lines inside the Reactor Building, but outside the secondary (missile) shield, to provide four injection paths to the Reactor Coolant System. The four connections to the Reactor Coolant System are located between the reactor coolant pump discharge and the reactor inlet nozzles. There are four injection lines penetrating the missile shield, minimizing the effect on injection flow in the unlikely event of missile damage to the injection lines inside the secondary shield.

Protection from missiles is given to the low pressure injection lines within the Reactor Building. The portion of the Low Pressure Injection System located in the Reactor Building consists of two redundant injection lines which are connected to injection nozzles located on opposite sides of the vessel. Both redundant suction lines from the sump are missile protected. The sump suction is located outside of the secondary shielding and is additionally protected by a grating.

The entire Core Flooding System is located within the Reactor Building. The core flooding tanks and two of the three valves in each core flooding line are located outside of the secondary shield.

## 6.3.2.8 Post-Accident Environmental Consideration

The major operating components of the Emergency Core Cooling System are external to the Reactor Building and will not be exposed to the post-accident building environment.

Electrical and mechanical equipment within the Reactor Building which are required to be operable during and subsequent to a LOCA and/or steam line break are:

The valves are then operated using air supplied through the solenoids. Proper valve travel, solenoid, and limit switch operation is verified.

3) ES Test (Both EMO and Piston Valves)

In checking the valve for ES actuation, the valve is placed in the position opposite to its ES position and then an ES signal is simulated. The valve moves to its ES position. Then the control room switch is turned to the position opposite of ES operation and the valve is verified as remaining in its ES position. Similarly, turning the circuit breaker panel switch to the position opposite of ES operation has no effect on the valve.

Acceptance criteria for these electrical tests are:

a) Valves must open, close, and travel in the proper direction in response to control and engineered safeguards signals.

b) The valve open and closed indicating lights must indicate correctly.

c) Valve electric motor operator resistance-to-ground readings must be within specification.

d) The specified valve travel time is within specification requirements.

4) System Engineered Safeguards Test

The purpose of these tests is to demonstrate actual valve performance for its intended engineered safeguard use. Initially, all valves are placed in their non-ES position prior to simulating an ES signal. Upon initiation of an ES signal, the tests for the subject valve demonstrate containment isolation and also emergency injection flow capability to the Reactor Coolant System from the Low Pressure Injection System and the High Pressure Injection System.

5) System Functional Testing

The purpose of this testing is to verify that the valves perform as intended for normal operation. Cycling the valves under conditions of specified differential pressure and/or flow that may be encountered during plant operation will verify that the valve operator does not exceed maximum cycle time.

6) Integrated ES Actuation Test

The purpose of this test, in which these valves are used, is to demonstrate the full operational sequence that would bring the Emergency Core Cooling Systems and the Containment Pressure Reducing Systems into action, including the transfer to alternate power sources.

General acceptance criteria for this test are:

a) The ES Systems operate as described in the FSAR.

b) Upon actuation of an ES signal, high pressure and low pressure injection to the Reactor Coolant System are supplied in accordance with FSAR requirements.

c) Upon loss of normal station power, the ES systems continue to perform their designed functions without interruption.

Following completion of the preoperational test program and issuance of an operating license for the facility, these valves are functionally tested as required by the FSAR.

7) System Hydrostatic Tests

Fluid systems hydrostatic tests are performed on the various systems to assure leak tight installation of the valve in the piping system.

8) Technical Specifications

conducted in August, 1970. During all of these tests, the operator was periodically cycled and was found to operate satisfactorily.

*6.3.2.6.3.3 Valve Purchase Specification*

In addition to a proven record as verified by the previous testing, the valve vendor must also comply with the purchase specification requirements. The purchase specifications for these seven valves require that they be hydrostatically tested, leak tested and cycled between the extremes of fully opened or closed. The hydrostatic test is in accordance with the Standard for Steel Pipe Flanges and Flanged Fittings (USAS B16.5) or FSAR Section 3.2.2.2, "System Piping Classifications" on page 3-41 which provides allowances for substitute codes. The leak test requires that with the disc closed tight, hydrostatic pressure shall be applied alternately on each side of the closed disc with the side opposite the pressure open for inspection. Acceptance criteria required that valves not show a leakage greater than 10 cubic centimeters per hour per inch of seat diameter, or permanent deformation when the valves are subjected to two times design pressure, except that the stress developed at test pressure shall not exceed 90 per cent of the specified minimum yield strength based on the minimum specified wall thickness.

Original valve vendors have submitted generic calculations to B&W which show that when similar valve assemblies are subjected to a 3g horizontal force and to a 2g vertical force, the stresses incurred are within the code allowable stresses. These calculations also verified that the first natural frequency is above 20 Hz for these valves. Replacement valves will meet or exceed these requirements.

*6.3.2.6.3.4 Preoperational Testing*

The testing procedures for valves that require operation to meet engineered safeguards requirements are quite extensive during the preoperational testing program. These tests demonstrate proper installation, strength and functional performance of valves. Subsequent to satisfactory preoperational testing, surveillance testing requirements have been established to assure continued satisfactory operation of these valves. Furthermore, if maintenance or repair of these valves is required, appropriate functional testing will be accomplished to assure proper operation subsequent to the maintenance or repair.

The following paragraphs summarize the preoperational testing and test criteria used for the seven valves listed in Table 6-16 that meet the definition of being active and also part of the reactor coolant pressure boundary in accordance with 10CFR50.

a. System Electrical Test

1) Electric Motor Operated Valves

The purpose of these tests is to verify electrical characteristics of valve operators in performing their function. Preliminary checkout of the operator valve assembly requires that the valve be free to move and that if the motor operated valve travels in the wrong direction from its mid-travel position, its breaker must be tripped immediately as there would be no torque limit protection. The valve can be operated manually with a handwheel to ascertain its freedom of movement.

The phase rotation of the operator is checked. During valve operation, verification that the valve travel and motor are stopped is done by closing the torque limit switch. Similarly, the opening of the valve is terminated by the opening of the limit switch.

2) Pneumatic Cylinder Operated Valves

The purpose of these tests is to verify proper operation of the piston operated valve and the solenoid controlling the air supply to the valve. Valves with handwheels are checked for freedom of movement prior to applying air to the pistons. Valve position limit switches are set during this check.

The test was successful in every respect. There were no malfunctions of the operator and upon inspection of the component parts used, there was no noticeable deterioration or wear.

c. Live Steam Testing

In January, 1969, a complete SMB series operator was set up for electrical operation and live steam was piped into the conduit taps on the top of the limit switch compartment. One of the bottom conduit taps was left open to drain off any condensate. The operator was set on a timer basis for operation every thirty minutes for two minutes per cycle over a period of approximately nine hours. During this test, the live steam in the switch compartment had no effect on the function of the limit switches in their control of the operator at the full open and full closed position of travel. In addition, the limit switches were wired to indicating lights which operated satisfactorily.

The test was successful and there was no noticeable effect on the function of any of the parts in the limit switch compartment.

d. Life Cycle Testing

In January, 1969, the operator was mounted on a stand inside a test chamber and a 150 cycle load test was made on the unit. This test cycle consisted of stroking a 2-3/8″ diameter valve stem at a speed of 6 inches per minutes for a total of approximately 12 inches in two minutes. The valve stem in the full closed position produced a thrust of 16,500 pounds on a rigid plate securely bolted to the test chamber. The unit was wired so that the open position geared limit switch stopped the unit in the full open position.

After the life cycle testing was completed, the unit was inspected and found to be in excellent condition. There was no noticeable wear on any of the parts.

e. Simulated Accident Environment Testing

In November, 1968, an electric motor operator was tested under conditions which simulated the temperature, humidity and chemical environments that could be expected in the containment following some postulated accident such as the rupture of a major reactor coolant pipe.

The operator was placed in an Autoclave type chamber and subjected to 90 psig saturated steam. At specified intervals, the operator was cycled to assure proper operation. Forty minutes after the introduction of steam, a 1.5 percent boric acid solution was sprayed on the operator assembly. The operator continued to operate satisfactorily. Later, the steam pressure was periodically reduced to simulate post-accident conditions. The boric acid spray was allowed to continue for four hours. The steam pressure was eventually reduced to 15 psig. The test continued for seven days.

During this time, the operation of the operator became erratic. The corrosive effects of the steam and boric acid spray caused electrical contact malfunctions which were bypassed by the use of an appropriate jumper. The valve continued to cycle during the seven day period.

A design change was made to the limit switch in order to correct the erratic operation, and it was tested under similar accident conditions and found to operate satisfactorily. This design change has been incorporated into all subsequent applicable models of this operator.

#### *6.3.2.6.3.2 Recent Testing*

More recent tests on Limitorque SMB series operator were conducted during the summer of 1972 by the Franklin Institute Research Laboratories. In these tests an operator was exposed to gamma radiation (200 megarads), a steam/ chemical environment (for twelve days), a steam environment at temperatures as high as 340°F during the first day (test consisted of a 30 day exposure) and a seismic test similar to those

| | | | | | |
|---|---|---|---|---|---|
| HP-26&27 | High Pressure Injection Isolation | 4″ | 2 | High Pressure Injection | Limitorque EMO |
| LP-17&18 | Low Pressure Injection Isolation | 10″ | 2 | Low Pressure Injection | Limitorque EMO/ROTORK |

4

All of these valves receive extensive preoperational testing prior to initial fuel loading. The electric motor operators are all of the Limitorque SMB series and have a history of qualification testing to verify their reliability and operability. Extensive testing has been carried out by Limitorque and by an independent institute. The testing has been done over a number of years, and the most recent testing in the summer of 1972 bears out the operability confirmed initially.

It is also noted that the insulation on HP-26 and-27 and LP-17 and-18 EMO motors is Class B rather than the Class H insulation which is used on EMO's inside the Reactor Building and on the EMO's on which the tests were run.

One valve with a pneumatic cylinder operator is noted above. Analyses have been performed assuming a 5g horizontal and 5g vertical seismic loading with the resultant acting in a direction to maximize deflection. It was shown that under these loading conditions, binding between the operator piston and cylinder tube or between the piston rod and the cylinder busing will not occur. The natural frequency of the cylinder structure has been analyzed and found acceptable.

The following paragraphs describe qualification testing on Limitorque electric motor operators.

*6.3.2.6.3.1 Electric Motor Operator Qualification Testing*

a. Shock and Vibration Testing

In August, 1970, a Limitorque SMB series operator was mounted on a test stand having a threaded valve stem driven by the operator simulating opening and closing a valve. The operator was electrically connected to stop at the full close position by means of a torque switch and stop at the full open position by means of a geared limit switch. The operator had a four-train geared limit switch installed and all contacts not being used for motor control were wired to electric indicating lights at a remote panel.

The unit successfully completed a 5.3g shock level at 32 Hz with no discrepancies noted. An exploratory scan of 5 Hz to 35 Hz was made and no critical resonant frequencies were noted on the operator. The unit was shocked and vibrated in each of three different axes a total of two minutes on, one minute off, three times per axis. The unit was operated electrically to both the full open and full close position and all torque switches and limit switches functioned properly. None of the auxiliary limit switches wired to indicating lights ever flickered or indicated they were opening or flickering. All electrical and mechanical devices on the operator performed successfully.

b. Heat Testing

In January, 1969, a completely assembled and operational SMB series operator was placed in an oven where the temperature was maintained at approximately 325 °F for a duration of 12 hours. The unit was electrically operated every thirty minutes for a period of approximately two minutes per cycle and the geared limit switches were used to stop the actuator at the full open and full closed position of travel. Indicating light circuits were also wired to the geared limit switches.

assumption that a major loss-of-coolant accident had occurred and coincidentally an additional malfunction or failure occurred in the Engineered Safeguards System. For example, the analysis included malfunctions or failures such as electrical circuit or motor failures, valve operator failures, etc. It was considered incredible that valves would change to the opposite position by accident if they were in the required position when the accident occurred. Table 6-11 also presents an analysis of possible malfunctions of the core flooding tanks that could reduce their post-accident availability. It is shown that these malfunctions result in indications that will be obvious to the operators so appropriate action can be taken. In general, failures of the type assumed in this analysis are considered highly improbable since a program of periodic testing will be incorporated in the station operating procedures. The adequacy of
0 equipment sizes in the ECCS is demonstrated by the post-accident performance analysis described in
0 Chapter 15, "Accident Analyses" on page 15-1.

#### 6.3.2.6.1 High Pressure Injection Operability

3 A cross connect line with electric operated valves (HP-409 and HP-410, both normally closed) is installed
3 between the "A" and "B" headers to ensure that two paths and two pumps can be aligned to inject to the
3 RCS. Two pumps through two trains must be available during an accident to ensure adequate flow
3 reaches the core. In the case of either of valves HP-26 or HP-27 failing to open during an accident
3 situation, the cross connect via valve HP-409 or HP-410 would be utilized to provide flow through the
3 train with the failed valve.

#### 6.3.2.6.2 Core Flood Tank Valve Operability

To assure that the Core Flood Tank isolation valves will not be accidentally closed while the reactor is at power, the circuit breaker supplying power to these valves will be kept open and under administrative control. Power to the starter controls comes from this same circuit breaker through a control transformer and will be disconnected when the circuit breaker is open.

Lights in the control room indicate valve position (open or closed). These lights have a power supply separate from the circuit breaker serving the Core Flood Tank isolation valves and are operated from limit switches on the valve operator. Another limit switch on the valve operator will cause an annunciator alarm in the control room anytime a Core Flood Tank isolation valve is away from the wide open position. The annunciator system has a power supply separate from that used to operate the valve or indicating lights.

#### 6.3.2.6.3 Active Valve Operability

In each of Oconee 1, 2 and 3, there are seven valves that meet the definition of being active and also part of the reactor coolant pressure boundary in accordance with 10CFR50. These valves are required to actuate upon an engineered safeguards signal and to either isolate the Reactor Building or to open an Engineered Safeguard System flow path. These valves and their design conditions are listed in Table 6-16. Actual system operating conditions are significantly less severe than design conditions, as shown in Table 6-16. A summary of these valves follows:

| | Mark Number | Service | Size | Qty. | System | Actuator |
|---|---|---|---|---|---|---|
| 4 | HP-3&4 | Letdown Cooler Isolation | 2" | 2 | High Pressure Injection | Limitorque EMO |
| 5 | HP-5 | Letdown Isolation | 2" | 1 | High Pressure Injection | Sheffer Pneumatic |

### 8.3.1.3 Physical Identification of Safety-Related Equipment

Detailed cable lists are developed for all cables. These cable lists identify each cable by cable type, specific cable routing by tray section number, and termination points. Protective system cables are identified as such on the cable lists. These lists are issued and are used by the field for cable installation. Each cable tray section, excluding cable trays inside the Reactor Building is identified by tags bearing the tray section number assigned to it.. Cables required for protective systems are identified as follows:

1. Power and control cables are color coded to identify their use and/or channel association. The color code is as follows:

| | |
|---|---|
| Gray | Swgr 1TC<br>Ld Ctr 1X8<br>MCC 1XS1<br>ESG channel 1, 3, 5, & 7<br>DC Pnlbd 1DIA<br>Vital Pwr Pnlbd 1KVIA<br>RPS Ch A |
| Yellow | Swgr 1TD<br>Ld Ctr 1X9<br>MCC 1XS2<br>ESG channel 2, 4, 6, & 8<br>DC Pnlbd 1DIB<br>Vital Pwr Pnlbd 1KVIB<br>RPS Ch B |
| Blue | Swgr 1TE<br>Ld Ctr 1X10<br>MCC 1XS3<br>DC Pnlbd 1DIC<br>Vital Pwr Pnlbd 1KVIC<br>RPS Ch C<br>ESG Channel Even-Odd |
| Orange | DC Pnlbd 1DID<br>Vital Pwr Pnlbd 1KVID<br>RPS Ch D |

2. All cables have their identifying number permanently affixed to both ends.

### 8.3.1.4 Independence of Redundant Systems

The physical locations of electrical distribution system equipment shown in Figure 8-1, Figure 8-3 and Figure 8-4 are arranged to minimize vulnerability of vital circuits to physical damage as a result of accidents.

#### 8.3.1.4.1 Auxiliary Transformers

Auxiliary transformers, startup transformers, and the 100 kV transformer are located out of doors and physically separated from each other. Transformer CT4, fed from the on-site Keowee Hydro Station is
3 physically separated from the other transformers and located in a Class I enclosure Reference Section
3 3.2.1, "Seismic Classification" on page 3-37. Surge arresters are used where applicable for lightning

one ESG switchgear bus and one main feeder bus, leaving two ESG switchgear buses and the other main feeder bus to supply the required loads which are sufficient to perform the intended safety functions.

d. Each ESG switchgear feeder breaker is provided with redundant trip coils, supplied from separate dc supplies, assuring positive trip action.

With the above protective features plus their metalclad construction and the physical separation maintained, failure of any one of the three redundant ESG switchgear buses or components will not affect the ability of the other two ESG switchgear buses to supply their engineered safeguards loads.

5. Reliability of the main feeder buses and the standby buses is assured by the following protective features:

   a. Each main feeder bus and each standby bus is protected independently by differential current relays. These relays will sense any fault condition in the zone between the source side of the incoming bus feeder breakers to the load side of the outgoing feeder breakers. The outgoing feeder breakers on the standby bus are the breakers connecting to the main feeder buses and they have overlapping differential protection from both buses. The outgoing feeder breakers of the main feeder buses are the feeder breakers to the engineered safeguards switchgear buses. If a fault condition occurs, the relays will function to isolate the affected bus from all sources of supply by opening all circuit breakers associated with that bus. The other redundant bus still provides the required power to all three engineered safeguards switchgear buses.

   b. Each feeder breaker to each of the buses is protected with phase overcurrent and ground fault overcurrent protective relaying. These relays function to open the breaker and isolate the main feeder bus from the power source upon the occurrence of these overcurrent conditions. This thereby maintains the integrity of the power source and allows the continued supply of power of the other bus and all three engineered safeguards switchgear buses. The comparable condition on a split bus concept would cause the loss of one engineered safeguards bus.

   c. Each feeder breaker is also provided with breaker failure protective relaying. This feature will initiate action to isolate the breaker from any source of supply if the breaker fails to open on a protective relay trip. The maximum loss on this condition would be the connected source of supply and the associated bus. The other bus would transfer by the redundant transfer logic to the alternate source of supply and continue supplying power to all three engineered safeguards switchgear buses. The maximum loss under the split bus concept would not only be the source of supply, but also the associated engineered safeguards switchgear bus.

   d. Each feeder breaker is provided with redundant trip coils supplied from separate dc supplies, assuring positive trip action.

   With the above protective features, their metal-enclosed construction and their physical separation, failure of any one of the redundant bus sections or components will not affect the ability of the other buses to supply the engineered safeguards loads.

6. The emergency power sources are independent of each other and switched on to the main feeder buses such that this independency is maintained. Paralleling of emergency power sources is prevented by redundancy in transfer logic equipment and interlocking.

   Redundant systems of emergency power switching equipment are provided to switch the emergency power to the unit's 4160 volt redundant main feeder buses. The redundant transfer logic will seek the most available source of power and when it becomes available close into it. If this source is then
2 subsequently lost, the switching logic and equipment will transfer to the other source automatically if
2 power is available.

The failure analysis covering the emergency electrical systems is outlined in Table 8-4.

condition is 20,628 kVA as shown in Table 8-1. The smallest of the emergency power sources is the connection to the 100 kV transmission system through Transformer CT-5 which has maximum continuous rating of 22,400 kVA. The significant effect of these large sources of emergency power is that a greater number of plant auxiliaries may be run and used to help cope with an incident as well as shutdown and maintain the other nuclear units in safe shutdown conditions.

2. The Keowee hydroelectric units are inherently reliable sources of power as proven by years of operating experience with similar generating units. Since they are stored energy type machines, their ability to start is very reliable.

2 Except for the penstock, and cooling water supply pipe to the first valve, shared air supply, static
2 inverter and regulation, standby battery charger, 4160V and 600V underground power supply to
2 Keowee through CX, 230 kV main transformer, fire protection system, ACB air system, each unit is entirely independent of the other, consisting of its own turbine, governor system, generator, exciter, voltage regulator, generator circuit breaker, synchronizing equipment, protective relaying, automatic startup control equipment, manual controls, unit dc control battery, etc.

If one hydro unit is out for maintenance, the other unit is available for service. The two units are served by a common tunnel-penstock, and unwatering for tunnel or scroll case maintenance will make both units unavailable. Based upon Duke's experience since 1919 with a hydro station similarly arranged, it is expected that unwatering frequency will be about one day per year plus four days every tenth year.

During all periods when the Keowee units are available for emergency power service, the Keowee Hydro Headgate will be rigidly fastened to assure that failure of the hoist system will not permit the gate to move into the closed position.

The independent Keowee units, along with the alternate circuits, provide the required redundancy to assure reliable emergency power. Storage capacity of the Keowee reservoir and naturally occurring minimum streamflow are such that the generating units can provide continuous emergency power following an accident. The Keowee reservoir, between its normal elevation and maximum planned drawdown, has sufficient storage which, when combined with minimum recorded streamflow on the Keowee River will permit a hydro unit to carry continuously one nuclear unit's emergency auxiliary loads for 126 days.

The failure analysis covering the Keowee Hydro Station is outlined in Table 8-3.

3. Each electric power distribution system is designed with redundant full capacity buses to match the capacity of the large emergency power source. This thereby provides two continuous sources of supply from the two full capacity main feeder buses to each of the three engineered safeguards switchgear buses.

4. Reliability of the engineered safeguards switchgear buses is assured by the following protective features:

   a. 4160 V engineered safeguards (ESG) switchgear bus overload and bus fault conditions are protected for by both ground fault overcurrent relays and phase overcurrent relays. These relays are provided on each ESG switchgear bus breaker and function to open the associated breaker to isolate the ESG switchgear bus from the main feeder buses, thereby maintaining the integrity of the main feeder buses.

   b. Each ESG switchgear feeder breaker is also included in the zone of protection afforded by the main feeder bus differential current relays which would function to isolate a faulted breaker from any source of supply.

   c. Each ESG switchgear feeder breaker is provided with breaker failure protective relaying. This feature will initiate action to isolate the breaker from any source of supply if the breaker fails to open upon a protective relay trip. The maximum equipment this would remove from service is

#### 8.3.1.1.6 Tests and Inspections

Remote startup of the Keowee generators is provided in each of the control rooms of the nuclear station.
2 Provisions are made in the control rooms to manually initiate an emergency start of both of the two hydroelectric generators connecting the generator to the nuclear station's 4160 volt buses. Testing of this system may be scheduled any time the Keowee hydro units are not running.

The 100 kV, 230 kV and 500 kV circuit breakers are inspected, maintained and tested as follows:

1. 100 kV transmission line circuit breakers are tested on a routine basis.
2. 230 kV and 500 kV transmission line circuit breakers are tested on a routine basis. This is accomplished without removing the transmission line from service.
2 3. 230 kV and 500 kV switchyard generator circuit breakers may be tested with the generator in service.

Transmission line protective relaying is tested on a routine basis.

Generator protective relaying is tested when the generator is off-line.

The 4160 volt circuit breakers and associated equipment are tested in service by opening and closing the circuit breakers in a manner that does not interfere with the operation of the station. The circuit breakers are "jacked out" to a test position and operated without energizing the circuits, if necessary.

The 600 volt circuit breakers, motor contactors, and associated equipment are tested in service by opening and closing the circuit breakers or contactors so as not to interfere with operation of the station.

Emergency transfers to the various emergency power sources are tested on a routine basis to prove the operational ability of these systems. Associated normal, startup, and standby circuit breakers on one bus can be "jacked out" into test position and initiated manually for an emergency transfer test.

### 8.3.1.2 Analysis

The emergency electric power system provided for each nuclear generating unit possesses certain inherent design features which improve its reliability over limited capacity split-bus arrangements usually provided in nuclear power plants.

2 The basic design criterion for the electrical portion of the emergency electric power system of a nuclear unit, including the generating sources, distribution system, and controls is that a single failure of any component, passive or active, will not preclude the system from supplying emergency power when required. Special provisions have been employed to accomplish this which include a double bus - double breaker distribution system, redundant circuit breaker trip coils and circuits, diverse protective relaying for each circuit breaker, redundant load shedding and transfer logic equipment, physical separation and other features.

The reliability afforded by the split bus concept is included in the design of the double bus - double breaker system employed here. Consideration has been given to the capacity of the emergency power sources, the method of switching, redundancy utilized and the protective features. For example, the electrical system together with the sources of electric power which are installed to supply emergency power to a nuclear unit possesses the following design features:

1. Each electric power source is extremely large for the requirements. For example, each of the redundant on-site Keowee hydroelectric units is rated 87,500 kVA while the maximum combined load demand on one nuclear unit with a LOCA and the other two nuclear units in a hot shutdown

Both Keowee hydro units are started immediately and the unit not connected to the 13.8 kV underground feeder is connected automatically to the 230 kV Yellow Bus by closing its respective generator circuit breaker and the 230 kV Power Circuit Breaker (PCB)-9 when the 230 kV Yellow Bus is isolated from the system network.

The 230 kV switching station is isolated automatically by energizing the dual trip coils of the 230
2 kV PCBs 8, 12, 15, 17, 21, 24, 26, 28, and 33.

The startup transformers No. CT1, CT2, and CT3 remain connected to the 230 kV switching station.

The 13.8 kV underground circuit from Keowee becomes energized as the hydro unit to which it is connected is started.

In the event of an accident and the simultaneous loss of the external transmission network, the engineered safeguard switchgear buses are supplied emergency power through both 4160 volt main feeder buses from either the 4160 volt startup transformers through their respective feeder breakers or from both of the redundant standby power buses. The standby power buses receive emergency power from either the Keowee Hydro Station or the 100 kV transmission line described in Section 8.3.1.1.3, "4160 Volt Auxiliary System" on page 8-10 (2 on page 8-11). In the event of a Loss of Coolant Accident (LOCA) any breakers supplying the engineered safeguards loads are closed automatically. In the event of a LOCA and the simultaneous loss of both the normal auxiliary source and the startup source, the non-essential load breakers are tripped. Redundant engineered safeguards load-shedding logic equipment assures positive shedding of non-essential equipment by energizing separate trip coils provided in their circuit breakers. Redundant engineered safeguards actuation channels initiate closing of the essential equipment feeder breakers.

#### 8.3.1.1.4 600 Volt Auxiliary System

Each unit's 600 volt auxiliary system is similar and arranged into multiple bus sections. Each bus section is fed from a separate load center transformer which is connected to one of the three 4160 volt switchgear bus sections. Various 600 volt non-engineered safeguard motor control centers are located throughout the station to supply power to equipment within the related area. The three engineered safeguards load centers and associated motor control centers as shown in Figure 8-4 are redundant and are supplied independently from the three 4160 volt engineered safeguards load buses. Load center X8 and X9 have an alternate feeder with manual transfer to be used when the normal source of power is not available. Each engineered safeguard motor control center has an alternate feeder with manual transfer to be utilized only for maintenance. No common failure mode exists for this system.

#### 8.3.1.1.5 208 Volt Auxiliary System

For each unit, a system is provided to supply instrumentation, control, and power loads requiring unregulated 208Y/120 volt ac power. It consists of motor control centers, distribution panels, and transformers fed from 600 volt motor control centers.

The redundant engineered safeguards 208 volt motor control centers for a unit are shown in Figure 8-4. Each of these motor control centers have redundant supply feeders from separate transformers and redundant 600 volt motor control centers. The feeder breakers have mechanical interlocks and manual transfers.

The 208 volt auxiliary system is similar in arrangement for each of the three units.

The 4160 volt auxiliary system as shown in Figure 8-1 and Figure 8-3 is similar in arrangement for all three units.

On loss of their normal sources of power the 4160 volt main feeder buses are transferred as described for the 6900 volt system to alternate sources of power in the following preferential sequence:

1. Transfer to startup transformer where:
   a. Power is supplied from the 230 kV transmission system, or
   b. Power is supplied from one of the two Keowee hydro units via the 230 kV switchyard.
2. Transfer to 4160 volt standby power buses where:
   a. Standby power is supplied from one Keowee hydro unit via the 13.8 kV underground feeder, or
   b. Standby power is supplied from the 100 kV transmission line.

The control system is designed to prevent the paralleling of two sources during the switching operation and is similar to the transfer systems Duke has used for many years in their fossil-fired plants.

Upon loss of the unit auxiliary transformer source and startup transformer source, and in the absence of an engineered safeguards (ESG) signal, the following occurs:

The turbine-generator and reactor are tripped and the main feeder buses become deenergized. Control power is still available from the dc and vital power systems.

Both of the Keowee hydro units are started and either is connected to the standby power buses from which power can be supplied to the shutdown auxiliaries.

The non-essential loads are shed.

The equipment required to bring the reactor to a hot shutdown is energized.

Logic and control circuits will be fed without interruption from dc sources and vital power buses.

In the event of a loss of coolant accident requiring engineered safeguards action, the following action takes place:

Both Keowee hydro units are started immediately. The unit not connected to the underground feeder is run on standby and connected to the 230 kV Yellow Bus when the bus is isolated.

The underground circuit from Keowee becomes energized as the hydro unit to which it is connected is started.

The 4160 volt redundant main feeder buses of the unit with the accident are switched to the emergency power sources in the preferential order as described in Section 8.3.1.1.3, "4160 Volt Auxiliary System" on page 8-10 (1) and (2).

The engineered safeguards of the unit with the accident are started and the non-essential loads are shed when power is unavailable from the normal or startup sources.

In the event the external transmission network is lost, the following action takes place:

On normal automatic startup, each unit is automatically connected and supplies power to the Oconee 230 kV switching station through the stepup transformer by its respective generator circuit breaker. This is accomplished by the automatic synchronizing equipment of each unit. On emergency automatic startup, both units are started; the unit with the underground feeder connected to it supplies that feeder and the other unit is available to supply the Oconee 230 kV switching. If there is a system disturbance, this unit is connected automatically to the Oconee 230 kV Yellow Bus only after the Oconee 230 kV Yellow Bus is
2 isolated automatically from the system and the preset time delay has elapsed. Redundant External Grid Trouble Protective Systems are provided to isolate the 230 kV switching station on failure of the external transmission network. Therefore, on loss of the external transmission network, both of the Keowee hydro units can provide emergency power to any of the Oconee units through either the 230 kV switching station to the unit's respective startup transformer or the underground feeder and Transformer CT4 at Oconee.

Power from the hydro units is available except when:

1. Both units are out of service, or
2. There is a coincident failure of the underground feeder circuit and a complete outage of the 230 kV feeder circuit through the switching station.

#### 8.3.1.1.2 6900 Volt Auxiliary System

The 6900 volt auxiliary system for each unit is designed to supply electric power to the 9000 horsepower reactor coolant pump motors. This system is arranged into two bus sections. Both bus sections feed into two switchgear bus sections, each feeding two motors. Each switchgear bus supplies one motor for each of the two reactor coolant piping loops. Either the unit auxiliary or the startup transformer is capable of feeding both switchgear buses. During startup, shutdown and after shutdown, the switchgear buses are supplied from the startup transformer. During normal operation, the switchgear buses are supplied from the unit auxiliary transformer. Normal bus transfers between the two sources are initiated at the discretion of the operator from the control room, while emergency transfer from the unit auxiliary to the startup transformer is initiated automatically by protective relay action. Normal bus transfers used on startup or shutdown of a unit are "live bus" transfers, i.e., the incoming source feeder circuit breaker is closed onto the energized bus section and its interlocks will trip the outgoing source feeder circuit breaker which results in transfers without power interruption. Emergency bus transfers used on the loss of the normal unit source are rapid bus transfers, i.e., the outgoing source feeder circuit breaker is tripped and its interlocks close the incoming source feeder circuit breaker which results in a transfer within a maximum of nine cycles. An exception to this occurs when the main generator has been supplying in-plant loads while separated from the switching station. In this instance there is a one-second transfer delay when the normal unit source is lost.

The 6900 volt auxiliary system as shown in Figure 8-1 and Figure 8-3 is similar in arrangement for each of Oconee 1, 2, and 3.

#### 8.3.1.1.3 4160 Volt Auxiliary System

The 4160 volt auxiliary system for each unit is arranged into a double bus - double circuit breaker switching arrangement. The three power sources, (1) the unit's auxiliary transformer, (2) the startup transformer and (3) the standby power buses, feed each of the main feeder buses by this double circuit breaker arrangement. Each of the two redundant main feeder buses provide power to each of the three redundant engineered safeguards switchgear bus sections that serve the engineered safeguards auxiliaries. The engineered safeguards auxiliaries are arranged so that a failure of any single bus section does not prevent the respective systems from fulfilling their protective functions.

# 8.3 ONSITE POWER SYSTEMS

2

## 8.3.1 AC POWER SYSTEMS

### 8.3.1.1 System Descriptions

The station distribution system consists of various electrical systems designed to provide reliable electrical power during all modes of station operation and shutdown conditions. The systems are designed with sufficient power sources, redundant buses, and required switching to accomplish this. Engineered safeguard equipment for each unit is arranged onto three load group buses such that the loss of a single bus section for any reason results in only the loss of equipment fed from that bus leaving redundant equipment to perform the same function. In general, the equipment related to unit operation is connected to its respective unit auxiliary electrical buses, whereas equipment common to and serving all units is distributed between the three unit auxiliary electrical buses. The control of power sources and switching for Oconee 1 and 2 is accomplished from the Oconee 1 and 2 control room while control of power sources and switching for Oconee 3 is from the Oconee 3 control room.

#### 8.3.1.1.1 Keowee Hydro Station

The Keowee Hydro Station contains two units rated 87,500 kVA each, which generate at 13.8 kV. Upon loss of power from the Oconee generating unit and 230 kV switchyard, power is supplied from both Keowee units through two separate and independent routes.

One route is a 4000 ft. underground 13.8 kV cable feeder to 12/16/20 MVA Transformer CT4 which supplies the redundant 4160 volt standby power buses. The underground emergency power feeder is arranged with double air circuit breakers so that it can be connected to either Keowee generator bus. The connection to the generator bus is made with metal-enclosed bus. This under ground feeder is connected at all times to one hydroelectric generator on a predetermined basis and is energized along with Transformer CT4 whenever that generator is in service in either emergency or normal mode. The underground feeder and associated transformer are sized to carry full engineered safeguards auxiliaries of one unit plus auxiliaries for safe shutdown of the other two units.

The second route is a 230 kV transmission line to the 230 kV switching station at Oconee which supplies each unit's startup transformer. Each Keowee generator is connected to a common 230 kV stepup transformer through a 13.8 kV metal-enclosed bus and synchronizing air circuit breaker.

Each Keowee unit is provided with its own automatic startup equipment located in separate cubicles within the Keowee control room. The initiation of emergency startup is accomplished by control signals from either Oconee control area. Normal startup of either unit is by operator action while emergency startup is automatic. Both units are started automatically and simultaneously and run on standby on
1 either of three conditions: 1) external grid trouble protection system actuation, 2) engineered safeguards
1 actuation or 3) main feeder bus monitor undervoltage actuation. If the units are already operating when either of the above conditions occur, they are separated from the network and continue to run on standby until needed. Each unit's voltage regulator is equipped with a volts-per-cycle limiting feature which permits it to accept full emergency power load as it accelerates from zero to full speed within 23 seconds from receipt of the emergency startup initiation signal.

b. Any single circuit breaker can be isolated for maintenance without affecting any circuit.

c. Short circuits of a single main bus will be isolated without interrupting service to any circuit.

d. Short circuit failure of the tie breaker will result in the loss of its two adjacent circuits until it is isolated by disconnect switches.

e. Short circuit failure of a bus side breaker will result in the loss of the associated bus until it is isolated.

f. Failure of either the primary protective relaying or the backup protective relaying will not result in the loss of circuit protection.

With the above protection features, the probability of loss of more than one source of 230 kV or 525 kV power from credible faults is low; however, in the event of an occurrence causing loss of all the 230 kV and 525 kV connections, the station is supplied from one or more of six sources of power, i.e., 2 the three nuclear units, the two hydro units or the 100 kV line supplied by either the Lee combustion 2 turbines or the Central Tie Substation.

6. The 100 kV transmission line is located above the level of any flood that is postulated on the Keowee River. On the Duke system, wind and ice loadings are more severe than seismic loadings and govern the structural design of transmission lines, including this 100 kV line.

7. As shown in Table 8-2, the 125 volt dc switching station power system is arranged such that a single fault within the system does not preclude the protective relaying and control in the switching station from performing its intended functions.

2 Substation or from Lee Steam Station via a 100 kV transmission line connected to 12/16/20 MVA Transformer CT5 located on the opposite side of the station from the 230 kV facilities. This single 100 kV circuit is connected to the 100 kV transmission system through the substation at Central located eight miles from Oconee. Central Substation is connected to Lee Steam Station twenty-two miles away
2 through a similar 100 kV line. If an emergency occurs that would require the use of the 100 kV
2 transmission system, this line can either be isolated from the balance of the transmission system to supply
2 emergency power to Oconee from Lee Steam Station, or emergency power can be supplied directly from
2 the 100 kV system from the Central Tie Substation.

Located at Lee Steam Station are three 44.1 MVA combustion turbines. One of these three combustion turbines can be started in one hour and connected to the 100 kV line. Transformer CT5 is sized to carry all the engineered safeguards auxiliaries of one unit plus the shutdown loads of the other two units. This source of power is available except:

1. When the 100 kV line or transformer is out of service, or
2. Temporarily after a complete system blackout of all transmission facilities.

### 8.2.1.5 Switching Station 125 Volt DC Power System

The switching station dc system consists of two 125 volt dc, two conductor, metalclad distribution center assemblies; three battery chargers; and two 125 volt dc batteries as shown in Figure 8-7. A bus tie with breakers is provided between the switchgear bus sections to "backup" a battery when it is removed for servicing. One standby 125 volt dc battery charger is also provided between the two 125 volt dc batteries for servicing. One battery supplies power through panelboards for primary control and protective relaying, and the second battery supplies power through panelboards for backup control and protective relaying. Dual feeds from the redundant panelboards are provided to each Power Circuit Breaker (PCB) for closing and tripping control. Separate dual trip coils are provided for each PCB. For the 230kV switching station PCBs isolating diodes are provided for the redundant power feeds to the common closing coil circuit.

## 8.2.2 ANALYSIS

Reliability considerations to minimize the probability of power failure due to faults in the network interconnections and the associated switching are as follows:

1. Redundancy is designed into the network interconnections by installing two full capacity transmission circuits for each connection to the 230 kV grid.
2. The two single 230 kV transmission circuits are installed on the same line of double circuit towers. Each line of double circuit towers is separated a safe distance from the others and in most cases installed over a different route.
3. One of the circuits on a line of 230 kV transmission towers is insulated at a higher insulation level than the other, thus minimizing the probability of double outages due to flashovers.
4. Each circuit is protected from lightning and switching surges by an overhead electrostatic shield wire and in addition, lightning arresters are installed at both terminals.
5. The breaker-and-a-half switching arrangement in the 230 kV and 525 kV switching stations includes two full capacity main buses which feed each circuit through a circuit breaker connected to each bus. Completely redundant primary and backup relaying is provided for each circuit along with circuit breaker failure backup protection. These provisions permit the following:
   a. Any circuit can be switched under normal or fault switching without affecting another circuit.

The normal power supply to a unit's auxiliary load is provided through the unit auxiliary transformer connected to the generator bus. This source of power is available except when:

1. The generating unit is in a normal shutdown condition, or
2. There is an in-station malfunction or failure of equipment preventing continued operation of the reactor-turbine-generator-auxiliary transformer combination.
3. A 230 kV system blackout occurs resulting in a turbine-generator trip.

If power is not available from the unit's generator through the unit's auxiliary transformer, power is supplied to the unit through its startup transformer fed from either or both of the buses in the 230 kV switching station. Power to the startup transformer can flow through the 230 kV switching station from any one of thirteen supplies. These include eight 230 kV transmission circuits, two nuclear generating units if operating, two hydroelectric units and the 500 kV switching station. Each unit's auxiliary startup transformer is sized to carry full load auxiliaries for one nuclear generating unit plus the engineered safeguards equipment of another unit. In addition, each unit's startup transformer can backup another unit's startup transformer through emergency startup buses and dual isolating disconnect switches.

This source of power is available except when:

1. Both of the 230 kV buses in the switching station are unavailable, or
2. There is a 230 kV system blackout, no nuclear generating unit is running, and neither hydro unit is capable of supplying power through the 230 kV connection; or
3. The startup transformer fails or their connection to the 230 kV switching station fails and the unit's auxiliary transformers or their backfeeding circuitry are not available.

2 **8.2.1.3.1 230KV Switching Station Degraded Grid Protection**

2 Two channels of Degraded (DGP) grid protection are provided to assure that the degradation of the
2 voltage from off-site sources does not adversely impact the safety function of safety-related systems and
2 components. Each channel of this system, upon indication of inadequate voltage, will provide an alarm
2 to alert control room personnel of the existence of inadequate voltage in the 230KV switchyard. If an ES
2 signal is sensed by the DGPS, while the voltage is sustained below acceptable levels, the DGPS will
2 initiate an isolation of the 230KV switchyard (yellow bus) and start Keowee so that the on-site emergency
2 overhead power path is available. The non-ES operating units will not be affected by this action. The
2 other units will continue to operate since their generators remain connected to the red bus. It is
2 anticipated that any degradation of the voltage in the 230KV switchyard will not last for an extended
2 period of time. It is recognized that the voltage in the yard needs to be maintained above acceptable
2 levels, and corrective measures would be taken to assure that timely actions are taken to restore the
2 voltage.

2 There are three single phase undervoltage relays installed to monitor the switchyard voltage at the line side
2 of each of the three startup transformers. Each of the undervoltage relays is connected to one of three
2 single phase coupling capacitor voltage transformers. The setpoint of the undervoltage relays considers
2 the minimum analyzed switchyard voltage and the accumulative tolerances of the undervoltage relays and
2 the voltage sensing devices. A time delay is provided to override transients in the offsite system and
2 prevent unnecessary actuation of this protection system.

### 8.2.1.4 100 kV Switching Station

Whenever there is inadequate power from the generating units, the 230 kV switching station and the
2 hydro units, power is available to the standby power buses either directly from the 100 kV Central Tie

# 8.2 OFFSITE POWER SYSTEM

2

## 8.2.1 SYSTEM DESCRIPTION

### 8.2.1.1 Utility Grid System

The primary transmission system of Duke consists of a highly integrated 525 kV and 230 kV loop network. Underlying the primary transmission system is an extensive 100 kV sub-transmission network integrated into the primary system by means of 230/100 kV tie stations.

### 8.2.1.2 525 kV Switching Station

Unit 3 generates electric power at 19 kV which is fed through an isolated phase bus to a unit step-up transformer where it is stepped-up to the transmission voltage of 525 kV. From the step-up transformer an overhead transmission line feeds power to the 525 kV switching station through two circuit breakers connecting the unit to the 525 kV transmission network.

Three transmission lines connect to the Oconee 525 kV Switching Station; one circuit goes east-northeast
2 to Jocassee, one east to the Newport Station and one southeast to the Georgia Power Co. In addition, a 230/525kV autotransformer connects the 525 kV switching station to the 230kV switching station. The 525 kV buses, disconnect switches, and circuit breakers are arranged into a breaker-and-a-half configuration.

### 8.2.1.3 230 kV Switching Station

Unit 1 and Unit 2 also generate electric power at 19 kV which is fed through isolated phase bus on each unit to its own step-up transformer, where it is stepped-up to the transmission voltage of 230 kV. From each step-up transformer, an overhead transmission line feeds power to the 230 kV switching station through two circuit breakers connecting each unit to the 230 kV transmission network. Eight transmission lines connect to the Oconee 230 kV Switching Station; two circuits are installed east-northeast to North Greenville, four east-southeast to Central, and two north-northwest to Jocassee. See Figure 8-1 and Figure 8-2 for arrangement of lines in the Oconee Station and on the site.

The 230 kV buses, disconnect switches, and circuit breakers are arranged into a breaker-and-a-half configuration.

Each unit is provided with two physically independent circuits from the switching station. One is the circuit from the 230 kV switching station through the startup transformer which is designed to be available within a few seconds following a loss of coolant accident. The second circuit is the path from the switchyard through the main step-up transformer, the main generator bus and the unit's auxiliary transformer with the generator disconnected from the main bus. This second circuit is designed to be available in time following a loss of coolant accident to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. Both the unit auxiliary transformer and the startup transformer are rated at 45/60 MVA and have two isolated secondary windings rated 6900 volts and 4160 volts each.

The onsite power systems and their interconnection with the offsite power system are shown in Figure 8-1.

The onsite power systems are described in detail in Section 8.3, "Onsite Power Systems" on page 8-9.

## 8.1.3 SAFETY-RELATED LOADS

The loads that require electric power to perform their safety function are identified in Table 8-1.

## 8.1.4 DESIGN BASES

The design of the electrical systems for this three unit nuclear station is based on providing the required electrical equipment and power sources to assure continuous operation of the essential station equipment under all applicable conditions.

1 A safety related valve with electric motor actuation will be assigned a safety related power source if the
1 valve is required to respond immediately in an accident scenario in order to assure safe shutdown of the
1 plant or to mitigate the consequences of the accident. Valves (with electric motor actuators) which are
1 not required to respond immediately for accident mitigation or safe shutdown may be powered from
1 safety related sources when readily available, or from non-safety related, non-loadshed sources when the
1 following conditions exist: a) the valve actuator is equipped with manual override to allow manual
1 actuation, b) the environment in the immediate vicinity of the valve will allow operator access, c)
1 adequate time exists for operator intervention to be effective, and d) operator training is such that there is
1 reasonable expectation that operator intervention will occur when required.

# 8.1 INTRODUCTION

An offsite power system and an onsite power system are provided for each unit at the Oconee Nuclear Station to supply the unit auxiliaries during normal operation and the Reactor Protection System and Engineered Safeguards Protection Systems during abnormal and accident conditions.

Each Oconee unit has six available sources of power to the Engineered Safeguards Systems as shown in Figure 8-1. These are:

1. The 230 kV transmission system and/or the 525 kV transmission system
2. Two Keowee hydro units
3. The 100 kV transmission system
4. The two other nuclear units

The normal arrangement is for three of these to serve any or all units and to be switched in the preferential order as follows: (1) the 230 kV transmission network through the unit startup transformers, (2) one Keowee hydro unit through an overhead 230 kV circuit, and (3) the other Keowee hydro unit through an underground circuit.

Whenever the underground circuit from Keowee is unavailable, a circuit from the 100 kV transmission network can be connected to the Standby Buses and serve as an emergency power source.

Any unit can provide power to another unit's Auxiliary System via the switchyard.

## 8.1.1 UTILITY GRID SYSTEM AND INTERCONNECTIONS

Duke Power Company is an investor-owned utility serving the Piedmont region of North Carolina and South Carolina. The Duke transmission system consists of interconnected hydro plants, fossil-fueled plants, combustion turbine units, and nuclear plants supplying energy to the service area at various voltages up to 525 kV. Duke is a member of the Virginia-Carolina (VACAR) Subregion of the Southeastern Electric Reliability Council (SERC). All the companies in the region are interconnected such that the combined networks operate as a single, integrated system.

A detailed description of the offsite power system is provided in Section 8.2, "Offsite Power System" on page 8-5.

## 8.1.2 ONSITE POWER SYSTEMS

The onsite power system for each unit consists of the main generator, the unit auxiliary transformer, the startup transformer, the Keowee Hydro Station, the batteries, and the auxiliary power system. Under normal operating conditions, the main generator supplies power through isolated phase bus to the unit step-up and unit auxiliary transformers. The unit auxiliary transformers are connected to the bus between the generator disconnect link and the associated unit step-up transformer. During normal operation, station auxiliary power is supplied from the main generator through these unit auxiliary transformers. During startup, during shutdown, and after shutdown station auxiliary power is supplied from the 230 kV system through the startup transformer.

# CHAPTER 8. ELECTRIC POWER

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Table 7-3. Engineered Safeguards Actuated Devices

| Channel 1 | Channel 2 | Channels 1 & 2 | Channel 3 | Channel 4 | Channels 3 & 4 |
|---|---|---|---|---|---|
| HP-P1A | HP-P1C | HP-P1B | LP-P1A | LP-P1B | LPSW-P1C (3) |
| HP-24 | HP-25 | | LP-17 | LP-18 | |
| HP-26 | HP-27 | | LP-21 | LP-22 | |
| HP-3 | HP-5 | | (also Chan. 7) | (also Chan. 8) | |
| HP-4 | HP-21 | | LPSW-4 | LPSW-5 | |
| HP-20 | KEOWEE START | | LPSW-PIA(1) | LPSW-PIB(2) | |
| KEOWEE START | (Channel B) | | | | |
| (Channel A) | LOAD SHED & | | | | |
| LOAD SHED & | STBY. BRK. 2 | | | | |
| STBY. BKR. 1 | Standby BUS | | | | |
| Standby BUS | FEED BKR. 2 | | | | |
| FEED BKR. 1 | RC-7 | | | | |
| RC-5 | FDW-106 | | | | |
| RC-6 | FDW-108 | | | | |
| FDW-105 | GWD-13 | | | | |
| FDW-107 | LWD-2 | | | | |
| GWD-12 | CS-6 | | | | |
| LWD-1 | PR-2 | | | | |
| CS-5 | PR-3 | | | | |
| PR-1 | PR-4 | | | | |
| PR-6 | PR-5 | | | | |
| PR-7 | PR-8 | | | | |
| PR-9 | PR-10 | | | | |
| | FDW-103 | | | | |
| | FDW-104 | | | | |

| Channel 5 | Channel 6 | Channels 5 & 6 | Channel 7 | Channel 8 |
|---|---|---|---|---|
| CC-7 | CC-8 | LPSW-15 | BS-1 | BS-2 |
| LPSW-18 | LPSW-24 | LPSW-6 | LP-21 | LP-22 |
| RBCU-F1A | RBCU-F1C | LPSW-21 | (also cha.3) | (also cha.4) |
| PR-E1A | PR-E1B | RBCU-F1B | BS-P1A | BS-P1B |
| | | LPSW-565 | | |
| | | LPSW-566 | | |

1. LPSW-P1C for Unit 2
   LPSW-P3B for Unit 3
2. LPSW-P1B for Unit 2
   LPSW-P3A for Unit 3
3. LPSW-P1A for Unit 2

# APPENDIX 7. CHAPTER 7 TABLES AND FIGURES

**Table 7-1. Reactor Trip Summary**

| | Trip Variable | No. of Sensors | Steady-State Normal Range | Trip Value or Condition for Trip |
|---|---|---|---|---|
| | Over Power | 4 Flux Sensors | 0-100% | 105.5 percent of rated power**** |
| 4 | **Nuclear Over Power Based on Flow and Imbalance | 4 Two-Section Flux Sensors<br>8 ΔP Flow | NA | 109.4 times flow minus reduction due to imbalance |
| | **Power/RC Pumps | 4 Pump Monitors | 2 to 4 Pumps | Loss of one operating coolant pump motor in each loop |
| 4<br>4 | | | | Loss of any two operating reactor coolant pumps |
| | | | 2 Pumps | Loss of one of two operating reactor coolant pump motors in one loop |
| | Reactor Outlet Temperature | 4 Temperature Sensors | 532-604 F | 618 F |
| | **Pressure/ Temperature | 4 Pressure Sensors<br>4 Temperature Sensors | NA | $(11.14T_{out}-4706) \geq P$ |
| | Reactor Coolant Pressure | 4 Pressure Sensors | 2,090-2,220 psig | 2,355 psig (high)***<br>1,800 psig (low)** |
| | Reactor Building Pressure | 4 Pressure Sensors | 0 psig | 4 psig |
| | Main Turbine Trip | 4 Pressure Sensors | NA | Turbine Trip |
| | Loss of Main Feedwater Trip | 8 Pressure Sensors | NA | Loss of both Main Feedwater Pumps or Feedwater discharge pressure |

**Note:**

**Bypassed by shutdown bypass.

***Reset to 1720 psig by shutdown bypass.

****Administratively reset to 5 percent during reactor shutdown.

### 7.8.2.3 Testing

Inputs are also provided from the ATWS test panel. The panel is resident in the PLC cabinet along with other ATWS equipment.

Periodic testing will use a Bypass/Enable switch located on the test panel for testing each channel of AMSAC and DSS logic in the PLC. Whenever this switch is not in the ENABLE position, a continuous indicator in the Control Room will be illuminated and a computer alarm will be generated for display in the Control Room on a CRT. Status indication of all inputs and outputs are on the test panel.

These systems are designed so that both are two out of two logic actuated systems, and provisions are incorporated which allow disabling of the system output when one of the channels is placed in test. This prevents accidental initiation of the systems during individual channel testing.

### 7.8.2.4 AMSAC and DSS I/O

Each input to the AMSAC and DSS logic is provided with complete indications and alarms that alert the operator to an off-normal status that might preclude an ATWS event. Each plant variable that inputs into the AMSAC and DSS is monitored as part of the existing plant indications and provide the operator with information relevant to the status of each variable prior to reaching the AMSAC or DSS set point.

Outputs from the PLC's are provided through interfacing relays located in the ATWS equipment cabinets. These relays provide the outputs to the Main Turbine, Turbine Bypass Valve Set Point, the Emergency Feedwater Pumps, and the Control Rod Drive System for Groups 5, 6, 7 and the Auxiliary programmer control assembly. The relays used are powered by the UPS. Each PLC channel output relays will be wired to the above devices in a manner such that both channels of AMSAC/DSS are required for the devices to trip, start, or drop. The relays also provide output status information to the operator.

**THIS IS THE LAST PAGE OF THE CHAPTER 7 TEXT PORTION.**

1 AMSAC interfaces with the following systems and devices:

| FROM | TO | ISOLATION |
|---|---|---|
| AMSAC PLC Interfacing Relays | Main Turbine Trip Solenoid | NE to NE |
| DSS Interfacing Relays | TBV's Control Setpoint | NE to NE |
| AMSAC PLC Interfacing Relays | EFDW Pump Start Circuits | NE to 1E |
| AMSAC Channels Actuation | Control Room Annunciator | NE to NE |
| NE = Non-Class 1E | 1E = Class 1E | |

Feedwater Pump Turbine Oil Pressure is sensed by pressure switches in the Feedwater Pump Turbine Control Console on the turbine standard. These switches are then multiplied using control relays for output to various plant control, monitoring and alarm circuits. AMSAC will be one of the end users of these signals.

Feedwater Pump Discharge Pressure is sensed by pressure switches in the discharge lines of each individual pump. These switches are then multiplied using control relays for output to various plant control, monitoring and alarm circuits. AMSAC will be one of the end users of these signals.

### 7.8.2.2 DSS

Each channel of DSS uses a Wide Range RCS Pressure signal supplied via an analog isolator from the Westinghouse supplied Reactor Vessel Level Indication System (RVLIS). These signal loops also provide the Regulatory Guide 1.97 wide range RCS pressure indications on the main control board. The DSS will utilize the signal conditioning equipment which is resident in the RVLIS cabinet through an isolation device that separates the Class 1E RVLIS from the Non-Class 1E DSS. DSS trip actuation is initiated at a setpoint of 2450 $\pm$ 25 psig using the logic in the PLC. Outputs from both channels of the PLC's are combined to make the required two-out-of-two logic. Upon actuation of both channels of DSS, relays will energize and interrupt the power to the Control Rod Drive System (CRDS) programmers in Control Rod Groups 5, 6 and 7 as well as the auxiliary programmer control assembly.

DSS interfaces with the following systems and devices:

| FROM | TO | ISOLATION |
|---|---|---|
| DSS Interfacing Relays | CRD Groups 5, 6, 7 | NE to NE |
| DSS Interfacing Relays | Auxiliary Rod Controls | NE to NE |
| DSS Channel Actuation | Control Room Annunciator | NE to NE |
| WR RCS Pressure (RVLIS) | DSS PLC Channels | 1E to NE |
| NE = Non-Class 1E | 1E = Class 1E | |

For each unit the Control Rod Drive System (CRDS) will also provide an input from the CRDS Diamond panel located in the main Control Room into the DSS logic for reset of the CRDS Solid State Programmers.

# 7.8 ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS) MITIGATION SYSTEM

## 7.8.1 DESIGN BASIS

The ATWS system that is installed at the Oconee Nuclear Station is based upon the B&WOG Generic ATWS Design Basis Document 47-1159091-00 dated October 9, 1985, subsequent B&WOG ATWS Committee submittal dated December 1, 1987, the Safety Evaluation Report on B&WOG 47-1159091-00 contained in the NRC letter to DPCo dated July 26, 1988, and the September 7, 1988 letter G. Holohan (NRC) to L. Stalter (B&WOG).

## 7.8.2 SYSTEMS DESIGN

The ATWS Mitigation System is composed of two parts, the ATWS Mitigating Systems Actuation Circuitry (AMSAC) and the Diverse SCRAM System (DSS).

The ATWS Mitigation System Actuation Circuitry (AMSAC) and Diverse Scram System (DSS) consist of two Programmable Logic Controllers (PLC's) for the logic control circuits and two Uninterruptible Power Sources (UPS) connected to offsite power. Inputs from the field sensors are wired to the PLC's and outputs to the final actuation devices are wired using interfacing relays housed with the ATWS equipment cabinets and powered from the UPS. The UPS's are powered from a 120 VAC local panelboard backed by the Oconee Station emergency source (Keowee Hydroelectric Generating Station). The 2 UPS's are isolated from the emergency source by individual fuses coordinated with the panelboard circuit breakers and the upstream distribution network.

The AMSAC/DSS System consists of a two channel energize-to-trip design with the AMSAC portion actuated on low Feedwater Pump Turbine (FDWPT) control oil pressure or low Feedwater Pump (FDWP) discharge pressure while the DSS portion is actuated upon high Reactor Coolant System (RCS) Pressure.

All AMSAC/DSS PLC's and UPS power supplies are located in a stand-alone cabinet located above the Control Room in what is called the Ventilation Room. This location is convenient to the Control Room and allows easy access for testing and maintenance. This location is a Mild Environment.

All AMSAC/DSS process monitoring inputs are provided by existing Oconee instrumentation and control systems. RCS pressure inputs to the DSS which are analog signals are currently displayed on the Main Control Boards. Annunciator alarms are provided in the Control Room to alert the operator that one channel for either AMSAC or DSS has actuated.

### 7.8.2.1 AMSAC

Each channel of AMSAC uses existing inputs from the Feedwater System which monitor FDWPTA(B) hydraulic control oil pressure and FDWPA(B) discharge pressure signals (one per pump to each channel) from pressure switches which are part of the original Oconee feedwater system design.

These signals are multiplied using relays to provide the contact inputs which will be wired directly to the PLC's. These signals are processed using programmable logic resident in the PLC to provide the outputs to the Main Turbine and the Emergency Feedwater System.

2 8. SSF Diesel Engine Service Water Pump Control
2 9. SSF Diesel Engine Service Water Pump Discharge Flow Meter
2 10. SSF Auxiliary Service Water Pump Control
2 11. SSF Auxiliary Service Water Pump Discharge Flow Meter
2 12. SSF Sump Pump Controls

2 SSF UNIT RELATED CONTROLS AND INSTRUMENTATION

2 1. Unit Annunciator
2 2. Unit Recorder
2 3. SSF RC Makeup System
2 a. Pump Controls
2 b. Valve Controls
2 c. Pump Suction Pressure and Temperature Indication
2 d. Pump Discharge Pressure, Flow, and Temperature Indication
2 4. Unit Process Indicators
2 a. Pressurizer Level
2 b. Pressurizer Pressure
2 c. RC Loop A and B Hot Leg Temperatures
2 d. RC Loop A and B Cold Leg Temperatures
2 e. RC Loop A and B Pressure
2 f. Steam Generator Level A and B
2 g. Steam Generator Auxiliary Service Water Flow
2 5. Unit Controls
2 a. Letdown Cooler A and B Outlet Valve
2 b. Pressurizer Water and Steam Space Samples
2 c. Steam Generator A and B Feedwater Control Valve
2 d. Boron Dilution Block Valve
2 e. Pressurizer Relief Block Valve
2 f. Pressurizer Heaters
2 g. Steam Generator A and B Emergency Feedwater Valves
2 6. Power Systems Alignment Indicating Lights

## 7.7.6 AUXILIARY CONTROL STATIONS

Auxiliary control stations are provided where their use simplifies control of auxiliary systems equipment such as waste evaporator, sample valve selectors, chemical addition, etc. The control functions initiated from local control stations do not directly involve either the Engineered Safeguards System or the Reactor Control System. Sufficient indicators and alarms are provided so that the Oconee control room operator is made aware of abnormal conditions involving remote control stations.

## 7.7.7 SAFETY FEATURES

Control room layouts provide the necessary controls to start, operate and shut down the units with sufficient information display and alarm monitoring to assure safe and reliable operation under normal and accident conditions. Special emphasis is given to maintaining control during accident conditions. The layout of the engineered safeguards section of the control board is designed to minimize the time required for the operator to evaluate the system performance under accident conditions.

shutdown panel located outside the control room. The following instrumentation and controls are available on the emergency shutdown panel:

1. Pressurizer Level Indicator
2. Pressurizer Heater Control
3. RC Pressure Indicator
4. RC Outlet Temperature Indicator
5. Turbine Steam Supply Header Pressure Indicator
6. Turbine Bypass Valve Loop "A" Station
7. Turbine Bypass Valve Loop "B" Station
8. Startup Feedwater Valve Loop "A" Station
9. Startup Feedwater Valve Loop "B" Station
10. Steam Generator "A" Startup Level
11. Steam Generator "B" Startup Level
12. Letdown Storage Tank Level Indicator
13. HP Injection Pump "B" Control Switch
14. Pressurizer Level Control Station

If HP Injection Pump "A" is in operation, it can be tripped from the 4.16 KV switchgear located on elevation 796′ + 6″. The operator has control of HP Injection Pump "B" at the emergency shutdown panel. Makeup to the letdown storage tank can be obtained, if desired, from one of the following sources:

1. RC Bleed Holdup Tank
2. Concentrated Boric Acid Storage Tank
3. Boric Acid Mix Tank

The necessary pumps can be controlled from the waste disposal control panel.

### 7.7.5.2 Standby Shutdown Facility

The Standby Shutdown Facility (SSF) provides a secondary alternate and independent means to achieve and maintain a hot shutdown condition for scenarios in which the Control Room is unavailable or equipment it controls is unavailable. The SSF was designed for safe shutdown during postulated fire, Turbine Building flooding, and physical security events. The following instrumentation and controls are available on the SSF:

<u>SSF DIESEL GENERATOR AND STATION RELATED CONTROLS AND INSTRUMENTATION</u>

1. Diesel Generator Annunciator Panel
2. Diesel Generator Recorder
3. Diesel Generator Controls
4. Diesel Generator Metering
5. Diesel Generator Syncroscope
6. SSF Power Systems Breaker Controls and Indicating Lights
7. SSF Power Systems Metering

### 7.7.4.2 Control Room to Outside Station

The commercial and microwave/fiber optic telephone network provides direct communication to points outside the station area. The commercial telephone network provides dial access through the PABX to the public telephone network. One part of the microwave/fiber optic network is integrated into the PABX and includes access to the General Office at Charlotte, the Spartanburg Disptacher, and Lee Steam Station. Another part of the microwave/fiber optics network is independent of the PBX and includes non-dial call ability to Charlotte Dispatcher, Spartanburg Dispatcher, the substation at Central, and Lee Steam Station. For an emergency situation involving loss of AC power, the microwave transmitter has its own battery for eight hours operation and propane-engine generator with fuel for at least one week.

The control room is also equipped with transmitter-receivers which operate on 47.98 megahertz and 47.84 megahertz and have ability to call the substation at Central as well as mobile receivers.

### 7.7.4.3 Exclusion Area Control

An emergency vehicle and a boat are provided, each of which contains a transmitter-receiver for communications with Oconee 1 and 2 control room and an amplifier speaker to be used for warning in the exclusion area.

## 7.7.5 OCCUPANCY

Safe occupancy of the control room during abnormal conditions is provided for in the design of the Auxiliary Building. Adequate shielding is used to maintain tolerable radiation levels in the control rooms for maximum hypothetical accident conditions. Each Control Room Ventilation System is provided with radiation detectors and appropriate alarms. See Section 9.4.1, "Control Room Ventilation" on page 9-53 for control room ventilation systems description. Emergency lighting is provided.

The potential magnitude of a fire in either control room is limited by the following factors:

1. The control room construction and furnishings are of noncombustible materials.
2. Control cables and switchboard wiring meet the flame test as described in Insulated Power Cable Engineers Association Publication S-61-402 and National Electrical Manufacturers Association Publication WC 5-1961.
3. Qualified trained personnel, adequate extinguishers, and accessibility to all control room areas are provided.

A fire, if started, would be of such a small magnitude that it could be extinguished by the operator using a hand fire extinguisher. The resulting smoke and vapors would be removed by the ventilation system.

Essential auxiliary equipment is controlled by either stored energy, closing-type, air circuit breakers which are accessible and can be manually closed in the event DC control power is lost, or by AC motor starters which have individual control transformers.

2 ### 7.7.5.1 Emergency (Auxiliary) Shutdown Panel

If temporary evacuation of the control room is required while operating at any power, the operator will trip the control rods and start the Keowee hydro units prior to evacuating the control room. This action can also be accomplished from the cable room located on elevation below the control room. After evacuation, the operator can establish and maintain a hot shutdown condition from the emergency

Reactor coolant pump controls located on Section 5 of the control boards consists of the pump controls and auxiliary instrumentation required for pump operation. Also mounted on this section are the Auxiliary Electrical System controls required for manual switching between the various power sources described in Section 8.2, "Offsite Power System" on page 8-5 and Section 8.3, "Onsite Power Systems" on page 8-9.

Controls and indications for all normal ventilation systems are located on Section 7 of the control boards.

In order to maintain the desired accessibility for control of the station, miscellaneous recorders not required for station control are located on the vertical recorder boards where they are visible to the operator. Radiation monitoring information is also indicated there.

## 7.7.3 SUMMARY OF ALARMS

Visible and audible alarm units are incorporated into the control boards to warn the operator if limiting conditions are approached by any system. Audible Reactor Building evacuation alarms are initiated from the Radiation Monitoring System and from the source range nuclear instrumentation. Audible alarms are sounded in appropriate areas throughout the station if high radiation conditions are present in that area. Alarms for the nuclear systems are indicated in process diagrams in Chapter 6, "Engineered Safeguards" on page 6-1, Chapter 7, "Instrumentation and Control" on page 7-1, and Chapter 9, "Auxiliary Systems" on page 9-1. Alarms are also provided to warn the operator of unauthorized entry into vital areas.

## 7.7.4 COMMUNICATIONS

### 7.7.4.1 Control Room to Inside Station

The telephones for the site are connected to a Private Automatic Branch Exchange (PABX) located inside the Administrative Annex building. The PABX has capability of up to 10,000 lines and provides access for communications and paging. The equipment provides 4-digit dialing, dial tone, ring-back tone and busy tone. The PABX is powered by 48VDC batteries, which are charged through an inverter/charger combination, fed by a 480VAC supply. Upon loss of normal AC power, the system batteries will provide required power for a minimum of four (4) hours. Alternate power is automatically provided from the emergency diesel generator provided for the building.

The public address system is accessible through plant telephones by a two digit access code. In the event of PABX failure, the PA system is operable through eleven handsets installed at strategic locations within the station.

A sound powered telephone system is provided and consists of a network of conductor pairs converted to jacks through the plant. Sound powered handsets will be plugged into the jacks to form talking paths. There are nine separate talking paths available for each unit. The system is completely independent from any other telephone system and involves no external power supply.

An additional radio transmitter/receiver communication system is provided between the control room and Reactor Building. This system would be used in the event of loss of telephone or sound powered phones. This system is used for security or fire situations and only when the containment is open for access.

# 7.7 OPERATING CONTROL STATIONS

Following proven power station design philosophy, all control station, switches, controllers, and indicators necessary to start up, operate, and shut down Oconee 1 and 2 are located in one control room. Controls for Oconee 3 are located in a separate control room. Control functions necessary to maintain safe conditions after a loss-of-coolant accident are initiated from the centrally located control rooms. Controls for certain auxiliary systems are located at remote control stations when the system controlled does not involve unit control or emergency functions.

## 7.7.1 GENERAL LAYOUT

The control room for Oconee 1 and 2 is designed so that one man can supervise operation of both units during normal steady state conditions. During other than normal operating conditions, other operators are available to assist the control operator. Figure 7-26 shows the control room layout for Oconee 1 and 2. Oconee 3 has similar accessibility to the various controls. The control boards are subdivided to show the location of control stations and to display information pertaining to various sub-systems.

## 7.7.2 INFORMATION DISPLAY AND CONTROL FUNCTIONS

Consideration is given in the control board layout to the fact that certain systems normally require more attention from the operator. The Integrated Control System is therefore located nearest the center line of the boards (Section 1 on Figure 7-26).

On Section 2 of the control board, one indicator will be provided for each control rod. Fault detectors in the Rod Drive Control System are used to alert the operator should an abnormal condition exist for any individual control rod. Displayed in this same area are limit lights for each control rod group and all nuclear instrumentation information required to start up and operate the reactor. Control rods are manipulated from the Section 2 bench position. Plant computer readout facilities for alarm monitoring and sequence monitoring are located here to aid the operator.

A plant computer is used on each unit to provide fuel management measurements and calculations. These computers also provide for alarm monitoring, performance monitoring, data logging, sequence monitoring, and facilitate control of some functions during start-up and shut-down of the turbine-generator. Monitoring and display functions of the plant computer which audit Nuclear Steam Supply System parameters of major interest are duplicated elsewhere in the control rooms. This type of computer application has been successfully applied to units presently in operation on the Duke system.

Variables associated with operation of the secondary side of the station are displayed and controlled from Section 1 and 3 of the control board. These variables include steam pressure and temperature, feedwater flow and temperature, electrical load, and other signals involved in the Integrated Control System. Section 3 of the control board also contains indication and controls of the Reactor Coolant System parameters.

The Engineered Safeguards System is controlled and monitored from Section 8 of the vertical boards. Indicating lights are provided as a means of verifying the proper operation of the Engineered Safeguards System. Control switches located on this panel allow manual operation of equipment that is not controlled elsewhere in the control room or test of individual units.

Control and display equipment for station auxiliary systems are located on Section 6 of the control board.

Each individual detector measures the neutron flux at its locality and is used to determine the local power density. The individual power densities are then averaged and a peak-to-average power ratio calculated. This information can be used to indicate possible power oscillations.

The location of the lower accelerometer was selected to measure the midspan vibratory motions of the perforated tube. The perforated section of the surveillance holder tube is expected to have the largest flow induced vibratory amplitudes relative to the other sections of the holder tube assembly.

The upper end of the perforated tube is connected to the thermal shield. Consequently, the upper accelerometer will measure the thermal shield mid-plane vibratory amplitudes.

The 1-inch penetrations in the reactor vessel head permit the addition of three accelerometers to measure the shell mode vibrations of the plenum cylinder. One accelerometer will be located at the lower end of each of three tubes which are welded to the outside of the cylinder adjacent to the outlet holes as shown in Figure 7-25.

Each of the four accelerometers in the surveillance holder tube is biaxial. Therefore, there will be eight separate channels, four channels for measuring the acceleration amplitudes of the lower section of the surveillance holder tube and four channels for detecting the accelerations of the thermal shield. The uniaxial accelerometers for the plenum cylinder will provide three channels for measuring the acceleration amplitudes of the cylinder.

The accelerometers, specially designed for the components, will be capable of measuring the frequency of the components over a range of 2 to 300 Hz at acceleration up to 30g's.

Analysis

The acceleration signals from the various components will be recorded on tape by a FM tape recorder. After the signals are recorded, the information on the tape will be digitized by use of a mini-computer which samples the data at preset time intervals. The digitized time history record will then be used as input to a computer program which will analyze the record.

A B&W proprietary computer program will be used to plot the time history of the fluctuating accelerations, determine the predominant frequencies, the autocorrelation of the signal and phase differences between signals.

Cyclic stress values will be determined from the measured acceleration amplitudes, frequency and mode shapes. These dynamic stresses will be combined with normal operational stresses. The combined stresses will be judged acceptable if they are less than the endurance limit for the materials used to manufacture the components.

Oconee 2 & 3

The reactor vessels and internals designed for Oconee 2 and 3 are essentially identical to Oconee 1. To confirm that the fabrication process has not altered the characteristics of the internals, one surveillance holder tube for Oconee 2 and one for Oconee 3 will be instrumented like Oconee 1. Measurements will be made as described for Oconee 1. The instrument cables will go through a control rod nozzle (requiring the removal of a control rod drive mechanism) because the reactor vessel heads for these units do not have the 1-inch penetrations. The results from each of these tests will be compared to those for Oconee 1 to confirm that the vibration characteristics are similar.

### 7.6.2.5 Detection and Control of Xenon Oscillations

Under normal operating conditions, the incore detectors supply information to the operator in the control room.

Flow in the outside annulus enters the plenum region in the bottom of the vessel, turns and then flows upward through the core. Approximately 1.5 percent of the upward flow passes through an annulus between the core barrel inside surface and the back side of the baffle plates. Velocity in this annulus is also limited to less than 1 ft/sec.

As the coolant exits from the core, it enters the plenum assembly. The plenum cylinder maintains the coolant flow parallel to the outside of the guide tube assemblies. Flow passes from the plenum to the two outlet nozzles through 34 inch and 22 inch diameter holes in the upper section of the plenum. The maximum flow velocity across the guide tube assemblies adjacent to the plenum outlets is approximately 19 ft/sec. At the two locations where a small amount of outlet flow passes through a cluster of twenty four 3-inch diameter holes, the flow across the adjacent guide tube assemblies is only 8 ft/sec.

The flow direction and velocity control where chosen to reduce the possibility of developing forces which would result in damaging vibrations in all regions of the core. The resulting velocities are low enough to preclude the necessity of measuring motions of the core barrel, control rod guide tube assembly (a part of the plenum assembly), and other upper plenum assembly components, as can be seen from the following:

1. The 19 ft/sec. flow velocity across the guide tube assemblies adjacent to the outlets in the plenum results in a vortex shedding frequency of only 6 cps. Since this shedding frequency is much lower than the 50 cps fundamental of the guide tube assembly, it was concluded that the assemblies will not have significant vibratory motion from the cross flow.
2. The flow velocity in the annulus between the core barrel and the thermal shield is less than 1 ft/sec. At this extremely small velocity, the vibratory motion of the shell modes will be negligible. Beam type motions of the core barrel can be measured by the upper accelerometer in the surveillance holder tube assembly. (The accelerometer instrumentation is described later.)
3. The plenum cover assembly is an extremely stiff assembly. Flow across the plenum cover occurs only at the outer edge of the assembly at a low velocity of 5 ft/sec. The force on the assembly due to flow is insignificant.
4. Since the coolant at 100 percent power operation is subcooled at the discharge of the fuel assembly, no steam bubbles exist which might induce vibration of the control rod guide tubes, plenum cylinder, or plenum cover assembly.

Pre-operational testing will yield results which are comparable to or more conservative than during operation for the following reasons:

1. The total flow is slightly greater during hot functional testing when the reactor core is not in place than during operation. This is particularly true for pump combinations of less than four pumps.
2. The velocities in areas of concern are not significantly influenced by the flow differences with or without the core.

Oconee 1

Instrumentation

The internal components which will be measured during pre-operational testing are the surveillance specimen holder tube, the thermal shield and the plenum cylinder. Details of the instrumentation follow.

A set of two accelerometer assemblies will be installed in each of two surveillance specimen holder tubes. The location of the holder tubes is shown in Figure 7-23. The accelerometer transducers will be located in the perforated section of the holder tube assembly as shown in Figure 7-24. In addition, two weights which simulate the surveillance capsules will be installed in each perforated tube.

## 7.6.2.4 System Evaluation

### 7.6.2.4.1 Operational Experience

Self-powered incore neutron detectors have been operated since 1962. Such detectors have been assembled and irradiated in a Babcock & Wilcox development program that began in 1964.

The B&W Development Program included these tests:

1. Parametric studies of the self-powered detector.
2. Detector ability to withstand PWR environment.
3. Multiple detector assembly irradiation tests.
4. Background effects.
5. Readout system tests.
6. Mechanical withdrawal-insertion tests.
7. Mechanical high pressure seal tests.
8. Relationship of flux measurement to power distribution experiments.

Conclusions drawn from the results of the test programs are as follows:

1. The detector sensitivity, resistivity, and temperature effects are satisfactory for use.
2. A multiple detector assembly can provide axial flux data in a single channel and can withstand reactor environment.
3. Background effects will not prevent satisfactory operation in a PWR environment.
4. Plant computer systems are successful as read-out system for in-core monitors.

For Incore Monitoring System development program results and conclusions, refer to B&W Topical Report BAW-10001A; "Incore Instrumentation Test Program."

### 7.6.2.4.2 Pre-Operational Testing

First-of-a-kind instrumentation which will measure flow induced vibrations at special locations during pre-operational testing was installed on the Oconee 1 internals. Confirmatory measurements were made on Oconee 2 and Oconee 3 internals.

General

The directions and velocities of the coolant flow are controlled by the design of the reactor internals and are primary criteria used to determine what internal components should not be measured. Consequently, a brief description of the coolant flow through the reactor as indicated in Figure 7-22 is given below.

Coolant for the core enters through the four reactor inlet nozzles. If is then directed downward in an outside annulus defined by the inside surface of the vessel, the core support shield, and the thermal shield. Approximately 99.6 percent of the downward flow enters an outside annulus at approximately 23 ft/sec. The remaining 0.4 percent enters an inside annulus between the inside surface of the thermal shield and the outside surface of the core barrel. The flow velocity in this annulus is limited to less than 1 ft/sec by orifices located in the bottom of the core barrel cylinder.

into calculations and are used to correct the output of the incore detectors for these factors. Operation of these detectors in both power and test reactors has demonstrated that this compensation program, when coupled with the initial sensitivity, provides detector readout accuracies sufficient to eliminate the need for a calibration system.

However, a calibration system has been installed in Oconee 1 to provide on-line confirmation of the experimentally derived compensation calculation methods. The calibration system consists of incore self-powered neutron detectors positioned by hand in selected detector assembly calibration tubes as shown in Figure 7-20.

If a loss of power event occurs, the ICS/NNI is designed to send the plant to a "Known Safe State" (KSS) by initiating a trip of both main feedwater pumps via the failsafe design of the high steam generator level monitoring circuits. These circuits are designed such that upon a loss of either "hand" or "auto" power they will initiate a trip of the main feedwater pumps which in turn will trip the main turbine and also the reactor via the Anticipatory Reactor Trip System (ARTS) circuitry. Emergency feedwater is also initiated upon loss of both feedwater pumps as described in Section 7.4.3, "Emergency Feedwater Controls" on page 7-37.

## 7.6.2 INCORE MONITORING SYSTEM

The Incore Monitoring System has been upgraded to meet the requirements of NUREG 0737 Item II.F.2.

### 7.6.2.1 Description

The Incore Monitoring System provides neutron flux detectors to monitor core performance. Incore self-powered neutron detectors measure the neutron flux in the core to provide a history of power distributions during power operation. Data obtained provides power distribution information and fuel burnup data to assist in fuel management decisions. The plant computer provides normal system readout and a backup readout system is provided for selected detectors.

### 7.6.2.2 System Design

The Incore Monitoring System consists of assemblies of self-powered neutron detectors and temperature detectors located at 52 preselected positions within the core. The incore monitoring locations are shown on Figure 7-20. In this arrangement, an incore detector assembly consisting of seven local flux detectors, one background detectors, one thermocouple and a calibration tube is installed in the instrumentation tube of each of 52 fuel assemblies. The local detectors are positioned at seven different axial elevations to indicate the axial flux gradient. The outputs of the local flux detectors are referenced to the background detector output so that the differential signal is a true measure of neutron flux. The temperature detectors located just above the top of the active fuel in the fuel assemblies measure core outlet temperature.

Multi-point recorder readouts of selected detectors are provided independent of the computer.

When the reactor is depressurized, the incore detector assemblies can be inserted or withdrawn through guide tubes which originate at a shielded area in the Reactor Building as shown in Figure 7-21. These guide tubes enter the bottom head of the reactor vessel where internal guides extend up to the instrumentation tubes of 52 selected fuel assemblies. The instrumentation tube serves as the guide for the incore detector assembly. During refueling operations, the incore detector assemblies are withdrawn approximately 13 feet to allow free transfer of the fuel assemblies. After the fuel assemblies are placed in their new location, the incore detector assemblies are returned to their fully inserted positions.

### 7.6.2.3 Calibration Techniques

The nature of the detectors permits the manufacture of nearly identical detectors which produces a high relative accuracy between individual detectors. The detector signals are compensated continuously for burnup of the neutron-sensitive material.

Calibration of detectors is not required. The incore self-powered detectors are controlled to precise levels of initial sensitivity by quality control during the manufacturing stage. The sensitivity of the detector changes over its lifetime due to such factors as detector burnup, control rod position, fuel burnup, etc. The results of experimental programs to determine the magnitude of these factors have been incorporated

*7.6.1.2.3.2 Loss-of-Load Considerations*

The nuclear unit is designed to accept 10 percent step load rejection without safety valve action or turbine bypass valve action. The combined actions of the control system and the turbine bypass valve permit a 40 percent load reduction or a turbine trip from 40 percent load without safety valve action. The controls will limit steam dump to the condenser when condenser vacuum is inadequate, in which case the safety valves may operate. The combined actions of the control system, the turbine bypass valve, and the safety valves permit separation from the external transmission system without reactor trip if there is no turbine trip.

The features that permit continued operation under load rejection conditions include:

Integrated Control System

During normal operation, the Integrated Control System controls the unit load in response to load demand from the system dispatch center or from the operator. During normal load changes and small frequency changes, turbine control is through the speed changer to maintain constant steam pressure.

4 During large load and frequency upsets, the turbine governor takes control to regulate frequency.

100 Percent Relief Capacity in the Steam System

This provision acts to reduce the effect of large load drops on the Reactor System.

Consider, for example, a sudden load rejection greater than 10 percent. When the turbine-generator starts
4 accelerating, the governor valves and the intercept valves begin to close to maintain set frequency. As the
4 governor valves close, the steam pressure rises and acts through the control system to reduce the feedwater
4 flow demand and reactor demand signals. In addition, when the load rejection is of sufficient magnitude, the turbine bypass valves open to reject excess steam to the condenser, and the safety valves open to exhaust steam to the atmosphere. The rise in steam pressure and the reduction in feedwater flow cause
4 the average reactor coolant temperature to rise, which reinforces the reactor power level demand reduction
4 to restore reactor coolant temperature to a set value.

As the turbine-generator returns to set frequency, the turbine controls revert to steam pressure control rather than frequency control. This feature holds steam pressure within relatively narrow limits and prevents further large steam pressure changes.

*7.6.1.2.3.3 Loss of Power Supply Considerations*

The ICS/NNI system power supply is arranged such that it is normally powered from a dedicated static inverter system, which receives a DC input from the Vital I & C batteries, and is backed by an AC input from one of the plants regulated non-load shed buses (Chapter 8, "Electric Power" on page 8-1). Both automatic and manual transfer switching is provided to select between these supplies.

In addition to the power supply reliability for the ICS, essential plant parameters necessary for shutdown have been arranged with their power supplies independent of the ICS source. Also, a "display group" has been developed and defined on the plant operator aid computer such that upon a loss of ICS power, the operator may quickly have full and complete information on key primary and secondary system parameters. Emergency procedures have also been developed to designate alternate sources of information on key plant parameters if the computer is unavailable, thus assuring the operator can obtain sufficient systems information.

The following additional features are provided with the reactor power controller:

1. A high limit on reactor power level demand ($N_d$).
2. An adjustable low limit on reactor power level demand ($N_d$).
3. A megawatt demand limit imposed by lack of feedwater flow capability from the steam generator controls.

The reactor controls incorporate automatic or manual rod control above 15 percent of rated power and manual rod control below 15 percent of rated power.

#### 7.6.1.2.3 System Evaluation

Redundant sensors for major system parameters are available to the Integrated Control System. The operator can select any of the redundant sensors from the control room.

Automatic signal selection is also provided for certain pairs of redundant sensors as described in Section 7.4.2.2.2, "Non-Nuclear Process Instrumentation in Regulating Systems" on page 7-31. The operator can select a redundant sensor from the pair; however, if a failure occurs, the automatic signal selector (SASS) will transfer the output signal from the failed device to the remaining valid input. The SASS also will not allow the operator to select the failed sensor if the failure occurred on the nonselected sensor.

The list of redundant pairs of sensors is contained in Section 7.4.2.2.2, "Non-Nuclear Process Instrumentation in Regulating Systems" on page 7-31.

Manual reactivity control is available at all power levels. Loss of electrical power to automatic control stations reverts the control system to manual, ultimately placing ICS in "load tracking."

Maloperation or failure of the ICS or any of its subsystems places no automatic limitations on reactor operation because the ICS reverts to the manual mode, ultimately placing the ICS in "load tracking," i.e., following the actual generated load.

The design of the NNI/ICS System in conjunction with procedures and training allow the operator to cope with various loss of power situations. Also, alarm indications provide information to the operator of various instrument and control functions. Emergency procedures provide assurance of positive responses by the operator.

The Steam Generator Level Indication System is a part of the Integrated Control System (ICS) and is not required to function during a steam line rupture accident. Failure of the ICS does not diminish the safety of the reactor. None of the functions provided by the ICS are required for reactor protection or for actuation of the ESPS. The reactor protection criteria, used in the analysis of the steam line rupture accident presented in Chapter 15, "Accident Analyses" on page 15-1 can be met irrespective of ICS action.

##### *7.6.1.2.3.1 Modes of Control*

The Integrated Control System is designed to revert to a "Load Tracking" mode of control to tie the unit to the subsystem on manual or to the subsystem being limited. In the startup control mode, the reactor is prevented from automatic rod withdrawal below 15 percent power.

In startup control mode, the controls are arranged so that the Steam System follows reactor power rather than Turbine System power demand. The controls will limit steam bypass to the condenser when condenser vacuum is inadequate.

Steam Generator Pressure Limit

Individual steam generator pressure limits respective feedwater demands whenever pressure increases in the steam generators.

Auxiliary Feedwater Header

Upon loss of all reactor coolant pumps, the ICS positions valves to direct main feedwater flow to the auxiliary feedwater header.

Feedwater Valve Control

Valve position demand for each steam generator is applied to both the startup and the main feedwater valves, through control stations. These valves are sequenced into operation so that the startup valve opens first (from zero to 15 percent load) followed by the main feedwater valve.

Feedwater Pump Control

Feedwater pump speed is controlled to maintain a constant differential pressure drop across feedwater valves.

*7.6.1.2.2.4 Reactor Control*

The reactor control is designed to maintain a constant average reactor coolant temperature over the load range from 15 to 100 percent of rated power. The steam system operates on constant pressure at all loads. The average reactor coolant temperature decreases over the range from 15 percent to zero load. Figure 7-18 shows the reactor coolant and steam temperatures and the steam pressure over the entire load range.

The reactor control consists of analog computing equipment with inputs of megawatt demand, core power, and reactor coolant average temperature. The output of the controller is an error signal that causes the control rod drive to be positioned until the error signal is within a deadband. A block diagram of the reactor control is shown on Figure 7-19.

First, reactor power level demand ($N_d$) is computed as a function of the megawatt demand ($MW_d$) and the reactor coolant system average temperature deviation ($\overline{\Delta T}$) from the set point, according to the following equation:

$$N_d = K_1 MW_d + K_2 (\overline{\Delta T} + \frac{1}{\tau} \int \overline{\Delta T}\, dt)$$

Megawatt demand is introduced as part of the demand signal through a proportional unit having a adjustable gain factor ($K_1$). The temperature deviation is introduced as a part of the demand signal after proportional plus reset (integral) action is applied. For the temperature deviation, $K_2$ is the adjustable gain and $\tau$ is the adjustable integration factor.

The reactor power level demand ($N_d$) is then compared with the reactor power level signal ($N_i$), which is derived from the nuclear instrumentation. The resultant error signal ($N_d$ - $N_i$) is the reactor power level error signal ($E_N$).

When the reactor power level error signal ($E_N$) exceeds the deadband settings, the control rod drive receives a command that withdraws or inserts rods depending upon the polarity of the power error signal.

*7.6.1.2.2.3 Steam Generator Control*

Control of the steam generator is based on matching feedwater flow to megawatt demand with bias provided by the error between steam pressure set point and steam pressure. The pressure error increases the feedwater flow demand if the pressure is low. It decreases the feedwater flow demand if the pressure is high. Figure 7-17 illustrates the steam-generator feedwater controls.

The basic control actions for parallel steam generator operation are:

1. Megawatt demand converted to feedwater demand.
2. Steam pressure compared to set pressure, and the pressure error converted to feedwater demand.
3. Total feedwater demand computed from sum of 1 and 2.
4. Total feedwater flow demand split into feedwater flow demand for each steam generator.
5. Feedwater demand compared to feedwater flow for each steam generator. The resulting error signals position the feedwater flow controls to match feedwater flow to feedwater demand for each steam generator.

For operation below 15 percent load, the steam generator control acts to maintain a preset minimum downcomer water level. The conversion to level control is automatic and is introduced into the feedwater control train through an auctioneer. At electrical loads below 15 percent, the turbine bypass valves will operate to control steam pressure rise.

The steam generator control also provides ratio, limit, and runback actions as shown in Figure 7-17 which include:

Steam Generator Load Ratio Control

Under normal conditions, the steam generators will each produce one-half of the total load. Steam generator load ratio control is provided to balance reactor inlet coolant temperatures during operation with more reactor coolant pumps in one loop than in the other.

Water Level Limits

A maximum water level limit prevents overpumping of feedwater and assures superheated steam under all operating conditions.

A minimum water limit is provided for 15 percent low load control in the downcomer section.

4

4 Reactor Outlet Low Temperature Limit

4 This limiter reduces feedwater demand when the reactor outlet temperature is low.

Feedwater Cross Limits

Reactor demand is limited by feedwater error if feedwater demand exceeds feedwater flow by 5 percent or more.

Feedwater demand is limited by neutron error when error exceeds 5 percent. When neutron error is greater than ± 5 percent, feedwater demand is modified to bring the error back within ± 5 percent.

4 Rate limiting is designed as a function of load, so transients are limited as shown in Table 7-6.

The limiter acts to runback and/or limit the load demand under any of the following conditions:

1. Loss of one or more reactor coolant pumps.
2. Total feedwater flow lags total feedwater demand, or reactor power lags reactor power demand, by more than 5 percent.
3. Loss of one feedwater pump.
4. Asymmetric rod withdrawal patterns exists in reactor.
5. The generator separates from the 230 KV bus.

The output of the limiters is a megawatt demand signal which is applied to the turbine control, steam-generator feedwater control, and reactor control in parallel.

The controlling subsystems of the ICS (turbine control, steam-generator feedwater control, and reactor control) normally operate in the automatic mode in response to a demand signal from the ULD. The subsystems control function is kept within pre-established bounds under other than normal automatic operation by a "load tracking" feature built into the ICS. The system will switch to the load tracking mode if either of the following conditions exists:

One or more of the subsystems are in manual.

Errors greater than preset limits develop between the demand and the variable.

In this mode, the load demand is made to follow the manual or limited control subsystem. Load tracking continues until the limiting condition is brought back to within the pre-established deadband or the subsystem is returned to automatic operation.

*7.6.1.2.2.2 The Integrated Master*

The Integrated Master has been designed to receive the megawatt demand signal from the Unit Load Demand Subsystem and convert this signal into a demand for the feedwater, turbine, and reactor control. A functional diagram of the Integrated Master Control is shown in Figure 7-16. The megawatt demand is compared with the generator megawatt output, and the resulting megawatt error signal is used to change the steam pressure set point. The turbine valves then change position to control steam pressure. As the megawatt error reduces to zero, the steam pressure set point is returned to the steady state value. By limiting the effect of megawatt error on the steam pressure set point, the system can be adjusted to permit controlled variations in steam pressure to achieve the desired rate of turbine response to megawatt demand.

Unit load demand is utilized as the feed-forward demand to the steam generator, reactor, and turbine while operating in the integrated control mode. This demand is compensated for deviations in the steam header pressure.

The Turbine Bypass System operates from the header pressure error or individual steam generator pressures as an overpressure relief for the turbine header. The Turbine Bypass System permits a load drop of 40 percent, or a turbine trip from 40 percent load without safety valve operation.

The Integrated Control System maintains constant average reactor coolant temperature between 15 and 100 percent rated power, and constant steam pressure at all loads. Optimum unit performance is maintained by limiting steam pressure variations; by limiting the unbalance between the steam generator, the turbine, and the the reactor; and by limiting the total unit load demand upon loss of capability of the steam generator feed system, the reactor, or the turbine generator. The control system provides limiting actions to assure proper relationships between the generated load, turbine valves, feedwater flow, and reactor power.

The response of the Reactor Coolant System to increasing and decreasing power transients is limited by the Integrated Control System as indicated in Table 7-6. The Turbine Bypass System permits a load drop of 40 percent or a turbine trip from 40 per cent load without safety valve operation.

#### 7.6.1.2.2 Description

The Integrated Control System includes four independent subsystems as shown in Figure 7-14. The four subsystems are: the Unit Load Demand; the Integrated Master; the Steam Generator; and the Reactor. The system philosophy is that control of the plant is achieved through feed-forward control from the Unit Load Demand. The Unit Load Demand produces demands for parallel control of the turbine, reactor, and Steam Generator Feedwater System through respective subsystems.

The Steam Generator Control is capable of automatic or manual feedwater control from a startup to full output. The Integrated Master Control is capable of automatic or manual turbine valve control from minimum turbine load to full output, and of manual control below minimum turbine load. The Reactor Control is designed for automatic or manual operation above 15 percent output, and for manual operation below 15 percent.

The basic function of the Integrated Control System is matching megawatt generation to unit load demand. The Integrated Control System does this by coordinating the steam flow to the turbine with the rate of steam generation. To accomplish this efficiently, the following basic reactor/steam-generator requirements are satisfied:

The ratios of feedwater flow and BTU input to the steam generator are balanced as required to obtain desired steam conditions.

BTU input and feedwater flow are controlled:

1. To compensate for changes in fluid and energy inventory requirements at each load.
2. To compensate for temporary deviations in feedwater temperature resulting from load change, feedwater heating system upsets or final steam pressure changes.

##### *7.6.1.2.2.1 Unit Load Demand*

The Unit Load Demand (ULD) is designed of accomplish three objectives related to the operation of the plant. First, the ULD conditions the load demand signal from the system dispatcher to make it compatible with the power level of the plant and its ability to change load. Second, the ULD permits the operator to separate the plant from the dispatch system and manually establish the power output. Third, the ULD initiates load limiting and runback functions to restrict operation within prescribed limits. Figure 7-15 illustrates the functions incorporated in the subsystem.

The Unit Load Demand obtains a load demand signal from the system dispatcher, the plant computer, or the operator. The load demand is restrained by a maximum load limiter, a minimum load limiter, a rate limiter, and a runback limiter.

Circuit Operation - The sequence monitor continuously compares the group average (reference) signals for each regulating rod group with the allowable sequence patterns. Bistable amplifiers and digital logic are used for this purpose. In addition, the rod group "enable" circuits are monitored to determine if a group is enabled for motion out-of-turn. The safety rod groups' out limit signals serve as a permissive to automatic sequencing: the sequence monitor prevents automatic control until the safety rods are fully withdrawn.

Corrective Action - When an out-of-sequence condition is detected and operation is in the automatic control mode, the automatic mode disengages and an alarm lamp alerts the operator to the malfunction. Control reverts to manual and remains in manual until the fault is corrected and the system is reset by the operator.

Trip Fault Detector

Design Basis - To sense faults which may affect operation of the trip circuits, such as one trip breaker in the tripped position during normal operation.

Circuit Operation - The circuit contains elements which sense the state of each trip device as well as the state of each of the four trip channels. If the state of a device fails to agree with the state of its associated trip channel, a trip fault will be alarmed. Other independent circuits sense and confirm that a trip, if commanded, has actually occurred.

Corrective Action - Alarms are provided.

Safety Rods Not Withdrawn

Design Basis - To prevent, on plant startup, withdrawal of the regulating rods until the safety rods are fully withdrawn.

Circuit Operation - The circuit continuously monitors the group "out" limit for the four safety rod groups. When the four groups are all fully withdrawn, signals are sent to the sequencer and the sequence monitor which permit automatic control.

Corrective Action - Alarms are provided.

2

Rod Position Sensor Faults

All rod position sensor faults lead to false asymmetric, stuck, or dropped rod symptoms which are acted upon by the Asymmetric Rod Monitor previously described.

## 7.6.1.2 Integrated Control System

### 7.6.1.2.1 Design Basis

The Integrated Control System (ICS) provides the proper coordination of the reactor, steam generator feedwater control, and turbine under all operating conditions. Proper coordination consists of producing the best load response to the unit load demand while recognizing the capabilities and limitations of the reactor, steam-generator feedwater system, and turbine. When any single portion of the plant is at an operating limit or control selection is on manual, the Integrated Control System design uses the limited or manual section as a load reference.

Faults serious enough to warrant immediate action produce automatic correction commands from the fault detection circuits, and manual bypass in not possible. Status indicators on the operator's console provide monitoring of control modes.

A description of each fault detector follows:

Asymmetric Rod Monitor

Design Basis - To detect and alarm if any rod deviates from its group reference position by more than a maximum of nine inches true position.

Circuit Operation - There are 69 asymmetric rod pattern monitors, one assigned to each control rod. These monitors continuously compare the individual rod absolute position signal with the absolute group reference (average) signal. The absolute value of the difference between the two signals is computed, and if this difference is less than the maximum value set by the circuit calibration, no output results. If, however, the difference is greater than the setpoint, a relay is actuated which alarms the asymmetric condition. Two alarm channels are provided in each monitor which are identical except for the setpoints. One setpoint is calibrated for a 3-inch signal differential (maximum 7-inch true position separation and initiates an alarm. The other setpoint is at 5-inch signal differential (maximum 9-inch true position separation) and initiates the action described below.

Corrective Action - Action taken upon detection of an asymmetric rod fault depends upon the control mode and the power level in effect at the time the fault is detected. Corrective action is the same for any asymmetric condition including "stuck-in," "stuck-out," or dropped control rods.

Detection of a 3-inch signal differential is defined as an "asymmetric rods alarm." Actuation of this alarm causes the fault indicator lamp for that rod to be energized and an alarm signal to be sent to the plant computer and annunciator.

If the condition is not corrected and the separation increases to a 5-inch signal difference, the following actions occur:

"Asymmetric fault" lamp on the operator's console is energized. If operation is in the manual control mode, operator action is required by administrative control.

If operation is in the automatic mode, a "runback fault" signal is sent to the Integrated Control System. The ICS will impose a maximum reactor power level of 60 percent of rated power if power is initially less than 60 percent.

When reactor power is greater than 60 percent of rated power, the Rod Drive Control System generates an "Out Inhibit" signal which disables the "Out" command circuits to all drives and the ICS initiates a runback to 60 percent reactor power. "Out Inhibit" alarms are sent to the ICS, plant annunciator and plant computer.

Reactor power remains limited to 60 percent maximum in automatic control until the fault is corrected.

2

Sequence Monitor

Design Basis - To detect any motion of the regulating rod groups outside of the predetermined automatic sequence patterns, and to prevent further automatic motion when such conditions occur.

Startup Considerations

The rod drive controls receive interlock signals from the ICS and nuclear instrumentation (NI). These inputs are used to inhibit automatic mode selection when a large error exists in the ICS reactor control subsystem and to inhibit out motion for high startup rates, respectively.

In addition to the startup considerations, dilution controls, to permit removal of reactor shutdown concentrations of boron in the reactor coolant, are provided. This control bypasses the normal reactor coolant dilution controls, described in Section 7.6.1.1.6, "System Design" on page 7-68, providing all safety rods are withdrawn from the core and the operator initiates a continuous feed and bleed cycle. An additional interlock on rod Group 5 inhibits the use of this circuit when rod Group 5 is more than 80 percent withdrawn.

Operational Considerations

The control rod assembly positioning system provides the ability to move any rod to any position required consistent with reactor safety. As noted in Section 7.6.1.1.8, "System Evaluation" on page 7-71, a uniform speed is provided by the drive system. A fixed rod position when motion is not required is obtained by the power supply ability to energize two adjacent windings of the CRA motor stator. This static energizing of the windings maintain a latched stator and fixed rod position.

Position Indication

As previously described, two separate position indication signals are provided. The absolute position sensing system produces signals proportional to CRD position from the reed switch matrix located on each CRD mechanism. The relative position indication system produces a signal proportional to the number of CRD motor power pulses from a stepping motor and precision potentiometer for each CRD mechanism.

Position indicating readout devices mounted on the operator's console consist of 69 single rod position
2 meters. The operation of a selector switch permits either relative or absolute position information to be displayed on the single rod meters.

Indicator lights are provided on the single-rod meter panel to indicate when each rod is (1) fully inserted, (2) fully withdrawn, (3) under control, and (4) whether a fault is present. Indicators on the operator's console show full insertion, full withdrawal, under control, and fault indication for each of the eight control rod groups.

Failures which could result in unplanned control rod withdrawal are continuously monitored by fault detection circuits. When failures are detected, indicator lights and alarms alert the operator. Fault indicator lights remain on until the fault condition is cleared by the operator. A list of indicated faults is shown below:

1. Asymmetric rod patterns (indicator and alarm).
2. 2. Sequence faults (indicator and alarm).
2. 3. Trip faults (indicator and alarm).
2. 4. Safety rods not withdrawn (indicator only).
2. 5. Rod position sensor faults.

mechanisms downstream from the reactor trip points such that the CRA are held in their withdrawn positions after a trip is not considered credible for the following reasons:

1. The secondary trip devices in the Rod Drive Control System remove all DC power from the drives.
2. Control rod drive power cables are terminated at only three points between the Rod Drive Control System cabinets and the drive mechanisms.

   Two of these terminations are made outside and inside the Reactor Building electrical penetrations
2 inside junction boxes containing only control rod drive power cables. The third termination is made in bulkhead connectors (one per drive) in the area of the reactor. The only other cables terminated in this area are the control rod drive instrumentation cables. The instrumentation cables are terminated in bulkhead connectors of a different size and configuration, therefore mismating of connectors could not be accomplished.
3. No cable splices are permitted between termination points described.
4. DC systems from the batteries at Oconee are not grounded and are equipped with ground detecting circuitry.

In summary, series redundant trip devices having adequate rating, testability and a "split bus" arrangement insure safety of reactor trip circuits.

Reactivity Rate Limits

The desired rate of change of CRA reactivity insertion and uniform reactivity distribution over the core are provided for by the control rod drive and power supply design, and the selection of rods in a group. The CRA motor, lead screw, and power supply designs are fixed to provide a uniform rate of speed of 30
2 in./min. The speed is determined by the solid state programmer, which digitally controls speed. The reactivity change is then controlled by the rod group size. To insure flexibility in this area, a patch panel has been included in the power supply to enable the interchange of rod worth between rod groups. Any rod may be patched into any group with the exception of Group 8.

Uniform and symmetrical group insertion rate is provided for by synchronous withdrawal of all rods in that group. Such synchronous withdrawal is achieved by the design of the power supply. A group power supply operates synchronously by having its load (4 to 12 CRA motor windings) connected in parallel on the output of the SCR's. As the programmer gates on the SCR's, all rods in a group have the same motor winding energized simultaneously producing synchronous motion of the entire group.

Each control rod is provided with a rod position indication monitor to sense asymmetric rod patterns by comparing the individual rod position with its group average position. When the rod moves out-of-step from its group by a preset amount, the monitor alarms the condition to the operator, the plant computer, and the ICS. Depending on the power setting and the control mode, action is initiated by the ICS to insert rods and reduce power.

As the lead screw (and the control rod assembly) moves, the switches operate sequentially, producing an
2 analog voltage proportional to position. Other reed switches included in the same tube with the position indicator matrix provide full-in and full-out limit indications.

The relative position transducer is a small pulse-stepping motor, driven from the power supply for the rod
2 drive motor. This small motor is coupled to a potentiometer which produces an output signal
2 proportional to rod position demand as indicated by the number of power pulses received by the rod drive
2 motor.

Rod Drive Control System trip breakers are provided to interrupt power to the control rod drive mechanisms. When power is removed, the roller nuts disengage from the lead screw allowing a gravity trip of the CRA.

The Group 8 drive mechanisms are modified to prevent rod drop into the core when power is removed from the stators. In this type of mechanism, the roller nuts are mechanically restrained to remain engaged with the lead screw at all times. Thus, the mechanical "trip" action has been removed from these AS PR, and they remain at the position they occupied immediately before trip was initiated. When a reactor trip is initiated, power to the Group 8 power supply is interrupted in the same manner as for the other regulating power supplies. Two series trip methods are provided for removal of power to the CRD mechanisms. First, a trip is initiated when Reactor Protective System logic interrupts power to the undervoltage (UV) coil of the main AC feeder breakers. Secondly, a trip is initiated when the Silicon Control Rectifier gating power and the DC holding power is interrupted. As parallel power feeds are provided on both holding and gating power, interruption of both feeds is required for trip action in either method of trip. Trip circuitry is shown in Figure 7-4 and Figure 7-13.

AC power feed breakers are of the three-pole, stored-energy type and are equipped with instantaneous undervoltage trip coils. Each AC feed breaker is housed in a separate metal clad enclosure. The secondary trip breakers are also of the stored-energy type with two parallel-connected instantaneous undervoltage trip coils consisting of two 2-pole breakers mechanically ganged to interrupt DC buses. All breakers are motor-driven-reset to provide remote reset capability. Each undervoltage trip coil is operated from the Reactor Protective System.

#### 7.6.1.1.8 System Evaluation

Safety Considerations

A reactor trip occurs whenever power has been removed from the rod drive motors. The design provides two stored energy breakers which do not require power to interrupt the electrical feeds to rod drive control power supplies and a second set of circuit-interrupting devices in the series on the output of the power supplies. All devices have interrupting capacity of sufficient rating to open under any group load configuration. Reactor trip is further assured by providing series trip devices, split buses, and provisions for periodic testing. Trip redundancy is provided by series breakers while availability and testability are provided through dual power sources. Redundant power supplies permit testing of the trip action of each power-interrupting device without loss of plant availability.

Reactivity shutdown margin provided by the safety rods is assured by diversification of their power buses. This feature, as shown in Figure 7-4 utilizes four separate buses, each having a separate trip device, to power the safety rods. A failure in one bus does not reflect into the other buses, therefore, a single failure in the distribution system for the safety rods does not prevent a plant shutdown.

The minimum voltage required to hold a drive in a withdrawn position is 42 volt DC per coil (2 coil "hold" mode). The probability of an external DC source being applied to the control rod drive

trip function. Major components of the logic system are the Operator's Control Panel, CRA position indication panels, automatic sequencer, and relay logic.

Switches are provided at the operators control panel for selection of desired rod control mode. Control modes are: (1) Automatic mode--where rod motion is commanded by the Integrated Control System; and (2) Manual mode--where rod motion is commanded by the operator. Manual control permits operation of a single rod or a group of rods. Alarm lamps on the RDC panel alert the operator to the systems status at all times. The Group 8 control rods can only be controlled manually even when the remainder of the system is in automatic control.

The sequence section of the logic system utilizes rod position signals to generate control interlocks which regulate rod group withdrawal and insertion. The sequencer operates in both automatic and manual modes of reactor control, and controls the regulating groups only. Analog position signals are generated by the reed switch matrix on the CRA, and an average group position is generated by an averaging network. This average signal serves as an input to electronic trip units which are activated at approximately 25 and at 75 percent of group rod withdrawal. Two bistable units are provided for each regulating group. Outputs of these bistables actuate "enable" relays which permit the rod groups to be commanded in automatic or manual mode.

The automatic sequencer circuit can control only rod Groups 5, 6, and 7. The safety rod groups, Groups 1, through 4, are controlled manually, one group at a time. In addition, the operator must select the safety group to be controlled and transfer it to the auxiliary power supply before control is possible. There is no way in which the automatic sequencer can affect the operations required to move the safety rods.

In addition to the sequencer, relay logic monitors are provided in the "enable" circuits which prohibit out of sequence conditions. The selection of manual control mode and sequence bypass mode functions permit intentional out-of-sequence conditions. This condition is indicated to the operator. If automatic control is selected, "sequence" operation cannot be bypassed.

"Sequence bypass" operation Permits selection of any rod group or any single rod for control. It will not permit selection of more than one rod group at any given time. Motion of more than one group at any given time is also not possible when this operation is selected.

4 Inputs to the system logic from the Nuclear Instrumentation and the Integrated Control System provide interlock control over rod motion. These interlocks cause rod motion command lines and control mode selection to be inhibited.

Under certain conditions, the nuclear instrumentation generates an "out inhibit" signal. When this signal is received by the Rod Drive Control System, all out command circuits are disabled, thus preventing withdrawal of all rods in either automatic or manual control.

Automatic operation of rods can only be commanded by the ICS when the Rod Drive Control System is in the automatic mode. These commands can only affect rod Groups 5, 6, and 7.

In the Rod Drive Control System, two methods of position indication are provided: an absolute position indicator and a relative position indicator. Either position signal is available to the control board indicator through a selector switch. The absolute position transducer consists of a series of magnetically operated reed switches mounted in a tube parallel to the motor tube extension. Each switch is hermetically sealed. Switch contacts close when a permanent magnet mounted on the upper end of the lead screw extension comes in close proximity.

During startup, the safety rod groups are withdrawn first, enabling withdrawal of the regulating control groups. The sequence allows operation of only one regulating rod group at a time except where reactivity insertion rates are low (first and last 25 percent of stroke), at which time two adjacent groups are operated simultaneously in overlapped fashion. These insertion rates are shown in Figure 7-11.

As fuel is used, dilution of soluble boron in the reactor coolant is necessary. When Group 6 is more than 95 percent withdrawn, interlocks permit dilution. The reactor controls insert Group 6 to compensate for the reduction in boron concentration by dilution. The dilution is automatically terminated by a pre-set volume measuring device. Interlocks are also provided on Group 6 rod position to terminate dilution at a pre-set insertion limit.

#### 7.6.1.1.7 System Equipment

The Rod Drive Control System consists of three basic components: (1) control rod drive motor power supplies; (2) system logic; and (3) trip breakers. The power supplies consist of four group power supplies, an auxiliary power supply, and two holding power supplies. See Figure 7-4, Figure 7-12, and Figure 7-13. The group power supplies are of a redundant six-phase half-wave rectifier design.

In each half of a group power supply, rectification and switching of power is accomplished through the use of Silicon Controlled Rectifiers (SCR). This switching sequentially energizes first two, then three, then two of the six CRA motor stator windings in stepping motor fashion, to produce a rotating magnetic field for the control rod assembly motor to position the CRA. Switching is achieved by gating the six SCR on for the period each winding must be energized. As each of the six windings utilize SCR to supply power, six gating signals are required.

Gating signals for the group power supplies are generated by a solid state programmer consisting of a microcomputer which is programmed to accept operational commands from the input interface relays and convert them to sequential outputs which cause the mechanism motors to step at the proper speed and direction of rotation. The microcomputer is also programmed to provide the 3-2 hold control, which ensures a two-coil hold when there are no commands. The programmer is redundant (except for microprocessor power supply) thus providing separate but synchronized gating signals to the dual power supply units.

Identical power supplies are used for the regulating (control) groups and for the auxiliary power supply. Each half of each group power supply is capable of driving up to 12 drive mechanisms which is the maximum number that may be in any one group. The power supplies have dual power inputs, each half fed from separate power sources and each half being capable of carrying the full load.

Unlike the control group power supplies, the holding power supply is used to maintain the safety rods fully withdrawn; consequently, switching is not required. A six-phase d-c power supply is used for this purpose. Two holding power supplies are provided. Each is rated to furnish power to one winding of 48 mechanisms; normal load would be 41 coils for each power supply.

The auxiliary power supply is used to position the safety rod groups and to provide single rod control. The safety rod groups are maneuvered with the auxiliary power supply, and then when fully positioned, are transferred to the holding busses described above. After positioning the safety rods, the auxiliary power supply is available to the regulating groups, through transfer relays, to serve either as a single rod controller, should repositioning of a single rod be necessary, or, as a spare group controller, should one of the group control power supplies require maintenance. The auxiliary power supply cannot be used to control more than one group at one time.

The system logic encompasses those functions which command control rod motion in the manual or automatic modes of operation, including group sequencing, safety and protection features, and the manual

#### 7.6.1.1.4 Startup Considerations

The Rod Drive Control System design bases for startup are as follows:

Reactor regulation during startup is a manual operation.

4 Control rod "out" motion is inhibited when a high startup rate in the wide range is detected.

#### 7.6.1.1.5 Operational Considerations

For operation of the reactor, functional criteria related to the rod drive control system are:

CRA Positioning

The Rod Drive Control System provides for controlled withdrawal, controlled insertion, and holding of the control rod assemblies (CRA) to establish and maintain the power level required for a given reactor coolant boron concentration.

Position Indication

Continuous rod position indication, as well as full-in and full-out position indication, are provided for each control rod drive.

System Monitoring

The Rod Drive Control System design includes provisions for routinely monitoring conditions that are important to safety and reliability.

#### 7.6.1.1.6 System Design

The Rod Drive Control System provides for withdrawal and insertion of the control rod assemblies to maintain the desired reactor output. This is achieved either through automatic control by the Integrated Control System discussed in Section 7.6.1.2, "Integrated Control System" on page 7-75 or through manual control by the operator. As noted previously, this control compensates for short term reactivity changes. It is achieved through the positioning in the core of sixty-one control rod assemblies and eight axial power-shaping rod assemblies. The sixty-one rods are grouped for control and safety purposes into seven groups. Four groups function as safety rods, and three groups serve as regulating rods. An eighth group serves to regulate axial power peaking due to xenon poisoning. Seven of the eight groups may be assigned from four to twelve control rod assemblies. Eight rod assemblies are used in Group 8.

Control rods are arranged into groups at the control rod drive control system patch panel. Typically,
5 thirty-six rods, including the axial power shaping rods, are assigned to the regulating groups, and
5 thirty-three rods are assigned to the safety rod groups. A typical rod grouping arrangement is shown below:

| Safety Rods | Regulating Rods | Axial Power Shaping Rods |
|---|---|---|
| Group 1 - 8 | Group 5 - 12 | Group 8 - 8 |
| 5 Group 2 - 8 | Group 6 - 8 | |
| 5 Group 3 - 8 | Group 7 - 8 | |
| 5 Group 4 - 9 | | |

# 7.6 CONTROL SYSTEMS NOT REQUIRED FOR SAFETY

## 7.6.1 REGULATION SYSTEMS

Reactor output is regulated by the use of movable control rod assemblies and soluble boron dissolved in the coolant. Control of relatively fast reactivity effects, including Doppler, xenon, and moderator temperature effects, is accomplished by the control rods. The control response speed is designed to overcome these reactivity effects. Relatively slow reactivity effects, such as fuel burnup, fission product buildup, samarium buildup, and hot-to-cold moderator reactivity deficit, are controlled by soluble boron.

Control rods are normally used for control of xenon transients associated with normal reactor power changes. Chemical shim shall be used in conjunction with control rods to compensate for equilibrium xenon conditions. Reactivity control may be exchanged between rods and soluble boron consistent with limitations on power peaking.

Reactor regulation is a composite function of the Integrated Control System and Rod Drive Control System. Design data for these subsystems are given in the following sections.

### 7.6.1.1 Rod Drive Control System

The Rod Drive Control System (RDC) includes drive controls, power supplies, position indicators, operating panels and indicators, safety devices, and enclosures.

#### 7.6.1.1.1 Design Basis

The Rod Drive Control System design bases are categorized into safety considerations, reactivity rate limits, startup considerations, and operational considerations.

#### 7.6.1.1.2 Safety Considerations

The control rod assemblies (CRA) are inserted into the core upon receipt of protective system trip signals. Trip command has priority over all other commands.

No single failure shall inhibit the protective action of the Rod Drive Control System.

#### 7.6.1.1.3 Reactivity Rate Limits

The speed of the mechanism and group rod worth provide the reactivity change rates required. For design purposes, the maximum rate of change of reactivity that can be inserted by any group of rods has been set at $1.1 \times 10^{-4}$ $\Delta K/K/s$. The drive controls, i.e., the drive mechanism and rods combination, have an inherent speed-limiting feature.

Speed-limiting is accomplished through the use of 60 Hz synchronous programmer motors. These motors are powered, through transformers, from the same 600 VAC source as the remainder of the Rod Drive Control System. Thus, the speed of rod motion is locked to the plant's AC power frequency which, in turn, is limited to 64 Hz maximum as controlled by the plant and system frequency control system. At 64 Hz, the speed of rod motion is $(64/60) \times 30 = 32$ in./minute.

2 **7.5.2.57 Atmospheric Stability**

2 The indicated range for atmospheric stability is -4° to 8°C for 44.7 meter interval. Loop accuracy is at
2 least +0.15°C. This range is adequate for Oconee site meteorological conditions.

3 **7.5.2.58 Low Pressure Service Water Flow to Low Pressure Injection Coolers**

3 Two QA Condition 1 instrumentation channels are provided (one per train) for post accident monitoring
3 of Low Pressure Service Water (LPSW) flow to the Low Pressure Injection (LPI) coolers in response to
3 Regulatory Guide 1.97. Each instrument channel is seismically and environmentally qualified and
3 powered from a safety grade power source. Each instrument channel signal inputs to a qualified indicator
3 and recorder. These channels monitor LPSW flow to the LPI Coolers over a range of 0-8000 gpm which
3 envelopes the 0-110% of design flow criteria for Regulatory Guide 1.97.

3 Two non-safety instrument channels are provided, one per train, for indication of LPSW flow to LPI
4 Cooler and control of valves LPSW-251, 252 (Unit 1 and 2) and LPSW-404, 405 (Unit 3). Each
3 instrument signal inputs to a controller which monitors flow and valve control. These channels monitor
3 LPSW flow to the LPI Cooler over a range of 0-6000 gpm. These instrument channels are not required
3 for Regulatory Guide 1.97 and are used for normal operation.

3 LPSW flow to LPI Coolers is a Type A variable at Oconee because the operator relies on this
3 information following a design basis event (LOCA) to throttle LPSW flow to LPI Coolers to maintain
3 proper flow balance in the LPSW System.

3 (RE: NSMs ON-1/2/32923)

2 and 1 to $10^7$ cpm for the normal range channel corresponding to approximately $10^{-7}$ to 3 x $10^3$ μCi/ml
2 Xe-133eq. This instrumentation is installed in a mild environment.

### 2 7.5.2.53 Unit Vent Flow

2 The installed instrumentation indicates flow in the unit vent stack over the range of 0 to 110% of design
2 flow. The design flow for the Unit 1 stack is 110,340 SCFM (112,145 for Unit 2; 127,250 for Unit 3).
2 The indicator and recorder, Units 1, 2 and 3 respectively, encompass the following dual ranges:

| | | |
|---|---|---|
| 2 Unit 1&2 | - | 0 to 100 x $10^3$ SCFM |
| 2 | | 0 to 124 x $10^3$ SCFM |
| 2 Unit 3 | - | 0 to 100 x $10^3$ SCFM |
| 2 | | 0 to 140 x $10^3$ SCFM |

2 The primary instrument loop which contains the transmitter, the plant computer and the retransmitter is
2 powered by a highly reliable battery backed bus. The secondary instrument loop contains the
2 retransmitter, indicator and recorder, and is powered by a highly reliable auxiliary bus. The
2 instrumentation is located in a mild environment and envelopes the Regulatory Guide 1.97, Rev. 2 range
2 criteria of 0 to 110% of design flow.

### 2 7.5.2.54 Main Steam Line Radiation Monitors

2 Area radiation monitors are located adjacent to the main steam lines to detect radioactivity emitted from
3 main steam. The monitors for all 3 units are located upstream of the main steam relief valves.
3 Correlation curves allow conversion of the monitor readings in mR/hr to μCi/cc. The indicated range for
3 the monitors is $10^{-2}$ to $10^7$ mR/hr. The monitors are powered from a highly reliable non load shed power
3 bus capable of receiving power from the on-site emergency power sources. This instrumentation is rated
2 to withstand the environmental conditions that would exist during accidents in which it is intended to
2 operate. A steam line break in the vicinity of this instrumentation may cause the environment to exceed
2 the rated temperature, however, the instrument is not required to remain operational for this event.

3

### 2 7.5.2.55 Wind Direction

2 Oconee has two channels of wind direction instrumentation. The indicated range is 0 to 540°. The
2 starting speed 0.7 mph and the damping ratio is 0.4 both at 10°. Loop accuracy is approximately +/- 6°.
2 The distant constant is 3.7 ft. (1.1 meter).

2 The recommended accuracy of +/- 5° over a range of 0 to 360° equates to +/- 1.39% full scale. The
2 Oconee wind direction loop accuracy is approximately +/- 1.11% full scale which exceeds the equivalent
2 recommended accuracy. Duke has determined the existing instrumentation is adequate for the intended
2 monitoring function.

### 2 7.5.2.56 Wind Speed

2 Oconee has two channels of wind speed instrumentation. The indicated range is 0 - 60 mph. The loop
2 accuracy is 0.5 mph or better under 25 mph and the starting threshold is 0.63 mph. The range of the
2 installed instruments is adequate for Oconee site meteorological conditions.

2 ### 7.5.2.48 RC Bleed Holdup Tank Level

2 The indicated range for this variable is 0 to 180 inches for the RC Bleed Holdup tank. This level
2 indication corresponds to a tank volume of approximately 1% to 99%. Although the range is not in
2 complete compliance with the recommendation of RG 1.97, Rev. 2 (top to bottom), the tap to tap range
2 of the installed instruments is adequate to provide tank level information for all design basis events. Duke
2 considers the installed instrumentation adequate for the intended monitoring function.

2 ### 7.5.2.49 Waste Gas Decay Tank Pressure

2 Oconee utilizes two tanks per unit for radioactive waste gas storage. The maximum operating pressure for
2 these tanks is approximately 100 psig (per Oconee FSAR, Section 11.3, "Gaseous Waste Management
2 Systems" on page 11-9). The indicated range is 0 to 150 psig for each tank, which is adequate for the
2 intended monitoring function.

2 ### 7.5.2.50 Emergency Ventilation Damper Position

2 There are three Emergency Ventilation Systems at Oconee; Reactor Building Purge, Penetration Room
2 Ventilation, and Reactor Building Cooling. Each system has indication that the required emergency
2 alignment has been achieved in the control room.

2 For the Reactor Building Purge System direct indication of damper position is provided. For the
2 in-containment damper limit switches in this system environmental qualification is provided. This
2 instrumentation is powered from safety grade emergency power. For the out-of-containment damper limit
2 switches, environmental qualification documentation may not be available and power is from a highly
2 reliable battery backed bus.

2 For the Penetration Room Ventilation System, positive indication of proper system operation is provided
2 by the Penetration Room Pressure Instrumentation. This instrumentation is pneumatic and is supplied
2 by normal Station Air System. The Unit 1 and 2 instruments are located in mild environments; however,
2 the Unit 3 instrumentation may be in a harsh environment and qualification documentation may not be
2 available.

2 For a description of the instrumentation required to determine proper operation of the Reactor Building
2 Cooling System see FSAR Section 7.5.2.40, "Emergency Feedwater Flow" on page 7-61.

2 ### 7.5.2.51 Emergency Power System Status

2 All safety-grade emergency or battery backed control busses have undervoltage alarms in the Control
2 Room with local diagnostic capabilities to enable an expedient assessment of abnormal situations. In
2 addition, the 125 VDC distribution centers have analog indicators of voltage level in the Control Room.
2 All of the Control Room alarms are on highly reliable battery backed busses. All of the sensing relays
2 and alarm electronics are located in a mild environment. See FSAR Chapter 8, "Electric Power" on
2 page 8-1.

2 ### 7.5.2.52 Unit Vent Radioactive Discharge Monitors

2 Oconee has a normal range, high range and high-high range channel of unit vent radioactivity
2 instrumentation. These channels are powered from a highly reliable non load shed power bus with the
2 exception of the Unit 2 high-high range channel which is powered from a highly reliable battery backed
2 bus. The indicated range is 1 to $10^8$ R/hr gross gamma for the high-high range monitor which envelopes
2 the upper end of the recommended range. The indicated range is 1 to $10^6$ cpm for the high range channel

2 The installed instrumentation is adequate for the intended monitoring function. For accidents in which
2 harsh environments are a result, the portion of the system containing this instrumentation is not required
2 for the mitigation of these accidents and is automatically isolated upon an ESF Actuation. Therefore,
2 Letdown Flow is not a key variable for accident monitoring and is considered to be Category 3
2 instrumentation. The level of environmental qualification provided for the instrumentation in this system
2 is consistent with the performance expectations of the system and meets the recommendations of Category
2 3 in Duke's interpretation of RG 1.97, Rev. 2.

### 2 7.5.2.45 Letdown Storage Tank Level

2 The existing instrumentation for this variable provides continuous monitoring of the letdown storage tank
2 level. The loop range is 0 to 100 inches which covers the linear portion of the tank (approximately 16 to
2 84% of tank volume). This instrument loop is powered from a highly reliable battery backed bus. This
2 instrumentation is located in a mild environment.

2 Minimum and maximum letdown storage tank levels are maintained within the range of the instrument.
2 Extending the range into the domed portions of this tank would result in nonlinear readings at each
2 extreme of the scale. The installed range is adequate for measuring letdown storage tank level and Duke
2 considers the installed instrumentation adequate for the intended monitoring function. Although Category
2 2 recommendations are met, this tank is not required to be utilized during an accident. Therefore, a
2 classification of Category 3 for this instrumentation is appropriate for Oconee.

### 2 7.5.2.46 Low Pressure Service Water Temperature to ESF System

2 The Oconee system for providing cooling water to ESF components is the Low Pressure Service Water
2 System (LPSW). The inlet temperature of the LPSW by design is based on a maximum temperature of
2 75°F from near the bottom of Lake Keowee. There is no control over the temperature of the LPSW;
2 therefore, there is no need to indicate the LPSW temperature in the control room since no operator action
2 is taken based on this temperature and, by design, no useful information would be provided to the
2 operator by such instrumentation.

### 2 7.5.2.47 Low Pressure Service Water Flow to ESF Systems (Pressure)

2 The Oconee system for providing cooling water to ESF components is the Low Pressure Service Water
2 System (LPSW). Primary indication of proper LPSW system and pump operation is line pressure
2 measured in each of the two LPSW headers. The indicated range is 0 to 100 psig for a system design
4 pressure of 100 psig. These instruments are located in a mild environment and powered by a highly
2 reliable battery backed source which meets Category 2 requirements. Additional instrument loops provide
2 backup indication in the Control Room of proper system operation. These include LPSW pump motor
2 amperage, valve position indication on valves operated in the control room, inlet and/or outlet cooling
2 water flow for certain ESF coolers, and flow and pressure alarms.

2 LPSW header pressure is a valid measurement of system and pump operation and Duke considers the
2 existing indications to meet the intent of Regulatory Guide 1.97, Rev. 2. For backup variables, a design
2 qualification of Category 3 is adequate for the intended monitoring functions and consistent with the
2 performance expectations of the instrumentation.

4 (RE: NSMs ON-1/3/32590)

2 Safety Related indication of each Reactor Building Cooler Fan motor starter status (high and low speed
2 lights), each Fan motor starter status on the computer, indication of each Fan motor amperage, indication
2 of inlet and outlet cooling water flow to each cooler, and inlet and outlet air temperature indication for
2 each cooler. All of the above indications are provided in the Control Room. The installed
2 instrumentation is adequate for the intended monitoring functions. For backup indications, the level of
2 environmental qualification provided for the instrumentation is consistent with the performance
2 expectations of the instrumentation and meets the recommendations of Category 3 in Duke's
2 interpretation of RG 1.97, Rev. 2.

### 2 7.5.2.42 Reactor Building Air Temperature

2 Thirteen dual element thermocouples are provided to measure Reactor Building air temperature. One
2 element of each T/C provides an input to the plant computer and the second element of each T/C
2 provides an input to the multipoint recorder. The plant computer displays a range of 0 to 390°F, the
2 recorder displays a range of 0 to 300°F. The plant computer and the recorder are powered by highly
2 reliable battery backed busses.

2 The displayed ranges are adequate for the intended monitoring function. The worst case DBA
2 temperature in the Reactor Building is 286°F. For accidents in which harsh RB environments are a result,
2 pressure and temperature are coupled such that as RB pressure is reduced the temperature is also reduced.
2 Therefore, RB pressure is considered the priority variable with temperature as a Category 3 backup
2 variable. The level of environmental qualification provided for this instrumentation is consistent with its
2 performance expectations and meets the recommendations of Category 3 in Duke's interpretation of RG
2 1.97, Rev. 2.

### 2 7.5.2.43 Makeup Flow

2 The existing instrumentation for this variable provides continuous monitoring of reactor coolant makeup
2 flow. The loop range is 0 to 160 gallons per minute which encompasses the Regulatory Guide 1.97,
2 Rev.2 criteria of 0-110% of design flow. Design flow is 35 GPM. The instrument is pneumatic and is
2 supplied by the normal Station Air System. The instrumentation is located in a mild temperature
2 environment.

2 The transmitter for this variable is not rated to withstand the anticipated maximum design basis accident
2 radiation dose for the installed location. The installed instrumentation is adequate for the intended
2 monitoring function. For accidents in which harsh environments are a result, the portion of the system
2 containing this instrumentation is not required for the mitigation of these accidents and is automatically
2 bypassed upon an ESF Actuation. Therefore, Makeup Flow is not a key variable for accident monitoring
2 and is considered to be Category 3, instrumentation. The level of environmental qualification provided
2 for the instrumentation in this system is consistent with the performance expectations of the system and
2 meets the recommendations of Category 3 in Duke's interpretation of RG 1.97, Rev. 2 as clarified in
2 Section 5.5.

### 2 7.5.2.44 Letdown Flow

2 The existing instrumentation for this variable provides continuous monitoring of reactor coolant letdown
2 flow. The loop range is 0 to 160 gallons per minute which envelopes the Regulatory Guide 1.97, Rev. 2
2 criteria of 0-110% of design flow. Design flow is 70 GPM. This instrument loop is powered from a
2 highly reliable battery backed bus. The instrumentation is located in a mild temperature environment.

2 The transmitter for this variable is not rated to withstand the anticipated maximum design basis accident
2 radiation does for the installed location.

2 function. No useful information would be gained by measuring tank volume from 0-15%. Normal level
2 (pre-accident) is maintained above 15% and post-accident condition will only increase tank level.
2 Therefore, the existing range is adequate for the intended monitoring function.

### 2 7.5.2.36 Quench Tank Temperature

2 The indicated of Quench Tank temperature is from 50° to 350°F. The design temperature of the Quench
2 Tank is 300°F which is greater than the maximum temperature reached in the tank during a design
2 transient. The tank design pressure is 55 psig, which is greater than the calculated pressure of
2 approximately 50 psig (rupture disc pressure) attained after the most severe transient. The saturation
2 temperature for 50 psig is 297°F. Thus, the indicated range of 50°-350°F will adequately measure the
2 expected maximum temperature as well as saturation temperature for the Quench Tank.

2 (RE: NSMs ON-1/2/32593)

### 2 7.5.2.37 Quench Tank Pressure

2 The indicated range of the Quench Tank pressure is from 0 to 60 psig. The tank rupture disc pressure is
2 50 psig and the tank design pressure is 55 psig. Therefore, the installed instrumentation is adequate for
2 the intended monitoring function.

### 2 7.5.2.38 Main Steam Safety Valve Position

2 This variable is not monitored directly. The position of the Main Steam Safety Valves (MSSV) are not
2 required to mitigate the consequences of a design basis accident. Direct indication of safety valve position
2 is not provided but indirect indication is provided via control room indication of steam generator pressure.
2 During Duke's Control Room Design Review, a specific Task Analysis Evaluation of MSSV indication
2 was undertaken. This evaluation dealt with steam leak transients with and without MSSV indication. As
2 a result of this evaluation, direct MSSV indication was found not necessary. Also, sound emitted from
2 the valves provides an audible indication to the operators when the valves lift. Duke feels that this is
2 adequate indication for the intended monitoring function.

### 2 7.5.2.39 Main Feedwater Flow

2 Oconee has four main feedwater flow channels, two channels per steam generator feedline. The indicated
2 range for this variable is 0 to $6.0 \times 10^6$ lbs/HR which corresponds to 0 to 113% of design flow.

### 2 7.5.2.40 Emergency Feedwater Flow

2 Oconee has four QA Condition 1 flow transmitters, two per steam generator monitoring Emergency
2 Feedwater Flow from all EFDW pumps to each steam generator. The indicated range for this variable is
2 0 to 1200 GPM which corresponds to a range of 0 to 115% design flow. This instrumentation is powered
2 from a safety grade emergency power source. The flow transmitters are located in a mild environment.
2 Seismic qualification methodology for these transmitters is as described in the Oconee FSAR, Section
2 3.10, "Seismic Qualification of Instrumentation and Electrical Equipment" on page 3-177. The indicators
2 are located in the control room which is classified as a mild environment. Emergency Feedwater flow is
2 recorded on a dedicated chart recorder in the Control Room for EFW flow to each steam generator.

### 2 7.5.2.41 Reactor Building Fan Heat Removal

2 The key variable for monitoring Reactor Building Cooler performance is Reactor Building Pressure
2 instrumentation which is Category 1 (See variable sheet B-13). Backup instrumentation includes Nuclear

2 **7.5.2.30 Boric Acid Charging Flow**

2 Oconee NSSS does not include a charging system as part of the Emergency Core Cooling System
2 (ECCS). Flow paths from the ECCS and the RCS include high pressure injection (HPI) and low
2 pressure injection (LPI) with the BWST or the RB Sump as the suction source, and the Core Flood
2 Tank injection. HPI and LPI flow rates are monitored, and BWST, Reactor Building Sump, and Core
2 Flood Tank levels are monitored by RG 1.97 variables. Therefore, Boric Acid Charging Flow monitoring
2 is not applicable to the operation of the ECCS and is not a Type D variable for Oconee.

2 **7.5.2.31 Reactor Coolant Pump Status**

2 The indicated range for RCP motor current is from 0 to 1200 amps. The instrumentation derives power
2 from the monitored source and is adequate for the intended monitoring function.

2 **7.5.2.32 Power Operated Relief Valves Status**

2 An acoustical leak detection monitoring system is the primary instrumentation for determining PORV
2 position. It is a single channel system powered from a highly reliable battery backed bus. It provides the
2 operator with positive indication of valve position by indicating fractional flow through the valve in ten
2 steps from 0.01 to 1.0. Backup indication of PORV position is provided by limit switch operated
2 indicating lights and PORV outlet temperature indication. The system was specified and is rated to
2 operate in all environmental conditions for its location.

2 (RE: NSMs ON-1/2/32594)

2 **7.5.2.33 Primary System Safety Relief Valve Positions (Code Valves)**

2 Acoustical leak detection monitoring systems are the primary instrumentation for determining code valves
2 position. Each code valve has a single channel system powered from highly reliable battery backed bus.
2 It provides the operator with positive indication of valve position by indicating fractional flow through the
2 valve in ten steps from 0.01 to 1.0. Backup indication of code valve position is provided by valve outlet
2 temperature indication. The system was specified, and is rated to operate in all environmental conditions
2 for its location.

2 (RE: NSMs ON-1/2/32594)

2 **7.5.2.34 Pressurizer Heater Status**

2 Control indicating lights are used for indication of the ON/OFF status of the pressurizer heater groups.
2 Indicating lights are powered by highly reliable battery backed busses. This monitoring instrumentation is
2 located in a mild environment.

2 ON/OFF status of the pressurizer heaters provides the operator adequate information for Design Basis
2 events. Additionally, RCS pressure can be monitored to determine the effectiveness of the heaters to
2 maintain system pressure. Duke feels that this is adequate for the intended monitoring function, and that
2 monitoring of electric current per Regulatory Guide 1.97, Revision 2, recommendations is not necessary.

2 **7.5.2.35 Quench Tank Level**

2 The indicated range of Quench Tank Level is from 0 to 125″ corresponding to tank volume of
2 approximately 15-96%. This range is not in complete compliance with RG 1.97, Rev. 2, which
2 recommended top to bottom tank monitoring, however, the upper range meets the intended monitoring

5

2 ### 7.5.2.27 Core Flood Tank Level

2 Oconee has two channels of tank level instrumentation on each of the two core flood tanks. Power for
2 these channels is provided by highly reliable battery backed buses. The indicated range for Unit 2 is 1.5
2 to 14 feet which corresponds to approximately 15% to 83% of the core flood tank volume. The indicated
2 range for Units 1 and 3 is 1.5 to 14 feet which corresponds to approximately 22% to 83% of the core
2 flood tank volume. The equipment is located in a harsh environment.

2 The range and environmental qualification of this instrumentation is not in total compliance with the
2 recommendations of RG 1.97, Rev. 2, which recommends a range of 10% to 90% volume and Category
2 2 classification.

2 The primary function of this instrumentation is to monitor the pre-accident status of the core flood tanks
2 to assure that this passive safety system is prepared to serve its safety function. The indicated range
2 envelopes the Technical Specification level requirements and Duke Power considers the range adequate to
2 meet the intended monitoring function. This instrumentation plays no significant role in the subsequent
2 management of an accident. Therefore, Core Flood Tank Level is not a key variable for accident
2 monitoring and is considered to be Category 3 instrumentation. The level of environmental qualification
2 provided for the instrumentation in this system is consistent with the performance expectations of the
2 system and meets the recommendations of Category 3 in Duke's interpretation of RG 1.97, Rev. 2.

2 ### 7.5.2.28 Core Flood Tank Pressure

2 Oconee has two channels of core flood tank pressure instrumentation on each of the two core flood tanks.
2 Power for these channels is provided by highly reliable battery backed buses. The indicated range is 0 to
2 700 psig. The tanks are pressurized to 600 psig under normal operating conditions.

2 The primary function of this instrumentation is to monitor the pre-accident status of core flood tanks to
2 assure that this passive safety system is prepared to serve its safety function. This instrumentation plays
2 no significant role in the subsequent management of an accident. Therefore, Core Flood Tank Pressure is
2 not a key variable for accident monitoring and is considered to be Category 3 instrumentation. The
2 installed system meets the Duke interpretation of Category 3 recommendations. Regulatory Guide 1.97,
2 Revision 2, classifies this variable as Category 2.

2 The range of this instrumentation is not in total compliance with the recommended 0 to 750 psig range of
2 Regulatory Guide 1.97, Revision 2. However, the indicated range covers approximately 0 to 117% of the
2 operating pressure of the tanks. Because the purpose of this variable is to monitor and maintain Core
2 Flood Tank pressure during normal operating to Technical Specification (TS) limits, the range of this
2 variable should provide some margin above that TS limit. Since the Oconee TS limit is 600 $\pm$ 25 psig, a
2 high range value of about 700 psig will provide greater than 10% excess range measurement capability and
2 will therefore be sufficient. Duke Power considers the instrumentation adequate for the intended
2 monitoring function.

2 ### 7.5.2.29 Core Flood Tank Isolation Valve Position

2 The core flood tank isolation valves are provided with control switches on the main control board.
2 During normal plant operation, power is removed from the valve operations to prevent a spurious signal
2 from inadvertently closing the valves. The indicating lights are powered from a separate highly reliable
2 battery backed bus and give actual valve position of both Closed-Not Closed and Open-Not Open.
2 Environmentally qualified limit switches are provided for the core flood tank isolation valves.

2 The use of diluted samples is in part for maintaining personnel exposures ALARA and is within the
2 guidelines provided in NUREG 0737 and its clarifications. Although Regulatory Guide 1.97, Revision 2
2 recommends a range of 10 uCi/gm to 10 Ci/gm or TID 14844 source term in coolant volume grab
2 samples. Duke considers the use of diluted samples in compliance with Regulatory Guide. The Criterion
2 5 of NUREG 0737 allows 96 hours to perform a chloride analysis which will be met by Duke Power.
2 The 24 hour time limit applies only to BWR's on sea or brackish water sites, and plants which use sea or
2 brackish water in essential heat exchangers. The existing chloride measuring capabilities are considered
2 adequate by Duke. Further discussion of these subjects is contained in the PASS response referenced
2 above.

2 (Re: FSAR 9.3.6.1, "Post-Accident Liquid Sampling System" on page 9-48, FSAR 9.3.6.2,
2 "Post-Accident Containment Air Sampling System" on page 9-49).

### 2 7.5.2.23 Reactor Building Area Radiation - High Range

2 Oconee has two redundant QA Condition 1 channels of Reactor Building high range radiation monitoring
2 instrumentation. Each channel is powered by safety grade emergency power. The indicated range is 1 to
2 $10^7$ R/hr. Diversity is provided by portable instrumentation or by sampling and analysis. The
2 instrumentation is seismically and environmentally qualified.

### 2 7.5.2.24 Airborne Process Radiation Monitors

2 Airborne process radiation monitors exist for monitoring ventilation exhausts and the condenser air
2 ejector exhaust (see Oconee FSAR, Section 11.5, "Process and Effluent Radiological Monitoring and
2 Sampling Systems" on page 11-17 and Table 11-7). However, in accordance with RG 1.97, Rev. 2 these
2 individual airborne process radiation monitors are not required for accident monitoring due to the fact
2 that ventilation systems and the condenser air ejector exhaust to the common unit vent (See Oconee
2 FSAR, Section 7.5.2.52, "Unit Vent Radioactive Discharge Monitors" on page 7-64).

### 2 7.5.2.25 Area Radiation

2 Oconee has an extensive Area Radiation Monitoring System installed for personnel protection. Channel
2 detector locations were selected based on areas normally having free access and low radiation dose rates
2 with the potential of having abnormal radiation levels. These channels have an indicated range of $10^{-1}$ to
2 $10^7$ mr/hr. Redundant indication can be provided by portable instrumentation. The channels are
2 powered by a highly reliable non load shed power bus capable of receiving power from the on-site
2 emergency power sources. See the Oconee FSAR, Section 12.3.3, "Area Radiation Monitoring System"
2 on page 12-11.

2 The environmental qualification of some of the instrumentation is not in compliance with the
2 recommendations of Regulatory Guide 1.97, Revision 2. However, the qualification is within the guidance
2 provided for Category 3 instrumentation which Duke considers adequate for the intended monitoring
2 function. Also note, this is in compliance with the recommendations of RG 1.97, Rev. 3. Continuous
2 recording is not required for the intended monitoring function.

### 5 7.5.2.26 Decay Heat Cooler Discharge Temperature

2 Each train of the Oconee LPI system contains instrumentation to monitor decay heat cooler discharge
2 temperature which is referred to in Regulatory Guide 1.97, Revision 2, as RHR Heat Exchanger Outlet
2 Temperature. The range for this instrumentation is 0 to 400°F, and the power supply is a highly reliable
2 battery backed control bus. Each train is environmentally qualified per the IEB-79-01B submittal
2 methodology and envelopes the Regulatory Guide 1.97, Rev. 2 range of 32° to 350°F.

2 ### 7.5.2.19 Reactor Building Pressure

2 Two redundant QA Condition 1 channels of instrumentation are provided for monitoring Reactor
2 Building Pressure. The instrumentation channels are environmentally and seismically qualified and
2 powered by safety grade emergency power buses. The indicated range is -5 to 175 psig with the reactor
2 building design pressure being 59 psig. This instrumentation range covers nearly 99% of the
2 recommended Regulatory Guide 1.97, Revision 2, range of 10 psig to 3 times the design pressure (177
2 psig). Duke considers the indicated range adequate for the intended accident monitoring function.

2 ### 7.5.2.20 Reactor Building Isolation Valve Position

2 All electrically controlled reactor building isolation valves are provided with control switches on the main
2 control boards. Actual valve position is provided by QA Condition 1 limit switches on the valves which
2 operate both Closed-Not Closed, and Open-Not Open control switch indicating lights. These valves are
2 powered by safety grade emergency power buses. Additional indication is provided by the computer.
2 Redundancy is not necessary on a per valve basis since redundant barriers are provided for all fluid
2 penetrations as discussed in the Oconee FSAR Section 6.2.3.2, "System Design" on page 6-28.
2 Environmental qualification of the limit switches is described in the Oconee FSAR section 3.10, "Seismic
2 Qualification of Instrumentation and Electrical Equipment" on page 3-177 and the Oconee Nuclear
2 Station Seismic Design Criteria (OSDC-0193.01-00-00001).

2 ### 7.5.2.21 Radiation Level in Primary Coolant

2 Oconee has one channel of primary coolant radiation level instrumentation which monitors the Reactor
2 Coolant and Letdown Line and is isolated upon ESF actuation signal. The channel is powered from a
2 highly reliable battery backed bus. The indicated range is $10^1$ to $10^6$ counts per minute which covers
2 reactor coolant concentration of approximately $10^{-3}$ uCi/ml to $10^3$ uCi/ml (see the Oconee FSAR,
2 Section 11.5, "Process and Effluent Radiological Monitoring and Sampling Systems" on page 11-17).
2 Although the Regulatory Guide 1.97, Revision 2, recommended range of 1/2 the Technical Specification
2 limit to 100 times the Technical Specification limit is not met, the indicated range is considered adequate
2 for the intended monitoring function.

2 This monitor was not installed to quantify accident conditions nor as a Category 1 instrument. It is
2 isolated following an accident. The level of environmental qualification provided for this instrumentation
2 is consistent with its performance expectations and meets the recommendations of Category 3 in Duke's
2 interpretation of RG 1.97, Rev. 2. Information for this variable is obtained by sampling and analysis
2 which is considered adequate for the intended monitoring function.

2 Section II.B.3 of NUREG-0737 required that the capability exist at each nuclear plant to sample the RCS
2 to access the magnitude of fuel failures during post-accident conditions. As such, this method should be
2 the primary means of determining clad breach. (Re: FSAR 9.3.6.1, "Post-Accident Liquid Sampling
2 System" on page 9-48)

2 ### 7.5.2.22 Primary Coolant and Reactor Building Pressure

2 The existing design of the sampling system for the primary coolant, the Reactor Building sump and
2 Reactor Building air allows samples to be taken for laboratory analysis. Samples from other plant
2 systems including various auxiliary building sumps can be obtained from sample points on system piping
2 and/or storage tanks. Capabilities for making the recommended measurements (some use diluted samples)
2 are provided. Detailed information concerning the Post-Accident Sampling Systems and the laboratory
2 capabilities available at Oconee is described in the NUREG 0737, II.B.3 Post-Accident Sampling System
2 (PASS) response.

2 ### 7.5.2.15 Reactor Coolant System Cold Leg Water Temperature

2 Oconee has indication of Reactor Coolant System (RCS) Cold Leg Temperature for each of the four cold
2 legs. The instrumentation is powered from a highly reliable battery backed source. The indicated range is
2 50° to 650°F. Additional diversity is provided by the Hot Leg Water Temperature and Core Exit
2 Temperature Instruments.

2 The RCS Cold Leg Water Temperature is used as a backup for the key variable of Hot Leg Temperature
2 and Core Exit Temperature. Because the Hot Leg and Cold Leg RTD's are located in the RCS loops
2 and not in the reactor vessel, either forced or natural circulation is required through the steam generators
2 for their indication to be representative of actual core conditions. When circulation is present, the 650°F
2 high end of the range provides 18% excess measurement capability based on a steam generator design
2 pressure of 1050 psig and a saturation temperature of approximately 553°F for the Oconee design.
2 Because the RCS Cold Leg Temperature is not used in the ATOG guidelines and functions as backup to
2 the other two variables, it is appropriate to classify this variable as a Category 3. The existing design is
2 adequate for the intended monitoring function.

2 ### 7.5.2.16 Reactor Coolant System (RCS) Hot Leg Water Temperature

2 Two qualified, QA condition 1 channels, are provided for post-accident monitoring Wide Range RCS
2 Hotleg Water Temperature in response to Regulatory Guide 1.97 Rev. 2. These instrument channels are
2 powered from safety grade emergency power sources. The indication readouts are located in the Control
2 Room in a mild environment. This variable inputs to the plant computer through isolation buffers and is
2 recorded on a dedicated chart recorder in the Control Room. (RE: NSMs ON-1/2/32401). The range of
2 the readouts is 50 to 700°F which Duke considers adequate for the intended monitoring function. Also
2 note, this range is in compliance with the recommendations of Revision 3 to RG 1.97. Control room
2 display is through the inadequate Core Cooling Monitoring system.

2 ### 7.5.2.17 Reactor Building Sump Water Level Narrow Range

2 Two channels of instrumentation monitor both the Normal Sump Level (0 to 2 feet, approximately 350
2 gallons) and the Emergency Sump Level (0 to 3 feet, approximately 4000 gallons). This instrumentation
2 is environmentally qualified and powered from safety grade emergency power buses. Qualified backup
2 indication is provided by the Wide Range Sump Level instrumentation.

2 (Re: FSAR 3.4.1.1.2, "Flood Protection Measures Inside Containment" on page 3-48).

2 (RE: NSMs ON-1/2/32248)

2 ### 7.5.2.18 Reactor Building Sump Water Level

2 Two redundant QA Condition 1 channels of level instrumentation are provided for measuring reactor
2 building sump water level from the bottom of the Reactor Building to approximately five feet above the
2 maximum flood elevation which exceeds the 600,000 gallon level. The indicated range is 0 to 15 feet.
2 Redundancy/diversity is provided by the Borated Water Storage Tank Level and the Narrow Range Sump
2 Level indicators. The instrumentation channels are environmentally and seismically qualified and powered
2 by safety grade emergency power buses.

2 (Re: FSAR 3.4.1.1.2, "Flood Protection Measures Inside Containment" on page 3-48).

### 7.5.2.12 Neutron Flux

Oconee has four channels of neutron flux for the source range, and four wide range QA Condition 1 channels of full range neutron flux instrumentation which are environmentally qualified for post-accident monitoring. Five neutron flux channels exist for the power range. The indicated ranges are: Source Range $10^{-1}$ to $10^{5}$ cps, -1.0 to +7.0 decade/min. rate of change; Wide range (Post-Accident Monitoring channels) $10^{-8}$ to 200% power, -1 to +7 decade/min. rate of change; and Power Range, 0 to 125%.

NI-1,-2,-3, and -4 channels are environmentally qualified and powered from safety grade busses and encompass the $10^{-6}$ to 100% Full Power range in response to Regulatory Guide 1.97, Rev. 2. All other NI channels are designed for the normal Reactor Building Environment for the safety function of overpower reactor trip but they are not environmentally qualified for post-accident operation.

Operator information is provided as follows:

- Seventeen Control Room indicators (Four source, four wide, five power)
- Twenty-one computer points (Eight source, eight wide range, and five power)
- Trend recording on demand
- One QA Condition 1 Wide Range channel recorded on a post-accident operation recorder. One source range, wide range, and power range channel recorded, four (two power range) channels accessible on a Non-QA Condition recorder.

RE: NSMs ON-1/2/32596 and 1/2/32909)

### 7.5.2.13 Control Rod Position

Each control rod's position is indicated on an analog display which has two switchable input modes for the full 0 to 139 inch range. In addition, separate Full In and Full Out indicating lights are provided for each control rod. Analog computer points are provided for each control rod's position. Analog computer points are also provided for control rod groups 5, 6, 7 and 8 rod position for the full 0 to 139 inch range. This instrumentation is powered from a highly reliable battery backed source. (Re: FSAR 4.5.3, "Control Rod Drives" on page 4-59).

Operator information is provided as follows:

- Indicating lights for Full In or Not Full In for all control rods.
- Analog display full range for all control rods.
- Computer inputs for all control rods and all control rod groups 5, 6, 7, and 8. Trend recording on demand.

### 7.5.2.14 RCS Soluble Boron Concentration

This variable is monitored by sampling and laboratory analysis. Primary system boron concentration is controlled manually with the sampling frequency determined by plant conditions and operating procedures. In addition post-accident sampling of the RCS is available (Re: FSAR 9.3.6.1, "Post-Accident Liquid Sampling System" on page 9-48). Neutron flux indication also provides indication of reactor subcriticality (Re: FSAR 7.5.2.12, "Neutron Flux"). Duke considers these measures adequate for the intended monitoring functions.

2 The ICCM cabinets also provide non-safety inputs to the Operator Aid Computer (OAC), annunciator,
2 and a non-safety indicator located in the Control Room. Safety train intergrity is maintained through the
2 use of isolation buffers provided by the ICCM system. Also provided is two non-safety instrument
2 channels which provide non-safety inputs to the OAC.

2 RBS Flow is a Type A variable at Oconee because the operator relies on this information following a
3 design basis event (LOCA, SB LOCA) to throttle RBS flow.

2 (RE: NSMs ON-1/2/32588)

### 2 7.5.2.10 Reactor Building Hydrogen Concentration

2 Two redundant channels of nuclear safety related instrumentation monitor reactor building hydrogen
2 concentration. The indicated range is from 0 to 10% concentration which envelopes the Regulatory
2 Guide 1.97 range requirements.

2 Both channels are powered by safety grade emergency buses. Control of the sample line switching valves
2 and sample selector solenoid valves is accomplished at the analyzer remote control panel. These
2 instruments are seismically and environmentally qualified. (RE: FSAR 9.3.7, "Containment Hydrogen
2 Monitoring System" on page 9-50)

### 2 7.5.2.11 Upper Surge Tank and Hotwell Level

2 Oconee's Emergency Feedwater System (EFDW) draws condensate grade suction from the Upper Surge
2 Tanks and the Condenser Hotwell. Condensate may also be provided from the Condensate Storage Tank
2 (CST) and the Makeup Demineralizers. Additional backup of the two normal condensate sources is
2 provided by these same locations associated with the other two units. The level transmitters which
2 monitor Upper Surge Tank and Hotwell level are located in the Turbine building which is a mild
2 environment.

2 Category 3 instrumentation is available to monitor Hotwell level in the Control Room. One continuous
2 recorder and computer monitoring point is provided to monitor this variable.

2 Two QA Condition 1 instrumentation channels are provided for monitoring Upper Surge Tank (UST)
2 level in response to Regulatory Guide 1.97. These instrument channels are environmentally and
2 seismically qualified and powered from a safety grade source. Each instrument channel, train A and B
2 respectively, input to the Inadequate Core Cooling Monitoring (ICCM) system cabinets. The ICCM
2 Train A cabinet provides safety inputs to a dedicated qualified recorder and to a qualified indicator located
2 in the Control Room which provides UST level indication. The ICCM Train B cabinet also provides a
2 safety input to a qualified indicator located in the Control Room. The range of UST level indication is 0
2 - 12 feet.

2 The ICCM cabinets, Train A and B respectively, also provide non-safety inputs to two computer alarm
2 points and one annunciator window. Safety train integrity is maintained through the use of isolation
2 buffers provided by the ICCM system.

2 Upper Surge Tank level is a Type A variable at Oconee because the operator relies on this information
3 following a design basis event.

2 (RE: NSMs ON-1/2/32449)

2 ### 7.5.2.7 High Pressure Injection System and Crossover Flows

2 Two channels of QA condition 1 instrumentation are provided for post accident monitoring of High
2 Pressure Injection (HPI) flow in response to Regulatory Guide 1.97. Each channel is seismically and
2 environmentally qualified and powered from a safety grade source. Each channel signal, A and B
2 respectively, inputs to a dedicated qualified recorder and qualified indicator via the Inadequate Core
2 Cooling Monitoring (ICCM) system cabinets. Two channels of QA condition 1 instrumentation is also
2 provided for monitoring HPI crossover flow. These instrument channel signals directly input to qualified
2 indicators on the Control Board. HPI System and Crossover Flow instrumentation channels monitor
2 flow over the range 0 - 750 gpm which envelopes the 0 to 110% design flow criteria of Regulatory Guide
2 1.97, Rev. 2.

2 The ICCM cabinets also provide non-safety inputs to the Operator Aid Computer (OAC) and
2 annunciator points. Safety channel integrity is maintained through the use of isolation buffers provided in
2 the ICCM.

5 HPI System flow is a Type A variable at Oconee because the operator relies on this information following
2 a design basis event (LOCA, SB LOCA, MSLP, OTSG Tube Rupture) to throttle HPI and initiate HPI
2 bypass (if necessary).

2 (RE: NSMs ON-1/2/32589)

3 ### 7.5.2.8 Low Pressure Injection System Flow

2 Two QA Condition 1 instrumentation channels are provided for normal and post accident monitoring
2 Low Pressure Injection (LPI) flow in response to Regulatory Guide 1.97. Each channel is seismically and
2 environmentally qualified and powered from a safety grade source. Each channel signal, train A and B
2 respectively, inputs to a qualified indicator and qualified recorder via the Inadequate Core Cooling
2 Monitoring (ICCM) system cabinets. These channels monitor LPI flow over the range 0-6000 gpm
3 which envelopes the 0-110% of design flow criteria for Regulatory Guide 1.97.

2 The ICCM cabinets also provide non-safety inputs to the Operator Aid Computer (OAC) and
2 annunciator points. Alarms generated in the ICCM cabinets provide high and low LPI flow and low
2 Decay Heat removal flow for each train. Safety train integrity is maintained through the use of isolation
2 buffers provided by the ICCM. Two non-qualified transmitters, one per train, also provide non-safety
2 inputs to the OAC.

2 LPI System is a Type A variable at Oconee because the operator relies on this information following a
3 design basis event (LOCA, SB LOCA) to throttle LPI flow.

2 (RE: NSMs ON-1/2/32587)

2 ### 7.5.2.9 Reactor Building Spray Flow

2 Two QA Condition 1 instrumentation channels are provided for post accident monitoring Reactor
2 Building Spray flow in response to Regulatory Guide 1.97. Each instrumentation channel is seismically
2 and environmentally qualified and powered from a safety grade source. Each instrument channel signal,
2 train A and B respectively, inputs to a qualified indicator and qualified recorder via the inadequate core
2 cooling monitoring (ICCM) cabinets. These channels monitor Reactor Building Spray flow over the
2 range 0-2000 gpm which envelopes the Regulatory Guide 1.97 range requirement of 0-110% of design
2 flow.

2 on the Main Control Board in the Control Room. One channel per steam generator also provides a
2 safety input to a qualified recorder located in the Control Room. The ICCM system cabinets, channels A
2 and B respectively, also provide non-safety inputs to the Operator Aid Computer (OAC). Safety train
2 integrity is maintained by isolation buffers provided by the ICCM system cabinets.

2 Each steam generator also has two Non-QA channels of steam generator pressure instrumentation. A
2 duplex gauge is mounted on the Main Control Board for indication of each channel. The indicated range
2 is 0 to 1200 psig corresponding to 14 above the lowest safety valve setting and 8% above the highest
2 safety valve setting. All channels of S/G Pressure are input to the OAC and trend recording is available
2 on demand. Instrumentation is powered by highly reliable battery backed buses.

2 The main steam lines are provided with safety relief valves, atmospheric dump valves and condenser dump
2 valves to prevent over pressurization of the lines as well as pressure control. Operability of the main
2 steam safety valves ensures that the secondary system pressure will be limited to within its design pressure
2 (1050 psig) during the most severe anticipated system operating transient. With an assumed 3%
2 accumulation when these safety valves are operating, the maximum pressure while they are relieving will
2 be less than 10% above design pressure. Also Technical Specifications limit the maximum allowable
2 plant power and thus steam flow in order to maintain that excess relief capacity. Therefore, based on the
2 facts that the, highest safety valve setting is 1104 psig, the steam relief capacity is 17% above the expected
2 steam flow rate and that excess relief capacity is maintained when safety valves are inoperable, the existing
2 range of 0 to 1200 psig is sufficient for this variable.

2 Steam Generator Pressure is classified a Type A variable at Oconee because the operator relies on this
2 information following a design basis event (MSLB, OTSG Tube Rupture) to isolate affected OTSG.

2 (RE: NSMs ON-1/2/32247)

### 2 7.5.2.6 Borated Water Storage Tank Level

2 Three QA Condition 1 channels of level instrumentation are provided for normal and post accident
2 monitoring the Borated Water Storage Tank (BWST) level. Each channel is seismically and
2 environmentally qualified and powered from a safety grade source. Signals to the Control Board are
2 processed through the Inadequate Core Cooling Monitoring (ICCM) system cabinets. The range for the
2 readouts, 0 to 50 ft (13%-100% of volume), is in compliance with Regulatory Guide 1.97, Rev. 2.

2 Two of the three QA Condition 1 instrumentation channels provide inputs to the ICCM system cabinets,
2 Train A and B respectively. The ICCM cabinets provides safety inputs to qualified indicators on the
2 Control Board and non-safety inputs to the Operator Aid Computer (OAC). Safety train integrity is
2 maintained through the use of isolation buffers provided by the ICCM system.

2 The third channel of qualified instrumentation provides a safety input from train B to a dedicated
2 qualified recorder. This channel also provides input to the computer and various annunciators via an
2 optical isolator which maintains safety train B integrity.

2 BWST level is classified a Type A variable at Oconee because the operator relies on this information
2 following a design basis event (LOCA, SB LOCA) to realign LPI to take suction from RB sump.

2 (RE: NSMs ON-1/2/32450)

2 requirements and is adequate for the intended monitoring function, including monitoring to ensure
2 continued safe operation of pressurizer heaters.

2 The qualified instrument channels are powered by safety grade emergency power sources. Continuous
2 recording is provided for one channel. The range for the instrumentation channels is 0 to 400 inches
2 which Duke considers adequate for the intended monitoring function as referenced in the above
2 paragraph.

2 Pressurizer level is classifed a Type A variable at Oconee because the operator relies on this information
2 following a design basis event (SBLOCA, OTSG Tube Rupture, MSLB) to throttle HPI.

2 (RE: NSMs ON-1/2/32248)

### 2 7.5.2.4 Steam Generator Level

2 Oconee has several different methods of Steam Generator level measurement and indication, as follows:

2 1. Start-up Range - Four transmitters (two per S/G) feed one dual gauge with ranges of 0″ to 250″. The
2 four channels are switch selectable for feeding the gauges.

2 2. Operate Range - Four transmitters (two per S/G) are combined with temperature compensation to
2 feed two recorders with ranges of 0-100% (96″-388″). The four channels are switch selectable for
2 feeding the recorders.

2 3. Full Range - Two transmitters (one per S/G) feed one dual gauge with ranges of 0 to 100%
4 (0-650″).

4 4. Extended Startup Range - Four transmitters (two per S/G) feed four gauges with ranges of 0″ to 388″.

2 Items 1 thru 3 are used for normal plant operating conditions and are not required, although they may be
2 used as backup verification, for accident conditions.

2 The instrumentation in item 4 is safety related and is used for post-accident monitoring. This
2 instrumentation is powered by safety grade emergency power sources and the transmitters are seismically
2 and environmentally qualified.

2 During accident conditions, the required range for a B&W once through steam generator is based on that
2 level in the steam generator needed to mitigate the effects of a small break LOCA. That range is based on
2 current assumed or known instrumentation errors and is 0″ to 120″. The installed range of 0″ to 388″ is
2 therefore adequate during accident conditions for measuring S/G level.

2 Steam Generator Level is classified a Type A variable at Oconee because the operator relies on this
2 information following a design basis event (MSLB, OTSG Tube Rupture) to isolate affected OTSG.

2 (RE: NSMs ON-1/2/32447)

### 2 7.5.2.5 Steam Generator Pressure

2 Four QA Condition 1 channels, two channels per steam generator, are provided for post-accident
2 monitoring steam generator outlet steam pressure in response to Regulatory Guide 1.97. Each instrument
2 channel is seismically and environmentally qualified and powered from a safety grade source.

2 Each instrument channel inputs to the Inadequte Core Cooling Monitoring (ICCM) cabinets. The
2 ICCM cabinets, Channel A and B respectively, provide safety inputs to two qualified indicators located

2 (CET's) out of twelve inputs to each train of ICCM. This average value is then used with the RCS
2 pressure measurement to calculate core subcooling margin.

2 The degrees of subcooling is also input to the plant computer through isolation buffers and is recorded on
2 a dedicated chart recorder in the Control Room. The range of the degrees of subcooling readouts is
2 200°F subcooled to 50° superheat which envelopes the Regulatory Guide 1.97 range of 200°F subcooling
2 to 35°F superheat.

2 (RE: NSMs ON-1/2/32401)

2 **7.5.2.2.3 Reactor Vessel Head and Hotleg Levels**

2 The Reactor Vessel Head Level indicating system (RVHLIS) and Hotleg (HL) system are an adaptation
2 of the Westinghouse RVLIS to the Babcock and Wilcox nuclear steam supply system. The HL and
2 RVHLIS monitor the RCS for voids and loss of level conditions only under natural circulation.

2 The HL and RVHLIS uses two sets of two d/p (differential pressure) cells to measure both vessel and hot
2 leg levels under natural circulation conditions. These cells are used to measure the pressure drop from the
2 hot leg decay heat drop line connection to the top of the vessel, and from the hot leg decay heat drop line
2 connection to the top of the candy cane on each hot leg. This differential pressure measuring system uses
2 cells of differing ranges to cover natural circulation conditions.

2 This is a two train system containing Trains A and B which are physically separate and electrically
2 isolated from each other. The trains perform the same function using identical but redundant inputs from
2 differential pressure transmitters, impulse line temperature sensors, reactor coolant temperature sensors
2 and wide range reactor coolant system pressure.

2 Software algorithms automatically perform compensation calculations required for variations in impulse
2 line temperatures. Software also calculates and provides the necessary compensation for reactor coolant
2 density.

2 Whenever the Reactor Coolant Pumps (RCPs) are running, the subcooling margin monitors and RCP
2 monitor current meters are used to detect possible void conditions. Computer inputs are provided for
2 both trains of level measurement. The Train A level measurements are recorded on a continuous recorder
2 on the Main Control Board. The plasma displays for each train provide indication of both HL and
2 RVHLIS in the Control Room.

2 **7.5.2.3 Pressurizer Level**

2 Three channels (2-Train A and 1-Train B) of QA 1 instrumentation are provided for post accident
2 monitoring the Pressurizer Level in response to Regulatory Guide 1.97, Revision 2. The indicated range
2 is 0 to 400 inches which represents 11% to 84% level as a percentage of volume. Duke considers this
2 range adequate for the intended monitoring function.

2 In order to determine the range or level that should be monitored for the pressurizer, it is important to
2 understand how the pressurizer is sized and how the level taps are located. The pressurizer water volume
2 is chosen such that the reactor coolant system can experience a reactor trip from full power without
2 uncovering the level sensors in the lower shell and to maintain system pressure above the High Pressure
2 Injection (HPI) system actuation setpoint. The steam volume is chosen such that the reactor coolant
2 system can experience a turbine trip without uncovering level sensors in the upper shell. Oconee has a 0
2 to 400 in range for pressurizer level based on these criteria. Although the installed range of
2 instrumentation is not in complete compliance with the recommendation of Regulatory Guide 1.97,
2 Revision 2, that pressurizer level be monitored from bottom to top, it is consistent with B&W NSSS

2 • Indicates loss of subcooling margin.

2 • Assists in detecting presence of a gas bubble or void in the reactor vessel head.

2 • Assists in the detection of the approach to inadequate core cooling.

2 The ICCM system consists, on a per train basis of centrally located electronics/microprocessor cabinet,
2 display electronics package, display selector key pad, and the plasma display unit on the main control
2 board.

2 A description of each of the process sub-systems are described as follows.

2 **7.5.2.2.1 Core Exit Temperature**

2 There are a total of 52 Core Exit Thermocouples (CETs) per Oconee Unit. Twenty-four (12 per train)
2 have been upgraded for accident monitoring and to meet seismic and environmental qualification
2 requirements.

2 The plant computer is the primary display for all 52 CETs. The ICCM plasma displays (1 per train)
2 located in the Control Room serve as safety related backup displays for the twenty-four nuclear safety
2 qualified CETs. The range of the readouts is 50°F to 2300°F.

2 The ICCM CET function uses inputs from twelve incore thermocouples per train to calculate and display
2 temperatures of the reactor coolant as is exits the core and to provide indication of thermal conditions
2 across the core at the core exit.

2 Each of the twelve qualified thermocouples per train is displayed on a spatially oriented core map on the
2 plasma display. The distribution of the monitored CETs in both trains assure minimum monitoring of at
2 least four per core quadrant. Trending of CET temperature is available continuously on the plasma
2 display. The average of the five hottest CETs is trendable for the past forty minutes.

2 Inputs to the plant computer for thermocouples used in the ICCM backup display is through qualified
2 isolation devices. Power for the backup display is from safety grade emergency power sources, and power
2 for the non-safety Operator Aid Computer (OAC) portion is from a highly reliable battery backed control
2 bus. The plant computer and ICCM backup display are installed in a mild environment.

2 Core exit temperature is classified a Type A variable at Oconee because the operator relies on this
2 information following a design basis event (LOCA) to secure HPI and throttle LPI, (SBLOCA) to
2 throttle HPI and begin forced HPI cooling if needed, (MSLB, OTSG Tube Rupture) throttle HPI and
2 isolate affected OTSG.

2 (RE: NSMs ON-1/2/32401)

2 **7.5.2.2.2 Degrees of Subcooling Monitoring**

2 The margin to saturation for the hotlegs and the reactor core are calculated from Reactor Coolant System
2 (RCS) pressure and temperature measurements. The hotleg subcooling margin is calculated from wide
2 range RCS pressure measurements and individual hotleg RTD temperature measurements. The hotleg
2 subcooling margins are displayed in the Control Room on the ICCM plasma display unit. Train A
2 displays the RCS Loop A hotleg subcooling margin while the Train B display provides RCS Loop B
2 hotleg subcooling margin. Computer inputs are also provided for both hotlegs.

2 The reactor core subcooling margin is displayed in the Control Room in an identical manner. The core
2 subcooling margin is calculated from the average of the five highest qualified Core Exit Thermocouples

displays. The Duke Control Room Review Team made recommendations as to the type and location of displays, for added instrumentation.

- To the extent practicable, the instrumentation is designed to facilitate the recognition, location, replacement, repair, or adjustment of malfunctioning components or modules.
- To the extent practicable, monitoring instrumentation inputs are from sensors that directly measure the desired variables.
- To the extent practicable, the same instruments which are used for accident monitoring are used for the normal operations of the plant to enable the operators to use, during accident situations, instruments with which they are most familiar. However, where the required range of monitoring instrumentation results in a loss of necessary sensitivity in the normal operating range, separate instruments are used.
- Periodic checking, testing, calibration, and calibration verification are in accordance with the applicable portions of the Oconee FSAR Chapter 7, "Instrumentation and Control" on page 7-1.

## 7.5.2 DESCRIPTION

Display instrumentation provided for Oconee operators is described below.

### 7.5.2.1 Reactor Coolant System Pressure

Three channels of Reactor Coolant System (RCS) Pressure indication are available through the plant operator computer (OAC), which receives the RCS Pressure signals through the Engineered Safety Features Actuation System (ESFAS) cabinets. This instrumentation is powered from a highly reliable battery backed source. Two channels are recorded. These instrumentation channels monitor RCS pressure over the range 0 to 2500 psig.

Two upgraded QA Condition 1 channels of Wide Range RCS Pressure indication are provided for post accident monitoring in response to Regulatory Guide 1.97. These instrumentation loops are seismically and environmentally qualified and are powered from safety grade emergency power sources. Signals to the Control Board readouts are processed through the Inadequate Core Cooling Monitoring (ICCM) system cabinets. The range for the readouts, 0-3000 psig, is in compliance with Regulatory Guide 1.97 specifications.

RCS pressure is a Type A variable at Oconee, since the operator relies on this indication to determine when to switch from high pressure injection to low pressure injection.

### 7.5.2.2 Inadequate Core Cooling Instruments

The Inadequate Core Cooling Monitor (ICCM) is of Westinghouse design. The ICCM system monitors hotleg level, reactor vessel head level, loop subcooling margin, core subcooling margin and core exit temperature and provides advanced warning of the approach to inadequate core cooling. The ICCM is a redundant two train Nuclear Safety-Related system powered by the vital instrumentation and control power system.

The microprocessor-based monitoring trains provide essential information to the control room operator so that conditions inherent to or leading to Inadequate Core Cooling (ICC) can be recognized and addressed.

The functions performed by the ICCM are as follows:

- Assists in detecting a void or loss of level in the hotleg during natural circulation.

2 *7.5.1.4.2.3 All Category 2 Instrumentation*

2 For both Nuclear Safety Related and Non Nuclear Safety Related Category 2 instrumentation:

2 The out-of-service interval should be based on normal Technical Specification requirements for the system
2 it serves where applicable or where specified by -other requirements.

2 The instrumentation signal may be displayed on an individual instrument or it may be processed for
2 display on demand by CRT or by other appropriate means.

2 The method of display may be by dial, digital, CRT, or stripchart recorder indication. Effluent
2 radioactivity monitors and meteorology monitors will be recorded. Where direct and immediate trend or
2 transient information is essential for operation information or action, the recording is continuously
2 available on dedicated recorders. Otherwise, it may be continuously updated, stored in computer
2 memory, and displayed on demand.

2 **7.5.1.4.3 Design and Qualification Criteria - Category 3**

2 These instruments do not play a key role in the management of an accident but they do add depth to the
2 Category 1 and 2 instrumentation to the extent that they remain operable. The instrumentation is of high
2 quality commercial grade and is selected to withstand the normal power plant service environment.

2 The method of display may be by dial, digital, CRT, or stripchart recorder indication. Effluent
2 radioactivity monitors and meteorology monitors will be recorded. Where direct and immediate trend or
2 transient information is essential for operator information or action, the recording is continuously
2 available on dedicated recorders. Otherwise, it may be continuously updated, stored in computer
2 memory, and displayed on demand.

2 **7.5.1.4.4 Additional Criteria for Categories 1 and 2**

2 In addition to the criteria of Duke Position 7.5.1.3, the following criteria apply to Categories 1 and 2:

2 • For Nuclear Safety Related (QA1) signals which are transmitted to non-safety related (non QA1)
2 equipment, isolation devices are utilized.

2 • Dedicated control board displays for the instruments designated as Types A, B, and C, Category 1 or
2 2 and qualified for use throughout all phases of an accident will be specifically identified on the
2 control panels so that the operator can discern that they are available for use under accident
2 conditions.

2 **7.5.1.4.5 Additional Criteria for All Categories**

2 In addition to the above criteria, the following criteria apply to all instruments identified in this document:

2 • Servicing, testing, and calibration programs are specified to maintain the capability of the monitoring
2 instrumentation. For those instruments where the required interval between tests will be less than the
2 normal time interval between generating station shutdowns, the capability for testing during power
2 operation is provided.

2 • Whenever means for removing channels from service are included in the design, the design facilitates
2 administrative control of the access to such removal means.

2 • The monitoring instrumentation design minimizes the development of conditions that would cause
2 meters, annunciators, recorders, alarms, etc., to give anomalous indications which are potentially
2 confusing to the operator. Human factors guidelines are used in determining type and location of

2 6. Continuous indication display is provided. Where two or more instruments are needed to cover a
2 particular range, overlapping of instrument span is provided.

2 7. Recording of instrumentation readout information is provided for at least one of the redundant
2 channels. Recorders which are utilized as the primary display device will be seismically qualified.
2 Where direct and immediate trend or transient information is essential for operator information or
2 action, the recording is continuously available on dedicated recorders. Otherwise, it may be displayed
2 on non-seismically qualified recorders or continuously updated, stored in computer memory, and
2 displayed on demand. Intermittent displays such as data loggers and scanning recorders may be used
2 if no significant transient response information is likely to be lost by such devices. All analog
2 variables which are wired to the plant computer may be displayed on trend recorders upon demand to
2 provide hard-copy trend information.

2 **7.5.1.4.2 Design and Qualification Criteria - Category 2**

2 *7.5.1.4.2.1 Nuclear Safety Related (QA1) Category 2 Instrumentation*

2 For instrumentation loops that are installed as nuclear safety related (QA1), environmental qualification is
2 provided per the methodology described in the Oconee Nuclear Station IEB 79-01B submittal and the
2 Resolution of Safety Evaluation Reports for Environmental Qualification of Safety Related Electrical
2 Equipment. Seismic qualification is in accordance with the Oconee Nuclear Station Licensing basis as
2 specified in the Oconee FSAR and Duke Power Seismic Design Criteria (OSDC-0193.01-00-0001).
2 Quality Assurance of these QA Condition 1 instrumentation systems is described in the Duke Power
2 Company Topical Report "Duke 1A" and Oconee FSAR Chapter 17, "Quality Assurance" on page 17-1.
2 These instruments are powered from the safety grade Emergency Power sources (as described in
2 Chapter 8, "Electric Power" on page 8-1 of the Oconee FSAR) and are backed by batteries where a
2 momentary power interruption is not tolerable.

2 *7.5.1.4.2.2 Non Nuclear Safety Related (Non-QA1) Category 2 Instrumentation*

2 For instrumentation loops of lesser importance which are not nuclear safety related, appropriate
2 qualification is provided. Environmental qualification is provided per the methodology described in the
2 Oconee Nuclear Station IEB 79-01B submittal and the Resolution of Safety Evaluation Reports for
2 Environmental Qualification of Safety Related Electrical Equipment.

2 Category 2 instrumentation which is of primary use during one phase of an accident need not be qualified
2 for all phases of the event. For example, an instrument of primary importance prior to attained the
2 recirculation mode need not be demonstrated to withstand post-recirculation radiation.

2 For non-QA1 Category 2 instrumentation, seismic qualification is not required unless seismic induced
2 failure of the instrumentation would unacceptably degrade a safety system.

2 These instrumentation systems are designed, procured, and installed per Duke Power Company standard
2 practices. Duke Power considers that this is adequate to assure the quality of the subject instrumentation.

2 Isolation devices are provided to interface between Nuclear Safety Related (QA1) and Non Nuclear Safety
2 Related (non QA1) portions of any of the subject instrumentation loops.

2 The instrumentation is energized from a highly reliable power source, not necessarily safety grade
2 Emergency Power, and is backed by batteries where momentary interruption is not tolerable.

- Obtain required information through backup or diagnosis channel where a single channel may be likely to give ambiguous indication.

## 7.5.1.4 Design and Qualification Criteria

Design and qualification criteria used by Duke Power Company for plant instrumentation are provided below. The category designations are provided for reference to the Regulatory Guide 1.97 (Revision 2) document.

### 7.5.1.4.1 Design and Qualification Criteria - Category 1

Accident monitoring instrumentation which comprise this design and qualification category are considered by Duke Power to be Nuclear Safety Related and thus are classified as Quality Assurance Condition 1 (QA1).

1. QA1 instrumentation is environmentally qualified as described in the Oconee Nuclear Station IEB-79-01B Duke Power Company submittal and the Resolution of Safety Evaluation Reports for Environmental Qualification of Safety Related Electrical Equipment. Seismic qualification is in accordance with the Oconee Nuclear Station licensing basis as specified in Oconee FSAR Chapter 3, "Design of Structures, Components, Equipment, and Systems" on page 3-1 and the Duke Power Seismic Design Criteria (OSDC-0193.01-00-0001).

2. No single failure within either the accident monitoring instrumentation, its auxiliary supporting features, or its power sources, concurrent with the failures that are a condition or result of a specific accident, will prevent the operators from being presented the information necessary to determine the safety status of the plant and to bring the plant to and maintain it in a safe condition following that accident. Where failure of one accident-monitoring channel results in information ambiguity (i.e., the redundant display disagree) that could lead operators to defeat or fail to accomplish a required safety function, additional information is provided to allow the operators to deduce the actual conditions in the plant. This is accomplished by providing additional independent channels of information of the, same variable (an identical channel) or by providing an independent channel to monitor a different variable that bear a known relationship to the multiple channels (a diverse channel). The information provided to the operator to eliminate ambiguity between redundant channels is needed only during a failure of one of the instrument loops. Therefore, it is considered acceptable to use installed instrumentation of equal design and qualification category, installed instrumentation of a lessor design and qualification category, temporary or portable instrumentation, or sampling to allow the operators to deduce the actual conditions in the plant. Redundant QA1 channels are electrically independent and physically separated from each other per the separation criteria described in Chapter 7, "Instrumentation and Control" on page 7-1 of the Oconee FSAR.

   At least one channel of QA1 instrumentation is displayed on a direct indicating or recording device. (Note: Within each redundant division of a safety system, redundant monitoring channels are not needed.)

3. The instrumentation is energized from the safety grade Emergency Power sources (as described in Chapter 8, "Electric Power" on page 8-1 of the Oconee FSAR) and is backed by batteries where momentary interruption is not tolerable.

4. The instrumentation channel will be available prior to an accident except as provided in Paragraph 4.11, "Exception" as defined in IEEE Standard 279-1971 or as specified in Technical Specifications.

5. The following documents pertaining to quality assurance are referenced:

   - Duke 1A, Duke Power Company Topical Report, "Quality Assurance Program"
   - Oconee FSAR Chapter 17, "Quality Assurance" on page 17-1

The Critical Safety Functions are:

- Subcriticality

The subcriticality fault tree monitors the reactor core to assure that it is maintained in a subcritical condition following a successful reactor trip.

- Inadequate Core Cooling

The inadequate core cooling fault tree monitors those variables necessary to evaluate the status of fuel clad heat removal.

- Heat Sink

The heat sink fault tree monitors the ability to transfer energy from the reactor coolant to an ultimate heat sink.

- Reactor Coolant System Integrity

The Reactor Coolant System integrity fault tree monitors those variables indicating a challenge to or a breach of the Reactor Coolant System pressure boundary.

- Containment Integrity

The containment integrity fault tree monitors those variables which would indicate a threat to containment integrity or other undesirable conditions within containment.

- Reactor Coolant System (RCS)

The RCS inventory fault tree monitors for indications of off-normal quantities of reactor coolant in the primary system.

### 7.5.1.3 System Operation Monitoring (Type D) and Effluent Release Monitoring (Type E) Instrumentation

#### 7.5.1.3.1 Definitions

Type D: Those variables that provide information to indicate the operation of individual safety systems.

Type E: Those variables to be monitored as required for use in determining the magnitude of the release of radioactive materials and in continually assessing such releases.

The Type D and E variables are selected on the basis of individual plant specific system design requirements.

#### 7.5.1.3.2 Operator Usage

The plant design has included variables and information display channels required to enable the Control Room operating personnel to:

- Ascertain the operating status of each individual safety system to the extent necessary to determine if each system is operating or can be placed in operation to help mitigate the consequences of an accident. (Note: Type D and E are not always safety systems)
- Monitor the effluent discharge paths to ascertain if there have been significant releases (planned or unplanned) of radioactive materials and to continually assess such releases.

# 7.5 DISPLAY INSTRUMENTATION

## 7.5.1 CRITERIA AND REQUIREMENTS

### 7.5.1.1 Type A Variables

Type A variables are defined as those variables which are monitored to provide the primary information required to permit the Control Room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for design basis accidents. Primary information is defined as that which is essential for the direct accomplishment of the specified safety functions; it does not include those variables associated with contingency actions which may also be identified in written procedures.

Emergency Procedures provide the lead guidance for selection of Type A variables. The following variables are those determined to be Type A for Oconee Nuclear Station, as defined above:

- Reactor Coolant System Pressure
- Core Exit (Thermocouples) Temperature
- Pressurizer Level
- Degrees of Subcooling
- Steam Generator Level
- Steam Generator Pressure
- Borated Water Storage Tank Level
- High Pressure Injection Flow
- Low Pressure Injection Flow
- Reactor Building Spray Flow
- Reactor Building Hydrogen Concentration
- Upper Surge Tank Level
- Low Pressure Service Water (LPSW) Flow to Low Pressure Injection (LPI) Coolers.

### 7.5.1.2 Type B and C Variables

Type B and C variable selection is based on the Safety Parameter Display System (SPDS) Critical Safety Functions. The SPDS is provided as an aid to the Control Room operating crew in monitoring the status of the Critical Safety Functions. The Critical Safety Functions monitored are those defined in the SPDS Critical Safety Function Fault Trees. The SPDS provides continuous status updated at regular intervals of the Critical Safety Functions.

Since these Critical Safety Functions constitute the basis of the Oconee SPDS, it is Duke Power's position that they should also be identified as the plant safety functions for accident monitoring (i.e., the basis for Type B & C variable selection).

Using the SPDS Critical Safety Functions as the basis for defining the accident monitoring instrumentation incorporates the concept of monitoring the multiple barriers to the release of radioactive material. The Critical Safety Functions monitored are those which assure the integrity of these barriers. The Fault Tree provides an explicit, systematic mechanism for organizing the plant data required to evaluate a Critical Safety Function. The prioritization of the Critical Safety Functions is consistent with the concept of multiple barriers to radiation release.

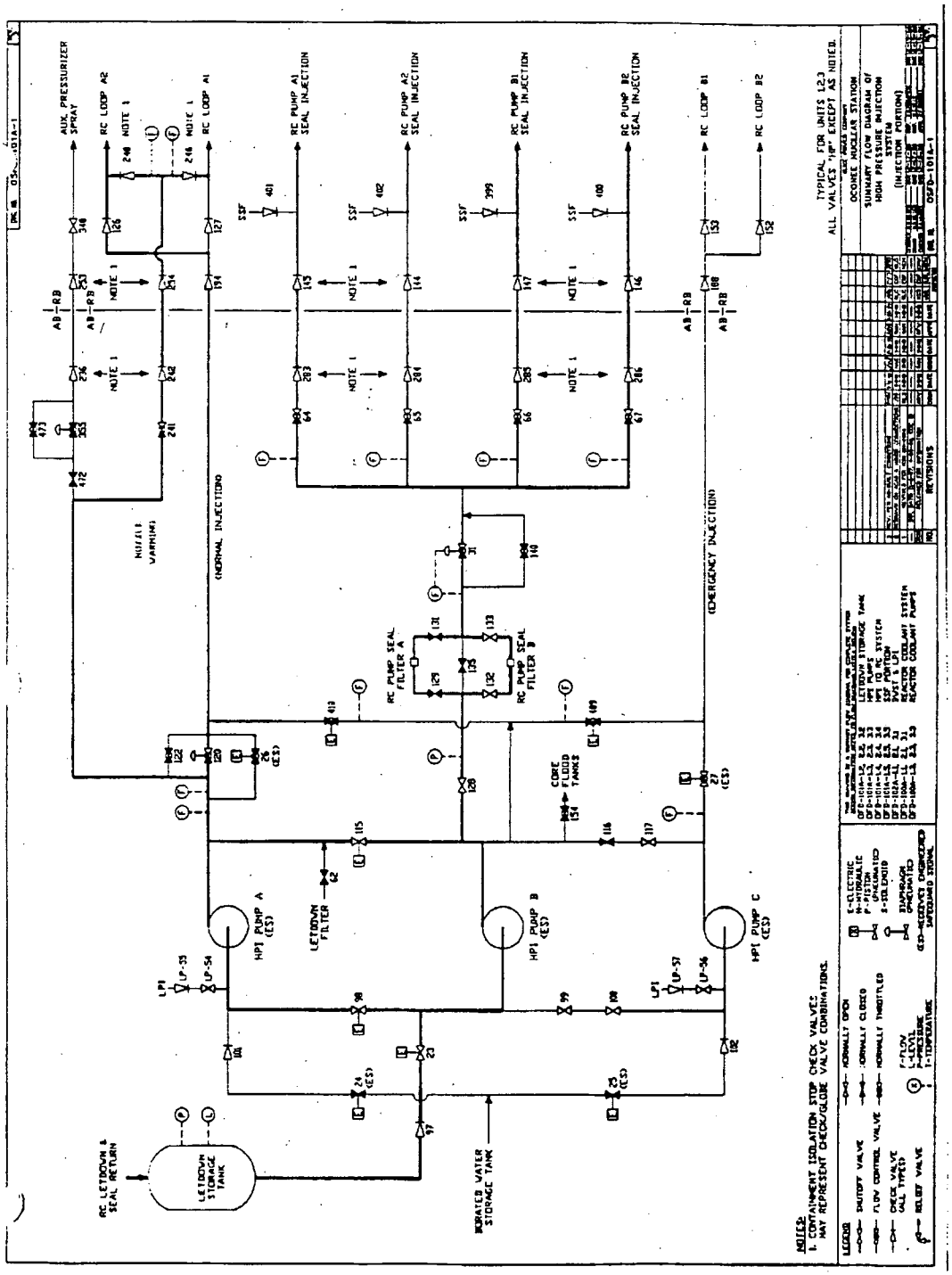


Figure 9-17.
High Pressure Injection System

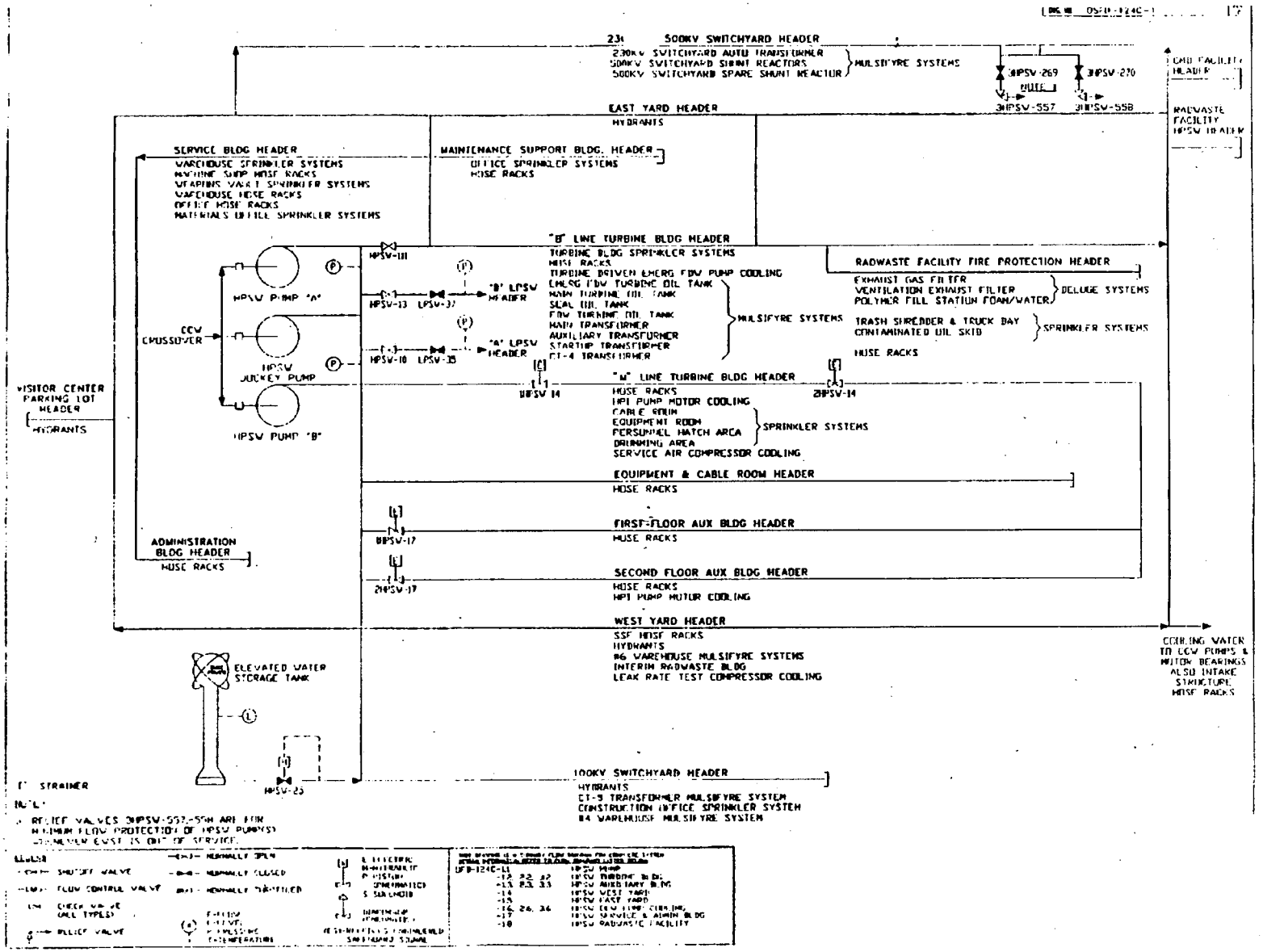


Figure 9-10.
High Pressure Service Water System

Table 9-4 (Page 2 of 2). Cooling Water Systems Component Data (Component Data on a Per Unit Basis)

| Parameter | Value |
|---|---|
| Recirculating cooling water outlet temperature, °F | 89 |
| Condenser circulating water inlet temperature, °F | 80 |
| Design pressure, psig | 150 |
| Design temperature, °F | 150 |
| Plate material | SA-240 |
| Shell material | Carbon steel (SA-285C) |

Table 9-4 (Page 1 of 2). Cooling Water Systems Component Data (Component Data on a Per Unit Basis)

| Parameter | Value |
|---|---|
| Condenser Circulating Water Pumps | 4 per unit |
| Flow (per pump), gal/min | 177,000 |
| Design temperature, °F | 75 |
| Design pressure, psig | 15.7 |
| High Pressure Service Water Pumps | 2 for all units |
| Flow (per pump), gal/min | 6,000 |
| Design temperature, °F | 75 |
| Design pressure, psig | 117 |
| High Pressure Service Water Jockey Pump | 1 for all units |
| Flow (per pump), gal/min | 500 |
| Design temperature, °F | 75 |
| Design pressure, psig | 117 |
| Low Pressure Service Water Pumps | 5 for all units |
| Flow (per pump), gal/min | 15,000 |
| Design temperature, °F | 75 |
| Design pressure, psig | 65 |
| Recirculated Cooling Water Pumps (Units 1 & 2) | 4 shared |
| Flow (per pump), gal/min | 2,400 |
| Design temperature, °F | 105 |
| Design pressure, psig | 100 |
| Recirculated Cooling Water Pumps (Unit 3) | 2 |
| Flow (per pump), gal/min | 2,050 |
| Design temperature, °F | 150 |
| Design pressure, psig | 100 |
| Recirculated Cooling Water Heat Exchangers (Units 1 & 2) | 4 shared |
| Type | Shell and tube |
| Recirculating cooling water flow, each (shellside), gal/min | 1,800 |
| Recirculating cooling water inlet temperature, °F | 105 |
| Recirculating cooling water outlet temperature, °F | 90 |
| Condenser circulating water inlet temperature, °F | 75 |
| Design pressure, shell/tube, psig | 100/50 |
| Design temperature, shell/tube, °F | 200/200 |
| Tube material | Admiralty metal (SB-111) |
| Recirculated Cooling Water Heat Exchangers (Unit 3) | 2 |
| Type | Flat plate |
| Recirculating cooling water flow, each, gal/min | 1,800 |
| Recirculating cooling water inlet temperature, °F | 109 |

5 ## 9.6.6 REFERENCES


1. Safety Evaluation by the Office of Nuclear Reactor Regulation Oconee Nuclear Station Standby Shutdown Facility, Docket Nos. 50-269, 50-270, and 50-287, April 28, 1983
2. Safety Evaluation for Station Blackout (10 CFR 50.63) - Oconee Nuclear Station, Units 1, 2, and 3 (TACS M68574/M68575/M68576), Docket Nos. 50-269, 50-270, 50-287, March 10, 1992
3. Safety Evaluation for Station Blackout (10 CFR 50.63) - Oconee Nuclear Staton, Units 1, 2, and 3 (TACS M68574/M68575/M68576), Docket Nos. 50-269, 50-270, 50-287, December 3, 1992
4. Safety Evaluation Report on Effect of Tornado Missiles on Oconee Emergency Feedwater System (TACS 48225, 48226, and 48227), July 28, 1989
5. Safety Evaluation Report for Implementation of Recommendation for Auxiliary Feedwater Systems, August 25, 1981
6. Evaluation of the Oconee, Units 1,2,&3 Generic Safety Issues (GSI-23 & GSI-105) Resolution, March 24, 1995
7. Letter from WO Parker (Duke) to EG Case (NRC), dated 1/25/78, Response to NRC Questions
8. Letter from WO Parker (Duke) to EG Case (NRC), dated 2/1/78, SSF System Description
9. Letter from WO Parker (Duke) to EG Case (NRC), dated 6/19/78, Response to Staff Questions Concerning Oconee Nuclear Station Safe Shutdown System
10. Letter from WO Parker (Duke) to HR Denton (NRC), dated 3/28/80
11. Letter from WO Parker (Duke) to HR Denton (NRC), dated 2/16/81, Response to NRC Request for Information
12. Letter from WO Parker (Duke) to HR Denton (NRC), dated 3/18/81, Modifications Needed to Meet Appendix R Requirements
13. Letter from WO Parker (Duke) to HR Denton (NRC), dated 3/31/81, Response to NRC Request for Information
14. Letter from WO Parker (Duke) to HR Denton (NRC), dated 4/30/81, Cable Routing and Separation
15. Letter from WO Parker (Duke) to HR Denton (NRC), dated 1/25/82, Response to NRC Concerns for Source Range Instrumentation and Steam Generator Pressure
16. Letter from HB Tucker (Duke) to HR Denton (NRC), dated 9/20/82, Response to NRC Request for Information
17. Letter from HB Tucker (Duke) to HR Denton (NRC), dated 12/23/82, Requested Supplemental Information
18. Letter from HB Tucker (Duke) to HR Denton (NRC), dated 7/15/83, Request for Exemption from 10CFR50 Appendix R, Section III.L.2
19. Letter from JF Stolz (NRC) to HB Tucker (Duke), dated 8/31/83, Exemption from Source Range Flux and Steam Generator Pressure Instrumentation for the SSF


**THIS IS THE LAST PAGE OF THE CHAPTER 9 TEXT PORTION.**

2 The structure meets the requirements of GDC 2, and the guidelines of Regulatory Guide 1.102 with
2 respect to protection against flooding.

## 2 9.6.5 OPERATION AND TESTING

2 The SSF will be placed into operation following total loss of equipment due to the following:

2 1. Flooding
2 2. Fire
2 3. Sabotage

2 If the normal shutdown equipment is inoperable, operators will be sent to the SSF. When directed by the
2 shift supervisor, the operator will start the diesel and establish service water to the diesel generator, start
5 the Auxiliary Service Water Pump and the RCM system as needed and close all of the Reactor Building
2 isolation valves that are controlled from the SSF.

2 Damage control measures, if necessary, will be taken to restore limited operability to the Low Pressure
2 Injection System, Low Pressure Service Water System, and the HP Injection System to bring a RC
2 System to a cold shutdown condition following the initiating event. Pump motors for each of the above
2 systems may be restored to an operable status and the valves will be manually operated to re-establish the
2 above systems to operation.

2 In-service testing of pumps and valves will be done in accordance with the provision of ASME Section XI
2 except for the Submersible Pump which is used to supply makeup water to the Unit 2 embedded
2 condenser circulating piping. This pump should be tested every other year to verify flow capability.

2 The electrical power system components will be tested consistent with current accepted industry practice
2 for safety related equipment.

2 The SSF is designed to achieve and maintain a hot shutdown condition for one or more of the three
2 Oconee units. The SSF is not designed to independently bring the reactor from hot shutdown to cold
2 shutdown. Cold shutdown will be achieved and maintained through the use of normal plant systems and
2 equipment.

2 **SOURCE RANGE FLUX MONITOR**

2 The SSF is designed to achieve and maintain hot shutdown conditions for any or all of the Oconee units.
2 Prior to leaving the Unit 1/2 or Unit 3 control room, all control rods for the unit under consideration are
2 required to be inserted. No non-borated sources tie into the SSF makeup/boration flow path. RCS
2 makeup and boration following transfer of control to the SSF RCM is from the spent fuel pool. Thus,
2 boron dilution events are highly unlikely.

2 Oconee Units 1, 2, and 3 can achieve and maintain controlled cooling to hot shutdown conditions safely
5 from the SSF without the need for remote SG pressure instrumentation or a remote source range
5 monitor. Thus, this instrumentation for the Oconee Nuclear Station is not required. The objectives of
2 Sections III.G.3 and III.L.2 of Appendix R to 10CFR Part 50 are met and the exemption from the
2 requirement to provide remote steam generator pressure and source range monitor instrumentation in the
2 SSF has been granted.

2 **9.6.4.6.3 Instrumentation Guidelines**

2 10CFR 50, Appendix R Section III.L.6 requires that, "Shutdown systems installed to ensure post-fire
2 shutdown capability need not be designed to meet seismic Category I criteria, single failure criteria, or
2 other design basis accident criteria, except where required for other reasons, e.g., because of interface with
2 or impact on existing safety systems, or because of adverse valve actions due to fire damage." Since the
2 monitors for the above listed parameters, in Section 9.6.4.6.2, "Performance Goals" on page 9-91, will
2 not interface with or impact on existing safety systems, the monitors need not be "safety grade".

2 **9.6.4.6.4 Repairs within the 72 Hour Requirement**

2 The use of the dedicated shutdown method for hot shutdown permits the capability of achieving all
2 necessary repairs to achieve cold shutdown within 72 hours after a fire accident. Repairs, including
3 replacement of power cabling, pump motors, valve operators, and switchgear associated with LPI, HPI, or
2 LPSW may be required for cold shutdown. Stored on-site are all components necessary to achieve all
2 repairs. Guidelines are available to implement the required repairs and replacements.

2 **9.6.4.6.5 Fire Protection Conclusion**

2 The ONS design has provided one train of systems necessary to achieve and maintain safe shutdown
2 conditions by utilizing either the main unit's control room or the SSF in conjunction with undamaged
2 systems in the fire-affected unit, and thus will meet the requirements of Appendix R to 10CFR 50,
2 Sections III.G.3 and III.L with respect to safe shutdown in the event of a fire.

2 ### 9.6.4.7 Flooding Review

2 The SSF will not be affected by the following postulated flood events:

2 1. Turbine Building Flood caused by a break in the non-seismic condenser circulating water (CCW)
2 piping system.
2 2. Infiltration of normal groundwater.

4

2 **9.6.4.6.1 Safe Shutdown Systems**

2 Safe shutdown of the reactor is initially performed by the insertion of control rods from the control room.
2 Insertion can also be accomplished by removing power to the control rod drive mechanisms. When
2 normal and emergency systems are not available, reactor coolant inventory and reactor shutdown margin
3 are maintained, from the SSF Control Panel, by the SSF RC makeup pump taking suction from the
2 spent fuel pool. Primary system pressure can be maintained by the pressurizer heaters or by use of
2 charging combined with letdown. Should the pressurizer heaters be unavailable (caused by fire inside
2 containment), progression towards cold shutdown may be initiated as soon as hot shutdown is achieved.
2 Decay heat removal may be accomplished by releasing steam from the steam generators via the
2 atmospheric main steam code relief valves. Makeup to the steam generators can be provided by the
2 Emergency Feedwater System. For fires affecting all banks of pressurizer heaters inside containment,
2 shutdown can be achieved from the unit's main control room.

2 Depressurization to cold shutdown can be achieved by bypassing steam to the turbine, use of the manual
2 atmospheric dump valves, or pressurizer spray. The low pressure injection (LPI) pumps will be used to
2 remove decay heat. Any damage to either the HPI or LPI power cabling or pump motors can be repaired
2 or replaced within 72 hours.

2 Also required for cold shutdown are the low pressure service water (LPSW) pumps. Only the one pump
2 for Unit 1 and 2 and one pump for Unit 3 is required for emergency plant operations. Five LPSW
2 pumps of equal capacity are provided - three for Units 1 and 2, and two for Unit 3. These pumps are
2 separated such that a single fire on any unit should not affect all pumps. The piping is separated such
2 that a single fire cannot affect all pumps. The piping for these pumps are interconnected so that they may
2 feed any of the three units. Any damage to the pump motors or associated power cabling can be repaired,
2 or if necessary, replaced within 72 hours.

2 **9.6.4.6.2 Performance Goals**

2 The performance goals for post-fire safe shutdown can be met using the SSF and undamaged/repaired
3 systems and equipment. Cold shutdown can be achieved within 72 hours of a fire by implementing
3 damage control measures including replacement of cables, pump motors, valve operators and use of
3 emergency switchgear. The control of these functions can be then accomplished using the SSF or the
2 control room, in the fire affected unit, depending on the location of the fire. The transfer of control
2 capability between the control room and the SSF is accomplished via a keyed interlock. Annunciation
2 will occur in the SSF control room upon transfer of control.

5 The process monitoring instruments to be used for a post fire shutdown include reactor coolant hot leg
5 and cold leg temperatures, reactor coolant pressure, pressurizer level and pressure, steam generator level,
5 SSF RC makeup pump flow, and SSF ASW system flow to each unit.

2 **STEAM GENERATOR PRESSURE**

2 Reactor coolant system (RCS) heat removal for hot shutdown can be directly monitored by RCS
2 parameters and controlled by SG level without SG pressure indication, provided that SG pressure is
2 regulated.

2 SG pressure should be regulated by the main steam code safety valves, which will relieve at their setpoints.
2 RCS conditions can be monitored by primary coolant temperature and pressure, pressurizer level and SG
2 level. Should RCS overcooling occur, corrective actions can be taken from the SSF to reinstate proper
5 cooling by controlling the SSF ASW flow rate provided to a unit's SGs in order to restore T-cold.

2 The SSF auxiliary service water buried piping is seismically designed for stresses resulting from SSE and
2 OBE events. The design and analysis were based on the current state-of-the-art for initial effects and the
2 effects of static resistance of the surrounding soil.

### 2 9.6.4.4 Dynamic Testing and Analysis of Mechanical Components

2 Procedures were established for the startup testing of the Class B and C piping in the SSF to verify the
2 following information under different operating modes:

2 • Physical Compliance with Piping Design: An "as built" verification procedure is utilized to verify that
2 piping, components and support-/restraints have been erected with design tolerance.

2 • Vibration Monitoring for Equipment: The purpose of this monitoring program is to verify that
2 vibration levels for system components are within acceptance criteria. Pump vibration is monitored
2 during testing in accordance with IWP-3210 to verify vibrations are less than or equal to the
2 maximum allowable per the specific vendor's requirements.

### 2 9.6.4.5 ASME Code Class 1, 2, and 3 Components, Component Supports and Core
### 2 Support Structures

2 Piping systems for the SSF are designed in accordance with the appropriate ASME Code based on the
2 Quality Group classifications outlined in Regulatory Guide 1.26. The SSF RC Makeup System is under
2 Quality Group B and is designed in accordance with ASME Code Section III Class 2. The remainder of
2 the SSF ASW system is under Quality Group C and is designed in accordance with ASME Code Section
2 III Class 2, Section III Class 3, or ANSI B31.7, 1969. The 1974 Edition of the ASME Boiler and
2 Pressure Vessel Code with addenda through the Summer of 1975 addenda was used. The load
2 combinations and stress limits contained in the requirements of SRP 3.9.3.II and referenced in Regulatory
2 Guide 1.48 are met, except Code Case 1606 is used for the faulted load combination.

2 The loads from pressure relief valves with an open discharge are evaluated in accordance with Code 1569,
2 "Design of Piping for Pressure Relief Valve Station", assuming multiple valves on the same pipe open in
2 the most conservative sequence. A dynamic load factor of two is used to determine the transient loads
2 unless a lower value is justified by analysis.

2 Relief valves discharging into a closed system or a system with long discharge piping are reviewed to
2 identify any significant transient loadings. Any significant loading is analyzed using dynamic analyses to
2 include the effects of changes in momentum due to fluid flow changes of direction and any potential water
2 slugs. The piping will be adequately supported such that piping stresses associated with the defined
2 transient loads satisfy applicable Code requirements.

2 The loading combinations and stress limits contained in the requirements of SRP 3.9.3.II.4 and referenced
2 in Regulatory Guide 1.48 are met. However, ASME Code Section III Subsection NF did not provide
2 faulted condition allowable stress limits for Class 2 and 3 component supports until the 1977 edition. The
2 allowables for Class 1 components in the 1974 edition of Subsection NF and subsequent applicable
2 addenda for its Class 2 and 3 component supports faulted stress allowables were utilized.

### 2 9.6.4.6 Fire Protection

2 The SSF will be used when the existing plant systems or facilities of any of the three units are unavailable
2 due to a fire. The SSF is not designed to independently bring the reactor from hot shutdown to cold
2 shutdown. Cold shutdown will be achieved and maintained through the use of existing plant systems and
2 equipment as discussed below. No repairs or modifications are required to achieve hot shutdown utilizing
2 the SSF shutdown method. Repairs for cold shutdown may be required depending upon the fire area.

2 ### 9.6.4.2 Structure Design

2 The SSF is statically and dynamically analyzed and designed as a three-dimensional space frame subjected
2 to the applicable loads summarized in Section 9.6.3.1, "Structure" on page 9-82. The Structural Design
2 Language (STRUDL) computer program is used to perform the analyses. The design is in accordance
2 with the codes and criteria listed in Table 9-19. Design loads and loading combinations are in accordance
2 with the NRC Standard Review Plan, Section 3.8.4.

2 The SSF is designed to withstand the effects of wind and tornado loadings, without loss of capability of
2 the systems to perform their safety functions. The basis for the selected wind velocity is reference 1 on
2 page 3-45 of Section 3.3, "Wind and Tornado Loadings" on page 3-43. Buildings and structures with a
2 height to minimum horizontal dimension ratio exceeding five should be dynamically analyzed to
2 determine the effect of gust factors (ref. American National Standard, "Building Code Requirements for
2 Minimum Design Loads in Buildings and Other Structures," ANSI A58.1-1972, New York, New York).
2 The SSF has a height/width ratio of less than five, and therefore, the gust factor of unity is used for
2 determining wind forces. The design tornado used in calculating tornado loadings is in conformance with
2 Regulatory Guide 1.76 except as noted in Section 9.6.3.1, "Structure" on page 9-82.

2 The relatively small surface area of the structure and its location result in an extremely low probability
2 that a turbine missile would strike the facility. Turbine missile impact is not considered a viable load
2 condition due to the location of the SSF with respect to the turbine. All postulated missiles are per the
2 NRC Standard Review Plan Section 3.5.1.4 Rev. 1 and Regulatory Guide 1.76. The barrier thicknesses
2 for the structure are such that they preclude any perforation and/or scabbing from the postulated tornado
2 generated missiles. Minimum barrier thickness is three times the postulated missiles calculated depths of
2 penetrations (see Table 9-18).

2 The dynamic analysis is made utilizing the STRUDL-DYNAL computer program. The design response
2 spectra were developed in accordance with the procedures of Regulatory Guide 1.60. It corresponds to
2 the expected maximum bedrock acceleration of 0.1g. Damping values are per Regulatory Guide 1.61.

2 The structure will withstand the specified design conditions without impairment of structural integrity or
2 safety function.

2 ### 9.6.4.3 Seismic Subsystem Analysis

2 The seismic analysis of Category I pipe is performed using dynamic modal analysis techniques. No static
2 seismic analysis is used for SSF ASME Code piping. Modal response spectrum methods are used.
2 Response of individual modes is combined by the Grouping Method of Regulatory Guide 1.92. An
2 adequate number of masses or degrees of freedom are included in the model to determine the response of
2 significant modes. The response due to each of three components of earthquake motion is combined by
2 the square-root-of-the-sum-of-the-square rule as described in Regulatory Guide 1.92. Pipe supported
2 from multiple levels or structure is designed for an envelop of the response spectra for all supporting
2 structures.

2 Constant vertical static factors are not used. Vertical response is obtained from a dynamic modal analysis.
2 Modal damping ratios are consistent with Regulatory Guide 1.61.

2 The location of the SSF non-Category I piping has been reviewed to determine those areas of proximity
2 to Category I piping or safety related equipment. Where Category I piping or safety related equipment is
2 in the proximity area, the non-Category I piping has been seismically qualified and supported or rerouted
2 out of the problem area.

2 **9.6.3.6.4 Heating Ventilation and Air Conditioning**

2 The SSF HVAC system consists of two subsystems, a ventilation system and an air conditioning system.
2 Both systems are powered by the SSF Power System. Sections of each system are shut down in event of
2 fire in the area served.

2 **VENTILATION SYSTEM**

2 The diesel generator room, switchgear room, pump room, and HVAC room do not require close control
2 of temperature, and the relatively high heat loads are dissipated with a variable volume ventilation system.
2 The purpose of the ventilation system is to provide filtered outside air which is tempered if necessary to
2 maintain a minimum temperature of 60°F and a maximum temperature of 104°F in each area.

2 **AIR CONDITIONING SYSTEM**

2 Certain rooms in the SSF require close control of temperature and have year-round heat loads of such
2 magnitude to necessitate continuous operation of mechanical refrigeration to maintain 72°F and a
2 maximum of 50 percent RH with a minimum of outside air for ventilation. The air conditioning system
2 supplies each area with a constant volume of air. A heat coil located in each area with a local control
2 tempers the air as required to maintain the desired temperature.

2 **9.6.3.6.5 SSF Sump System**

2 The SSF Sump System provides a collection and discharge function for normal equipment drainage
2 within the SSF. The main components of the system are the sump and two sump pumps which handle
2 the flow routed to the sump via the floor drain system located throughout the SSF.

## 2 9.6.4 SYSTEM EVALUATIONS

### 2 9.6.4.1 General

2 The design of the SSF was reviewed to meet the requirements of Appendix R of 10CFR 50, Sections
2 III.G.3 and III.L, and those requirements applicable for flooding and seismic events.

2 The SSF, the associated mechanical and electrical systems and power supplies meet or exceed the
2 applicable criteria contained in the Oconee FSAR Chapter 3, "Design of Structures, Components,
2 Equipment, and Systems" on page 3-1. Additionally, ASME and IEEE codes are utilized as appropriate,
2 in the design of various subsystems and components. The SSF and systems/components needed for safe
2 shutdown are designed to withstand the Safe Shutdown Earthquake (SSE). The SSF systems required for
2 safe shutdown are designed with adequate capacity to ensure safe hot shutdown conditions of all three
2 Oconee units.

2 The SSF power system is designed with adequate capacity and capability to supply the necessary loads,
2 and is physically and electrically independent from the station electrical distribution system power supply.
2 Additionally, the AC and DC power systems and equipment required for the SSF essential functions have
5 been designed and installed consistent with the Oconee QA program of Class 1E equipment.

2 These systems are not designed to meet the single failure criterion, but are designed such that failures in
2 the systems do not cause failures or inadvertent operations of existing plant systems. The electrical
2 systems in the SSF are manually initiated, that is, multiple actions must be performed to provide flow to
2 existing plant safety systems.

2 **9.6.3.6.1 SSF Lighting System Description**

2 Normal lighting for the SSF is provided by fluorescent and HID lighting units. These lighting units are
2 located to provide adequate levels of light with good distribution throughout the structure.

2 Emergency AC lighting for the SSF is provided by incandescent lighting units. These units are located to
2 provide adequate levels of lighting in all areas of the structure.

2 Emergency DC lighting for the SSF is provided by self-contained 12VDC battery pack lighting units.
2 These units are located to provide adequate levels of lighting for control panel operation and for entering
2 and leaving the structure. These battery pack lights are energized automatically upon an undervoltage in
2 the normal lighting system power supply.

2 **9.6.3.6.2 SSF Fire Protection and Detection**

2 The SSF contains two fire protection systems, a water system and a carbon dioxide system.

2 The water system is provided with manually valved hose reels in the stairwell at each floor elevation and
2 inside the entrance to the diesel room. From these locations the hose lengths are such that the entire SSF
2 can be served by the primary fire protection system.

2 The low pressure carbon dioxide system provided is actuated by thermal detectors to automatically flood
2 the diesel area. Carbon dioxide is stored in a refrigerated storage tank in sufficient quantity to provide
2 twice the required coverage for the area.

2 Portable carbon dioxide extinguishers are also provided.

2 Detection devices are located throughout the SSF and will annunciate with a single alarm to the Unit
2 Control Rooms, SSF Control Room, Security. Specific alarms annunciate on the Fire Alarm Control
2 Unit located in the SSF vestibule.

2 **9.6.3.6.3 SSF Service Water**

2 The SSF Service Water System consists of two subsystems: The HVAC Service Water System and the
2 Diesel Engine Service Water System.

2 The HVAC Service Water System, which operates continuously, contains two pumps and supplies
2 cooling water to the HVAC condensers. Only one pump will operate at any given time with the other
2 idle pump acting as a backup.

2 The Diesel Engine Service Water System, which operates only when the diesel is operating, contains one
2 pump and provides service water to the diesel engine jacket water heat exchangers.

2 This flow is monitored during periodic operational test or emergency operation. All three pumps take
2 their suction from the embedded CCW piping and return the flow to the CCW piping after passing
2 through their respective system. SSF Diesel Service Water is diverted to the yard drain during an SSF
2 event to avoid overheating the water contained in the SSF ASW supply piping.

2 The SSF Diesel Service Water System is shown on Figure 9-37.

2 **9.6.3.4.2 Diesel Generator**

2 The SSF Power System is provided with standby power from a dedicated diesel generator. This SSF
2 diesel generator is rated for continuous operation at 3500 kW, 0.8 pf, and 4160 VAC. The SSF electrical
2 design load does not exceed the continuous rating of the diesel generator. The auxiliaries required to
2 assure proper operation of the SSF diesel generator are supplied entirely from the SSF Power System.
2 The SSF diesel generator is provided with manual start capability from the SSF only. It uses a
2 compressed air starting system with four air storage tanks. Each set of two tanks will provide sufficient air
2 to start the diesel unit five successive times. An independent fuel system, complete with a separate
2 underground storage tank, duplex filter arrangement, a fuel oil transfer pump, and one-hour day tank, is
2 supplied for the diesel-electric generating unit.

2 The diesel generator protection system initiates automatic and immediate protective action to prevent or
2 limit damage to the SSF diesel generator. The following protective trips are provided to protect the diesel
2 generator at all times and are not bypassed when the diesel generator is in the emergency mode:

2 1. Engine Overspeed
2 2. Generator Differential Protection
2 3. Low-low Lube Oil Pressure

2 **9.6.3.5 Instrumentation**

2 **9.6.3.5.1 SSF Reactor Coolant Makeup System Instrumentation**

2 Each unit is provided with instrumentation to monitor RCM System flow, pressure and temperature; RC
2 Loop A and B pressure and temperature; pressurizer level, pressurizer and reactor incore temperature.
5 Five (5) Incore Thermocouples per unit may be used to monitor the incore temperature. Six (6) RTD's
5 per unit will be used to monitor Loop A and B RC System Hot & Cold Leg temperature. Readout is
2 displayed on the SSF control panel. Table 9-16 provides a listing of instrumentation.

2 **9.6.3.5.2 SSF Auxiliary Service Water Instrumentation**

2 Each unit is provided with Steam Generator A & B level instrumentation labeled as listed in Table 9-16.
2 Readout is displayed on the SSF control panel.

2 **9.6.3.6 Support Systems**

2 The Standby Shutdown Facility (SSF) Support Systems are designed to provide for the SSF:

2 • Lighting
2 • Fire Protection
2 • Fire Detection
2 • Service Water
2 • Heating Ventilation and Air Conditioning (HVAC)
2 • Sump Drainage

2 The diesel engine service water and the HVAC service water piping are designed in accordance with
2 ASME Section III, Class 3, which includes seismic design. The fire protection water, carbon dioxide,
2 potable water, and sewage piping systems are seismically restrained in areas above seismically designed
2 equipment. The lighting system, the fire detection system, and the sump drainage system, are not
2 seismically designed. The water and carbon dioxide fire protection systems and the fire detection system
2 are designed and constructed to meet or exceed National Fire Codes.

2 ### 9.6.3.3 Auxiliary Service Water (ASW) System

2 The SSF ASW System is designed to cool the RCS during a station blackout and in conjunction with the
2 loss of the normal and Emergency Feedwater System by providing steam generator cooling.

2 The SSF ASW pump is the major component of the system. One motor driven SSF ASW pump,
2 powered from OST1 Switchgear, serves all three units and is located in the SSF. The suction supply for
2 the SSF ASW pump is lake water from the embedded Unit 2 condenser circulating water piping. A
2 submersible pump is available to replenish the water supply in the embedded CCW pipe if siphon flow
2 through the CCW pipe is lost.

2 The SSF ASW flow rate provided to each unit's steam generators is controlled using the motor operated
4 valves on each unit's SSF ASW supply header. Manually operated bypass valves, installed in parallel with
4 the motor-operated valves, are also available to:

4 1. Provide SSF ASW Flow control at low SSF ASW Flow rates.

4 2. Provide more precise SSF ASW Flow control when used in parallel with the motor-operated valves.

2 The SSF ASW pump is sized to provide enough flow to all 3 Oconee units to adequately remove decay
2 heat from the RCS and maintain natural circulation in the RCS. An SSF ASW pump minimum flow
2 line is provided to ensure that the pump minimum flow requirements are met. The SSF ASW system,
2 pump and valves are operated and tested from the SSF only. The SSF ASW system is shown on
2 Figure 9-36.

2 Auxiliary service water enters the steam generators via the normal emergency feedwater ring headers.

2 ### 9.6.3.4 Electrical Power

2 #### 9.6.3.4.1 General Description

2 The Standby Shutdown Facility (SSF) Electrical Power System includes 4160VAC, 600VAC, 208VAC,
2 120VAC, and 125VDC power. This system supplies power necessary to maintain hot shutdown of the
2 reactors of each unit, in the event of loss of power from all other power systems. It consists of
2 switchgear, load center, motor control centers, panelboards, batteries, battery chargers, inverters, a
2 diesel-electric generator unit, relays, control devices, and interconnecting cable supplying the appropriate
2 loads.

2 The 120VAC power system in conjunction with the 125VDC instrumentation and control power system
2 supplies continuous control power to all loads that are required for a hot shutdown of each reactor.

2 Following the loss of all normal and emergency power, on-site and off-site, the diesel-electric generating
2 unit will be manually started by initiating its start signal from the SSF Control Panel in the SSF. The
2 diesel generator and its associated auxiliaries are housed in a Class 1 structure and are protected against
2 seismic events.

2 The 4160VAC SSF Power System bus will then be connected to its diesel-electric, backup source of
2 power by manually closing the appropriate 4160VAC generator breaker.

2 Schematics of the SSF electrical system are shown on Figure 9-40 and Figure 9-41.

2 resulting matrix is inverted to obtain the stiffness matrix, which is used together with the mass matrix to
2 obtain the eigenvalues and associated eigenvectors.

2 Having obtained the frequencies and mode shapes and employing the appropriate damping factors, the
2 spectral acceleration for each mode can be obtained from Design Ground Motion response spectra curves.
2 The standard response spectrum technique is used to determine inertial forces, shears, moments, and
2 displacements for each mode. The structural response is obtained by combining the modal contributions
2 of all the modes considered. The combined effect is represented by the square root of the sum of the
2 squares.

2 The analytical technique used to generate the response spectra at specified elevations is the time history
2 method. The acceleration time history of each elevation is retained for the generation of response spectra
2 reflecting the maximum acceleration of a single degree of freedom system for a range of frequencies at the
2 respective elevation. The structure will withstand the specified design conditions without impairment of
2 structural integrity or safety function.

### 2 9.6.3.2 Reactor Coolant Makeup (RCM) System

2 The SSF RCM System is designed to supply borated makeup to the Reactor Coolant System (RCS) to
2 provide Reactor Coolant Pump Seal cooling and RCS inventory. An SSF RCM Pump located in the
2 Reactor Building of each unit will supply makeup to the RCS should the normal makeup system and the
2 reactor coolant pumps become inoperative because of a station blackout condition caused by the loss of
2 all other on-site and off-site power. The system is designed to ensure that sufficient borated water is
2 available from the spent fuel pools to allow SSF to maintain hot shutdown conditions for all three units
2 for approximately 72 hours. This time period is based on drawing the water level in the spent fuel pool
2 down to a minimum of one foot above the top of the spent fuel racks. The SSF RCM System is
5 operated and/or tested from the Standby Shutdown Facility. The SSF RCM System is shown on
3 Figure 9-35. The SSF RCM Pump is capable of delivering borated water from the Spent Fuel Pool to
2 the RC pump seal injection lines. A portion of this seal injection flow is used to makeup for RC pump
2 seal leakage while the remainder flows into the RCS to makeup for volume shrinkage and other RCS
2 leakage.

2 The SSF RCM Pump is a positive displacement pump driven by an induction motor, powered from the
2 SSF Power System. The pump is located in the Reactor Building basement sufficiently below the spent
2 fuel pool water level to assure that adequate net positive suction head is available.

2 A SSF RCM Filter is supplied downstream of the SSF RCM Pump to collect particulate matter larger
2 than five microns that could be harmful to the seal faces. The filter is sized to accept three times the flow
2 output of the SSF RCM Pump. Fouling of this filter is not considered to be a problem since the filter
2 has been conservatively sized.

2 There is a select bank of pressurizer heaters that are normally controlled from the main unit's control
2 room, however, during SSF events this bank can be controlled should it become necessary from the SSF
2 Control Panel. Pressurizer level control can be accomplished from proper control of ASW flow to the
2 steam generators, control of the SSF RCM pump flow to the RCS, and proper control of the SSF RC
3 letdown line flow. Additional RCS inventory control can be accomplished using the RV head vent. SSF
3 D/G power can be connected to the RV head vent valves within 8 hours after the start of an SSF Event.
3 Control of the RV head vent valves will be accomplished using a portable control panel.

2 determining wind forces. The design tornado used in calculating tornado loadings is in conformance with
2 Regulatory Guide 1.76 with the following exceptions:

2 1. Rotational wind speed is 300 mph.
2 2. Translational speed of tornado is 60 mph.
2 3. Radius of maximum rotational speed is 240 ft.
2 4. Tornado induced negative pressure differential is 3 psi, occurring in three seconds.

2 The spectrum and characteristics of tornado-generated missiles is covered in a later section.

2 **FLOOD DESIGN**

2 Flood studies show that Lake Keowee and Jocassee are designed with adequate margins to contain and
2 control floods. The first is a general flooding of the rivers and reservoirs in the area due to a rainfall in
2 excess of the Probable Maximum Precipitation (PMP). The FSAR addresses Oconee's location as on a
2 ridgeline 100′ above maximum known floods. Therefore, external flooding due to rainfall affecting rivers
2 and reservoirs is not a problem. The SSF is within the site boundary and, therefore, is not subject to
2 flooding from lake waters.

2 The grade level entrance of the SSF is 797.0 feet above mean sea level (msl). In the event of flooding due
2 to a break in the non-seismic condenser circulating water (CCW) system piping located in the Turbine
2 Building, the maximum expected water level within the site boundary is 795.5 ft. Since the maximum
2 expected water level is below the elevation of the grade level entrance to the SSF, the structure will not be
2 flooded by such an incident.

3

2 The SSF will stabilize the plant at hot shutdown condition. Damage to any other (outside of
2 containment) equipment required to maneuver the plant to cold shutdown will need to be evaluated and
3 repaired following Turbine Building Flooding. As a PRA enhancement the SSF is provided with a five
3 foot external flood wall which is equipped with a water tight door near the south entrance of the SSF. A
3 stairway over the wall provides access to the north entrance.

2 **MISSILE PROTECTION**

2 The only postulated missiles generated by natural phenomena are tornado generated missiles. The SSF is
2 designed to resist the effects of tornado generated missiles in combination with other loadings. Table 9-17
2 lists the postulated tornado generated missiles.

2 Penetration depths are calculated using the modified NDRC formula and the modified Petry formula.
2 Table 9-18 lists the calculated penetration depths and the minimum barrier thicknesses to preclude
2 perforation and scabbing, hence eliminating secondary missiles.

2 **SEISMIC DESIGN**

2 The design response spectra correspond to the expected maximum bedrock acceleration of 0.1 g. The
2 design response spectra were developed in accordance with the procedures of Reg. Guide 1.60. The
2 seismic loads as a result of a base excitation are determined by a dynamic analysis. The dynamic analysis
2 is made utilizing the STRUDL-DYNAL computer program. The base of the structure is considered
2 fixed.

2 With the geometry and properties of the model defined, the model's influence coefficients (the flexibility
2 matrix) are determined. The contributions of flexure as well as shearing deformations are considered. The

2. Cold shutdown must be achievable within seventy-two hours following the fire accident. Credit can be taken for reasonable damage control measures.
3. No credit is allowed for fire protection equipment in developing shutdown scenarios.

**TURBINE BUILDING FLOOD CRITERIA**

Components of the SSF systems and the associated structures are designed to achieve and maintain hot shutdown conditions in the event the Turbine Building is subjected to flooding.

**ELECTRICAL SEPARATION CRITERIA**

Selected motor operated valves and the pressurizer heater bank are capable of being powered and controlled from either the normal station electrical systems or the SSF electrical system. Suitable electrical separation is provided in the following manner. Electrical distribution of the SSF is identified in Figure 9-40 and Figure 9-41 is provided by the SSF motor control centers (MCC's). These MCC's are capable of being powered from either an existing plant load center or the SSF load center through key interlocked breakers at the MCC's. These breakers provide separation of the power supplies to the SSF loads.

During normal operation, these loads are powered from a normal (non-SSF) load center via the SSF MCC's.

During operation of the SSF, these loads are powered from the SSF diesel generator via the SSF load center and SSF MCC's.

## 9.6.3 SYSTEM DESCRIPTIONS

### 9.6.3.1 Structure

The Standby Shutdown Facility (SSF) is a reinforced concrete structure consisting of a diesel generator room, electrical equipment room, mechanical pump room, control room, central alarm station (CAS), and ventilation equipment room. The general arrangement of major equipment and structures is shown in Figure 9-30, Figure 9-31, Figure 9-32, Figure 9-33 and Figure 9-34.

The SSF has a seismic classification of Category 1. The following load conditions are considered in the analysis and design:

1. Structure Dead Loads
2. Equipment Loads
3. Live Loads
4. Normal Wind Loads
5. Seismic Loads
6. Tornado Wind Loads
7. Tornado Missile Loads
8. High Pressure Pipe Break Loads
9. Turbine Building Flooding Potential

**WIND AND TORNADO LOADS**

The design wind velocity for the SSF is 95 mph, at 30 ft. above the nominal ground elevation. This velocity is the fastest wind with a recurrence interval of 100 years. A gust factor of unity is used for

# 9.6 STANDBY SHUTDOWN FACILITY

## 9.6.1 GENERAL DESCRIPTION

The Standby Shutdown Facility (SSF) is designed as a standby system for use under extreme emergency conditions. The system provides additional "defense in-depth" protection for the health and safety of the public by serving as a backup to existing safety systems. The SSF is provided as an alternate means to achieve and maintain hot shutdown conditions following postulated fire, sabotage, or flooding events, and is designed in accordance with criteria associated with these events. Loss of all other station power is assumed for each event. In that the SSF is a backup to existing safety systems, the single failure criterion is not required. However, failures in the SSF systems will not cause failures or inadvertent operations in existing plant systems. The SSF requires manual activation and would be activated under adverse fire, flooding or sabotage conditions when existing redundant emergency systems are not available.

The SSF is designed to:

1. Maintain a minimum water level above the reactor core, with an intact Reactor Coolant System, and maintain Reactor Coolant Pump Seal cooling.
2. Assure natural circulation and core cooling by maintaining the primary coolant system filled to a sufficient level in the pressurizer while maintaining sufficient secondary side cooling water.
3. Transfer decay heat from the fuel to an ultimate heat sink.
4. Maintain the reactor subcritical by isolating all sources of RCS addition except for Reactor Coolant Makeup Pump System which always supplies makeup of a sufficient boron concentration.

The SSF consists of the following:

1. SSF Structure
2. SSF Reactor Coolant Makeup (RCM) System
3. SSF Auxiliary Service Water (ASW) System
4. SSF Electrical Power
5. SSF Support Systems

System Main Components are listed in Table 9-14. SSF Primary Valves are listed in Table 9-15. SSF Instrumentation is listed in Table 9-16.

## 9.6.2 DESIGN BASES

### FIRE PROTECTION CRITERIA

Cabling for the two independent methods to achieve hot shutdown, the SSF, and the normal plant safety systems, are separated to the extent practical within containment and by three-hour fire barriers outside containment.

The following additional criteria was utilized for identifying problem areas and to design the necessary system changes to insure safe shutdown.

1. The hypothesized fire is to be considered an "event", and thus need not be postulated concurrent with non-fire-related failures in safety systems, other plant accidents, or the most severe natural phenomena.

#### 9.5.1.7.4 Materials Containing Radioactivity

Materials are stored in accordance with station documents.

## 2 9.5.2 INSTRUMENT AND BREATHING AIR SYSTEMS

### 2 9.5.2.1 Design Basis

2 The Instrument and Breathing Air Systems are designed to provide clean, dry, oil free instrument air to all
2 air operated instrumentation and valves, and breathing air at ANSI Z86.1 Grade D standards to minimize
2 personnel exposure in areas of airborne contamination.

### 2 9.5.2.2 System Description

2 The Instrument Air (IA) System consists of a) one primary IA compressor with two filter/dryer trains, b)
2 three backup IA compressors with two filter/dryer trains, c) distribution headers, d) receiver tanks and e)
2 components supply lines. The IA System is shared by all three Oconee Units; therefore, the IA System is
2 required to operate continuously.

2 Normal operation for the IA System is for the primary IA compressor to supply all IA demands. Should
2 the primary IA compressor trip, be required to be removed from service for maintenance, or the IA
2 System demand exceed the primary IA compressor capacity, the backup IA compressors and any available
2 Service Air System compressor capacity reserves are used in supplying IA System demands.

2 An Auxiliary Instrument Air (AIA) System provides a reliable auxiliary source of instrument air to key
2 plant components needed to reach and maintain safe shutdown of the plant during a loss of IA event.
2 This system is composed of three (one per unit) compressors, combination filters, and desiccant dryers.
2 Separate distribution headers and supply lines are provided to these key components to ensure AIA
2 availability.

2 The Breathing Air System consists of one primary and one backup compressor package. This package
2 consists of one a) two stage inlet air filter, b) compressor, c) air/oil separator, d) and oil cooler/aftercooler.
2 After the compressor the air is passed through a) an air/water separator, b) a filter package, c) two
2 purification packages in parallel, d) into two parallel receiver tanks, and e) finally into the breathing air
2 manifolds. Breathing air is supplied to all areas and elevations by headers and individual supply stations
2 where the pressure is regulated for personnel use. Units 1 & 2 have one primary and one backup
2 compressor total for both Units, and Unit 3 has one primary and one backup for its use. The breathing
2 air systems are cross connected in such a way that any of the compressors can supply either of the Units'
2 breathing air needs.

#### 9.5.1.6.14 Interim Radwaste Building

At Oconee, the Interim Radwaste Building is a separate building. Fire hydrants and portable extinguishers are located in and around the Interim Radwaste Building. A radwaste technician conducts routine tours whenever the Interim Radwaste Building is being utilized.

#### 9.5.1.6.15 Decontamination Area

Flammable liquids are not stored in the Decontamination Area.

#### 9.5.1.6.16 Safety Related Water Tanks

Storage tanks supplying water for safe shutdown are located in areas which contain a minimum of combustibles or located outside the building. Portable extinguishers are provided in the area.

#### 9.5.1.6.17 Cooling Towers

Not applicable to Oconee Nuclear Station.

#### 9.5.1.6.18 Miscellaneous Areas

Records storage areas, shops, warehouses and auxiliary boilers are located such that if a fire occurs in this area it will not affect safe shutdown. The fuel oil tanks for the auxiliary boiler are located outside the protected area and are provided with appropriately sized dikes.

#### 9.5.1.6.19 Radwaste Facility

An automatic foam-water spray system is provided over the polymer fill station. Automatic sprinklers are provided in the trash storage and shredder area, truck bay, over contaminated oil pump skid, decon skid, contaminated oil storage area. Polymer tanks are vented through roof with a flame arrestor on the discharge. Smoke detectors are located over hazards in the facility and alarm in the Unit 3 control room. Vapor detectors are located in polymer fill station and near the tank fill line in the truck bay where vapors may accumulate. Portable fire extinguishers are provided in the area.

### 9.5.1.7 Special Protection Guidelines

#### 9.5.1.7.1 Welding and Cutting, Acetylene - Oxygen Fuel Gas Systems

Stations procedures cover storage and use of this equipment. A permit system is required to utilize this equipment.

#### 9.5.1.7.2 Storage Areas for Dry Ion Exchange Resins

Dry Ion exchange resins are not stored near vital areas. Resin storage in other areas is maintained at the minimum practical.

#### 9.5.1.7.3 Hazardous Chemicals

A program has been implemented concerning the storage and use of hazardous chemicals at Oconee Nuclear Station.

#### 9.5.1.6.8 Turbine Lubrication and Control Oil Storage and Use Area

The main turbine oil tanks are located on the Mezzanine Floor (Elevation 796+6). Each tank and distribution system is protected by an ionization detector and an automatic deluge sprinkler system.

The Turbine/Auxiliary Building wall is constructed of three hour fire rated materials except where flood control measures are required at door openings in the basement of the Turbine Building and mechanical penetrations within fifty feet from safety related cable trains.

#### 9.5.1.6.9 Diesel Generator Area

Emergency power for Oconee is the Keowee Hydro Station. As stated in Section 9.5.1.5.5, "Carbon Dioxide Suppression System" on page 9-74, the Diesel Generator located in the SSF is protected by a total flooding carbon dioxide system.

Keowee is a two-unit hydro plant with a combined output of 175,000 KVA which is connected to Oconee's 230 KV switching station through a single circuit overhead transmission line and to the Oconee startup transformers through a single circuit 13.8 KV underground line.

The Main Transformer and Lube Oil Storage Room at Keowee are protected by automatic deluge systems and the station is equipped with detection which alarms and annunciates in the Keowee and Oconee control rooms.

#### 9.5.1.6.10 Diesel Fuel Oil Storage Areas

The bulk Diesel Fuel Oil Storage tank is buried. A smaller volume day tank is located in the SSF and is protected by the total flooding carbon dioxide system.

#### 9.5.1.6.11 Safety Related Pumps

1. High Pressure Injection Pumps: Located in the Auxiliary Building at Elevation 758+0. One pump is separated from the others by masonry walls.
2. Low Pressure Injection Pumps: See (1) above.
3. Low Pressure Service Water Pumps: Located in the Turbine Building at Elevation 775+0.
4. Turbine Driven Emergency Feedwater Pump: Located in the Turbine Building at Elevation 775+0. Detection which alarms and annunciates in the control room is provided in addition to an automatic water spray system.
5. Motor Driven Emergency Feedwater Pumps: Located in the Turbine Building at elevation 775+0.

#### 9.5.1.6.12 New Fuel Area

At the present time new fuel is put into the spent fuel pool prior to installation in the reactor. No specific area has been designated as new fuel area. The spent fuel pool area is protected by portable extinguishers.

#### 9.5.1.6.13 Spent Fuel Pool Area

See Section 9.5.1.6.12, "New Fuel Area."

Breathing apparatus for control room operators is provided for each control room. Two self-contained Breathing Apparatus units are provided in each complex with additional air supply available as described in Section 9.5.1.4.4, "Ventilation" on page 9-69 (8 on page 9-71).

Power and control cables are not located in concealed floor and ceiling space and cables entering the control room terminate there, except for the SPDS computer power cable as stated above.

#### 9.5.1.6.3 Cable Spreading Room

The cable rooms at Oconee have ionization detectors and portable extinguishers available.

Manual hose stations are located adjacent to the cable rooms, which would be used in the event of a cable room fire. Portable extinguishers are also provided.

Cable rooms are separated from other plant areas by fire rated barriers and have at least two remote and separate entrances to provide access by fire brigade personnel.

The Cable Rooms and adjacent Cable Shafts are protected with manually actuated fixed water spray systems, hydraulically designed to provide a density of 0.10 GPM per square foot. The design of each Cable Room water spray system requires the manual operation of a HPSW full size pump to provide the above spray density.

#### 9.5.1.6.4 Plant Computer Room

Each control room has a computer room adjoining it. These computers are not control computers and perform no safety functions.

#### 9.5.1.6.5 Switchgear Rooms

Equipment rooms at Oconee are separated from other plant areas by adequate barriers. Automatic fire detection alarms and annunciates in the control room. Fire hose stations and portable extinguishers are readily available. Switchgear for equipment is located in the Turbine Building at Elevation 796 + 6.

The Equipment Rooms and adjacent Cable Shafts are protected by manually actuated fixed water spray systems, hydraulically designed to provide a density of 0.10 GPM/square foot. The design of each Equipment Room water spray system requires the manual operation of a HPSW full size pump to provide the above spray density.

#### 9.5.1.6.6 Remote Safety Related Panels

Combustible materials except cabling associated with the panels are not located in the area of the remote shutdown panels. Hose stations and portable extinguishers are available.

#### 9.5.1.6.7 Station Battery Rooms

Battery rooms are separated from the Turbine Building by three hour fire rated walls. Ventilation systems in the battery rooms are designed to maintain the hydrogen concentration below two percent volume concentration. Portable fire extinguishers are provided in each battery room in addition to extra extinguishers in adjacent areas.

#### 9.5.1.6.1 Primary and Secondary Containment

1. The Reactor Coolant Pumps, which are not required to operate but must maintain pressure boundaries for safe shutdown, have been provided with seismically qualified oil collection systems to prevent oil spillage reaching areas which may be above the flash point of the lubricating oil. The upper and lower oil pots have been modified with a shield to catch oil and carry it through a properly sized drain to a collection tank.

   Station procedures assure that during a refueling outage the oil collection system will be subjected to a routine preventative maintenance program and the collection tank will be verified to be empty prior to unit startup.

   The cable used at Oconee is constructed such that internal faults will not be a source of ignition.

   Carbon filters are treated as described in Section 9.5.1.4.4, "Ventilation" on page 9-69 (d). Portable extinguishers are available in the area in case a fire should start.

2. During refueling and maintenance periods, an excess of materials and personnel are in areas which are normally clear (i.e., work areas outside personnel hatches). During these periods, security personnel are on duty 24 hours at the personnel hatch entrance and are in a position to observe maintenance activities. These areas are protected by an automatic sprinkler system.

   Station Directives require permits for any welding and cutting operations and ensure that proper precautions are taken prior to allowing work to begin.

   Portable fire extinguishers are provided within the containment. In the event of a fire, fire brigade personnel would bring additional fire extinguishers to the area for fire fighting.

   Self-contained Breathing Apparatus are provided for the fire brigade personnel and additional air supply is available on site as previously described.

#### 9.5.1.6.2 Control Room

The control room is isolated from other areas of the plant by three hour fire barriers except for the wall adjacent to the lobby around the entrance door, where a steel plate was provided to satisfy concerns other than fire protection. Since the lobby is free of combustible material (and transient combustibles are not expected to obstruct the door), a fire would not propagate into the Control Room via the steel plate. There are two doors to stairways with 1½ hour rated doors which were found to be acceptable based on low combustible loading and constant attendance of the Control Room.

Fire hazard evaluations indicated that a fixed extinguishing system is not required in the Control Rooms, however, smoke detection devices have been located inside cabinets and consoles. Hose stations are located adjacent to each control room area with portable extinguishers provided in and around the general area. Guidance has been provided for the fire brigade concerning use of water in the control room.

Nozzles used on the hose stations in the area are the adjustable type which would cope with actual fire fighting needs, satisfy electrical safety and minimize physical damage to electrical equipment from hose stream impingement.

As previously stated, smoke detection is provided in the control room. The ionization detectors alarm on the control panel along with alarms from other areas of the plant. In addition to adding detectors in the cabinets and consoles, the location of several detectors have been modified in accordance with applicable NFPA Standard to provide effective coverage.

4 not a Selected Licensee Commitment. The largest Selected Licensee Commitment demand is 1738 gal/min. required by the Unit 3 Cable Room Water Spray System. This demand includes a 1238
0 gal/min system demand plus a 500 gal/min. non-fire related service water demand. A hose stream allowance is not included for this system based upon its expected usage.

6. Water for the fire protection system is provided from Lake Keowee. Full pond elevation is 800+0 with maximum drawdown at elevation 775+0.

   Between elevation 800+0 (full pond) and elevation 775+0 (maximum drawdown) there are 391,679 acre-feet of water available.

7. Fire hydrants are installed at a maximum of every 300 feet. Hose supplies are adequate to provide fire protection to all perimeter areas. Post indicator valves are provided and sections of the fire loop can be isolated for maintenance or repairs. Hose houses are located at several yard hydrants and are equipped with at least 200 feet of 1½ inch hose, 200 feet of 2½ inch hose, one-2½ inch gated wye and two-1½ inch nozzles and one-2½ inch to 1½ inch reducer.

#### 9.5.1.5.3 Water Sprinklers and Hose Standpipe Systems

1. Each automatic sprinkler system and hose station header has an independent connection to the plant HPSW System. LPSW System supplies the source water for hose stations in the Reactor Building.
2. Valves for the HPSW system are not electrically supervised. A program at the station requires fire protection valves to be sealed or locked in the normal open position. A periodic recorded inspection is conducted to ensure that there has been no tampering with the fire protection valves.
3. The automatic sprinkler systems were designed to conform to requirements of appropriate NFPA Standards.
4. Hose stations installations are equipped with a maximum of 100 feet of 1½ inch fire hose with an adjustable nozzle.

   Hose stations are located on elevations 771 + 0, 783 + 9 and 809 + 3 in the Auxiliary Building, and on all three levels in the Turbine Building. Six hose stations are located in each Reactor Building. The LPSW system supplies the source water in the Reactor Building.

5. Adjustable nozzles are provided on hoses for fighting fires. These nozzles are appropriate for the type fires which might occur.
6. The only fire suppression system at Oconee which uses foam is located in the Radwaste Facility. This system covers fires around the Polymer used in the solidification system.

#### 9.5.1.5.4 Halon Suppression System

The Administration Building Record Storage Vault is protected by a Halon 1301 total flood fire suppression system.

#### 9.5.1.5.5 Carbon Dioxide Suppression System

The SSF Diesel Generator Room is protected by a low pressure total flooding carbon dioxide suppression system.

#### 9.5.1.5.6 Portable Extinguishers

4 Portable fire extinguishers are provided in accordance with NFPA 10, "Standard For Portable Fire
4 Extinguishers."

### 9.5.1.6 Guidelines for Specific Plant Areas

#### 9.5.1.5.2 Fire Protection Water Supply Systems

1. An underground fire loop (16 inch cement-lined, ductile iron pipe) is provided around the perimeter of the plant site. Post indicator valves are provided and are sealed or locked open to prevent inadvertent closing of valves required open for fire protection. Monthly recorded inspections of fire protection valves and key control procedures will back up the availability of water for fire protection. Post indicator valves are arranged to provide isolation to portions of the main for maintenance or repair without shutting off the complete system.

   Valves will allow other service water systems to be removed from the HPSW system without compromising the fire protection system.

2. As indicated above, a 16 inch loop is provided around the perimeter of the plant. Connections from this header to the units are redundant. Auxiliary Building headers are fed from a 16 inch line coming from the yard and a four inch line from the Turbine Building.

3. Two 6000 gal/min and one 500 gal/min (jockey) high pressure service water pumps supply the HPSW system.

   The 500 gal/min pump will normally operate to keep pressure on the fire headers. In the event of a fire, one full size pump provides adequate capacity for fire protection service. The second full size pump is considered to be a spare.

   A 100,000 gallon elevated storage tank is provided as described in Section 9.5.1.2, "System Description and Evaluation" on page 9-65.

   Each pump has a motor with power taken from separate sources, i.e.: HPSW Pump A power from Bus No. 2, Unit No. 1; HPSW Pump B power from Bus No. 1, Unit No. 1.

   Since both HPSW pumps are powered directly from the 4160 V Buses which are interconnected between Oconee 1, 2 and 3 and to the emergency power from Keowee Hydro, adequate backup power is provided.

   The HPSW pumps are located in separate concrete block structures with power cable to the motors being embedded in concrete floor. Separation is by fire rated wall assemblies.

   HPSW system alarms received in the Control Room are:

   - Jockey pump stopped
   - Low HPSW system header pressure
   - Low elevated water storage tank level
   - Elevated water storage tank overflow.

4. Water is supplied to the HPSW pumps from the CCW system piping. Intake for this water is through the twelve pumps located at the intake structure. If power is lost to the CCW pumps, the system can continue to function as an unassisted syphon. 3

   Intake for the HPSW pumps is located at the CCW cross-connect header in Oconee 1 for HPSW pump A and Oconee 2 for HPSW pump B and the jockey pump. The CCW headers can be connected between Oconee 1, 2 and 3 to allow flow from either or all units as required.

   A 100,000 gallon elevated water storage tank has inventory which is also available to provide water for fire protection.

5. The total water supply using one 6000 gal/min pump for two hours is 720,000 gallons.

   The greatest demand for fire protection water is based on 1000 gal/min for fire hose plus 2571 gal/min (all sprinkler heads opened and flowing in the Unit 3 Turbine Building Mezzanine Level Sprinkler System) plus 500 gal/min non-fire related service water for a total of 4071 gal/min. This demand is

5. Radio communication is available with base stations in the Unit 1/2 control room and the security office. Portable radios are available at each of these locations. The fixed repeater is located above the Unit 1/2 Control Room in the Ventilation Equipment Room. If the repeater is lost to fire, the Communication System can function on the radio to radio channel.

## 9.5.1.5 Fire Detection and Suppression

### 9.5.1.5.1 Fire Detection

1. Deviations from NFPA 72D are identified and justified by paragraph number per National Fire Code,
2 1975:
2

1221, 1223 - At Oconee, the alarm comes into the control room. The operator then notifies plant personnel of the fire location.

1231 - Alarms on the control board are tested on each 12-hour shift by operator procedures.

1232 - Procedures require annual testing of transmitters and water flow actuated devices. Water spray systems for safety-related equipment are tested on an annual basis.

2110 - The Oconee fire detection system is cabled using steel/aluminum sheathed #16 AWG cable for signal transmission. This cable meets or exceeds the requirements for physical and electrical protection as defined in NEC, Article 760.

2222, 2223 - The fire detection system at Oconee is powered from a battery backed power supply through a static inverter to provide 240/120 VAC. These batteries which are Class 1E but utilized as Non-Class 1E are continuously charged from normal station power. On loss of normal station power, the system is designed to provide power to the fire detection system for one hour. In addition, a transfer switch is provided for transferring the inverter loads to regulated normal station AC power should a malfunction of the battery inverter supply occur.

2521 - The annunciator audible alert at Oconee serves the fire detection system as well as other plant systems. The visual indicator provided by the visual display prevents operator confusion regarding source of the alarm.

2. The fire detection system provides an audible and visual alarm and annunciation in the control room. Local audible alarms do not sound at the location of the fire. The operator receives the alarm in the control room, dispatches plant personnel to the location of the alarm to ascertain the local conditions and then, if necessary, summons the fire brigade by the PA and a radio paging system. By using the PA system the chance of misinterpretation of the alarm is minimized.

3. As stated in (2) above, with the use of the PA system, the possibility of confusion of the fire alarm with any other plant system alarms is negligible.

4. The fire detection system is powered from a battery-backed power supply through a static inverter to provide 240/120 VAC.

   These batteries which are Class 1E, but utilized as Non-Class 1E are continuously charged from normal station power. If normal station power is lost the system is designed to provide power to the detection system for one hour. In addition, a transfer switch is provided for transferring the inverter loads to regulated normal station AC power should a malfunction of the battery/inverter supply occur. This system design provides a power supply as dependable as the emergency power sources.

   Locations of detection devices are shown in Table 9-12. Detector locations in Equipment Rooms, Battery Rooms, Penetration Rooms and other areas exceed recommended spacing; however, spacing is in accordance with NRC commitments. Detector locations are selected based on engineering judgement to monitor areas containing vital equipment.

6. Escape and access routes will be established by pre-fire plan and practiced in drills by operating and fire brigade personnel.
7. Due to construction arrangement and discharge limitations, smoke and heat vents are not applicable.
8. Self-contained breathing apparatus, using full face positive pressure masks, approved by either NIOSH or US Bureau of Mines are provided for fire brigade, damage control and control room personnel.

   Air for refilling the air packs is provided from a breathing air compressor with a cascade system. The compressor is powered from a non-load shed source.

#### 9.5.1.4.5 Lighting and Communication

In addition to the normal ac lighting system, for each unit two separate emergency lighting systems are provided. These are an emergency 250V dc lighting system and a separate engineered safeguards 208Y/120 volt ac lighting system. These two systems are separate and distinct.

1. Fixed emergency dc lighting is fed from batteries that supply power for at least one hour.

   Since two ac lighting systems and one dc emergency lighting system are provided this is considered adequate.

   The engineered safeguards lighting system (ac), which is normally de-energized, provides lighting in the Auxiliary Building to enable personnel to leave or enter as necessary. Power is provided from two engineered safeguards 600 volt ac control centers through two 600/208Y/120 volt ac dry type transformers which in turn feed each of two panel boards located in the equipment room area. The engineered safeguard lighting is energized automatically by undervoltage sensing relays monitoring the normal 600 volt ac feeder voltage.

   The 250 volt dc lighting system, which is normally de-energized, provides operating level lighting in the control room and lighting at selected stairs and corridors in the Auxiliary, Turbine and Reactor Buildings. The emergency lighting is energized automatically by an undervoltage sensing relay mounted on individual panel boards located in their associated areas. Control power for the under-voltage transfer circuit is provided from the 250 volt dc station batteries. A test button is also provided at each panelboard to test the operability of the system without affecting normal lighting. Associated lighting units are incandescent.
2. Sealed beam portable lights are provided for fire brigade personnel.
3. Pathways from the Units 1/2 and Unit 3 control rooms to the SSF and from the Control Rooms to valve FDW-315 for each unit have normal and emergency lighting and, in addition, have 8-hour battery backed emergency lighting units. This includes at least one stairwell and corridor from each control room leading to an outside door of the Auxiliary Building, which leads to the SSF.

   Outside the Auxiliary Building, the area is normally well lit by daylight or by security lighting, powered from several sources.

   The SSF has 1½ hour battery backed emergency lighting which is backed by the SSF Diesel Generator.

   In addition, flashlights and spare batteries are available in the Control Rooms for operators to use.
4. The primary method of communication is an IBM CBX Telephone system with outside as well as in-plant connections. In conjunction with the telephone, a page system is used for calls throughout the plant. Each telephone is marked with the emergency reporting numbers.

   Sound powered telephones are available throughout the plant in addition to the telephone-page system.

In Oconee 1 and 2 pressurizing and ventilation air is brought from the outside, sent through one of the two redundant fan and filter units after which it mixes with return air. The mixed air is then conditioned and conveyed to the control room. Air handling units condition recirculated air for the cable and equipment rooms. A connecting cable shaft between the cable and equipment rooms is used to convey cool air to the equipment rooms to supplement cooling from smaller cooling units dedicated to the equipment rooms.

Approximately 1500 CFM passes from the Oconee 1 equipment room to Oconee 2 equipment room to maintain air balance conditions.

A purge fan is located in the wall of Oconee 2 equipment room to purge air to the Auxiliary Building corridor where it would be transported by the Auxiliary Building HVAC equipment, monitored and exhausted through the unit vent.

In Oconee 3, two purge exhaust ducts are furnished for the equipment room and kitchen area of the control room. These exhaust ducts enable the equipment room, cable room and the control room to be purged in the event of a fire.

The equipment and cable rooms of Oconee 3 are equipped with individual air handling units and are balanced. In the event of purging, the cable shaft would be used to carry smoke from the cable room to the equipment room.

The fan for purging Oconee 3 is located on Elevation 838+0 with HVAC equipment and would exhaust to that area enabling the Auxiliary Building System to pick up, monitor and discharge products of combustion through the unit vent. The Oconee 1 and Oconee 2 control room would be purged with portable equipment.

The Reactor Building's ventilation systems are designed to remove normal heat loss from equipment. A Reactor Building purge system is provided to purge the containment with fresh air when circumstances dictate. The purge equipment (fans, filters, etc) for the Reactor Building, except for interior ducts, is located outside the Reactor Building. The purge exhaust is filtered, monitored and alarmed prior to discharge to the atmosphere to prevent releases exceeding acceptable limits.

2. In Oconee 1 and 2 the purge fan is located in the Oconee 2 equipment room wall. This 3000 CFM fan is available to purge smoke to the Auxiliary Building corridor where it can be exhausted by the Auxiliary Building HVAC system. This purge fan would remove smoke from the Oconee 1 and 2 cable rooms and Oconee 1 and 2 equipment rooms.

   The Oconee 3 purge fan is designed to remove smoke from the control room through the kitchen and from the equipment room. This smoke would be exhausted to the HVAC level of the Auxiliary Building and handled by the HVAC system at that point.

   Neither a single failure nor an inadvertent operation of the purge systems would adversely affect plant operations. A single failure would require portable equipment be used to purge individual areas.

   The operation of the Reactor Building purge system is monitored from the control rooms. Monitors and alarms are provided to indicate the status of the system. Triple isolation valves are provided at the Reactor Buildings' penetrations. In the unlikely event of an inadvertent operation, the fact that the air is filtered, monitored and alarmed prior to discharge, assures that the protection for the public would be maintained.

3. The power and controls for the HVAC units serving the control rooms, cable rooms and equipment rooms are located at the air handling units.

4. The filters will have isolation valves at their inlet which will be closed unless the fans are energized. The fans run only during an emergency.

5. The fresh air intakes for the air handling units are located on the Auxiliary Building roof. The outside air is used only to make-up lost air in the system in that return air is circulated to cool areas.

Armstrong Armaflex Sheet Insulation is a low-to-medium temperature insulation with high resistance to transmission of water vapor. Flame spread - 25 in accordance with ASTM E-84.

5. Cable routings and separation are adequate to preclude the loss of safe shutdown capability by a single fire hazard. Current carrying capability of cables is derated by 30 percent of manufacturer's rating as a design criterion.

6. The cable used at Oconee is classified as either power, control or instrumentation.

   The 5 and 8 KV cables are three conductor power cables. The tinned copper conductors are covered with a semi-conductive extruded strand shield, insulated with ethylene propylene rubber (EPR) and wrapped with a tinned copper shield tape. The three conductors are then twisted with a flame retardant non-hygroscopic filler, bound together with binders tape, encased in a 25 mil galvanized steel interlocked armor jacket and covered with a flame retardant polyvinyl chloride (PVC) jacket.

   The three conductor 2KV power cable, which is used for 600 V systems, is constructed the same as the 5 and 8 KV cable except that a hypalon or neoprene jacket has been applied over the EPR insulation in lieu of the tinned copper shield tape.

   Control cables are multi-conductor cables. The tinned copper conductor has EPR insulation with the hypalon or neoprene jacket over the singles; the singles have been twisted with the flame retardant non-hygroscopic fillers and covered with an asbestos mylar binder tape. This is encased in 25 mil galvanized steel interlocked armor with a polyvinyl-chloride jacket.

   Instrumentation cable (outside the containment) is single or multipaired cable consisting of #16 AWG copper conductor with PVC insulation. The singles are paired and twisted with an aluminum mylar shield with PVC jacket and overall served wire armor encased in a flame retardant PVC jacket.

   Instrumentation (inside containment) is multi-conductor and paired cables consisting of #16 AWG, tinned copper conductors insulated with EPR and hypalon jacket. This is twisted with flame retardant fillers, wrapped with an asbestos mylar binder tape and encased in 25 mil galvanized steel interlocked armor with a flame retardant PVC jacket overall.

   The use of armor on cables ensures they are more resistant to fire, mechanical damage and electrostatic and electromagnetic interferences. The armor also provides protection from short circuits and overloads.

7. Any new cable installed at Oconee will be constructed similar to the cable presently used at the station.

8. Cables are located in dedicated cable trays and trenches. Piping is not routed in cable trays or trenches and cables are not routed in pipe trenches (except for one cable tray in the pipe trench to the Interim Radwaste Building, which is outside the plant). Miscellaneous storage is not permitted in the cable trays or trenches. There are no other fire hazards present in cable trays.

9. The cable rooms, the equipment rooms and the cable shafts are provided with smoke venting capabilities. Portable purge fans would augment installed equipment.

10. Only those cables which are required are routed to the control room. Cables entering the control room terminate there. There are no power and control cables in concealed floor or ceiling space, except for one 220V-40A power cable in the ceiling space supplying the SPDS computer. Therefore, a fire suppression system is not required in this area.

#### 9.5.1.4.4 Ventilation

1. At Oconee a separate ventilation system serves the Unit 1/2 and Unit 3 control rooms. A common ventilation system serves the cable and equipment rooms for Units 1 and 2. In Unit 3, each of these areas has a separate conditioning system.

7. Transformers installed in buildings containing safety related systems are not oil-filled, except for CT-4, which is located in the Unit 1/2 Blockhouse.
8. Transformers which are oil-filled and within 50 feet of a building containing safety related systems are protected with an automatic water spray system.
9. Floor drains are sized to remove fire protection water in locations where suppression systems are present, with exception of the cable spread rooms, equipment rooms, cable shafts and personnel hatch areas.
10. Redundant systems and equipment essential for a dedicated safe shutdown are separated by fire rated barriers.

#### 9.5.1.4.2 Control of Combustibles

1. Safe shutdown systems are separated from combustible materials except for those required for operation.
2. There is no bulk gas storage in areas affecting safe shutdown equipment.
3. Power and control cable at Oconee is covered with a PVC jacket. Refer to Section 9.5.1.4.3, "Electric Cable Construction, Cable Tray and Cable Penetrations" (6 on page 9-69) for discussion of construction and use of cable at Oconee.
4. Storage of flammable liquids comply with NFPA 30, "Flammable and Combustible Liquids Code." The governing edition of NFPA 30 is the edition current when the storage area is designed.

   An exception is the SSF Diesel Generator Fuel Oil Day Tank, which is as follows:

   - NFPA-30, Section 2.4.4.3 requires a "normally closed remotely activated valve... on each liquid transfer connection below the liquid level...to provide for quick cutoff of flow in the event of a fire in the vicinity of the tank."

   Since the fuel oil transfer pumps are positive displacement type and all piping connected to the storage tank is Duke Class B (seismic design), the intent of this section is met.

#### 9.5.1.4.3 Electric Cable Construction, Cable Tray and Cable Penetrations

1. Cable trays are constructed from non-combustible materials.
2. See Section 9.5.1.6.3, "Cable Spreading Room" on page 9-76.
3. Cable splices in raceways are not permitted; refer to Section 8.3.1.5.2, "Cable Tray Fill" on page 8-19. Current carrying capacity in cables is designed at 70 percent of manufacturer recommendation. On this basis, the potential of internally generated faults with ensuing fires is considered remote; therefore, protection of the cable insulation and jacketing from an internally initiated fire is not required.
4. Penetrations in fire barriers, horizontal and vertical, have been sealed. Mono-Kote is the trade name for a fire protection material developed by Construction Products Division, W R Grace and Company. It is a cementitious (plaster), mill-mixed material requiring only the addition of water and applied directly to surfaces requiring sealing for fire protection.

   Mono-Kote has been fire tested and rated by Underwriters' Laboratories, Inc. (UL) in accordance with ASTM E-119. Fire ratings of up to 4 hours have been achieved. Testing in accordance with ASTM E-84 by UL demonstrated Flame Spread - 10; Fuel Contribution - 5; and Smoke developed - 0.

   Dow Corning 3-6548 silicone RTV Foam is also used to seal penetrations in fire barriers. RTV Foam, when tested in accordance with ASTM E-119, has achieved fire ratings of up to 3 hours.

reporting fires are stationed during and for thirty minutes after the welding or burning takes place.
4 The Site Fire Protection Engineer audits the welding and burning program to assure its proper implementation.

2. Leak testing and similar procedures such as air flow determination use commercially available aerosol techniques.

3. Provisions for the bulk storage of combustible material such as HEPA filters, carbon filters, and dry
3 ion exchange resins are described in Site Directive 3.2.7. Procedures have been developed to control the transient storage of these materials during periods of replacement. The use of wood inside buildings containing safety-related systems or components is permitted only when suitable non-combustibles are not available. If wood is required, only fire retardant treated wood is used.

The Oconee Nuclear Station organization has been staffed and equipped to be self sufficient regarding
3 fire situations which might arise in the protected area. The Emergency Coordinator has the authority to utilize offsite fire departments. The local fire departments receive annual training.

In order to assure the capability to successfully contain a fire it is necessary to perform testing and maintenance of fire detection, fire fighting, fire protection equipment, emergency lighting and communications equipment. This is accomplished through the station periodic test program in which the performance of equipment is verified through actual testing as in the case of pumps or inspection and verification for such items as hoses and fire extinguishers. The responsibility for completion of these tests is not assigned to specific individuals but rather are assigned to responsible groups within the station organization to be performed at established frequencies. Testing is not considered necessary for the communication system since it also serves as the normal system and is in continuous use. Deficiencies identified as a result of inspections and use are reported and corrected.

3 Site Directive 3.2.9 has been developed to provide for the reporting and appropriate corrective action to be taken in the event fire protection/detection equipment has been determined to be, or is scheduled to be inoperable. Specific priorities have been established for expediting repairs to this equipment. Additional surveillance is specified as well as other necessary supplementary actions. Fire
4 protection/detection equipment will not be taken out of service without notifying the Site Fire
4 Protection Engineer.

3 The Oconee Fire Brigade organization is addressed by Site Directive 3.2.8 which describes the functions and duties of each position and identifies individuals by title to fill these positions. The organization provides for a Fire Chief, Assistant Fire Chief, and Shift Coverage.

Fire brigade training is provided in accordance with the Nuclear Production Department Fire Protection Training and Qualification Manual or commitments made to NRC.

## 9.5.1.4 General Guidelines for Plant Protection

### 9.5.1.4.1 Building Design

1. Plant layout separates safe shutdown systems from unacceptable fire hazards.
2. The Fire Protection Review analysis will be reviewed and updated as necessary.
3. See Section 9.5.1.6.3, "Cable Spreading Room" on page 9-76 for cable room comments.
4. Interior wall and structural components and radiation shielding are non-combustible. Coatings are non-combustible with flame spread and fuel contribution of 50 or less.
5. There is no metal roof deck construction related to safe shutdown systems at Oconee, with the exception of the Turbine Building.
6. Suspended ceilings and their supports are non-combustible. Combustibles in this area are minimal.

HPSW Pump A is located in Oconee 1 and power is furnished directly from 4160V Bus. No. 2 Unit No. 1.

Power to these pumps may be fed from Oconee 1, 2 or 3; through the 230 KV and/or 525 KV systems; Unit 1 or 2 of Keowee Hydro Units; or from Combustion Turbines located at Lee Steam Station. These alternate sources, including the emergency power from Keowee, assure that the fire protection system will not be lost due to a single failure.

Failure or inadvertent operation of an automatic fire suppression system will not incapacitate redundant safe shutdown systems or functions.

New reactor fuel is stored in the Spent Fuel Pool prior to installation in the reactor. Hazards evaluations performed on this area (Spent Fuel Pool Area) revealed an insignificant amount of combustibles ordinarily in the area and that fire protection by portable fire extinguisher is acceptable.

Equipment required for safe shutdown is not shared between units except for the LPSW and CCW Systems. This arrangement has been evaluated and is acceptable.

### 9.5.1.3 Administrative Procedures and Controls

Administrative procedures consistent with the need for maintaining the performance of the Fire Protection System and personnel have been established at the Oconee Nuclear Station. These procedures have been
4 established at the Oconee Nuclear Station through the use of Site Directives and other station documents. Guidance incorporated in the following publications has been utilized as much as practical:

NFPA-4 - Organization for Fire Services

NFPA-6 - Industrial Fire Loss Prevention

NFPA-7 - Management of Fire Emergencies

NFPA-8 - Management Responsibility for Effects of Fire on Operations

NFPA-27 - Private Fire Brigades

NRC Document - Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance.

3 Site Directive 3.2.7, "Control of Combustible Materials," states the parameters of allowable storage of
5 combustible materials for Oconee. Periodic inspections by insurance representatives and station personnel assure adherence to the Directive.

Reviews are conducted of work requests by the Oconee Planning Section to determine the effects of these activities on station fire barriers or stops. This identification than alerts personnel to special precautions which must be taken.

1. Work involving ignition sources such as welding and burning is performed under closely controlled
3 conditions. Site Directive 3.2.10 has been written to cover these activities in areas which have not
4 been approved specifically by the Site Fire Protection Engineer. This directive has been developed by personnel experienced in fire protection to provide guidance and precautions for fire protection when welding or burning. Provisions of the directive require the establishment of clear zones, preparation of floors, considerations of the ventilation systems and openings to adjacent rooms. Proper use of shielding materials is also required to protect equipment from the possibility of stray sparks. Prior to the initiation of welding or burning, the area must be inspected by a trained individual and written permission must be given. Fire watches, trained in the use of fire fighting equipment and methods of

# 9.5 OTHER AUXILIARY SYSTEMS

0 **Note**

0 This section of the FSAR contains information on the design bases and design criteria of this
0 system/structure. Additional information that may assist the reader in understanding this system is
0 contained in the design basis document (DBD) for this system/structure.

## 9.5.1 FIRE PROTECTION SYSTEM

### 9.5.1.1 Design Bases

The overall fire protection program is based on an evaluation of the potential fire hazards in the Auxiliary, Reactor Buildings, adjacent areas of the Turbine Building and the effect of postulated design basis fires relative to maintaining the ability to perform safe shutdown functions and minimize radioactive releases to the environment.

Total reliance is not placed on a single automatic fire suppression method. Fire hose stations and portable extinguishers are provided.

### 9.5.1.2 System Description and Evaluation

The High Pressure Service Water System (HPSW) provides water for the fire protection system at Oconee. Two 6,000 gal/min (at 117 psig) pumps and one 500 gal/min (at 117 psig) jockey pump supply the HPSW System.

The 500 gal/min jockey pump normally operates to maintain the system pressure on the fire protection headers. In the event of a fire, one 6000 gal/min pump will automatically provide sufficient water for maintaining elevated water storage tank inventory. The second 6000 gal/min pump is considered to be a spare.

5 Pump suctions are connected to the Condenser Circulating Water (CCW) crossover header. Service water
5 may be supplied by the CCW inlet headers for any Oconee Unit.

If power to the CCW pumps is lost, the emergency discharge to the Keowee hydro tailrace will
3 automatically open and the system would continue to function as an unassisted syphon. CCW normal flow is 177,000 gal/min with each of the twelve (12) pumps.

A 100,000 gallon elevated storage tank is connected to the HPSW system. This tank serves as a source of water should the demand of the HPSW system exceed the capacity of the HPSW jockey pump. If the tank level should drop to 70,000 gallons, one HPSW pump will start. If the tank level continues to drop to 60,000 gallons, the second HPSW pump will start. The HPSW pumps will continue to run until the tank level reaches the 90,000 gallons (approximate) level and then cut off.

Power to the jockey pump is provided from 600V Motor Control Center 1XE located at Elevation 775 which is normally fed from 600V load center 1X3 located at Elevation 796 with an alternate supply from 600V load center 1X2 at Elevation 796. The jockey pump is located in Unit 2 along with HPSW pump B. Power to HPSW pump B is furnished directly from 4160V Bus No. 1 Unit No. 1.

The system may be actuated by an operator during normal operation for testing.

Particulate filtration is achieved by a medium efficiency pre-filter and a high efficiency (HEPA) filter.

The pre-filter consists of multiple horizontal tubular bags attached to a vertical metal plate header. The bags are made of ultra fine glass fibers and are supported so that adjacent bags do not touch and reduce the flow area. At the filter train design flow of 1000 cfm, the pre-filter is operating at one-half its rated flow.

The HEPA filter will intercept any particulates that pass through the pre-filter. The filter consists of a single cell of fiber glass media mounted in a metal frame. The cell has face dimensions of 24 inches x 24 inches and a depth of 11 ½ inches and is rated at 1150 scfm.

0 Adsorption filtration is accomplished by an activated charcoal filter. The filter consists of three horizontal removable type double tray carbon cells. Flow through the trays is essentially vertical. Each tray has a face area of 4.2 sq ft and a bed depth of 2 inches. At rated flow (167 cfm), the average face velocity is 40 ft/min and the residence time is 0.25 seconds. Each tray contains 40 lbs of carbon. The carbon is
0 impregnated so that it will adsorb methyl iodide as well as elemental iodine.

### 9.4.7.3 Safety Evaluation

The fans and filter trains for the system are redundant and only one fan and one filter train is required for emergency operation. Refer to Table 6-19 for a failure analysis of the system.

### 9.4.7.4 Inspection and Test Requirements

The Penetration Room Ventilation System is not normally in operation, but the equipment is accessible for periodic inspection. The entire system can be tested during normal operation. Testing and inspection of the system shall be as required by the Technical Specifications.

### 9.4.7.2 System Description

The Penetration Room Ventilation System is provided with two fans and two filter assemblies. Both fans
0 discharge through a single line to the unit vent. A schematic of the system is shown in Figure 6-4.

During normal operation, this system is held on standby with each fan aligned with a filter assembly. The
0 engineered safeguards signal from the Reactor Building pressure will actuate the fans. The Control room,
0 as well as remote instrumentation, monitors operation.

The design flow rate from the penetration room far exceeds the maximum anticipated Reactor Building
1 leakage. The design leak rate of .125 volume percent per day from the Reactor Building to the
1 penetration room (this is one-half of the total design leak rate out of the Reactor Building referenced in
1 Section 6.2.1, "Containment Functional Design" on page 6-9 amounts to approximately 7.8 scfm compared to a design evacuation rate of 1000 scfm for each half of the system. The three valves in each purge line penetration will be closed by Reactor Building isolation signal. The Reactor Building Purge Equipment, if running, will be shut down from an interlock on the Reactor Building Purge isolation valves. After closing of the external valves, a small normally open valve vents the leakage, if any, from the two outermost valves into the penetration room. The Reactor Building Purge Equipment is not activated when the reactor is above cold shutdown conditions.

Following a loss-of-coolant accident, a Reactor Building isolation signal will place the system in operation by starting both full-size fans. Two power-operated butterfly valves which open when the fans start are provided at the discharge of each fan. This valve will be closed to prevent recirculation if one fan fails. A check valve is also provided at the discharge of each fan to prevent recirculation on failure of a fan. In the event of a fan failure, the normally closed tie valve (PR-20) can be opened from its remote manual station to maintain adequate cooling air through the idle filter train.

The system utilizes remote manual control valves PR-13 and PR-17 in conjunction with constant speed fans to provide the proper negative pressure in the penetration room. Locations of penetrations and openings in the penetration room are shown on Figure 6-23 and Figure 6-24. The system is designed so
1 that each filter train will maintain a maximum negative pressure of 1.73 inches $H_2O$ with respect to the
1 outside atmosphere, assured by the two redundant 8 inch vacuum relief valves. If during operation the
0 leakage increases causing a decrease in negative pressure below 0.06 inches $H_2O$ with respect to the
0 outside atmosphere, the remote manual control valve will be adjusted or leaks will be repaired to bring the negative pressure to .06 inches $H_2O$ or greater.

The remote manual control valve is also used to compensate for filter loading. Initially, it will be partially closed; and as the filter loads up causing a decrease in flow and negative penetration room pressure, the valve will gradually be opened so that the pressure drop across the filter-valve combination remains constant. By periodically adjusting the remote manual control valve to offset the effect of increased leakage and filter loading, the system characteristic remains constant.

The communicative paths between various parts of the penetration room are very large in comparison with the minute leakage that might exist due to imperfect seals. It therefore can be assumed that no pressure differentials exist in the room so that an instrument string sensing pressure at a single point can be used. Penetration room pressure is displayed in the control room and excessive and insufficient vacuum are annunciated.

Fan status and radiation level of filter effluent are displayed in the control room and excessive radiation is annunciated. Filter $\Delta P$ is displayed locally. Filter flow is displayed remotely adjacent to the remote manual control valves PR-13 and PR-17 remote control stations.

Inside the Reactor Building, the cooling units are located outside the secondary shield at an elevation above the water level in the bottom of the Reactor Building during post-accident conditions. In this location, the units are protected from being flooded.

The major equipment of the Reactor Building Cooling Units is arranged in three independent strings with three duplicate service water supply lines. In the unlikely event of a failure in one of the three cooling units, the Reactor Building Spray System independently, or half of the Reactor Building Spray System capacity combined with the remaining two cooling units, will provide cooling capacity in excess of that required. Fan-motor operation under design LOCA conditions has been demonstrated by prototype test.

A failure analysis of the cooling units is presented in Table 6-6.

### 9.4.6.4 Inspection and Testing Requirements

The equipment, piping, valves, and instrumentation are arranged so that they can be visually inspected. The cooling units and associated piping are located outside the secondary concrete shield. Personnel can enter the Reactor Building during power operations to inspect and maintain this equipment. The service water piping and valves outside the Reactor Building are inspectable at all times. Operational tests and
0 inspections are performed prior to initial startup after each refueling outage.

0 In addition, the cooling units will be tested periodically as follows:

0 1. the fans will be started and inspected for proper operation.

2. The return line service water valves will be opened, and the lines checked for flow.

## 9.4.7 REACTOR BUILDING PENETRATION ROOM VENTILATION SYSTEM

### 9.4.7.1 Design Bases

This system is designed to collect and process potential Reactor Building penetration leakage to minimize environmental activity levels resulting from post-accident Reactor Building leaks. Experience has shown that Reactor Building leakage is more likely at penetrations than through the liner plates or weld joints.

The main function of the system is to control and minimize the release of radioactive materials from the Reactor Building to the environment in post-accident conditions. When the system is in operation, a
0 negative pressure with respect to surrounding areas will be maintained in the penetration room to ensure inleakage.

Leakage into each of the penetration rooms is discharged to the unit vent through a pair of filter assemblies each consisting of a prefilter, an absolute filter, and a charcoal filter in series. The entire system is designed to operate under negative pressure up to the fan discharge.

The Penetration Room Ventilation System is not vulnerable to control malfunctions since it is controlled manually. Instrumentation is used only to monitor system performance and has no control function other than to guide the operator in adjusting the final control elements.

More detailed information concerning radiation levels and leakage requirements are discussed in Section 6.5.2, "Containment Spray Systems" on page 6-58.

pressure service water supply and return lines and isolation valves are provided on these lines at the penetrations.

### 9.4.6.2 System Description

0 The Reactor Building Cooling System shown in Figure 9-29 consists of the following subsystems and components:

4 1. Three Reactor Building Cooling Units (RBCUs), each consisting of a 2-speed vane axial fan, four cooling coils and distribution ductwork. These three cooling units are Engineered Safety Systems.

2. Four Reactor Building Auxiliary Cooling Units each consisting of a 2-speed vane axial fan, four cooling coils, and distribution ductwork.

During normal plant operation, the A and C Reactor Building Cooling Units operate in the high speed
0 mode. These units circulate Reactor Building air over low pressure service water supplied cooling coils and distribute the cool air throughout the lower portion of the Reactor Building. Low pressure service
0 water normally supplied to the B RBCU is diverted to four Auxiliary Cooling Units. Two EMO-ES valves (LPSW-565 and LPSW-566) provided in the Low Pressure Service Water System divert the water
0 from the B RBCU to the Auxiliary Cooling Units. This low pressure service water supplies the four cooling coils that comprise each Auxiliary Cooling Unit. The four auxiliary cooling unit fans are operated
0 in the high speed mode. The Auxiliary Cooling Units distribute the cool air via a duct system to the upper portion of the Reactor Building. The temperature in the Reactor Building can be controlled by varying the number of Auxiliary Cooling Units running.

During an emergency, the Reactor Building Cooling System mode of operation changes automatically. Upon receipt of the signal from the Engineered Safeguards Actuation System, the Reactor Building
0 Cooling Units A and C change to low speed operation and the B RBCU unit is energized at low speed. The fans are run at the slower speed because of the changed horsepower requirements generated by the denser building atmosphere. Also on the ES signal, the EMO valves, in the Low Pressure Service Water
0 System, which diverted water from the B RBCU to the Auxiliary Cooling Units are re-aligned. Valve
0 LPSW-565 closes, stopping water flow to the Auxiliary Coolers, and LPSW-566 opens, allowing water
0 flow to the B RBCU. Additionally, all Low Pressure Service Water valves at the discharge of the three
0 RBCUs go to the full open position.

0 The accident may impose severe stresses on the lower portion of the duct work, causing possible collapse
0 or deformation. Therefore, the fusible links holding the dropout plates provided in the duct work below the coils melt and drop off, assuring that a positive path for recirculation of the Reactor Building atmosphere is available.

0

### 9.4.6.3 Safety Evaluation

0 The three Reactor Building Cooling Units (RBCUs) are an engineered safety feature. These units provide the design heat removal capacity following a loss-of-coolant accident with all three coolers operating by continuously circulating the steam-air mixture past the cooling tubes to transfer heat from the containment atmosphere to the low pressure service water.

Heat removal capacity of the Reactor Building Cooling Units at various Reactor Building temperatures is
4 in Figure 6-6. Figure 6-7 shows how the Reactor Building cooling rate varies with the air-steam mixture flow rate. The cooling capacity decreases by less than 7 percent even if the mixture flow rate decreases by 40 percent.

There are three modes of operation possible for the "Reactor Building Purge System": 1) the normal purge, 2) the mini-purge, and 3) the recirculation mode.

The normal purge mode purges the Reactor Building with 35,000 cfm of fresh air which enters by way of the supply portion and leaves by way of the exhaust portion described above. The filtered exhaust air is all released to the atmosphere via the unit vent.

The mini-purge mode of operation provides a means to purge the Reactor Building at a reduced flow rate when activity levels are higher than desired for full purging. A 10,000 cfm vane-axial fan is provided to by-pass the normal purge exhaust fan. A series of pneumatically operated dampers provide isolation and control. During mini-purge, flow from the Reactor Building is through the purge filter train and can be modulated up to a maximum of 10,000 cfm. The vane-axial mini-purge fan is constant volume and to maintain 10,000 cfm flow, Reactor Building air is mixed with outside air, i.e., the more air being purged from the Reactor Building, the less air drawn from the outside air make-up intake. The mini-purge fan and normal purge fan cannot operate simultaneously.

The recirculation mode of operation provides a means of filtering the Reactor Building atmosphere to reduce the level of airborne radiation therein without discharging contaminants to the environment. Through a series of pneumatically operated dampers, the normal purge exhaust fan recirculates the Reactor Building air through the purge filter train and returns the air back to the Reactor Building.

### 9.4.5.3 Safety Evaluation

Each Reactor Building Purge System supply and exhaust penetration of the Reactor Building wall is equipped with dual isolation valves. The valves inside the Reactor Building are electrically operated and
0 the valves outside the Reactor Building have pneumatic actuators. The valves operate independently of one another and are in the closed position unless the purge is in operation.

The Purge System discharge to the unit vent is monitored and alarmed to prevent the release from exceeding acceptable limits.

### 9.4.5.4 Inspection and Testing Requirements

The Reactor Building Purge System is normally not in operation. The equipment and component are accessible for periodic maintenance. Parts of the system are maintained and tested in accordance with the Technical Specifications.

## 9.4.6 REACTOR BUILDING COOLING SYSTEM

### 9.4.6.1 Design Bases

The Reactor Building Cooling Systems are designed to remove the heat in the containment atmosphere
2 during normal plant operation and post accident operation.

A portion of the Reactor Building Cooling System is described in Section 6.2.2, "Containment Heat Removal Systems" on page 6-22 as an Engineered Safety Feature.

The Reactor Building Cooling System is composed of two subsystems: Reactor Building Coolers and Reactor Building Auxiliary Coolers.

All components of the Reactor Building Cooling System are inside the Reactor Building. The only penetrations into and out of the Reactor Building that are related to the cooling system are the low

#### 9.4.4.3 Safety Evaluation

The Turbine Building Ventilation System operates to maintain suitable environmental conditions in the Turbine Building during normal plant operation.

#### 9.4.4.4 Inspection and Testing Requirements

The Turbine Building Ventilation System is in continuous operation during normal plant operation and is readily accessible for periodic inspection and maintenance.

### 9.4.5 REACTOR BUILDING PURGE SYSTEM

#### 9.4.5.1 Design Bases

0 The Reactor Building Purge System purges the Reactor Building with fresh air during and just prior to
0 unit outages or acts as a recirculation clean-up system to reduce airborne contaminant levels inside the Reactor Building.

In the purge mode of operation, outside air is introduced into the Reactor Building through a supply system which has dual isolation valves at the containment wall. Outside air is circulated throughout the Reactor Building by the normal Reactor Building Ventilation System. Air is then exhausted from the Reactor Building by the Reactor Building purge exhaust filter train.

0 The filter train consists of prefilters, HEPA filters, and charcoal filters. A centrifugal fan is positioned
0 downstream of the filter train. There are double isolation valves in the piping running from the Reactor Building to the filter train.

The isolation valves are automatic, are normally closed, and are opened only for the purging operation. The valves are arranged so the purge supply piping and the purge exhaust piping each have a electrically actuated valve inside the Reactor Building and a pneumatically actuated valve outside the Reactor Building.

There are three modes of operation possible for the Reactor Building Purge System; normal purge, mini-purge, and by-pass purge (recirculation). The purge filter train can also be used to provide filtered exhaust as discussed in Section 9.4.2, "Spent Fuel Pool Area Ventilation System" on page 9-55.

#### 9.4.5.2 System Description

0 The "Reactor Building Purge System" (Figure 9-29) purges the Reactor Building with fresh air or acts as a recirculation clean-up system to reduce airborne contaminant levels inside the Reactor Building.

The supply portion of this system consists of an outside air intake louver, roughing filters, a steam heating coil, associated ductwork and dual isolation valves at the reactor building wall. The exhaust portion of this system consists of a filter train, fans, associated ductwork, and dual isolation valves at the Reactor Building wall. The filter train consists of prefilter, HEPA filter, and charcoal filter. The isolation valves are automatic, normally closed and are opened only for the purging operation. The valves are so arranged that the supply portion and exhaust portion of the system each have an electrically actuated isolation valve inside the Reactor Building and two (2) pneumatically operated valves outside the Reactor Building (one is an isolation valve). A bleed valve between the two (2) outer valves vents any leakage from the Reactor Building into the penetration room.

0 The Hot Machine Shop air is supplied by two recirculating local cooling units. Each unit consists of
0 roughing filters, a compressor, evaporator and condenser coils, and centrifugal fan. These units supply
0 recirculated air with a small amount of make-up air throughout the Hot Machine Shop via a low pressure
0 duct system. Air is exhausted from the Hot Machine Shop via exhaust duct and filter train and is
0 discharged to the atmosphere through an independent vent stack.

0 Table 9-11 is a list of the primary equipment which comprises the Auxiliary Building Ventilation System
0 and the Hot Machine Shop Ventilation System. The list includes capacities and normal operation requirements.

Temperatures are maintained in the Auxiliary Building by throttling steam to the steam coils or low
0 pressure service water to the cooling coils as required. Temperatures are maintained in the Hot Machine
0 Shop by electric unit heaters in the supply ductwork. The Hot Machine Shop uses direct expansion (DX)
0 cooling.

Remote recirculating fan-coil type units provide standby spot cooling in the pump rooms and other high heat load areas. The fan coil units are also served by the Low Pressure Service Water System.

#### 9.4.3.3 Safety Evaluation

Under normal operating conditions, the Auxiliary Building Ventilation System supply fans and exhaust fans are balanced such that the exhaust air flow exceeds the supply air flow in order to minimize outleakage.

All exhaust air from the Auxiliary Building is directed to the unit vents where it is monitored prior to
0 being released to the atmosphere. All exhaust air from the Hot Machine Shop is monitored prior to being
0 released to the atmosphere through an independent vent stack.

#### 9.4.3.4 Inspection and Testing Requirements

0 The Auxiliary Building Ventilation System and the Hot Machine Shop Ventilation System are in
0 continuous operation and are readily accessible for periodic inspection and maintenance.

### 9.4.4 TURBINE BUILDING VENTILATION SYSTEM

#### 9.4.4.1 Design Bases

The Turbine Building Ventilation System is designed to provide a suitable environment for the operation of equipment and personnel access as required for inspection, testing and maintenance.

#### 9.4.4.2 System Description

The Turbine Building is ventilated using 100 percent outside air. Air is supplied through wall openings along the east wall and is exhausted by fans mounted in the roof and along the west wall.

3 There are twelve roof mounted exhaust fans. Eighteen additional exhaust fans are located along the west wall. Each of the thirty fans are independently operated so that all or a portion of the fans can run as needed to maintain conditions within the Turbine Building.

0 Table 9-11 is a list of the primary equipment which includes the Turbine Building Ventilation System
3 Exhaust Fans. The list includes number installed and normal operation requirements.

There are two 100 percent capacity vane axial fans which direct the Spent Fuel Pool air through the Reactor Building Purge Filter Train prior to being released to the unit vent. Only one fan is required for operation. The fans are manually energized by the operators should it become necessary to filter the exhaust air from the Fuel Pool Area. The automatic control sequence is such that the damper alignment, to redirect air flow through the Reactor Building Purge Filters, is automatically done when one of the fans is energized.

An alarm is provided when the fuel pool filtered flow drops below 70 percent of design flow.

A radiation monitor is provided to continuously monitor the fuel pool air and will alarm on a high radiation level.

### 9.4.2.4 Inspection and Test Requirements

The normal mode of the Spent Fuel Pool Area Ventilation System is in continuous operation and is
0 accessible for periodic inspection. The filtering mode of the Spent Fuel Pool Area Ventilation system is tested periodically to demonstrate its readiness and operability as required by the Technical Specifications.

## 9.4.3 AUXILIARY BUILDING VENTILATION SYSTEM

### 9.4.3.1 Design Bases

The Auxiliary Building Ventilation System is designed to provide a suitable environment for the operation, maintenance and testing of equipment and also for personnel access.

The Auxiliary Building Ventilation System serves all areas of the Auxiliary Building with the exception of the Control Room Area and the Penetration Rooms. The ventilation system is designed to maintain temperature limits during normal plant operation of 104°F and 60°F during summer and winter respectively.

Ventilation air is supplied to both clean and potentially contaminated areas within the Auxiliary Building. The flow path of the ventilation air in the Auxiliary Building is from clean or low activity areas towards areas of progressively higher activity.

0 All air from the Auxiliary Building is directed to the unit vent stacks at which point it is exhausted and continuously monitored by a radiation monitor which alarms on high radiation levels. In addition, a radiation monitor samples air throughout the Auxiliary Building Ventilation System. The detector output
0 is logged on a recorder in the Control Room. All air from the Hot Machine Shop is exhausted to the
0 atmosphere after being measured by an air flow monitor. Periodically, radiation levels are checked in the
0 air flow using an air flow totalizer and particulate sampler.

The exhaust fans and supply fans are manually balanced such that the exhaust flow exceeds the supply air flow to minimize outleakage.

### 9.4.3.2 System Description

0 The Auxiliary Building Ventilation System is comprised of the Auxiliary Building Ventilation System
0 proper and the Hot Machine Shop as shown in Figure 9-27 and Figure 9-28. Air is supplied to the
0 Auxiliary Building by a low pressure fan duct system. Air is taken in through outside air intake louvers by supply units consisting of roughing filters, steam coil, and cooling coil supplied by low pressure service
0 water. There are six main supply fans, each required for normal plant operation. Auxiliary Building air is
0 exhausted from the building, via exhaust duct and exhaust fans, through three unit vent stacks.

## 9.4.2 SPENT FUEL POOL AREA VENTILATION SYSTEM

### 9.4.2.1 Design Bases

The Spent Fuel Pool Area Ventilation System is designed to maintain a suitable environment for the operation, maintenance and testing of equipment and also for personnel access. The ventilation system is designed to maintain the Spent Fuel Pool Area at a maximum inside temperature of 104°F and a minimum temperature of 60°F.

The path of ventilating air in the Spent Fuel Pool Area is from areas of low activity toward areas of progressively higher activity for discharge to the unit vent.

An air handling unit consisting of roughing filters, steam heating coil, cooling coil supplied by low pressure service water, and a centrifugal fan supply 100 percent outside air to the Spent Fuel Pool Area. Two methods of exhausting air from the Fuel Pool Area are provided, a filtered exhaust system and an unfiltered exhaust system. Normal operation is with the unfiltered system in operation. In the filter mode, the Fuel Pool Area ventilation air passes through a filter train consisting of prefilters, high efficiency particulate (HEPA) filters, charcoal filter and two 100 percent vane axial fans. The filtered exhaust system is operable whenever fuel handling operations above or in the fuel pool are in progress.

The Spent Fuel Pool Area air is continuously monitored by radiation monitor, RIA-41.

### 9.4.2.2 System Description

Ventilation air for the Spent Fuel Pool Area is supplied by an air handling unit which consists of roughing filters, steam heating coil, cooling coil supplied by low pressure service water, and a centrifugal fan. Temperature is maintained in the Spent Fuel Pool Area by throttling steam to the heating coil or low pressure service water to the cooling coil.

In the normal mode of operation, the air from the Spent Fuel Pool Area is exhausted directly to the unit vents by the general Auxiliary Building exhaust fans. When fuel handling operations are in progress, the filtered exhaust system must be operable so in the event of an emergency the air leaving the Fuel Pool Area can be filtered.

The filtered exhaust system consists of a single filter train and two 100 percent capacity vane axial fans. The filter train utilized is the Reactor Building Purge Filter Train. The filter train is comprised of
0 prefilters, HEPA filters, and charcoal filters. An attempt to start the main Reactor Building purge fan will stop the Spent Fuel Pool filtered ventilation.

To control the direction of air flow, i.e., to direct the air from the Fuel Pool Area to the Reactor Building
0 Purge Filter Train, a series of pneumatic motor operated dampers are provided along with a crossover duct from the Fuel Pool to the filter train.

0 Figure 9-25 and Figure 9-26 are detailed diagramatics of the Spent Fuel Pool Area Ventilation System. The flow paths as well as air quantities are given in the diagram.

### 9.4.2.3 Safety Evaluation

Prior to handling fuel in the Spent Fuel Pool Area, the Spent Fuel Pool Ventilation System must be made operable as required by the Technical Specifications.

0 Cooling is provided to the Cable Rooms and Electrical Equipment Rooms by four small air handling units located in the vicinity of the rooms.

0 Table 9-11 is a list of the air handling units and operation requirement for the Control Room and Control Room Zone air conditioning system. Figure 9-24 is a schematic description of the ventilation and air conditioning systems for the Control Room and Control Room Zone.

#### 9.4.1.2.2 Control Room Oconee 3

4 The Oconee 3 Control Room Zone is comprised of the Control Room, the Cable Room, and the
4 Electrical Equipment Room. These areas are served by six air handling units. Two 100 percent air
4 handling units serve the Control Room, two 100 percent air handling units serve the Cable Room, and two 100 percent air handling units serve the Electrical Equipment Room. The air handling units consist
0 of roughing filters, chilled water cooling coils, and centrifugal fans. Chilled water is supplied to the air handling units by the Plant WC Chilled Water System.

4 Outside air is supplied to the Control Room for pressurization purposes from an intake on the Auxiliary Building roof. The outside air passes through a filter system composed of a prefilter, 99.5 percent efficient
3 HEPA filter, 90 percent efficient charcoal filter beds and centrifugal fan. Outside air is supplied to the return air intake of the air handling units. The outside air system is started by the plant operators.

A radiation monitor is provided to sample the return air entering the Control Room and Control Room Zone air handling units. The monitor alarms on a high radiation signal and alerts the operators to energize the outside air filter system to minimize the infiltration of unfiltered air into the Control Room.

0 Table 9-11 lists the air handling unit and operation requirements. Figure 9-24 is a schematic representation of the air conditioning system.

### 9.4.1.3 Safety Evaluation

The Control Room is served by redundant air handling units. The chilled water for the air handling units is supplied from the Plant WC Chilled Water System which is capable of supplying sufficient chilled water for all necessary systems with 50 percent of the system out of service.

Return air from the Control Room is continuously monitored by a radiation monitor before recirculating back to the Control Room. A high radiation level will alert the operators to energize the outside air filter
3 trains. The filter trains are 50 percent, each train consisting of a prefilter, HEPA filter, 90 percent efficient charcoal filter bed and centrifugal fan. The filters act to filter particulate matter from the outside air supplied to minimize uncontrolled infiltration into the Control Room.

### 9.4.1.4 Inspection and Testing Requirements

0 The Control Room Ventilation System is in continuous operation and is accessible for periodic
0 inspection. The Control Room pressurization portion of the system is tested periodically to demonstrate its readiness and operability as required by the Technical Specifications.

# 9.4 AIR CONDITIONING, HEATING, COOLING AND VENTILATION SYSTEMS

0 ## 9.4.1 CONTROL ROOM VENTILATION

### 9.4.1.1 Design Bases

0 The Control Room Ventilation and Air Conditioning Systems are designed to maintain the environment
4 in the Control Room, Cable Room and Electrical Equipment Rooms as indicated on Figure 9-24 within acceptable limits for the operation of unit controls as necessary for equipment and operating personnel.

Design conditions for the Control Room are 74°F and 50 percent maximum relative humidity. The Equipment Room is designed for 86°F and all other areas, i.e., the Control Room Zone and Cable Room are designed for 74°F. Outdoor design conditions are 95°F dry bulb and 76°F wet bulb. The ventilation and air conditioning systems are designed for continuous operation.

The radiation monitor, RIA-39, has a continuous sample of control room air pumped through the detector. High radiation level and loss of sample flow are annunciated at which time the operator energizes the outside air filter trains. The outside air filter trains act to filter particulate matter from the outside air to minimize uncontrolled infiltration into the Control Room.

### 9.4.1.2 System Description

#### 9.4.1.2.1 Control Room Oconee 1 and 2

The Control Room for Oconee 1 and 2 is shared for the operation of both units. The Control Room is
3 primarily served by two large air handling units. The units are 100 percent capacity and only one unit is required to operate at a time. Cooling is provided to the Unit 1 Cable Room, Unit 2 Cable Room, Unit 1 Equipment Room, and Unit 2 Equipment Room by air handling units 34, 35, 22, and 23 respectively.

4 The Control Room Zone is comprised of the Control Room and the Unit 1 and 2 Electrical Equipment
4 & Cable Rooms. These areas are served by a total of eight air handling units located on the third, fourth,
4 fifth, and sixth floor. All of the units are required to operate to maintain environmental conditions in the
4 areas, except for in the Control Room area where redundant units exist.

0 All of the air handling units described above consist of roughing filters, chilled water cooling coils, and centrifugal fans. Chilled water is supplied to the units from the plant WC chilled water system. Electric duct heaters are installed in the ductwork to provide heat to the different areas when necessary.

4 Outside air is supplied to the Control Room for pressurization purposes, from an intake on the Auxiliary Building roof. Air passes through filter trains which consist of pre-filters, 99.5 percent efficient HEPA
3 filters, 90 percent efficient charcoal filter beds, and a centrifugal fan. There are two 50 percent filter trains and the system is capable of operating with one train or both trains. During normal plant operations, the filter trains are not energized and require operator action to start. The outside air is supplied to the return
0 air intake of the large air handling units which serve the Control Room. A radiation monitor is provided in the return air intake of the air handling units to alert the operators in the Control Room on a high radiation reading at which time the operators start the outside air filter trains. The filter trains are designed for a flow of 1350 cfm each.

5. The radiation monitor/hydrogen recombiner outlet header, Elevation 824′ + 0″

Hydrogen Measurement

Analysis is accomplished by using the well established principle of thermal conductivity measurements of gases. This technique utilizes a self-heating filament fixed in the center of a temperature-controlled metal cavity. The filament temperature is determined by the amount of heat conducted by the presence of gas from the filament of the cavity walls. Thermal conductivity varies with gas species, thereby causing the filament temperature to change as the gas in the cavity changes. Filament resistance changes with temperature therefore, by using two filaments in separate cavities and connecting them in an electrical bridge, the difference in thermal conductivity of gases in the separate cavities may be determined electrically.

Electrical zero is set by first introducing the same gas to both cavities, then adjusting the electrical bridge to balance, resulting in a zero output. As different gases are introduced to the two individual cavities, the bridge will become unbalanced, and the electrical output will amplify with increasing differences in thermal conductivity of the gases used.

The measurement of hydrogen in the presence of nitrogen, oxygen and water vapor is possible because the thermal conductivity of hydrogen is approximately seven times higher than nitrogen, oxygen or water vapor, which have nearly the same thermal conductivities (at the filament operational temperature of approximately 550°K). The measurement is accomplished by using a thermal conductivity measurement cell and a catalytic reactor. The sample first flows through the reference section of the cell, then passes through the sample section of the measuring cell that includes the catalyst. The catalyst is chosen so that post-LOCA iodine will not poison the catalyst bed. The change in sample composition, due to the catalytic reaction is therefore indicated by the difference in thermal conductivity of the sample hydrogen content, as measured in the sample and reference sides of the cell.

If an excess amount of oxygen does not exist in the sample for recombining all the hydrogen, oxygen can be provided ahead of the hydrogen analyzer. The amount of oxygen added is determined by the highest range of the analyzer.

Alarms

Alarms are provided for high hydrogen concentration, cell failure and loss of power. These alarms are available on the analyzer itself and as signals to the control room annunciator. Additional alarms on the analyzer itself include low instrument temperature, low sample flow, low gas pressure and common failure.

### 9.3.7.3 Safety Evaluation

The Containment Hydrogen Monitor System (CHMS) meets the requirements of NUREG-0737, Item II.F.1.6. The CHMS has both indicator and recorder readouts in the control room on one of the two redundant channels and a indicator readout on the second channel. The CHMS has a range of 0% to 10% of Hydrogen. The CHMS indicator loop has a system accuracy of 5.0% of the full scale. The CHMS hardwired recorder loop and all the CHMS plant process computer loops have a system accuracy of 2.6% of the full scale. These values will provide information over the intended range of the CHMS that is sufficiently accurate and useful to allow the plant operator to adequately assess the hydrogen concentration within containment. There are five ports to draw samples for each of the redundant hydrogen monitors. The system provides capability to rapidly detect Hydrogen from the reactor and determine its concentration throughout the containment.

Selection of the monitor is by switching on the hydrogen monitor control panel. The sampling sequence is described in Section 9.3.6.2.3, "Mode of Operation" on page 9-50.

The operator can complete the sampling sequence in 30 to 60 minutes. The combined time allotted for sampling and analysis will be well within 3 hours from the time a decision has been made to take a sample. Alternate power sources are provided to meet the 3-hour limit in case of a loss of off-site power.

#### 9.3.6.2.3 Mode of Operation

The operation of the Post-Accident Containment Air Sampling System is sequenced as follows:

1. The Post-Accident Containment Sampling System isolates a known quantity of containment atmosphere, moves this quantity through a thiosulfate solution for separation of iodine and particulates from the noble gases, and provides a sample of the diluted gas for analysis.
2. Dilution gas (nitrogen) is provided for dilution factor up to 10,000 to 1.
3. The sampler grabs a sample of thiosulfate solution for various offline analyses such as iodine and/or particulate concentration.
4. The sampling system flushes and purges all interior sample lines as soon as possible to reduce personnel exposure dose rates.

## 9.3.7 CONTAINMENT HYDROGEN MONITORING SYSTEM

### 9.3.7.1 Design Bases

The containment Hydrogen Monitoring System provides continuous indication of hydrogen concentration in the containment atmosphere. The measurement capability is provided over the range of 0% to 10% hydrogen concentration under both positive and negative ambient pressures. A continuous indication of the hydrogen concentration is not required in the control room at all times during normal operation. If continuous indication of the hydrogen concentration is not available at all times, continuous indication and recording shall be functioning within 30 minutes of the initiation of the safety injection.

### 9.3.7.2 System Description

The Containment Hydrogen Monitor System withdraws a sample from the containment under normal, LOCA or Post LOCA conditions, the sample is analysed and returned to the containment. The monitoring system is designed to monitor containment gas for percentage volume of hydrogen.

A system of sample taking tubing is installed in the containment to draw air samples from 5 different levels or areas. Each of the sample intake lines has a solenoid valve which is remotely operated from a control panel in the ventilation room. At the control panel a selector solenoid valve is used to provide air flow to the Hydrogen Analyser from the selected intake port. The Hydrogen Analyser panels and associated remote control panels are located in the ventilation room. Remote alarm and indication is provided in the control room. There are two trains of equipment for each unit.

Ten Hydrogen Analyzer intake parts are installed, (two each) in the following locations:

1. The top of the Containment Building Dome, Elevation 983′ ± 5″
2. The operational level as close to the vessel as practical, Elevation 844 + 0′ ± 10′
3. The basement area, Elevation 788′ + 0″ ± 10′
4. The radiation monitor/hydrogen recombiner inlet header, Elevation 827′ + 4″

#### 9.3.6.1.3 Mode of Operation

The operation of the Post-Accident Liquid Sampling System is sequenced as follows:

1. The post-accident liquid sampling panel (sampler) isolates a reactor coolant sample at a system pressure up to 2500 psig and system temperature up to 650°F. The reactor coolant is cooled such that the isolated (pressurized) sample is below 200°F.
2. The pressurized liquid sample is depressurized into an evacuated gas collection bomb.
3. Gases are further stripped from the depressurized liquid sample by bubbling nitrogen through the liquid sample into the evacuated gas collection bomb.
4. Stripped gases are diluted with a known quantity of nitrogen to atmospheric pressure.
5. A diluted gas grab sample is taken for remote lab analysis.
6. pH of the depressurized, degassed liquid sample is measured.
7. A measured quantity of liquid sample is collected and taken for radioisotopic and chemical analysis.
8. Completely flush the sample lines from the sample isolation valves to the sampler, through the sampler, and from the sampler through the return line to the containment. Completely purge all gas sample paths (and around the diluted gas grab sampler) out the sample return line to containment. Completely flush all liquid sample paths (and around the diluted liquid grab sampler) out the sample return line into the containment. Completely drain and nitrogen purge all sample paths to the sampler sump and provide capability to pump the sampler sump through the sample return line to containment.

### 9.3.6.2 Post-Accident Containment Air Sampling System

#### 9.3.6.2.1 Design Bases

The system provides the capability to promptly (within three hours) obtain and analyze a containment air sample under accident conditions (Reg. Guide 1.3 or 1.4 release of fission products) without incurring a radiation exposure to any individual in excess of five (5) rems whole body dose or 75 rems to extremities. The system has the capability to:

1. Provide information related to the extent of core damage that has occurred or may be occurring during an accident.
2. Determine the types and quantities of fission products released to the containment atmosphere and which may be released to the environment

#### 9.3.6.2.2 System Description and Evaluation (Reference 1)

The Post-Accident Containment Air Sampling System consists of a sampler panel that houses the tubing, valving, instrumentation, and system components. The system is controlled and monitored remotely from the sampler control panel. The system is schematically illustrated in Figure 9-23. Oconee 1 and 2 share common sample and control panels. Oconee 3 has a completely separate system.

The sample panel is located in an area close to the containment that would normally have limited accessibility. The sampler control panel is located in an accessible area separate from the sampler panel location.

Existing sample lines to the online hydrogen monitors, through solenoid valves selection, become the sample source for the containment air sampling panel.

The deborating demineralizers may also be loaded with mixed bed resin and used as purification demineralizers to support normal purification and boron/lithium coordination programs.

The coolant bleed holdup tanks and the concentrated boric acid storage tanks are vented to the gaseous waste vent header to provide for filling and emptying without overpressurization or causing a vacuum to exist. In addition, each tank is equipped with a relief valve and a vacuum breaker. Pressurized nitrogen can be supplied to each tank to allow purging.

Instruments and controls for operation of this system are located in the control rooms. Instruments and controls for the coolant bleed holdup tanks and pumps and for the concentrated boric acid storage tanks and pumps are duplicated on the auxiliary control boards.

## 9.3.5 COOLANT TREATMENT SYSTEM

The Coolant Treatment System was originally designed and installed to both store reactor coolant bleed and to treat RC bleed for recycling. Since the boron recycling portion of the original Coolant Treatment System never functioned properly, the coolant storage portion is the only part of the system still in use at Oconee. The Coolant Storage System is described in Section 9.3.4, "Coolant Storage System" on page 9-47. Radwaste processing is described in Section 11.6.3, "Mechanical Systems" on page 11-26.

## 9.3.6 POST-ACCIDENT SAMPLING SYSTEM

### 9.3.6.1 Post-Accident Liquid Sampling System

#### 9.3.6.1.1 Design Bases

0 This system provides the capability to obtain and analyze a liquid Reactor Coolant System sample under accident (Reg. Guide 1.3 or 1.4 release of Fission products) conditions without incurring a radiation exposure to any individual in excess of five (5) rems whole body dose or 75 rems to extremities. The diluted liquid and dissolved gas samples obtained from the system have the capability to:

1. Provide information related to the extent of core damage that has occurred or may be occurring during accident.
2. Determine the types and quantities of fission products released to the containment in the liquid and gas phase and which may be released to the environment.
3. Provide information on coolant chemistry (i.e., boron concentration, dissolved gas, pH, etc.).

#### 9.3.6.1.2 System Description and Evaluation (Reference 1)

The Post-Accident Liquid Sample System consists of a sample panel that houses the tubing, valving, instrumentation and system components. The system is controlled and monitored remotely from the sample control panel. The system is schematically illustrated in Figure 9-22. There is one separate
5 system for each Oconee unit, located at elevation 771 + 0 in the auxiliary building.

The control panel actuates valves that are outside containment and before/after the sample enters/leaves the sampling panel. The remaining valves are controlled either by Operations from the control room or by Operations manually. All valves involving penetrations to the reactor building are normally closed except when sampling.

Selection of sample lines is provided on the sample control panel. The sampling sequence is described in Section 9.3.6.1.3, "Mode of Operation" on page 9-49.

## 9.3.4 COOLANT STORAGE SYSTEM

### 9.3.4.1 Design Bases

The Coolant Storage System for each unit is designed to accommodate the accumulated coolant bleed over a core cycle, including startup expansion and coolant letdown to storage for boric acid reduction.

Two coolant bleed holdup tanks, each with a capacity of 11,000 $ft^3$, are provided for each unit. One tank
2 provides storage for the reactor coolant bleed prior to treatment by the Radwaste Facility or makeup to
2 the Reactor Coolant System. The other tank provides additional storage and is used to store clean water
2 for use as feed to the Reactor Coolant System. An additional tank is provided for storage of the
2 concentrated boric acid from the boric acid mix tank. The RC Bleed Evaporator and associated
2 equipment is not used for coolant processing. Coolant processing is performed by the Radwaste Facility.

2 The storage of reactor coolant bleed requires approximately 55 percent of the volume of the bleed holdup tanks for each unit. The tanks for all three units are arranged so that they can be utilized to store liquid from the other units if so desired.

The design volume of coolant removed from one unit during heatup and dilution from cold shutdown is approximately 9600 $ft^3$. This occurs near the end of the core cycle when boric acid concentrations are reduced. Earlier in life the bleed quantity would be less.

An additional requirement for coolant storage is the partial drain which occurs during refueling. The coolant is removed in a batch of approximately 6100 $ft^3$. per unit and returned to the Reactor Coolant System upon completion of refueling. Thus, it occupies storage capacity only during the period of refueling. The required storage volume for refueling operations of 6100 $ft^3$. is less than 10 percent of the total available capacity.

A quench tank, located inside the Reactor Building, condenses and contains any effluent from the pressurizer safety valves. The quench tank is sized to condense one normal pressurizer steam volume without relieving to the Reactor Building atmosphere. A quench tank drain pump is provided for pumping the quench tank contents into the letdown storage tank. The reactor coolant which has leaked into the quench tank can be pumped directly back into the coolant system to avoid routing this leakage through the waste disposal system.

### 9.3.4.2 System Description and Evaluation

The Coolant Storage System is used for the collection and storage of reactor coolant liquid. The liquid is received from the High Pressure Injection System both as a result of reactor coolant expansion during startup and for boric acid concentration reduction during startup and normal operation. It is either conveyed to coolant bleed holdup tanks for storage or passed through deborating demineralizers for boric acid removal and returned as unborated makeup to the High Pressure Injection System. A spray nozzle in the coolant bleed tanks on the inlet line allow some of the gases to be released. Recirculating the tank allows further stripping action to occur. Liquid from the coolant bleed holdup tanks is pumped to the Coolant Treatment System for processing. This is schematically shown in Figure 9-21 and Figure 9-18. Component data is shown in Table 9-10.

The quench tank, located inside the Reactor Building, condenses and contains effluent from the pressurizer safety valves and various vents. Liquid in the quench tank can be circulated through a cooler
2 for temperature control, sampled and the excess liquid pumped to the Letdown Storage Tank, coolant bleed holdup tanks or the Liquid Waste Disposal System. This portion of the Coolant Storage System is shown schematically on Figure 9-20.

#### 9.3.3.2.4 System Isolation

The Low Pressure Injection System is connected to the reactor outlet line on the suction side and to the reactor vessel on the discharge side. The system is isolated from the Reactor Building on the suction side by two electric motor-operated valves located inside the Reactor Building and one electric motor-operated valve located outside the Reactor Building. The discharge side is isolated from the Reactor Building by a check valve inside and an electric motor-operated valve outside the Reactor Building. All of these valves are normally closed whenever the reactor is in the operating condition. In the event of a loss-of-coolant accident, the valve on the discharge side opens, but the valves between the reactor vessel and the suction side of the pumps remain closed throughout the accident.

#### 9.3.3.2.5 Leakage Considerations

During reactor power operation, all equipment of the Low Pressure Injection System is idle, and all isolation valves are closed. Under loss-of-coolant accident conditions, fission products may be recirculated in the coolant through the exterior piping system. Potential leaks have been evaluated to obtain the total radiation dose to the public due to leakage from this system. The evaluation is discussed in Chapter 12, "Radiation Protection" on page 12-1.

#### 9.3.3.2.6 Operational Limits

Alarms or interlocks are provided to limit variables or conditions of operation that might affect system or station safety. These variables or conditions of operation are as follows:

Decay Heat Removal Flow Rate

Low flow from the pumps during the decay heat removal mode of operation is alarmed to signify a reduction or stoppage of flow and cooling to the core.

Reactor Coolant Pressure Interlock

The first valve from the Reactor Coolant System in the suction line to the low pressure injection pumps is interlocked with the Reactor Coolant System pressure instrumentation to prevent inadvertent overpressurization of the Low Pressure Injection System piping while the Reactor Coolant System is still above Low Pressure Injection System design pressure.

Reactor Coolant Leaving Decay Heat Removal Coolers

High temperature of the reactor coolant discharging from the decay heat removal coolers is alarmed to signal a loss of cooling capability in the respective cooler.

4 #### 9.3.3.2.7 Failure Considerations

4 The effects of failure and malfunctions in the Low Pressure Injection System concurrent with a
4 loss-of-coolant accident are presented in Chapter 6, "Engineered Safeguards" on page 6-1, Section
4 9.3.3.2.4, "System Isolation." Redundant safety features are provided where required.

4 For pipe failures in the Low Pressure Injection System, the consequences depend upon the location of the
4 rupture. If the rupture were to occur between the first check valve upstream of the core flood nozzle and
4 the vessel, this would lead to a loss-of-coolant accident. The analysis of this loss-of-coolant accident is
4 included in Chapter 15, "Accident Analyses" on page 15-1. Section 15.14.4.3, "Small Break LOCA" on
4 page 15-61 addressed this failure as one of the limiting small break. Reference ECCS Analysis of B&W
4 177 FA LOWERED-LOOP NSS Rev. 3 (BAW-10103A, Rev. 3 Topical Report July 1977).

Borated Water Storage Tank

3 The borated water storage tank is located outside the Reactor Building and the Auxiliary Building. It
3 contains borated water with boron concentration maintained in accordance with the Core Operating
3 Limits Report. It is used for filling the fuel transfer canal during refueling and for filling the incore
3 instrumentation handling tank. The borated water storage tank also provides borated water for emergency core cooling and the Reactor Building Spray System. Liquid level in the borated water storage tank is monitored by redundant level instrumentation.

#### 9.3.3.2.1 Mode of Operation

2 Two pumps and two coolers normally perform the decay heat removal function for each unit. The steam
2 generators reduce the reactor coolant temperature to approximately 250°F and pressure to approximately
4 300 psig. These conditions represent upper limits for starting an LPI pump so as to avoid exceeding
2 system design limits. For Oconee Units 1 and 2, when these temperatures and pressures are reached,
2 decay heat removal will be initiated by aligning the system in one of two possible "switchover"
2 configurations. The first (preferred) path aligns A and C pumps to RCS through newly installed high
2 pressure piping. With either the A or C pump operating, fluid is returned to the RCS through the "A"
2 train of LPI. The second (alternate) path aligns the B cooler to the RCS and the outlet of the cooler is
2 routed to the suction of the A and C pumps. In this alignment, the pump in service will return fluid to
2 the RCS through the "B" train of LPI. After the RCS pressure has been reduced to approximately 125
2 psig, the system is aligned so that two pumps take suction from the reactor outlet line and discharge
2 through two coolers.

4 For Oconee 3 decay heat cooling is initiated at 290 psig/250°F by aligning pumps to take suction from the reactor outlet line and discharge through the coolers into the reactor vessel. The equipment utilized for decay heat cooling is also used for low pressure injection during accident conditions.

During refueling, the decay heat from the reactor core is rejected to the low pressure injection coolers in the same manner as it is during cooldown to 140°F. At the beginning of the refueling period, both coolers and both pumps are required to maintain 140°F in the core and fuel transfer canal. Later, as core decay heat decreases, one cooler and pump can maintain the required 140°F.

The fuel transfer canal may be filled by switching the suction of the decay heat removal pumps from the reactor outlet to the borated water storage tank. When the transfer canal is filled, suction to the pumps is switched back to the reactor outlet pipe. (Normally filled with the spent fuel cooling pumps as described in Section 9.1.3, "Spent Fuel Cooling System" on page 9-13.)

After refueling, the transfer canal is drained by switching the discharge of one of the pumps from the reactor injection nozzle to the borated water storage tank. The other pump will continue the recirculation mode of decay heat removal.

#### 9.3.3.2.2 Reliability Considerations

Since the equipment is designed to perform both normal and emergency functions, separate and redundant flow paths and equipment are provided to prevent a single component failure from reducing the system performance below a safe level. All rotating equipment and most valves are located in the Auxiliary Building to facilitate maintenance and periodic operational testing and inspection.

#### 9.3.3.2.3 Codes and Standards

Each component of this system will be designed to the code or standard, as applicable, as noted in Table 9-9.

3 b. Interlocks on the regulating control rod bank automatically terminates the dilution cycle regardless
3 of the mode of operation the controller is in, automatic or manual, if the regulating rod group
3 (Group 6) is inserted into the core beyond 25 percent.

3 c. The operator may manually terminate the dilution cycle at any time.

## 9.3.3 LOW PRESSURE INJECTION SYSTEM

### 9.3.3.1 Design Bases

The Low Pressure Injection System removes decay heat from the core and sensible heat from the Reactor
2 Coolant System during the latter stages of cooldown. It provides the means for filling and draining the
2 fuel transfer canal. The system maintains the reactor coolant temperature during refueling and reduced
2 inventory operation. The LPI and support system(s), selected components of the RCS and HPI are
2 dedicated to prevention and mitigation of loss of Decay Heat Removal events. (See Section 16.5.3 in the
2 Selected Licensee Commitments Manual.)

2 In the event of a loss-of-coolant accident, the system injects borated water into the reactor vessel for longterm emergency cooling. The emergency functions of this system are described in Chapter 6, "Engineered Safeguards" on page 6-1. Performance data is listed in Table 9-8.

### 9.3.3.2 System Description and Evaluation

The Low Pressure Injection System is shown schematically in Figure 9-19. An independent system is provided for each unit. The Low Pressure Injection System normally takes suction from the reactor coolant outlet line and delivers the water back to the reactor through the core flooding nozzles after passing through the low pressure injection pumps and coolers. The Low Pressure Injection System may be lined up when the reactor pressure is below the system suction piping design pressure for cooldown of the system to refueling temperatures. The decay heat is transferred to the Low Pressure Service Water System by the decay heat removal coolers. Component data are shown in Table 9-9.

The major system components are described as follows:

Decay Heat Removal Pumps

Three decay heat removal pumps are arranged in parallel with electric motor operated valves in the suction line to each pump. The two outboard pumps are normally available for emergency operation, and the center pump is valved off on both the suction and discharge sides of the pump. During decay heat removal, any two of the three pumps are lined up to the decay heat removal coolers.

The design flow is that required to cool the Reactor Coolant System from 250°F to 140°F in 14 hours. The steam generators are used to reduce the Reactor Coolant System from operating temperature to the 250°F temperature.

Decay Heat Removal Coolers

The decay heat removal coolers, during a routine shutdown, remove the decay heat from the circulated reactor coolant. Both coolers are designed to cool the circulated reactor coolant from 250°F to 140°F in 14 hours.

penetrating the Reactor Building. These lines each contain a check valve on the inside and on the outside for Reactor Building isolation.

#### 9.3.2.2.5 Leakage Considerations

Design and installation of the components and piping in the High Pressure Injection System considers the radioactive service of this system. Except where flanged connections have been installed for ease of
2 maintenance, the system is an all-welded system.

#### 9.3.2.2.6 Failure Considerations

The effects of failure and malfunctions in the High Pressure Injection System concurrent with a loss-of-coolant accident are presented in Chapter 6, "Engineered Safeguards" on page 6-1. Section 9.3.2.2.4, "System Isolation" on page 9-42. These analyses show that redundant safety features are provided where required.

For pipe failures in the High Pressure Injection System, the consequences depend upon the location of the rupture. If the rupture were to occur between the reactor coolant loop and the first isolation valve or check valve, it would lead to an uncontrolled loss-of-coolant from the Reactor Coolant System. The analysis of this loss-of-coolant accident is included in Chapter 15, "Accident Analyses" on page 15-1. If the rupture were to occur beyond the first isolation valve or outside the Reactor Building, the release of radioactivity would be limited by the small line sizes and by closing of the isolation or check valve.

A single failure will not prevent boration when desired for reactivity control, since several alternate paths are available for adding boron to the Reactor Coolant System. These are: (a) through the normal makeup lines, (b) through the reactor coolant pump seals, and (c) through the emergency injection lines. If pump suction is unavailable from the letdown storage tank, a source of borated water is available from the borated water storage tank during reactor power operation.

#### 9.3.2.2.7 Operational Limits

Alarms or interlocks are provided to limit variables or conditions of operation that could cause system upsets. The variables or conditions of operation that are limited are as follows:

1. Letdown Storage Tank Level

   Low water level in the letdown storage tank is alarmed and interlocked to the three-way bleed valve. Low water level will switch the three way valve from the bleed position to is normal position.

2. Letdown Line Temperature

   A high letdown temperature in the letdown line downstream of the letdown coolers is alarmed and interlocked to close the pneumatic letdown isolation valve, thus protecting the purification demineralizer resins.

3. Dilution Control

   The dilution cycle is initiated by the operator. Several safeguards are incorporated into the design to prevent inadvertent excessive dilution of the reactor coolant.

   3 a. The dilution valves have an automatic feature such that the operator may preset the desired
   3 quantity of dilution volume before initiating the dilution cycle. The dilution cycle will terminate
   3 when flow has integrated to the desired batch size. This interlock may be manually bypassed.
   3 Operation in the automatic mode is the preferred method of dilution.

System control is accomplished remotely from the control room with the exception of the reactor coolant pump seal return cooling. The letdown flow rate is set by remotely positioning the letdown flow control valve to pass the desired flow rate. The spare purification demineralizer can be placed in service by remote positioning of the demineralizer isolation valves. The letdown flow to the Coolant Storage System is diverted by remote positioning of the three-way valve and the valves in the Coolant Storage System. The reactor coolant volume control valve is automatically controlled by the pressurizer level controller.

A continuous cooling flow is maintained through the HPI nozzle warming lines. Flow is monitored via the Operator Aid Computer with signals from a flow transmitter on each warming line.

Auxiliary pressurizer spray is remote manually controlled from the control room. No means exists for directly monitoring auxiliary pressurizer spray flow. Instead, pressurizer level is utilized for process monitoring of auxiliary pressurizer spray.

For emergency operation as a High Pressure Injection System, the normal letdown coolant flow line and the normal pump seal return line are closed, and additional makeup flow is supplied through the high pressure injection emergency lines. The pumps and pump motors are designed to be able to operate at the higher flow rates and lower discharge pressures associated with emergency high pressure injection requirements. Emergency operation of this system is described in Chapter 6, "Engineered Safeguards" on page 6-1.

#### 9.3.2.2.2 Reliability Considerations

This system provides essential functions for the normal operation of the unit. Redundant components and alternate flow paths have been provided to improve system reliability.

Each unit has three high-pressure injection pumps, each capable of supplying the required reactor coolant pump seal and makeup flow. One is normally in operation while another is in standby status to be used as needed. The third pump is used only for emergency injection. There are two letdown coolers and two seal return coolers. One cooler in each group will perform the required duty while the other may be used as a spare.

One of the two letdown filters or reactor coolant pump seal filters in normally in use while the other is a spare.

#### 9.3.2.2.3 Codes and Standards

Each component of this system will be designed to the code or standard, as applicable, noted in Table 9-7.

#### 9.3.2.2.4 System Isolation

The letdown line and reactor coolant pump seal return line are outflow lines which penetrate the Reactor Building. Both lines contain electric motor-operated isolation valves inside the Reactor Building and pneumatic valves outside which are automatically closed by an engineered safeguards signal. The injection line to the reactor coolant pump seals is an inflow line penetrating the Reactor Building. This line contains a stop-check valve inside the Reactor Building and a remotely operated valve on the outside of the Reactor Building. Check valves in the discharge of each high pressure injection pump provide further backup for Reactor Building isolation. The two emergency coolant injection lines are used for injecting coolant to the reactor vessel after a loss-of-coolant accident. After use of the lines for emergency injection is discontinued the electric motor-operated isolation valves in each line outside the Reactor Building may be closed for isolation. The HPI nozzle warming lines and auxiliary pressurizer spray lines are inflow lines

Reactor Coolant Pump Seal Return Coolers

The seal return coolers are sized to remove the heat added by the high pressure injection pumps and the heat picked up in passage through the reactor coolant pump seals. Heat from these coolers is rejected to the Recirculated Cooling Water System. Two coolers are provided and one is normally in operation.

Letdown Storage Tank

The letdown storage tank serves as a receiver for letdown, seal return, chemical addition, and system makeup. The tank also accommodates temporary changes in system coolant volume.

#### 9.3.2.2.1 Mode of Operation

During normal operation of the Reactor Coolant System, one high pressure injection pump continuously supplies high pressure water from the letdown storage tank to the seals of each of the reactor coolant pumps and to a makeup line connection to one of the reactor inlet lines. Makeup flow to the Reactor Coolant System is regulated by a flow control valve, which operates on signals from the pressurizer level controller.

A control valve in the common injection line to the pump seals automatically maintains the desired total injection flow to the seals. Manual Throttle valves in each pump seal injection line provide a capability to balance the seal injection flow rates. A portion of the water supplied to the seals enters the Reactor Coolant System. The remainder returns to the letdown storage tank after passing through one of the two reactor coolant pump seal return coolers.

On Oconee 1 only, when the leakage rate past the No. 1 face seal on any operating reactor coolant pump is less than 1 gal/min, the isolation valve in the seal bypass line is opened allowing flow of injection water past the lower radial pump bearing for cooling and lubrication. Provision is also made for filling the No. 3 (vapor) seal standpipe from the No. 1 seal water return line to the Letdown Storage Tank for Oconee 1 only.

Seal water inleakage to the Reactor Coolant system requires a continuous letdown of reactor coolant to maintain the desired pressurizer level. Letdown is also required for removal of impurities and boric acid from the reactor coolant. The letdown is cooled by one of the letdown coolers, reduced in pressure by the letdown orifice, and then passed through the purification demineralizer to a three-way valve which directs the coolant to the letdown storage tank or to the Coolant Storage System.

Normally, the three-way valve is positioned to direct the letdown flow to the letdown storage tank. If the boric acid concentration in the reactor coolant is to be reduced, the three-way valve is positioned to divert the letdown flow to the Coolant Storage System. Boric acid is removed by directing the letdown flow through a deborating demineralizer with the effluent returned directly to the letdown storage tank, or by the feed and bleed method. Feed and bleed is the process of directing the letdown flow to a coolant bleed holdup tank and maintaining the level in the letdown storage tank with demineralized water pumped from a supply of unborated water. The flow of demineralized water is measured and totaled by inline flow instrumentation. The flow of demineralized or borated water returning to the letdown storage tank is controlled remotely by the makeup control valve. During normal operation the inline instrumentation or the control rod drive interlock will terminate makeup flow.

The letdown storage tank also receives chemicals for addition to the reactor coolant. A hydrogen overpressure is maintained in the tank to assure a slight amount of excess hydrogen in the circulating reactor coolant. Other chemicals are injected in solution into the tank.

### 9.3.2.2 System Description and Evaluation

The High Pressure Injection System is shown schematically on Figure 9-17 and Figure 9-18. Table 9-6 and Table 9-7 list the system Performance requirements and data for individual components. The following is a brief functional description of system components:

Letdown Cooler

The letdown cooler reduces the temperature of the letdown flow from the Reactor Coolant System to a temperature suitable for demineralization and injection to the reactor coolant pump seals. Heat in the letdown coolers is rejected to the Component Cooling System.

Letdown Flow Control

The letdown flow rate at reactor operating pressure is limited by a fixed block orifice. At reduced pressure a parallel, normally closed, remotely operated valve can be opened to maintain the desired flow rate. In addition there is a second parallel, normally closed valve which may be manually positioned for flow control.

3

Purification Demineralizer

The letdown flow is passed through the purification demineralizer to remove reactor coolant impurities other than boron. The design purification letdown flow is equal to one Reactor Coolant System volume in 24 hours. One demineralizer is provided for each unit. In addition, a spare demineralizer is shared between Oconee 1 and 2, and another spare is installed for Oconee 3. The spare demineralizer may be used to remove lithium from the reactor coolant system to maintain system chemistry and/or used to remove cesium from the reactor coolant system in the event of fuel defects. Chapter 11, "Radioactive Waste Management" on page 11-1 describes coolant activities, coolant handling and storage, and expected limits on activity discharge.

Letdown Filters

Two letdown filters in parallel are provided to prevent particulates from entering the Reactor Coolant System and subsequently the pump seal filters. One filter is normally in use.

High Pressure Injection Pumps

The high pressure injection pumps are designed to return coolant which is letdown for purification to the Reactor Coolant System, and to supply the seal water to the reactor coolant pumps. The pumps are sized to permit one pump to provide normal operating makeup and seal water flow.

Reactor Coolant Pump Seal Injection Filters

Two reactor coolant pump seal filters are provided to prevent particulates from entering the pump seals. One is normally in use.

Seal Return filter

A single filter is installed in the seal return line upstream of the seal return coolers to remove particulate matter. A bypass is installed to permit servicing during operation.

#### 9.3.1.2.7 Operational Limits

The Chemical Addition and Sampling System provides certain chemicals to several systems in proper quantities and concentrations and provides a capability to sample fluids in various systems. The limits that must be maintained on these operations are described below.

The boric acid mix tank solution is to be maintained above an average temperature of 105°F in order to maintain boric acid in solution at a concentration of 7 percent by weight. The capacity of the boric acid mix tank is 500 cubic feet. Approximately 216 cubic feet is required to borate the Reactor Coolant System for cold shutdown near the end of core life.

## 9.3.2 HIGH PRESSURE INJECTION SYSTEM

### 9.3.2.1 Design Bases

The High Pressure Injection System is designed to accommodate the following function during normal reactor operation:

1. Supply the Reactor Coolant System with fill and operational makeup water.
2. Provide seal injection water for the reactor coolant pumps.
3. Provide for purification of the reactor coolant to remove corrosion and fission products.
4. Control the boric acid concentration in the reactor coolant.
5. In conjunction with the pressurizer, the system will accommodate temporary changes in reactor coolant volume due to small temperature changes.
6. Maintain the proper concentration of hydrogen and corrosion inhibiting chemicals in the Reactor Coolant System.
7. Provides continuous flow for cooling the normal HPI nozzles (see FSAR Section 5.4.7.2, "High Pressure Injection" on page 5-61) to minimize thermal shock.
8. Provides auxiliary pressurizer spray control for cooldown when normal pressurizer spray is unavailable.

The specific design bases for various parts of the system are as follows:

Letdown Capability

The system will accommodate letdown required as a result of coolant volume expansion when heating the reactor coolant to operating temperature at a rate of 100°F/h while maintaining constant pressurizer level. The letdown is cooled before leaving the Reactor Building.

Purification

Filters and demineralizers are provided to remove reactor coolant impurities. The letdown filters and purification demineralizers are sized for full flow through the letdown orifice.

Makeup

The system will accommodate makeup requirements during design reactor coolant system transients and for Reactor Coolant System cooldown at the design rate.

b. High Activity Waste Tank.

c. RC Bleed Holdup Tanks.

d. RC Bleed Holdup Tanks (Demineralized Water).

e. Waste Gas Vent Header.

f. RC Bleed Evaporator. (out of service)

g. Miscellaneous Waste Evaporator. (out of service)

h. Waste Gas Decay Tanks.

0 i. Containment ($H_2$ Analyzer)

2. Sample Containers (to be analyzed for a variety of substances).

a. Letdown Storage Tank Gas Space.

b. Letdown Line Upstream of Purification Demineralizer.

c. In addition, all gas analyzer samples can be containerized for further analysis.

#### 9.3.1.2.2 Reliability Considerations

The Chemical Addition and Sampling System is not required to function during an emergency condition. Redundant boric acid pumps and flow paths are provided to guard against a single component failure rendering the system inadequate for boron addition. In addition to the boric acid mix tank, boric acid is also available for boration in 5-percent by weight solution from the concentrated boric acid storage tank. To prevent precipitation, heating/heat tracing is installed on components and lines used to transfer concentrated boric acid. The pumps, tanks, coolers, and instrumentation are located in the Auxiliary Building and are accessible for inspection and maintenance.

#### 9.3.1.2.3 Codes and Standards

The components of the Chemical Addition and Sampling System are designed to the codes and standards noted in Table 9-5.

#### 9.3.1.2.4 System Isolation

0 The pressurizer sample line, core flood sample line, and both steam generator sample lines are the only
0 system lines that penetrate the Reactor Building. All these lines contain electric motor-operated isolation
0 valves inside the Reactor Building and pneumatic valves outside, which are automatically closed by an
0 engineered safeguards signal (except for the core flood sample line which has a manual isolation valve).

#### 9.3.1.2.5 Leakage Considerations

Leakage of radioactive reactor coolant from this system within the Reactor Building will be collected in the Reactor Building normal sump. Leakage of radioactive gases from this system outside the Reactor Building is collected by placing the sampling stations under hoods exhausting to the unit vent.

#### 9.3.1.2.6 Failure Considerations

Since the system serves no engineered safeguards function, the only consideration following a loss-of-coolant accident is the operation of the isolation valves. Redundant isolation valves are provided to assure isolation of the Reactor Building as described in Section 9.3.1.2.4, "System Isolation."

acid reaction. The sampling portion of this system is used to take samples to assure that water qualities and boric acid concentrations are maintained. Sampling locations and the samples taken at each location are as follows:

Liquids

1. Steam Generator Sample Sink.
   a. Secondary Side of Steam Generator.
2. Reactor Coolant Sample Sink.
   a. Pressurizer Water Space.
   b. Pressurizer Steam Space.
   c. Low Pressurizer Injection Cooler Outlet.
   d. Core Flooding Tanks.
   e. Total Gas Sample
   f. Reactor Coolant
3. Auxiliary Systems Sample Sink.
   a. Purification Demineralizer Inlet and Outlet.
   b. Deborating Demineralizer Outlet.
   c. Letdown Storage Tank Water Space.
   d. RC Bleed Evaporator Feed Pump Discharge.
   e. Deborating Demineralizer Outlet (Regeneration).
   f. Waste Evaporator Feed Pump Discharge.
   g. RC Bleed Evaporator (Concentrate).
   h. Concentrated Boric Acid Transfer Pump Discharge.
   i. RC Bleed Evaporator (Distillate).
   j. Waste Evaporator (Concentrate).
   k. RC Bleed Transfer Pump Discharge.
   l. Waste Transfer Pump Discharge.
   m. High Activity Waste Transfer Pump Discharge.
   n. Low Activity Waste Transfer Pump Discharge.
   o. Condensate Test Tank Pump Discharge.
   p. RC Bleed Evaporator Demineralizer Outlet.
   q. Reactor Building Normal Sump
   r. Waste Evaporator (distillate)

Gaseous

1. Hydrogen and Oxygen Gas Analyzer.
   a. Miscellaneous Waste Holdup Tank.

Caustic Mix Tank

Caustic addition serves the primary purpose of adding sodium hydroxide to the LPI system following a LOCA to minimize the zinc-boric acid reaction. Previously, this system was used to control the pH in the RC bleed and miscellaneous waste evaporators and to regenerate the resins in the deborating demineralizers, but it is no longer used in this capacity. A single caustic mix tank is provided for Units 1 and 2, and one tank is provided for Unit 3. These tanks would be used only as an alternative to the caustic bulk storage containers.

Caustic Pump

The caustic pump transfers sodium hydroxide from caustic bulk storage containers or the caustic mix tank to the LPI system. A single pump is provided for Units 1 and 2 and one is provided for Unit 3.

Lithium Hydroxide Tank

Lithium hydroxide is mixed and added to the Reactor Coolant System for pH control from the lithium hydroxide tank. A single tank is provided for Units 1 and 2, and one tank is provided for Unit 3.

Lithium Hydroxide Pump

The lithium hydroxide pump transfers lithium hydroxide from the LiOH tank to the letdown line upstream of the letdown filters. A single pump is provided for Units 1 and 2, and one pump is provided for Unit 3.

Hydrazine Pumps

The hydrazine pump transfers hydrazine to the letdown line upstream of the letdown filters. The hydrazine pump, after sufficient demineralized water flushes, is also used to transfer hydrogen peroxide. A single pump is provided for units 1&2, and one pump is provided for unit 3. These pumps can also be used as a backup to the Lithium Hydroxide pump or to add other chemicals (as needed) to the RCS.

A separate hydrazine pump transfers hydrazine from a small container backwards through the pressurizer water space sample line to the pressurizer. Each unit has its own separate pump.

Pressurizer Sample Cooler

This cooler cools the effluent sample taken from the pressurizer steam or water space. One cooler is provided per unit.

Steam Generator Sample Cooler

This cooler cools the effluent sample taken from the secondary side of the steam generator. Two coolers are provided per unit.

#### 9.3.1.2.1 Mode of Operation

The chemical addition portion of this system delivers the necessary chemicals to other systems as required. Boric acid is provided to the spent fuel pool, borated water storage tank, letdown storage tank, and core flooding tanks as makeup for leakage or to change the concentration of boric acid in the associated systems. Sodium hydroxide is added to the LPI system following a LOCA to minimize the zinc-boric

# 9.3 PROCESS AUXILIARIES

## 9.3.1 CHEMICAL ADDITION AND SAMPLING SYSTEM

### 9.3.1.1 Design Bases

Chemical addition and sampling operations are required to change and monitor the concentration of various chemicals in the Reactor Coolant System and Auxiliary Systems. The Chemical Addition and Sampling System is designed to add boric acid to the Reactor Coolant System for reactivity control, lithium hydroxide for pH control, and hydrazine for oxygen control. The Chemical Addition and Sampling System can also be used for hydrogen peroxide additions to induce 'crud' bursts during unit shutdowns to enhance corrosion product removal and, therefore, reduce equipment/system/component dose rates. Following a LOCA, the chemical addition and sampling system can be used to add sodium hydroxide (caustic) to the reactor coolant system for pH adjustment.

### 9.3.1.2 System Description and Evaluation

The Chemical Addition and Sampling System is shown schematically on Figure 9-15 and Figure 9-16. The Sampling System has separate sampling stations for reactor coolant and steam generator sampling for each of the three units. Two auxiliary systems sampling stations are provided, one for Oconee 1 and 2 and one for Oconee 3.

Two chemical addition systems are also provided, one for Oconee 1 and 2 and one for Oconee 3. These systems permit chemical addition to and sampling of the Reactor Coolant System and other Reactor Auxiliary Systems during normal reactor operation.

4 The Chemical Addition and Sampling System performs no emergency functions (Refer to Section 9.3.6,
4 "Post-Accident Sampling System" on page 9-48 for information on Post-Accident Sampling System). Guidelines for maintaining feedwater and reactor coolant quality are derived from vendor recommendations and the current revisions of the EPRI PWR Secondary and Primary Water Chemistry Guidelines, respectively. Detailed operating specifications for the chemistry of these systems are addressed in the Chemistry Section Manual. A brief functional description of the major system components follows.

Boric Acid Mix Tank

Two boric acid mix tanks are provided as a source of concentrated boric acid solution. The volume of the tanks provides sufficient boric acid solution to increase the reactor coolant system boron concentration to that required for cold shutdown. Tank heaters and electrically heat traced transfer lines maintain the fluid temperature above that required to assure solubility of the boric acid.

Boric Acid Pumps

Six boric acid pumps, three shared between units 1 and 2, and three for Unit 3, are provided to transfer the concentrated boric acid solution from the boric acid tank to the borated water storage tank, letdown storage tanks, spent fuel storage pool, or the core flood tanks. Two pumps, each with a 1 gal/min capacity, supply boric acid to the core flood tanks. The other four pumps, which each have 10 gal/min capacities, supply boric acid to other tanks, systems, and locations (Figure 9-15 and Figure 9-16).

## 9.2.5 REFERENCES

5 1. Safety Evaluation Report for Oconee Units 2 and 3, dated July 6, 1973.

5 2. Letter from J. W. Hampton (Duke) to USNRC Document Control Desk, dated May 31, 1995,
5 Service Water Issues.

### 9.2.3.2 System Description

The Auxiliary Service Water System utilizes the plant CCW intake and discharge conduits as a source of raw cooling water for decay heat removal (Figure 9-14). These conduits are interconnected by crossovers and unwatering lines. An Auxiliary Service Water Pump located in the Auxiliary Building at Elev. 771 takes its suction from the Oconee 2 intake conduit and discharges into the steam generators of each unit
1 via separate lines into the emergency feedwater headers. The raw water is vaporized in the steam generator removing residual heat and dumped to the atmosphere.

The auxiliary service water pump is an end suction centrifugal pump with a rated capacity of 3000 gal/min at a total head of 176 feet.

It has been submitted to the following tests:

1. A non-witness ASME hydro test
2. Witnessed performance test
3. Sonic testing of shaft
4. Mill test certificates for casing, impeller, and shaft
5. Certified caliper measurements

The pump power supply is taken from the 4160 volt Standby Bus No. 1.

All valves required for operation of the Auxiliary Service Water System are either check valves or manually operated. The pump suction is equipped with a manually operated butterfly valve and the discharge with a check valve and manually operated gate valve. The pump bypass is equipped with a globe valve. The individual lines to each steam generator auxiliary feedwater header are equipped with a check valve and two normally closed gate valves which are used to control flow. All non-embedded piping is Duke Class F.

Atmospheric steam dumps on each main steam line are equipped with one normally closed gate valve and one normally closed control valve which must be opened to reduce steam generator shell side pressure before placing the Auxiliary Service Water System into operation.

## 9.2.4 ULTIMATE HEAT SINK

The Condenser Circulating Water System is the ultimate heat sink for Oconee Nuclear Station. This system is described in Section 9.2.2.2.1, "Condenser Circulating Water System (CCW)" on page 9-27.

The LPSW returned from the Auxiliary Building is monitored for radioactivity. Upon any indication of radioactivity in the effluent, the component suspected of leaking may be individually isolated.

The LPSW pumps are connected to the 4160 volt buses which supply power to engineered safeguards equipment. The emergency power supply is adequate to operate all LPSW pumps upon a loss of off-site power.

During normal operation, the cooling requirements are supplied by operating one LPSW pump per unit.
1 The LPSW requirement following a loss of coolant accident can also be supplied by one pump per unit. The spare pump is started by the engineered safeguards actuation signal to provide redundancy for single failure criteria.

#### 9.2.2.2.4 Recirculated Cooling Water System (RCW)

The RCW system for the Oconee station is shown schematically in Figure 9-13. This system provides inhibited closed cycle cooling water to various components outside the Reactor Building including:

1. RC pump seal return coolers
2. Spent fuel cooling
3. Sample coolers
4. Evaporator systems
5. Various pumps and coolers in the Turbine Building

The RCW system consists of two parallel loops which are normally isolated from each other. One loop supplies cooling for shared station loads, Unit 1 and 2 loads and secondary loads on Unit 3. It consists of four motor-driven pumps and four RCW heat exchangers. A 25,000 gallon surge tank provides a surge volume to accommodate temperature changes and leakage. Condenser circulating water is used to cool the RCW heat exchangers. The other loop supplies cooling for Unit 3 primary loads. It consists of two motor-driven pumps and two RCW heat exchangers. It contains a 7,700 gallon surge tank and also utilizes condenser circulating water to cool the RCW heat exchangers. RCW effluent from the Auxiliary Building is monitored for radioactivity. Leakage of radioactive fluids from any of the coolers in the Auxiliary Building will be indicated by these monitors. Separate monitors are provided on the return lines from the Oconee 1 and 2 Auxiliary Building and the Oconee 3 Auxiliary Building.

The number of RCW pumps and RCW heat exchangers in operation varies depending on the spent fuel heat load and lake water temperature. The isolation valves, which normally separate the two parallel loops, can be opened, however; it is not a necessary configuration.

The RCW provides no engineered safeguards functions and does not penetrate the Reactor Building.

## 9.2.3 AUXILIARY SERVICE WATER SYSTEM

### 9.2.3.1 Design Basis

The Auxiliary Service Water System is designed for decay heat removal following a concurrent loss of the
1 main feedwater system, Emergency Feedwater System, and Decay Heat Removal System. The system will maintain decay heat removal for a minimum of 37 days.

For Oconee 3, each of the two 15,000 gal/min LPSW pumps take their suction from the CCW crossover. These pumps provide cooling water via separate supply lines to engineered safeguards equipment in the Reactor Building and the Auxiliary Building similar to Oconee 1 and 2. The return lines from the Oconee
1 3 engineered safeguards maintain separation to a point beyond a remote-operated isolation valve.

The Turbine Building requirements for LPSW are supplied from other separate headers. The three pumps associated with Oconee 1 and 2 have a Turbine Building header serving the Turbine Building requirements for Oconee 1 and 2. The two pumps associated with Oconee 3 also have a Turbine Building header to supply the Oconee 3 requirements.

The separate flow paths serving the emergency safeguards equipment can be isolated by remote-operated isolation valves.

The LPSW system is monitored and operated from the control room. Isolation valves are incorporated in all LPSW lines penetrating the Reactor Building.

The three (per unit) Reactor Building coolers ("A," "B," and "C") are supplied by individual lines from the separate LPSW supply headers. Each inlet line is provided with a motor operated shutoff valve located outside the Reactor Building. Similarly, each discharge line from the coolers is provided with a motor operated valve located outside the Reactor Building. This allows each cooler to be isolated
1 individually. During normal operation, the "A" and "C" coolers receive throttled flow while flow through the "B" cooler is diverted to the four Reactor Building auxiliary cooling units to provide normal Reactor Building cooling. Flow to the RB auxiliary cooling units is automatically isolated by an engineered safeguards signal returning full flow to the RB cooling units. Also, on an engineered safeguards signal the outlet valves on the three RB cooling units fully open automatically to assure emergency flow through coolers.

2 The LPSW System provides sufficient flow to the Low Pressure Injection (LPI) coolers and Reactor
2 Building Cooling Units (RBCUs) to ensure sufficient heat transfer capability following a design basis
2 accident and a single active failure. The worst case design basis accident involves a LOCA/loss of offsite
3 power with a loss of instrument air. The worst case single failures for achieving desired flows to the
2 RBCUs and LPI coolers are 1) failure of a single LPSW pump, and 2) failure of a 4160 volt bus which
2 fails an LPSW pump, an RBCU fan, and an LPI cooler isolation valve. Analysis and testing have been
2 performed to demonstrate system performance under worst case conditions.

2 The LPSW System can provide sufficient flow to the required loads following a seismic event. Valves
2 LPSW-139 (Units 1 and 2) and 3LPSW-45 (Unit 3) are remotely-operated, seismically-qualified valves
2 which can isolate the non-seismic, non-essential header from the safety-related portions of the system.
2 Other non-seismic connections to the system exist which cannot be remotely isolated. Analyses have
2 demonstrated that, given a simultaneous failure of all non-seismic connections that cannot be remotely
2 isolated, the LPSW System can provide sufficient flow to the required loads.

4 LPSW flow to the LPI coolers is normally throttled using air-operated valves 1-,2-LPSW-251,
4 1-,2-LPSW-252 and 3LPSW-404, 3LPSW-405. During a design basis accident involving a loss of
2 instrument air, these valves fail open to their travel stops. Motor-operated valves LPSW-4 and LPSW-5
2 will be used to throttle LPSW flow to the LPI coolers under these conditions. Travel stops are in place
4 on 1-,2-LPSW-251, 1-,2-LPSW-252, 3LPSW-404, and 3LPSW-405 to ensure LPSW flow through an LPI
4 cooler does not exceed the design limit of 7500 gpm under worst case conditions.

2 The LPSW flow to and from each Reactor Building cooler is measured. Provisions are available to
2 indicate cooler leakage.

5 conservative assumptions than those used in the original design. Based on results obtained, the AEC staff
5 concluded that a rapid drawdown of Lake Keowee could cause considerable displacement of the riprap
5 used to face the weir. However, the AEC staff did not require Duke Power to redesign the weir, since the
5 water trapped in the condenser intake and discharge piping below elevation 791 ft. MSL is adequate to
5 supply the three Oconee units with steam generator boil off for safe shutdown for a period of 37 days
5 (Reference 1 on page 9-33). The Auxiliary Service Water (ASW) System is capable of using the
5 inventory trapped in the CCW piping for decay heat removal (Reference 9.2.3, "Auxiliary Service Water
5 System" on page 9-31). Therefore, the licensing basis does not rely on the weir nor recirculation of the
5 intake canal for decay heat removal after a loss of Lake Keowee event (Reference 2 on page 9-33).

#### 9.2.2.2.2 High Pressure Service Water System (HPSW)

The schematic arrangement of the HPSW system is shown on Figure 9-10. This system is used primarily for fire protection throughout the Oconee station. In the event of a loss of the normal LPSW supply, HPSW automatically supplies cooling water to the HPI pump motor coolers. For loss of AC power, HPSW via the elevated water storage tank automatically supplies cooling water to the turbine driven
4 emergency feedwater pump and its associated oil cooler, and maintains CCW pump bearing cooling water
4 and cooling water for the CCW pump motors. Refer to Sections 16.9.7 and 16.9.8 for specific
4 requirements to support the LPSW System.

Two full size (6000 gal/min at 117 psig) and one reduced size (500 gal/min at 117 psig) high pressure service water pumps supply the high pressure system. A 100,000 gallon elevated water storage tank provides inventory for a backup supply of water.

The 500 gal/min pump will normally operate to keep pressure on the fire headers. In the event of a fire, one full size pump provides adequate capacity for automatically maintaining the elevated water storage tank inventory. The second full size pump is an installed spare. The HPSW pumps take suction from the CCW system. The HPSW and LPSW pump suctions are connected to the 42 inch cross-connection between the condenser cooling water inlet headers for the three units. Manual isolation valves are provided so that service water may be supplied from any or all of the inlet headers.

4

#### 9.2.2.2.3 Low Pressure Service Water System (LPSW)

The schematic arrangement of the LPSW system is shown on Figure 9-11 and Figure 9-12. Oconee 1 and 2 share three 15,000 gal/min LPSW pumps. The LPSW pumps and the HPSW pumps take suction from the 42 inch crossover line between the condenser inlet headers; two LPSW pumps are supplied by one suction line and the other pump is supplied by the other suction line. The HPSW system is
0 connected to LPSW at the LPSW pump discharge to provide emergency back-up.

2 Suction is provided to the LPSW pumps via gravity flow or siphon flow from the CCW System (ECCW
2 mode) following a design basis accident where the CCW pumps are not running. Lake level is
2 administratively controlled to maintain sufficient NPSH for the LPSW pumps under these conditions.

0 The LPSW system provides cooling for components in the Turbine Building, the Auxiliary Building, and
0 in the Reactor Building. Two separate 24 inch lines provide LPSW to the components in the Auxiliary and Reactor Buildings. These two supply lines are further divided into four separate supply headers, two supplying the components in Oconee 1 and two supplying the components in Oconee 2. The decay heat removal coolers and the Reactor Building cooling units are supplied by separate LPSW supply lines. The return lines from the decay heat removal coolers and the Reactor Building coolers maintain separation to a point beyond a remote-operated isolation valve.

5 The suction of the condenser circulating water pumps extends below the maximum drawdown of the lake. The intake structure is provided with screens which can be manually removed for periodic cleaning.

5 The CCW system is designed to take advantage of the siphon effect so the pumps are required only to
5 overcome pipe and condenser friction loss.

2 The CCW system has an emergency discharge line to the Keowee hydro tailrace. This discharge line is connected to each of the three condensers of each unit. Under a loss-of-power situation, the emergency discharge line will automatically open and the CCW system will continue to operate as an unassisted siphon system supplying sufficient water to the condenser for decay heat removal and emergency cooling
2 requirements. This siphon system is the Emergency Condenser Circulating Water (ECCW) System and
3 can be divided into two distinct parts. The "first siphon" takes suction from the CCW intake canal and
3 supplies flow to the CCW crossover header in the Turbine Building basement, where the LPSW System
3 takes its suction. The "second siphon" takes suction from the condenser inlet piping, supplies flow
3 through the condenser, and discharges to the Keowee Hydro tailrace. A loss of function of the second
3 siphon would not affect the capability of the first siphon to perform its function.

3 In a loss of off-site power (LOOP) situation, the CCW pumps will be tripped by a load shed command
3 from the Engineered Safeguards System. The ECCW System first siphon is required to supply suction to
3 the LPSW System until a CCW pump can be manually restarted by the control room operator. Gravity
3 flow (without relying on the siphon) to the suction of the LPSW pumps is possible if the lake level is
3 sufficiently above the bottom of the CCW intake piping to maintain the required NPSH and flow
4 demand. Refer to Section 16.9.7, Selected Licensee Commitments Manual, for additional requirements
4 regarding the CCW Supply to the LPSW System.

3 During a loss of all AC power situation (Station Blackout), the CCW System is not required to supply
3 suction to the LPSW System since power to the LPSW pumps would not be available. The second
4 siphon is the preferred method for decay heat removal but is not required. Decay heat removal can also
3 be accomplished by venting steam to the atmosphere using the main steam safety valves or the manual
4 atmospheric dump valves. The CCW piping has sufficient inventory to cope with a four-hour Station
4 Blackout by supplying suction to the SSF Auxiliary Service Water System. (Reference 8.3.2.2.4, "Station
4 Blackout Analysis" on page 8-25.)

3

3 During normal operation, the continuous vacuum priming system removes noncondensible gases from
5 portions of the CCW System. An emergency steam air ejector (ESAE) is available to enhance operation
5 of the second siphon if the vacuum priming pumps are lost due to a loss of power. Since vacuum system
4 connections to the CCW inlet piping are normally isolated, the vacuum system is not required for the first
4 siphon to function.

4 Pursuant to the recommendations of the Oconee Probabilistic Risk Assessment study a pushbutton has been installed in the control room for sending a close signal to the CCW pump discharge valves. The capability to close the CCW valves is needed to protect against the possibility of CCW siphoning into the turbine building basement, causing flooding.

5 The intake canal that supplies water from Lake Keowee to the suction of the CCW pumps contains a
5 submerged weir. The purpose of this weir is to provide an emergency pond of cooling water if the water
5 supply from Lake Keowee were lost. This emergency pond could be recirculated through the condensers
5 and back to the intake canal for decay heat removal as long as the intake canal level remains sufficient.
5 However, during the operating license review of Oconee Units 2 and 3, the Atomic Energy Commission
5 (AEC) staff requested a reanalysis of the capability of the weir to withstand hydraulic forces using more

## 9.2.2 COOLING WATER SYSTEMS

### 9.2.2.1 Design Bases

The cooling water systems for the station are designed to provide redundant cooling water supplies to insure continuous heat removal capability both during normal and accident conditions.

4 The Low Pressure Service Water (LPSW) and portions of the Condenser Circulating Water (CCW) systems are designed so no single component failure will curtail normal station operation or impair emergency safeguards operation. Redundant pumping capability is provided, heat exchangers and pumps can be isolated and pressure reducing valves are provided with bypasses.

All cooling systems are designed to be operated and monitored from the control room. Component design parameters are given in Table 9-4.

The design purpose of each of the cooling water systems is outlined below:

Condenser Circulating Water (CCW) System - This system provides for cooling of the condensers
2 during normal operation of the plant. The system also serves as the ultimate heat sink for decay heat
4 removal during cooldown of the plant. The CCW System is the suction source for other service water
4 systems, including HPSW, LPSW, ASW, and SSF ASW. In addition, CCW provides a heat sink for
4 the RCW system. Following a design basis event involving loss of the CCW pumps, the Emergency
2 Condenser Circulating Water (ECCW) System supplies suction to the LPSW pumps.

High Pressure Service Water (HPSW) System - This system provides a source of water for fire protection throughout the station. In the event of a loss of the normal LPSW supply, HPSW automatically supplies cooling water to the HPI pump motor coolers. For loss of A.C. power, HPSW via the Elevated Water Storage Tank automatically supplies cooling water to the Turbine
4 Driven Emergency Feedwater Pump and its associated Oil Cooler, and maintains CCW pump bearing
4 cooling water and cooling water for the CCW pump motors.

Low Pressure Service Water (LPSW) System - This system provide cooling water for normal and emergency services throughout the station. Safety related functions served by this system are:

1. Reactor Building cooling units.
2. Decay heat removal coolers.
3. High pressure injection pump motor bearing coolers.
0 4. Motor-Driven Emergency Feedwater Pump motor air coolers.
5. Turbine Driven Emergency Feedwater Pump cooling water jacket

4 Recirculated Cooling Water (RCW) System - This is a closed loop system to supply inhibited cooling water to various components. This system has no direct safety related functions.

### 9.2.2.2 System Description and Evaluation

#### 9.2.2.2.1 Condenser Circulating Water System (CCW)

The Little River arm of Lake Keowee is the source of water for the CCW systems. Figure 9-9 shows the arrangement of the systems with respect to the two branches of Lake Keowee. Each unit has four condenser circulating water pumps supplying water via two 11 ft. conduits into a common condenser intake header under the turbine building floor. The discharge from the condenser is returned to the Keowee River arm of Lake Keowee.

provides a reservoir of component cooling water until a leaking cooling line can be isolated. Makeup water and corrosion inhibiting chemicals are added to the system in the surge tank.

### 9.2.1.4 Reliability Considerations

The Component Cooling System performs no emergency functions. Redundancy in active components is provided to improve system reliability. The pumps, coolers, surge tank, and most of the instrumentation are located in the Auxiliary Building and are accessible for inspection and maintenance.

### 9.2.1.5 Codes and Standards

2 The components of the system are designed to the codes and standards given in Table 9-13.

### 9.2.1.6 System Isolation

Since the Component Cooling System is not an engineered safeguards system, Reactor Building isolation valves are automatically closed on a high Reactor Building pressure signal to provide building isolation. The Reactor Building inlet lines are isolated by two check valves, one on the outside and one on the inside of the Reactor Building. The Reactor Building outlet line is isolated by an electric motor-operated valve on the inside and by a pneumatic valve on the outside of the Reactor Building.

### 9.2.1.7 Leakage Considerations

Water leakage from piping, valves, and other equipment in the system is not considered to be detrimental since the cooling water is normally nonradioactive. Welded construction is used throughout the system to minimize the possibility of leakage except where flanged connections are required for servicing.

In-leakage of reactor coolant to the system is detected by a radiation monitor (RIA-50) located in the recirculation line from the pumps to the surge tank and is also indicated by an increase in surge tank level. A defective coil of a coolant pump can be remotely isolated by an electric motor-operated valve on the outlet cooling line and a stop-check valve on the inlet line. A letdown cooler leak can be remotely isolated with motor-operated valves on the reactor coolant side of the cooler. The cooling water side can be completely isolated by closing a remotely operated, motor-actuated valve on the inlet of the cooler and the manual valves on the outlet cooling lines. Leakage from the quench tank cooler can be isolated by manual valves on the reactor coolant side. The cooling water side can be completely isolated by two manual valves.

### 9.2.1.8 Failure Considerations

Since the system serves no engineered safeguards function, the only consideration following a loss-of-coolant accident is the operation of the isolation valves. Redundant isolation valves are provided as described in Chapter 6, "Engineered Safeguards" on page 6-1. Failures and malfunction of components during normal operation were evaluated. Operation of the Component Cooling System is essential to normal reactor operation. In the event of loss of a component cooling pump, the standby pump will automatically start and maintain cooling water flow. The complete loss of cooling water flow does not require immediate reactor shutdown. However, procedures will require the operator to shutdown the reactor to protect the control rod drive coils. The reactor coolant pumps can be operated without component cooling water if seal injection flow is available.

# 9.2 WATER SYSTEMS

2

## 9.2.1 COMPONENT COOLING SYSTEM

### 9.2.1.1 Design Bases

The Component Cooling System is designed to provide cooling water for various components in the Reactor Building as follows: letdown coolers, reactor coolant pump cooling jacket and seal coolers, quench tank cooler, and control rod drive cooling coils. The design cooling requirement for the system is based on the maximum heat loads from these sources. The system also provides an additional barrier between high pressure reactor coolant and service water to prevent an inadvertent release of activity.

### 9.2.1.2 System Description and Evaluation

The Component Cooling System is shown schematically on Figure 9-8, and the performance requirements of the system are tabulated in Table 9-3. The following is a brief functional description of the major components of the system and their sharing between nuclear units of the station:

Component Cooler

Each component cooler is designed for the total Component Cooling System heat load for a reactor unit. Oconee 1 and 2 each have a single component cooler with a shared common spare. Oconee 3 has two coolers. The coolers reject the heat load to the Low Pressure Service Water System.

Component Cooling Pumps

Each component cooling pump is designed to deliver the necessary flows to the letdown coolers, reactor coolant pump cooling jackets and seal coolers, quench tank cooler, and control rod drive cooling coils. Each unit has one operating pump and one spare.

Component Cooling Surge Tank

This tank allows for thermal expansion and contraction of the water in this closed-loop system. It also provides the required NPSH for the component cooling pumps.

Control Rod Drive Filters

Two filters are provided in the cooling water circuit to the control rod drives to prevent particulates from entering the drive cooling coils. Only one filter is used at a time, with the second as a spare. A bypass is also provided.

### 9.2.1.3 Mode of Operation

During operation, one component cooling pump and one component cooler recirculate and cool water to accommodate the system heat loads for each reactor unit. The component cooling surge tank accommodates expansion, contraction, and leakage of coolant into or out of the system. The surge tank

## 9.1.5 REFERENCES


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9. U.S. Nuclear Regulatory Commission letter to All Power Reactor Licensees, from B.K. Grimes, April 14, 1978, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications".
10. Oak Ridge National Laboratory, "SCALE 4.2, A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation", Volumes I-IV, NUREG/CR-200, Revision 4, April 1995.
11. Letter from W. O. Parker, Jr. (DPC) to H. R. Denton (USNRC), dated July 1, 1980.
12. Letter from W. O. Parker, Jr. (DPC) to H. R. Denton (USNRC), dated July 25, 1980.
13. USNRC Technical Branch Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling".
14. Letter from H. B. Tucker (DPC) to H. R. Denton (USNRC), dated March 10, 1983.
15. NUREG-0800, Standard Review Plan Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System".

hoists to prevent vertical movement unless the grapples are either fully opened or fully closed. The fuel grapple is so designed that when loaded with the fuel assembly, the fuel grapple cannot be opened as a result of operator error, electrical, pneumatic, or hydraulic failure.

All operating mechanisms of the system are located in the fuel handling and storage area for ease of maintenance and accessibility for inspection prior to start of refueling operations. All electrical equipment, with the exception of some limit switches, is located above water for greater integrity and ease of maintenance. The hydraulic systems which actuate the fuel basket rotating frame use demineralized water for operation.

5 Suspected defective fuel is removed from the core and tested for leakage. Leakage verification utilizes an
5 ultrasonic test rig which can be used to detect the presence of water inside a fuel pin. If this method
5 indicates that the clad of the fuel pin has been breeched, the fuel assembly will be repaired or evaluated for
5 acceptability for use in future fuel cycle designs.

The fuel handling bridges are limited to handling of fuel assemblies and control rod and orifice rod assemblies only. All lifts for handling of reactor closure heads and reactor internal assemblies will be made using the Reactor Building Polar crane.

Travel speeds for the fuel handling bridges, hoists and fuel transfer carriages will be controlled to assure safe handling conditions.

4 Since 1990, Oconee has been involved in transferring spent fuel from the Unit 1 and 2 and the Unit 3
4 Spent Fuel Pools to an on-site Independent Spent Fuel Storage Installation. A specially designed transfer
4 cask and associated handling equipment is used for this operation. Cask handling accidents are addressed
5 in Chapter 15, "Accident Analyses" on page 15-1. More detailed information on cask loading and
4 handling activities can be found in the ONS ISFSI FSAR.

Fuel handling equipment is designed to minimize the possibility of mechanical damage to the fuel assemblies during transfer operations. If fuel damage should occur, the amount of radioactivity reaching the environment will present no hazard. The fuel handling accident is analyzed in Chapter 15, "Accident Analyses" on page 15-1.

5 All spent fuel assembly transfer operations are conducted underwater. The water level in the fuel transfer
5 canal provides a nominal water level of 9 feet over the active fuel line of the spent fuel assemblies during
5 movement from the core into storage to limit radiation at the surface of the water. The fuel storage racks
5 provide a nominal 23.5 feet of water shielding over the stored assemblies. The minimum water depth over
5 the stored fuel assemblies is equal to, or greater than 21.34 feet. The minimum depth of water over the
5 fuel assemblies and the thickness of the concrete walls of the storage pool are sufficient to limit radiation
5 levels in the working area. Dose rate information for fuel transfer conditions, and from the storage racks
5 in the Spent Fuel Pool, are provided in Chapter 12.3.1

Water in the reactor vessel is cooled during shutdown and refueling as described in Section 9.3.3, "Low Pressure Injection System" on page 9-44. Adequate redundant electrical power supply assures continuity of heat removal. The spent fuel pool water is cooled as described in Section 9.1.3, "Spent Fuel Cooling System" on page 9-13. A power failure during the refueling cycle will create no immediate hazardous condition due to the large water volume in both the transfer canal and spent fuel pool. With a normal quantity of spent fuel assemblies in the storage pool and no cooling available, the water temperature in the spent fuel pool would increase very slowly (Section 9.1.3, "Spent Fuel Cooling System" on page 9-13).

During reactor operations, bolted and gasketed closure plates, located on the reactor building flanges of the fuel transfer tubes, assure that spent fuel pool water will not leak into the transfer canal in the event of a leak through the transfer tube valves. Both the spent fuel pool and the fuel transfer canals are completely lined with stainless clad steel plate for leak tightness and for ease of decontamination. The fuel transfer tubes will be appropriately attached to these liners to maintain leak integrity. The spent fuel pool cannot be accidently drained by gravity since water must be pumped out.

During the refueling period the water level in both the fuel transfer canal and the spent fuel pool is the same, and the fuel transfer tube valves are open. This eliminates the necessity for interlocks between the fuel transfer carriages and transfer tube valve operations except to verify full-open valve position.

5 The fuel transfer canal and spent fuel pool water will have a boron concentration of at least 2210 ppm. Although this concentration is sufficient to maintain core shutdown if all of the control rod assemblies were removed from the core, only a few control rods will be removed at any one time during the fuel shuffling and replacement. Although not required for safe storage of spent fuel assemblies, the spent fuel pool water will also be borated so that the transfer canal water will not be diluted during fuel transfer operations.

The fuel transfer mechanisms permit initiation of the fuel basket rotation from the building in which the fuel basket is being loaded or unloaded. Carriage travel and fuel basket rotation are interlocked to prevent inadvertent carriage movement when the fuel basket are in the vertical position. Rotation of the fuel baskets is possible only when the carriages are in the rotating frame at the end of travel.

Interlocks are provided to prevent operation of the bridges or trolleys until the assemblies have been hoisted to the upper limit in the mast tube. Mandatory slow zones are provided for the hoisting mechanisms as the grapples approach the core and fuel baskets during insertion of fuel assemblies. The slow zones will be in effect during entry into the reactor core or fuel storage rack and just before and during bottoming out of the fuel assemblies. The controls are appropriately interlocked to prevent simultaneous movement of the bridge, trolley or hoists. The grapple mechanisms are interlocked with the

During operation of the reactors, the fuel transfer carriages are stored in the respective spent fuel pools, thus permitting a blind flange to be installed on the Reactor Building side of each tube.

Space is provided in each spent fuel pool to receive a spent fuel shipping cask as well as provide for required fuel storage. The layout of the fuel pool is shown on Figure 1-4 through Figure 1-8. The cask area is located at the north end of the fuel pools and adjacent to the fuel racks. Following a decay period, the spent fuel assemblies are removed from storage and loaded into the spent fuel shipping cask under water for removal from the site. The spent fuel shipping cask does not pass over fuel storage racks, or any systems or equipment important to safety when being moved to or from the spent fuel pool.

The spent fuel cask handling facility consists of a 100-ton capacity overhead bridge crane with an 11 foot 6 inch span. The hoist controls are five step magnetic, contactor reversing, secondary resistor type with time delay acceleration and a maximum speed of 9 feet per minute. The hoist is equipped with AC solenoid-operated brake system and an eddy-current brake. The bridge controls are the same as the hoist controls and are equipped with AC solenoid operated brake system and has a maximum speed of 50 feet per minute. The trolley is a single speed, four feet/minute, magnetic contactor reversing type controller with AC solenoid-operated brake system. The cranes were designed in accordance with Electric Overhead Crane Institute's Specification No. 61, Class A.

The cranes were tested in the shop by performing a running test, and load tested at the Oconee site to 98 percent of capacity. The running and load test results were satisfactory. Maintenance of the cranes is in accordance with ANSI B30.2. The structural and mechanical components of the crane are designed to have a minimum factor of safety of 2.5 based on yield strength and rated capacity. The hoist brake system consists of the dynamic AB 707 eddy-current control brake and a 13-inch solenoid-operated shoe brake (Whiting SESA). The bridge is equipped with a hydraulic brake for operating the crane from the cab and a solenoid-operated shoe brake for operating the crane by pendant control from the floor. The trolley is equipped with a solenoid-operated shoe brake. The hoist system is equipped with a 75 horsepower motor that produces 328 foot-pounds of torque at full load, 1200 rpm. The starting and instantaneous stalling torque is 902 foot-pounds. The hoist is equipped with a geared lower limit switch for block travel and a paddle-type upper limit switch to prevent a two-blocking situation from occurring.

The cranes are equipped with a sister type hook with safety latch. The hook was load tested and non-destructive tested in the shop. Bethanized wire rope with a safety factor of 6 was used. A lifting adaptor to be used between the yoke and the crane hook is also designed to support three times the load. The lifting adaptor is a stainless steel member approximately 24 feet long, used to lift the cask from the platform to the bottom of the spent fuel pool.

A decontamination area is located in the building adjacent to each spent fuel pool where the outside surfaces of the casks can be decontaminated prior to shipment by using water, detergent solutions and manual scrubbing to the extent required.

#### 9.1.4.2.3 Safety Provisions

Safety provisions are designed into the fuel handling system to prevent the development of hazardous conditions in the event of component malfunctions, accidental damage or operational and administrative failures during refueling or transfer operations.

All fuel assembly storage facilities employ neutron poison material and/or maintain an eversafe geometric spacing between assemblies to assure fuel storage arrays remain subcritical under all credible storage conditions. The fuel storage racks are designed so that it is impossible to insert fuel assemblies in other than the prescribed locations, thereby assuring the necessary spacing between assemblies. Fuel handling and transfer containers are also designed to maintain an eversafe geometric array. Under these conditions, a criticality accident during refueling or storage is not considered credible.

The upper plenum assembly is removed from the reactor and stored underwater on a stand on the fuel transfer canal floor using the head and internals handling fixture and adaptor attached to the polar crane with an internals handling extension.

Refueling operations are carried out from two fuel handling bridges which span the fuel transfer canal in each Reactor Building. One bridge, the Main Bridge, is used to shuttle spent fuel assemblies from the core to the fuel transfer station and new fuel assemblies from the transfer station to the core. During this operation, the second bridge, the Auxiliary Bridge, is occupied with relocating partially spent fuel assemblies in the core as specified by the fuel management program. Fuel shuffling may also be performed by completely unloading the core, shuffling the control components in the spent fuel pool using manual tools, and then reloading the core.

For Units 1 and 2, the main bridge is equipped with two trolley-mounted hoists. One hoist (fuel handling mechanism) is equipped with a fuel grapple and the second hoist (control rod handling mechanism) houses the control rod grapple. The Unit 3 main bridge has one trolly mounted multiple purpose hoist equipped with both fuel and control component grapples. Unit 3 main bridge also has provisions for an auto positioner system to position the bridge and trolly to index over canal/core positions automatically; however, the system is not fully operational (NSM-939). The auxiliary bridge has only one trolley-mounted hoist equipped with a fuel grapple and is used primarily for shuffling or rearranging partially spent fuel assemblies from one position in the core to another. The hydraulic systems which operate the grapple mechanisms use demineralized water for operation, with the exception of Unit 3 main bridge which uses a pneumatic system.

The main bridge moves a spent fuel assembly from the core underwater to the transfer station where the fuel assembly is lowered into the fuel transfer carriage fuel basket. The control rod handling mechanism is used to transfer the control rod or orifice rod assembly to a new fuel assembly waiting in the second fuel transfer carriage basket. This new fuel assembly with control rod or orifice rod assembly is carried to the reactor by the main bridge and located in the core while the spent fuel assembly is being transferred to the spent fuel pool.

Spent fuel assemblies removed from the reactors are transported to the spent fuel pool from the Reactor Building via fuel transfer tubes by means of the fuel transfer mechanism. The fuel transfer mechanisms are carriages that run on tracks extending from each spent fuel pool through the transfer tubes and into the respective Reactor Building. Each of the two independently operated fuel transfer mechanisms which serve Oconee 1 and 2 is designed to operate in two directions so that either of the two Reactor Buildings can be serviced by one or two mechanisms as required. A rotating fuel basket is provided on each end of each fuel transfer carriage to receive fuel assemblies in a vertical position. The hydraulically operated fuel basket is rotated to a horizontal position for passage through the transfer tube, and then rotated back to a vertical position in the spent fuel pool or Reactor Building for vertical removal or insertion of the fuel assembly.

The spent fuel assemblies are removed from the fuel transfer carriage fuel basket using a fuel handling bridge equipped with a fuel handling mechanism and fuel grapple. This bridge spans the spent fuel pool and permits the refueling crew to store or remove new and spent fuel assemblies in any one of the storage rack positions. Spent fuel assemblies may be moved within the spent fuel pools by use of the fuel handling bridge auxiliary hoist and appropriate remote handling tools. In addition, a Post Irradiation Examination jib crane, with associated grapple that may be used to move fuel, is installed in the Unit 1 and 2 spent fuel pool.

Once refueling is completed, the fuel transfer canal is drained through a pipe located in the deep transfer station area. The canal water is pumped to the borated water storage tank to be available for the next refueling.

a reinforced concrete structure lined with stainless clad plate to form a canal above the reactor vessel which is filled with borated water for refueling.

Space is available in the deeper portion of the fuel transfer canal for underwater storage of the reactor vessel internals upper plenum assembly. This portion of the fuel transfer canal can also be used for storage of the reactor vessel internals core barrel and thermal shield assembly by storing the upper plenum assembly in the upper end of the fuel transfer canal.

#### 9.1.4.1.6 Fuel Handling Equipment

This equipment consists of fuel handling bridges, fuel handling mechanisms, fuel storage racks, control rod handling mechanisms, fuel transfer mechanisms, and shipping casks. In addition to the equipment directly associated with the handling of fuel, equipment is provided for handling the reactor vessel closure head and the upper plenum assembly to expose the core for refueling.

### 9.1.4.2 System Description and Evaluation

#### 9.1.4.2.1 Receiving and Storing Fuel

New fuel assemblies are received in shipping containers, unloaded and stored in the appropriate spent fuel pool. After reactor shutdown, new fuel assemblies can be transferred from the spent fuel pool to the Reactor Building with the use of the fuel transfer mechanisms and the fuel transfer tubes.

#### 9.1.4.2.2 Loading and Removing Fuel

Following the reactor shutdown and Reactor Building entry, the refueling procedure is begun by removal of the reactor closure head. The first step in this operation is to uncouple the control rods from the drive mechanism. An auxiliary hoist is used for this and any other special purposes that may be required during refueling. The electrical and water connections to the head assembly are disconnected.

To close the annular space between the reactor vessel flange and fuel transfer canal floor, a seal plate supported by the head stand is lowered into position and bolted to the canal shield flange with appropriate gaskets.

Head removal and replacement time is minimized by the use of two stud tensioners which are also supported by the head assembly. The stud tensioners are hydraulically operated to permit preloading and unloading of the reactor vessel closure studs at cold shutdown conditions. The studs are tensioned to their operational load in two steps in a predetermined sequence. Required stud elongation after tensioning is verified by an elongation gage.

Following removal of the studs from the reactor vessel tapped holes, the studs and nuts are supported in
0 the closure held bolt holes with specially designed spacers. The studs and nuts are then removed from the
0 reactor closure head for inspection and cleaning using special stud and nut handling fixtures. Two special alignment studs are installed in stud location Nos. 15 and 45. The lift of the head and replacement after refueling is guided by these studs. These studs are also used to locate the index fixture used for aligning
0 the plenum assembly during removal and replacement. Storage racks are provided for the closure head
0 studs and the alignment studs.

The reactor closure head is lifted out of the canal onto a head storage stand on the operating floor by a head and internals handling fixture attached to the polar crane. The stand is designed to protect the
0 gasket surface of the closure head. The stud holes except for locations Nos. 15 and 45 are closed with special plugs that prevent water and/or other foreign substances from entering the holes. The fuel transfer canal is then filled with borated water.

performs no emergency functions. Alarms are provided to alert operator of abnormal pool level and temperature.

The Spent Fuel Cooling System has no process lines connecting to the Reactor Coolant System. Its major penetration to the Reactor Building is through the fuel transfer tube. The fuel transfer tube is isolated inside the Reactor Building by a blind flange connection in the fuel transfer canal.

## 9.1.4 FUEL HANDLING SYSTEM

### 9.1.4.1 Design Bases

#### 9.1.4.1.1 General System Function

The fuel handling system shown on Figure 9-7 is designed to provide a safe, effective means of transporting and handling fuel from the time it reaches the station in an unirradiated condition until it leaves the station after postirradiation cooling. The system is designed to minimize the possibility of mishandling or maloperations that could cause fuel assembly damage and/or potential fission product release.

Separate fuel handling equipment is provided for each reactor. A common fuel storage area serves Oconee 1 and 2, while a separate fuel storage area is provided for Oconee 3.

The reactors are refueled with equipment designed to handle the spent fuel assemblies underwater from the time they leave the reactor vessels until they are placed in a cask for shipment from the site. Underwater transfer of spent fuel assemblies provides an effective, economic, and transparent radiation shield, as well as a reliable cooling medium for removal of decay heat. Use of borated water assures reactor subcriticality during refueling.

#### 9.1.4.1.2 New Fuel Storage

New Fuel Storage is described in Section 9.1.1, "New Fuel Storage" on page 9-3.

#### 9.1.4.1.3 Spent Fuel Pool

Each spent fuel pool is a reinforced concrete pool located in its respective Auxiliary Building. The Oconee 1, 2 pool is lined with stainless clad plate. The Oconee 3 pool is lined with stainless steel plate. Each pool is sized to accommodate a full core of irradiated fuel assemblies in addition to the current storage of the largest quantity of new and spent fuel assemblies predicated by the fuel management program. Control rod assemblies requiring removal from the reactors are stored in the spent fuel assemblies, or in brackets suspended from the top of the fuel racks.

#### 9.1.4.1.4 Fuel Transfer Tubes

Two horizontal tubes are provided to convey fuel between each Reactor Building and the respective Auxiliary Building. These tubes contain tracks for the fuel transfer carriages, gate valves on the spent fuel pool side, and a means for flanged closure on the Reactor Building side. The fuel transfer tubes penetrate the spent fuel pool and the fuel transfer canal at their lower depth, where space is provided for the rotation of the fuel transfer carriage baskets.

#### 9.1.4.1.5 Fuel Transfer Canal

The fuel transfer canal is a passageway in the Reactor Building extending from the reactor vessel to the Reactor Building wall. It is formed by an upward extension of the primary shield walls. The enclosure is

5 burnups of 421 EFPDs. The Spent Fuel Cooling System was analyzed to predict the pool temperatures
5 which would result from these heat loads. Temperatures meet the design requirements as specified in
5 Section 9.1.3.1, "Design Bases" on page 9-13.

2 At the time that the Unit 3 spent fuel pool was re-racked, its spent fuel cooling system was upgraded to
2 handle the higher total heat load expected from the increased number of stored fuel assemblies. The heat
2 removal capability of the upgraded spent fuel cooling system has been sized to meet the design limits
2 specified in Section 9.1.3.1, "Design Bases" on page 9-13. A specific analysis of expected maximum
2 normal and abnormal heat loads was performed, based on 18 month cycles, with average fuel burnups of
5 440 EPFDs. Again, the Spent Fuel Cooling System was analyzed to predict the pool temperatures
5 resulting from these heat loads. These temperatures meet the design requirements as specified in Section
2 9.1.3.1, "Design Bases" on page 9-13.

5 During an actual refueling outage for any unit at ONS, it is now common practice to offload a full core
5 (177 fuel assemblies) into the pool. The resulting heat load under this condition will be less than the
5 abnormal heat load cases evaluated in Sections 9.1.3.1.1, "Units 1 and 2 Spent Fuel Pool Cooling
5 System" on page 9-13 and 9.1.3.1.2, "Unit 3 Spent Fuel Pool Cooling System" on page 9-14 for the
5 Units 1 and 2 fuel pool and Unit 3 fuel pool respectively. In addition, the resulting temperature will be
5 less than 205°F in the fuel pools in the abnormal heat load case, assuming a single active failure. The
5 seismic structural integrity of the storage racks, pools, and supporting structures has been evaluated at or
5 above this temperature, and found to be adequate. Also, the thermal-hydraulic analysis of the storage
5 racks indicates that localized boiling will not occur if water entering the storage cells reaches this
5 temperature, as long as normal pool level is maintained.

2 A bypass purification loop is provided to maintain the purity of the water in the spent fuel pool. This
2 loop is also utilized to purify the water in the borated water storage tank following refueling, and to
2 maintain clarity in the fuel transfer canal during refueling. Water from the borated water storage tank or
2 fuel transfer canal can be purified by using the borated water recirculation pump.

#### 9.1.3.3.2 Failure Analysis

An analysis of the maximum fuel cladding temperature has been performed for the postulated case of complete loss of coolant circulation to the pool. The analysis assumes maximum anticipated heat load in the pool, with the hottest assembly located in the least cooled storage area. The maximum cladding temperature will occur at the location of maximum heat flux. For a fuel assembly having the maximum value for decay heat power of 80 kw, and for an axial peak to average power density ratio of 1.2, the maximum local fuel rod heat flux is 1200 BTU/hr-ft$^2$. Natural circulation flow rates within the storage tubes have been calculated which give confidence that convection film coefficients in excess of 50 BTU/hr-ft$^2$ °F can be expected. Assuming this low value for conservatism, the clad surface temperature is 24°F above the coolant temperature. Because the heat flux is small, very large uncertainties in the film coefficient are acceptable without causing prohibitively high clad temperatures. For example, a reduction by a factor of five in the film coefficient would result in a clad surface temperature of 120°F above the coolant temperature. A reduction by a factor of ten, from 50 BTU/hr-ft$^2$ °F to 5 BTU/hr-ft$^2$ °F would result in a clad surface temperature of 240°F above the coolant temperature. These temperatures are below 650°F, which is the normal operating temperature of the fuel clad in the core.

### 9.1.3.4 Safety Evaluation

The Spent Fuel Cooling System provides adequate capacity and component redundancy to assure the cooling of stored spent fuel, even when large quantities of fuel are in storage. Multiple component failures or complete cooling failures permit ample time to assure that protective actions are taken. The system is arranged so that loss of fuel pool water by piping or component failure is impossible. The system

Spent Fuel Coolant Pumps

The spent fuel coolant pumps take suction from the spent fuel pool and recirculate the fluid back to the pool after passing through the coolers. A portion of the flow is demineralized and filtered depending on conditions. There are three pumps for Oconee Units 1 and 2, and three pumps for Oconee 3. The spent fuel coolant pumps are also used for filling the fuel transfer canal or incore instrumentation handling tank with borated water from the borated water storage tank.

Spent Fuel Coolant Demineralizers

One spent fuel coolant demineralizer will process approximately one-half of the spent fuel pool volume in 24 hours. There is one demineralizer for Units 1 and 2, and one for Unit 3.

Spent Fuel Coolant Filters

The spent fuel coolant filters are designed to remove particulate matter from the spent fuel pool water. They are sized for the same flow rate as the demineralizers (180 gpm). There are two filters for Units 1 and 2, and two for Unit 3.

Borated Water Recirculation Pump

This pump removes water from the borated water storage tank for demineralization and filtering. The pump may also be used while demineralizing and filtering the water in the fuel transfer canal during a transfer of fuel. It may also be used for emptying the fuel transfer canal if both spent fuel coolant pumps are unavailable for use. There is one pump for Units 1 and 2, and one for Unit 3.

### 9.1.3.3 System Evaluation

#### 9.1.3.3.1 Normal Operation

2 The normal operation of the Spent Fuel Cooling System provides two main functions. The first is to
2 maintain the pool temperature below the design bases limits specified in Section 9.1.3.1, "Design Bases"
2 on page 9-13. The second function is to maintain the pool inventory, clarity and chemistry at acceptable
2 levels.

2 Spent fuel pool heat removal is accomplished by recirculating spent fuel coolant water through heat
2 exchangers and then back to the pool. The spent fuel pumps take suction from the spent fuel pool and
2 transport the flow through the coolers, which are arranged in parallel. The waste heat is removed from
2 the shell side of the coolers by the Recirculated Cooling Water System. The cooled spent fuel pool water
2 is then directed back to the spent fuel pool.

2 The spent fuel pool water temperature is a direct function of the decay heat load produced by the fuel in
2 the racks, in conjunction with the heat removal capability of the spent fuel cooling system. The total heat
2 removal capacities are the same for the Units 1 and 2 and the Unit 3 spent fuel pool coolant systems.
2 Both systems use the same numbers of pumps and coolers, with the same design specifications and overall
2 equipment configurations. The expected decay heat loads vary with the number of fuel assemblies present
2 in the pool, the burnups of the various fuel assemblies, and the post-irradiation decay times.

2 At the time that the Units 1 and 2 spent fuel pool was re-racked, its spent fuel cooling system was
2 upgraded to handle the higher total heat load expected from the increased number of stored fuel
2 assemblies. The heat removal capability of the upgraded spent fuel cooling system has been sized to meet
2 the design limits specified in Section 9.1.3.1, "Design Bases" on page 9-13. A specific analysis of expected
2 maximum normal and abnormal heat loads was performed, based on 18 month cycles, with average fuel

2 **9.1.3.1.2 Unit 3 Spent Fuel Pool Cooling System**

2 The Unit 3 Spent Fuel Pool Cooling System duplicates the equipment used for the Units 1 and 2 system.
2 The Unit 3 system is designed to remove the decay heat from the stored fuel in the Unit 3 spent fuel pool.
5 The cooling system heat removal requirements are as set forth in NRC Standard Review Plan Section
5 SRP-9.1.3 (References 14 on page 9-23, 15 on page 9-23). Other system functions are to maintain the
2 pool inventory, clarity and chemistry at acceptable levels.

5

2 The Unit 3 system heat removal design requirements, as stipulated by Standard Review Plan 9.1.3, are:

2 1. For the maximum normal heat load with the normal cooling systems in operation, and assuming a
2 single active failure, the temperature of the pool water shall be maintained at or below 140°F and the
2 liquid level in the pool should be maintained.

2 2. For the abnormal maximum heat load with the normal cooling systems in operation, the pool water
2 temperature should be kept below boiling and the liquid level in the pool should be maintained. A
2 single active failure need not be considered.

5 The design basis maximum normal and abnormal decay heat loads are as defined in SRP 9.1.3 (Reference
5 15 on page 9-23), for fuel racks with greater than 1 1/3 core storage capacity. The decay heat predictions
5 are to be based on the methodology presented in Reference 13 on page 9-23.

5 It should be noted that, while both temperature conditions above represent design criteria associated with
5 specific analytical assumptions, only the boiling criterion represents an actual design limit. An operating
5 limit of 205°F is imposed for conservatism. Analyses have been performed to ensure that seismic and
5 structural integrity of the pool liner, supporting concrete, and fuel racks are not compromised at this
5 temperature limit. Thermal - hydraulic analysis of the racks also has shown that boiling within the fuel
5 cells does not occur with pool temperatures maintained at or below this limit, provided normal operating
5 pool level is maintained.

2 In addition to the primary function of decay heat removal, the system provides for purification of the
2 spent fuel pool water, the fuel transfer canal water, and the contents of the borated water storage tank, in
2 order to remove fission and corrosion products and to maintain water clarity for fuel handling operations.
2 The system also provides inventory makeup for the fuel transfer canal and the incore instrument handling
2 tank.

2 The system is designed to withstand the effects of a seismic event and meet the requirements of Quality
2 Group C classification.

### 9.1.3.2 System Description

0 The Spent Fuel Cooling System (Figure 9-5, Units 1, 2 and 3) provides cooling for the spent fuel pool to remove fission product decay heat energy. System performance data are shown in Table 9-1 (Units 1 and 2) and Table 9-2 (Unit 3). Major components of the system are briefly described below.

Spent Fuel Coolers

The spent fuel coolers are designed to maintain the temperature of the spent fuel pool as noted in Section 9.1.3.1, "Design Bases" on page 9-13. There are three coolers for Oconee 1 and 2, and three coolers for Unit 3, arranged in parallel.

## 9.1.3 SPENT FUEL COOLING SYSTEM

### 9.1.3.1 Design Bases

2 #### 9.1.3.1.1 Units 1 and 2 Spent Fuel Pool Cooling System

5 The primary function of Spent Fuel Pool Cooling System for Units 1 and 2 is to provide decay heat
2 removal for the spent fuel stored in the Units 1 and 2 spent fuel pool. The cooling system design
5 requirements are the criteria imposed by the 1980 re-racking (References 11 on page 9-23, 12 on
5 page 9-23). Other system functions are to maintain the pool inventory, clarity and chemistry at
2 acceptable levels.

5

5 Revised criteria have been imposed during the 1980 re-racking modification, pursuant to Amendments 90, 90, and 87 for License Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station. The
5 thermal-hydraulic analyses associated with the spent fuel pool racks assumes that the bulk spent fuel pool temperature remain at or below 150°F, for normal heat loads (Reference 10 on page 9-23). The Units 1 and 2 Spent Fuel Cooling System is designed to keep the pool bulk water temperature:

5 1. Below 150°F for normal heat loads and two or three pump-cooler configurations in operation
5 (Reference 11 on page 9-23)

5 2. Below 150°F for abnormal heat loads and three pump-cooler configurations in operation (Reference
5 11 on page 9-23)

5 3. Below 205°F for abnormal heat loads and any two pump-cooler configurations in operation
5 (Reference 11 on page 9-23).

For the Units 1 and 2 spent fuel cooling system, the design basis normal heat load assumes that Units 1 and 2 are refueled consecutively, and the rack positions are filled with previous discharges, except for 118
5 spaces reserved for a full core discharge (Reference 11 on page 9-23). The design basis abnormal heat load assumes that Units 1 and 2 are refueled consecutively, followed by a full core discharge after a short
5 period of operation. In this case, all rack positions contain spent fuel (References 11 on page 9-23 and 12
5 on page 9-23). The decay heat predictions are to be based on the methodology presented in Reference 13
5 on page 9-23.

5 It should be noted that, while all temperature conditions above represent design criteria associated with
5 specific analytical assumptions, only the higher temperature of 205°F represents an actual operating limit.
5 Analyses have been performed to ensure that seismic and structural integrity of the pool liner, supporting
5 concrete, and fuel racks are not compromised at this temperature limit. Thermal - hydraulic analysis of
5 the racks has also shown that boiling within the fuel cells does not occur with pool temperatures
5 maintained at or below this limit, provided normal operating pool level is maintained.

2 In addition to the primary function of decay heat removal, the system provides for purification of the
2 spent fuel pool water, the fuel transfer canal water, and the contents of the borated water storage tank, in
2 order to remove fission and corrosion products and to maintain water clarity for fuel handling operations.
2 The system also provides inventory makeup for the fuel transfer canal and the incore instrument handling
2 tank.

2 The system is designed to withstand the effects of a seismic event and meet the requirements of Quality
2 Group C classification.

5 **9.1.2.3.4 Interface of High Capacity Fuel Storage Rack and Spent Fuel Storage Pool**

The pool floor will support the high capacity storage rack as a free-standing structure during all design conditions. For installation into the pool, pairs of rack modules are field connected by welding coupling plates (two on each a side) to form four 8 x 12 assemblies and one 7 x 12 assembly. These assembled pairs have four floor beams attached to their bases prior to being lowered into the pool. Each floor beam attached to the assembled pairs has bearing pads which are shimmed to ensure contact with the pool floor corresponding to the model used for the seismic analysis. The assembled pairs are connected to adjacent assembled pairs at the tops.

For the free-standing rack structure, conservative analysis shows that under simultaneous forces from vertical and lateral seismic excitation, including pool fluid sloshing, the residual displacement of the rack relative to the pool floor is less than 1.3 inches for unloaded conditions and less than 0.7 inches for full-loaded condition (i.e., much less than minimum clearance of 6.1 inches to pool walls and installed equipment.)

The relative motion between the rack and pool floor is determined by the following procedure for both full and empty rack.

1. The amplitude of the most severe of the two lateral time histories as determined from an inspection of their response spectra is increased by a factor of $\sqrt{2}$ to conservatively account for simultaneous excitation in the two horizontal directions. This time history is then input to the lateral SHOCK code models for full and empty rack calculations.
2. The coefficients of friction used in the analyses, 0.2 static and 0.1 dynamic, are well below minimum values found in the literature for stainless on stainless and similar metals in a water environment. These low friction coefficient values account for the decrease in normal force between the rack base and pool floor due to vertical seismic response. The peak vertical seismic response was calculated to be 0.3 g based on the vertical natural rack frequency and vertical response spectrum.
3. The rack/pool floor normal force on which the lateral friction forces used in the analysis are based includes bouyancy effects due to submergence of the rack and fuel.

The maximum lateral seismic force exerted by each rack module on pool floor is 111,200 pounds and results in a stress of 1,650 psi in the floor liner and 2,330 psi in the weld connecting the floor liner to embedments in the concrete. The maximum combined seismic and thermal stress in the floor liner is 16,300 psi and 22,000 psi in the weld between liner and embedments. The maximum stresses are below the design allowable stress of 27,000 psi.

5 **9.1.2.4 Safety Evaluation**

The storage rack is designed and constructed to retain the integrity of the structure under all anticipated loads, including the Safe Shutdown Earthquake, with the maximum number of fuel assemblies occupying the storage locations.

The rack design provides protection against damage to the fuel and precludes the possibility of a fuel assembly being placed between cells. Although not required for safe storage of spent fuel assemblies, the
5 spent fuel pool water is normally borated to a concentration of at least 2210 ppm boron. The rack design also assures a $K_{eff}$ of less than 0.95 even when the entire array of fuel assemblies, assumed to be in their
5 most reactive condition and within the limits specified in the Technical Specifications, are immersed in
5 unborated water at room temperature. Furthermore, if the pools were filled with the most reactive fuel
5 allowed, which is clearly in violation of the Technical Specifications, $K_{eff}$ would be $\simeq$ 0.8 with credit for
5 soluble boron. Under these conditions a criticality accident during refueling or storage is not considered credible.

5 The presence of approximately 2000 ppm boron in the pool water will decrease reactivity by about 20
5 percent Δk. In perspective, this is nearly as much negative reactivity as in the poison plates, so $k_{eff}$ for the
5 rack would only be slightly greater than 0.95 even if the poison plates were not present. Thus, for
5 postulated accidents, should there be a reactivity increase, $k_{eff}$ would be less than or equal to 0.95 due to
5 the combined effects of the dissolved boron and the poison plates.

5 Increasing the $k_{eff}$ limit to 0.98 provides an additional 0.03 Δk margin for accident conditions. This still
5 provides 0.02 Δk margin to criticality as required in ANSI/ANS-57.2-1983 Section 6.4.2.2.3.

5 The "optimum moderation" accident is not a problem in the spent fuel storage racks because possible
5 water densities are too low (≤ 0.01 gm/cm³) to yield $k_{eff}$ values higher than for full density water and the
5 rack design prevents the preferential reduction of water density between the cells of a rack (e.g. boiling
5 between cells). Further, the presence of the poison plates removes the conditions necessary for "optimum
5 moderation" so that $k_{eff}$ continually decreases as moderator density decreases from 1.0 g/cm³ to 0.0 g/cm³
5 in poison rack designs.

5 *9.1.2.3.2.5 Acceptance Criteria for Criticality*

5 The neutron multiplication factor in the spent fuel pools shall be less than or equal to 0.95, including all
5 uncertainties. Under certain accident conditions $K_{eff}$ may be less than or equal to 0.98 as permitted by
5 ANSI/ANS-57.2-1983.

5 Generally, the acceptance criteria for postulated accident conditions can be $k_{eff} \leq 0.98$ because of the
5 accuracy of the methods used coupled with the low probability of occurrence. For instance, in ANSI
5 N210-1976 the acceptance criteria for the "optimum moderation" condition is $k_{eff} \leq 0.98$. However, for
5 storage pools which contain dissolved boron, the use of the realistic initial conditions provides additional
5 subcriticality margin below 0.98. Thus, the acceptance criteria for most conditions in the spent fuel pools
5 will be $k_{eff} \leq 0.95$. This assumes credit may be taken for soluble boron under accident conditions as
5 allowed by the double contingency principle in ANSI/ANS-57.2-1983, and that no credit is taken for
5 soluble boron under normal conditions.

5 For certain accident conditions, the less restrictive 0.98 limit on $k_{eff}$ is applied. Examples of this include
5 the cask drop accident in Section 15.11.2.5.1, "Criticality Analyses for Dry Storage Transfer Cask Drop
5 Scenarios" on page 15-37 and makeup to the spent fuel pool with unborated water following a Standby
5 Shutdown Facility event.

5 *9.1.2.3.2.6 Cask Drop Accident*

5 Cask drop accidents are analyzed for criticality consequences in Section 15.11.2.5.1, "Criticality Analyses
5 for Dry Storage Transfer Cask Drop Scenarios" on page 15-37.

5 **9.1.2.3.3 Material, Construction, and Quality Control**

The entire fuel assembly storage rack is constructed of type 304 stainless steel. All welded construction is used in the fabrication of the fuel assembly storage rack. The all-welded construction ensures the structural integrity of the storage modules and provides assurance of smooth, snag-free passage in the storage cavities so that it is highly improbable that a fuel assembly could become stuck in the rack.

The material, construction and quality control procedures are in the accordance with the quality assurance requirements of Duke Power Company, as described in Duke Power Company Topical Report, DUKE-1.

5 configuration. Each burnup versus enrichment curve shows the minimum amount of burnup required to
5 qualify fuel for storage in the applicable loading pattern as a function of the fuel's initial enrichment.
5 Additional details of the methods used can be found in Reference 8 on page 9-23.

5 The SCALE-4 system of computer codes (Reference 10 on page 9-23) was used to analyze the boundary
5 condition created between the restricted and unrestricted storage configurations to assure that the storage
5 configurations at the boundary do not cause an increase in the nominal $k_{eff}$ for the individual regions.
5 This analysis is performed to determine if there is a need for new administrative restrictions at the
5 boundaries.

5 This methodology utilizes Monte Carlo theory. Specifically, this analysis method used the CSAS25
5 sequence contained in Criticality Analysis Sequence No. 4 (CSAS4). CSAS4 is a control module
5 contained in the SCALE-4 system of codes. The CSAS25 sequence utilizes two cross section processing
5 codes (NITAWL and BONAMI) and a 3-D Monte Carlo code (KENO Va) for calculating the effective
5 multiplication factor for the system. The 27 Group NDF4 cross section library was used exclusively for
5 this analysis.

5 Acceptable interface boundary conditions between storage configurations were determined by varying the
5 boundaries between various storage regions to determine the worst case configurations for coupling
5 between assemblies in different regions. The boundaries were then reflected to simulate an infinite array.
5 The $k_{eff}$ of these infinite boundary arrays were compared to the base $k_{eff}$ of infinite arrays of either fuel
5 storage region creating the boundary. If the infinite boundary array $k_{eff}$ did not represent an increase in the
5 $k_{eff}$ of the regions making the boundary, then no storage restrictions were imposed at the interface. When
5 the worst case did represent an increase, conservative storage restrictions were applied.

5 These methods conform with ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary
5 Pressurized Water Reactor Plants," Section 5.7, Fuel Handling System; ANSI N210-1976, "Design
5 Objectives for LWR Spent Fuel Storage Facilities at Nuclear Power Stations," Section 5.1.12; ANSI
5 N16.9-1975, "NRC Standard Review Plan," Section 9.1.2, "Spent Fuel Storage" on page 9-3 and the
5 NRC guidance contained in Reference 9 on page 9-23.

5 *9.1.2.3.2.4 Postulated Accidents*

5 Most accident conditions will not result in an increase in $k_{eff}$ of the rack. Examples are loss of cooling
5 systems and dropping a fuel assembly on top of the rack. For the loss of cooling systems, the reactivity
5 decreases with decreasing water density for the Oconee spent fuel storage racks and the current analyzed
5 fuel designs. For an assembly dropped on top of the storage rack, the rack structure pertinent for
5 criticality is not excessively deformed and the dropped assembly has more than eight inches of water
5 separating it from the active fuel height of stored assemblies which precludes interaction. Although the
5 dropped assembly is more reactive outside rather than inside the poisoned storage cell, the assembly is no
5 more reactive dropped on top of the storage rack than located anywhere else in the pool outside the
5 storage rack.

5 However, accidents can be postulated which would increase reactivity. For accident conditions, two
5 techniques are employed to ensure that sufficient criticality margin exists, the double contingency principle
5 and increasing the $K_{eff}$ limit to 0.98. The acceptance criteria for criticality is further discussed in
5 9.1.2.3.2.5, "Acceptance Criteria for Criticality" on page 9-11.

5 The double contingency principle of ANSI/ANS-57.2-1983 states that it is not required to assume two
5 unlikely, independent concurrent events to ensure protection against a criticality accident. Thus, for
5 accident conditions, the presence of soluble boron in the storage pool water can be assumed as a realistic
5 initial condition since not assuming its presence would be a second unlikely event.

5 page 9-23). The B&W critical experiments used are specifically designed for benchmarking reactivity
5 calculation techniques. The experiments are analyzed, and the statistical accuracy of the calculated
5 reactivity results are assessed.

5 The bias associated with the benchmarks is -0.00189 Δk with a standard deviation of 0.00371 Δk. The
5 95/95 one-sided tolerance limit factor for 10 values is 2.911. Therefore, there is a 95 percent probability at
5 a 95 percent confidence level that the uncertainty in reactivity due to the method is not greater than
5 0.01080 Δk.

5 For burned fuel, the maximum reactivity occurs approximately 100 hours after shutdown due to the decay
5 of $Xe^{135}$. Therefore, all fuel assemblies in the spent fuel pool are modeled at no xenon conditions.

5 An additional bias and uncertainty are required to quantify the reactivity of burned nuclear fuel
5 assemblies. Two burnup uncertainties associated with this methodology are accounted for in the
5 criticality analysis. The first penalty accounts for uncertainties in the reactivity due to uncertainties in the
5 burnup of the assembly, while the second penalty accounts for the reactivity holddown effect of lumped
5 burnable absorbers.

5 The exposure reactivity uncertainty accounts for the uncertainty on the assembly burnup. Since the final
5 burnup qualification curves are based on a code calculated burnup, the uncertainty in that calculated
5 burnup must be considered. Rather than determining the uncertainty on the actual burnup, the
5 uncertainty on reactivity due to burnup was applied to account for the burnup uncertainty. A 95/95
5 one-sided tolerance was determined to account for the maximum reactivity error associated with the
5 burnup of the fuel.

5 As required by the standards, no removable poisons are accounted for in the criticality analyses. Thus, all
5 assemblies are modeled with no burnable poisons (BPs). However, this can be slightly non-conservative
5 due to the increase in reactivity associated with the removal of the BP. Thus a burnable poison removal
5 (BP-Pull) penalty is developed to account for this effect. BPs are used in the core design to hold down
5 reactivity, and hence peaking of fresh assemblies. Thus, the reactivity of the BPd assembly is less than the
5 non-BPd assembly. However, once the BP is removed from the previously BPd assembly, a reactivity
5 increase is seen due to the shadowing effect the BPs had on the assembly. This reactivity increase is large
5 enough such that the assembly with the BPs removed is more reactive than the assembly which never
5 contained BPs, once the BPs are removed. This difference in reactivity is applied as an additional bias on
5 reactivity.

5 The basic approach in the burnup credit methodology is to use reactivity equivalencing techniques to
5 construct burnup versus enrichment curves which represent equivalent and acceptable reactivity conditions
5 over an applicable range of burnups and initial enrichments. These burnup versus enrichment curves are
5 established for each type of storage, e.g. unrestricted and restricted storage.

5 Generation of the applicable burnup credit curves requires a two part calculation process. The first part is
5 to create two types of reactivity versus burnup curves. The first type of curve defines the maximum
5 reactivity for the spent fuel pool such that the appropriate design criteria are met including allowances for
5 both calculational uncertainties and manufacturing tolerances. The second type of curve represents the
5 reactivity versus burnup for a particular enrichment, and is generated for the range of enrichments. The
5 intersection of the maximum design reactivity curve with the multiple enrichment curves provides data
5 points for the second part of the process.

5 The second part of the process generates the burnup versus initial enrichment curves by plotting the
5 burnup where the maximum design reactivity equals the reactivity of a particular enrichment for each
5 enrichment. Two curves are generated which represent the qualification criteria for a particular storage

- Boraflex width
- Can ID
- Stainless steel thickness
- Center-to-center spacing
- Fuel enrichment
- Fuel pellet density
- Fuel pellet OD

Other applicable uncertainties and biases are discussed in Section 9.1.2.3.2.3, "Criticality Analysis Methodology."

6. No credit is taken for the assembly spacer grids.
7. No credit is taken for fuel assembly control components which can be removed (e.g. burnable poisons and control rods).
8. Credit is taken for the inherent neutron absorbing effect of some of the rack structure materials and in solid materials added specifically for neutron absorption in accordance with Section 6.4.2.2.8 of ANSI/ANS-57.2-1983.
9. A bias is included in the reactivity calculation to account for $B_4C$ particle self shielding.

#### *9.1.2.3.2.3 Criticality Analysis Methodology*

Criticality of fuel assemblies outside the reactor is precluded by adequate design of fuel transfer, shipping and storage facilities and by administrative control procedures. The two principal methods of preventing criticality are limiting the fuel assembly array size and limiting assembly interaction by fixing the minimum separation between assemblies and/or inserting neutron poisons between assemblies.

The design basis for preventing criticality outside the reactor is that, considering possible variations, there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor ($k_{eff}$) of the fuel assembly array will be less than or equal to 0.95 as recommended in ANSI N210-1976. The conditions that are assumed in meeting this design basis are outlined in Section 9.1.2.3.2.2, "Normal Storage" on page 9-7.

In order to justify storage of fuel up to 5.0 w/o, the burnup credit approach was utilized in the spent fuel pools. The burnup credit approach to fuel rack criticality analysis requires calculation and comparison of reactivity values over a range of burnup and initial enrichment conditions. In order to accurately model characteristics of irradiated fuel which impact reactivity, a criticality analysis method capable of evaluating arrays of these irradiated assemblies is needed. The advanced nodal methodology combining CASMO-3/TABLES-3/SIMULATE-3 is used for this purpose. CASMO-3 (Reference 4 on page 9-23) is an integral transport theory code, SIMULATE-3 (Reference 6 on page 9-23) is a nodal diffusion theory code, and TABLES-3 (Reference 5 on page 9-23) is a linking code which reformats CASMO-3 data for use in SIMULATE-3. This methodology permits direct coupling of incore depletion calculations and resulting fuel isotopics with out-of-core storage array criticality analyses. The variable effects of fission product poisoning, fissile material production and utilization and other related effects are accurately modeled with the CASMO-3/TABLES-3/SIMULATE-3 methodology. Applicable biases and uncertainties are developed and become inputs to the methodology.

The results for the criticality methodology are validated by comparison to measured results of fuel storage critical experiments. The criticality experiments used to benchmark the methodology were the Babcock and Wilcox close proximity storage critical experiments performed at the CX-10 facility (Reference 7 on

5 **9.1.2.3.2 Criticality Analysis**

5 The design methodology which ensures the criticality safety of the fuel assemblies in the spent fuel storage
5 rack is discussed in Section 9.1.2.3.2.3, "Criticality Analysis Methodology" on page 9-8 and in Reference
5 8 on page 9-23.

5 *9.1.2.3.2.1 Neutron Multiplication Factor*

5 Criticality of fuel assemblies in the spent fuel storage rack is prevented by the design of the rack which
5 limits fuel assembly interaction. This is done by fixing the minimum separation between assemblies and
5 inserting neutron poisons between assemblies.

5 The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95
5 percent probability at a 95 percent confidence level that the effective multiplication factor ($k_{eff}$) of the fuel
5 assembly array will be less than 0.95 for most conditions (0.98 for certain accident conditions) as
5 recommended in ANSI/ANS-57.2-1983 and in Reference 9 on page 9-23. The acceptance criteria for
5 criticality is further discussed in Section 9.1.2.3.2.5, "Acceptance Criteria for Criticality" on page 9-11.

5 *9.1.2.3.2.2 Normal Storage*

5 Under normal storage conditions, the following assumptions were used in the criticality analysis.

5 1. Credit is taken for the decrease in reactivity associated with the fuel assembly burnup.

5 2. The fuel assembly is the most reactive fuel assembly to be stored based on a minimum burnup. The
5 fuel designs analyzed are all 15x15 arrays and include up through the Babcock and Wilcox MkB11
5 design.

5 3. The moderator is pure water at the temperature within the design limits of the pool which yields the
5 largest reactivity. No dissolved boron is included in the water for normal storage in the spent fuel
5 pool racks. Credit is taken for soluble boron under postulated accident conditions and during fuel
5 movement. For accident conditions the double contingency principle of ANSI N16.1-1975 is applied.
5 This principle states that it shall require at least two unlikely, independent, and concurrent events to
5 produce a criticality accident. During fuel movement the presence of dissolved boron in the spent fuel
5 pool water is assumed since this is only a temporary condition and only a single assembly is handled
5 at a time.

5 4. The array is either infinite in the lateral extent or is surrounded by a conservatively chosen reflector,
5 whichever is appropriate for the design. The nominal case calculation is infinite in the lateral extent.
5 However, poison plates are not necessary on the periphery of the modular array and between widely
5 spaced modules because calculations show that this finite array is less reactive than the nominal case
5 infinite array. The assemblies are also infinite in the axial extent. The 2 dimensional infinite array
5 assumption is consistent with other burnup credit analyses performed by the spent fuel storage cell
5 vendor. The vendor studied the differences between a detailed 3-D model which included the effects
5 of axial burnup, and an infinite 2-D model which did not. The conclusion reached was that the
5 reactivity differences were relatively small and that the infinite 2-D model conservatively bounded the
5 results of the 3-D model with axial burnup effects for the typical range of minimum burnup
5 requirements. Therefore, the nominal case of a 2 dimensional infinite array of poison cells is a
5 conservative assumption.

5 5. Mechanical uncertainties and biases due to mechanical tolerances during construction are treated by
5 either using "worst case" conditions or by performing sensitivity studies and obtaining appropriate
5 values. The items included in the analysis are:

5 - Boraflex thickness

elastic columns acting as spring restraints. The base of the structure is considered fixed. The choice of the location of the mass-joints depends on the distribution of masses in the real structure.

The modes and frequencies of natural vibration are obtained. A time-history analysis is performed using the modal superposition method, in which the responses in the normal modes are determined separately, then superimposed to provide the total response. The structural damping values used are 4 percent for a ½ SSE and 7 percent for SSE. These values conform to the requirements of Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants."

The acceleration-time history response of the mass-joint at elevation 802 ft (fuel pool floor) was then used as input for the time history analysis of the fuel rack model (see 3 below).

Seismic response loads for determining stresses in the racks are determined by the following procedure:

1. A modal extraction analysis of a space frame analytical model of the rack modules minus fuel but including the effects of the water surrounding and within the cans is performed using the STARDYNE Computer Code, Reference 1 on page 9-23.
2. The modal parameters of the rack modules are used to derive a dynamically equivalent model of the rack modules for incorporation into a lateral non linear model which includes both the racks and the fuel assemblies.
3. A time history analysis of the nonlinear model is performed using the pool structure response time histories as input. This analysis is performed with the SHOCK computer code, Reference 2 on page 9-23, and the results include the impact force time histories between the fuel assemblies and cavities and the rack support reaction loads.
4. The fuel assembly/cavity impact load time histories and the rack support motion time histories are then applied to the STARDYNE space frame model of the rack module to obtain the detailed distribution of structural member loads.
5. The above analysis is consistent with the three directional component excitation requirements of Regulatory Guide 1.92 with the assumption that the base of the racks is restrained against lateral motion. This conservative assumption is applied to account for the high coefficient of friction that exists for stainless steel in contact with stainless steel. Damping values of 2 percent and 3 percent are assumed for the rack structural members and fuel respectively.

An analysis was performed to determine the effects of seismic excitation of the pool water using the procedure outlined in Reference 3 on page 9-23. This analysis showed that the rack is below the depth where sloshing forces are effective and therefore should not experience excitation from this source. However, for conservatism, drag forces on the rack were calculated using the peak pool fluid velocity due to sloshing combined with the peak rack response velocity. The results show total drag forces on the entire rack structure in the pool to be only three percent of the weight of the empty rack.

The maximum uplift load available from the fuel handling crane on the storage rack is limited to 3000 lbs or less by the hoist interlock. This loading represents a less severe loading on the fuel rack structure than that resulting from the design basis earthquake.

Structural design precludes placing a fuel assembly between cells, and the rack will withstand the loadings imposed by a postulated dropped fuel assembly.

#### 9.1.2.2.1 Design Bases

The Oconee 3 Spent Fuel Pool has the same Design Bases as the Oconee 1 and 2 pool described in Section 9.1.2.1.1, "Design Bases" on page 9-3.

#### 9.1.2.2.2 Design Description

The Oconee 3 Spent Fuel Pool storage racks are similar to the Oconee 1 and 2 racks described in Section 9.1.2.1.2, "Design Description" on page 9-4. The following information applies to Oconee Unit 3 spent fuel storage racks.

| | | |
|---|---|---|
| 5 | Number of Cells | 822 plus storage locations for 3 failed fuel containers |
| | Number Rack Arrays | 7 - 8 x 10 |
| | | 2 - 8 x 12<br>1 - 8 x 10 x/3 container locations |
| 5 | Poison Material | Boraflex 0.03 gm $B^{10}/cm^2$ Vented to pool environment |
| | Center-to-Center Spacing | 10.60 in. |
| 5<br>5 | Type of Fuel | B&W 15 x 15, MK B11 and earlier designs, 5.0 weight percent maximum nominal enrichment. |
| | Rack Assembly Dimension and Weights | 8 x 10 - 85.5 x 107 x 172 - 20,200 lbs. |
| | | 8 x 12 - 85.5 x 128 x 172 - 24,000 lbs. |

The pool outline and rack arrangements are shown in Figure 9-3 and Figure 9-4.

### 5 9.1.2.3 System Evaluation

#### 5 9.1.2.3.1 Structural and Seismic Analysis

Fuel assembly storage rack and associated structures are designed to withstand the maximum forces generated during normal operation combined with the Safe Shutdown Earthquake according to the requirements of a Seismic Class 1 structure. For these conditions, the storage rack design is such that all stresses fall within the allowable stress limits specified in the AISC Specifications for Design, Fabrication and Erection of Structural Steel.

Normal operating loads include dead weight (in air) and thermal expansion loads. Lateral and vertical seismic loads along with the fluid forces generated by seismically generated pool water sloshing are considered to be acting simultaneously.

The seismic input spectra conform to the requirements of Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants."

Reference is made to Project 81 PSAR, Docket Nos. STN50-488 through -493, Section 3.7, "Seismic Design" on page 3-61. The smoothed response spectra shown on Figure 2E-2A were normalized to 10 percent g for Safe Shutdown Earthquake (SSE). An earthquake acceleration-time history compatible with these spectra, as shown in Figures 2E-2B through 2E-2E, was used as a base motion on the model of the Auxiliary Building.

The seismic response of the Auxiliary Building to the base excitation is determined by a dynamic analysis. The dynamic analysis is made by idealizing the structure as a series of lumped masses with weightless

3. The racks are designed to allow coolant flow such that boiling in the water channels between fuel assemblies does not occur.
4. The racks are designed to Seismic Category 1 requirements, and are classified as ANS Safety Class 3 and ASME Code Class 3 Component Support structures.
5. The racks are designed to withstand loads which may result from fuel handling accidents and from the maximum uplift force of the fuel handling crane.
6. Each storage position in the racks is designed to support and guide the fuel assembly in a manner that will minimize the possibility of application of excessive lateral, axial and bending loads to fuel assemblies during fuel assembly handling and storage.
7. The racks are designed to preclude the insertion of a fuel assembly in other than design locations.
8. The materials used in construction of the racks are compatible with the storage pool environment and do not contaminate the fuel assemblies

#### 9.1.2.1.2 Design Description

The Oconee fuel storage racks are composed of individual storage cells made of stainless steel interconnected by grid assemblies to form integral module structures as shown in Figure 9-1. Each cell has a lead-in opening which is symmetrical and is blended smooth to facilitate fuel insertion. The cells are open at the top and bottom to provide a flow path for convective cooling of spent fuel assemblies through natural circulation. The fuel assembly storage cells are structurally connected to form modules through the use of channels, plates, and angles which limit structural deformations and maintain a nominal center-to-center spacing between adjacent storage cavities during design conditions including the Safe Shutdown Earthquake. The racks utilize a neutron absorber, Boraflex, which is attached to each cell. The modules are neither anchored to the floor nor braced by the pool walls. The following information applies to the Oconee 1 and 2 fuel storage pool.

| | |
|---|---|
| Number of Cells | 1312 |
| Number of Modules | 4 - 8 x 11 |
| | 10 - 8 x 12 |
| Poison Material | Boraflex 0.02 gm $B^{10}/cm^2$ Vented to pool environment |
| Center-to-Center Spacing | 10.65 in. |
| Type of Fuel | B&W 15 x 15, MK B11 and earlier designs, 5.0 weight percent maximum nominal enrichment |
| Rack Assembly Dimension and Weights | 8 x 11 85.5 x 117 x 176 - 24,200 lbs. |
| | 8 x 12 - 85.5 x 128 x 176 - 26,400 lbs. |

The pool outline and rack arrangements are shown in Figure 9-3 and Figure 9-4.

### 9.1.2.2 Spent Fuel Storage - Oconee 3

The Spent Fuel Pool serving Oconee Unit 3 has been re-racked to increase the spent fuel storage capacity to 822 fuel assemblies, plus 3 additional storage spaces for failed fuel containers, through the use of neutron absorbing racks. This modification is pursuant to License Amendment Nos. 123, 123, and 120 for License Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station.

# 9.1 FUEL STORAGE AND HANDLING

## 9.1.1 NEW FUEL STORAGE

New fuel will normally be stored in the spent fuel pool serving the respective unit.

New fuel may also be stored in the fuel transfer canal. The fuel assemblies are stored in five racks in a row having a nominal center-to-center distance of 2 ft 1-3/4 inches. One rack is oversized to receive a failed fuel assembly container. The other four racks are normal size and are capable of receiving new fuel assemblies.

New fuel may also be stored in shipping containers.

## 9.1.2 SPENT FUEL STORAGE

### 9.1.2.1 Spent Fuel Storage - Oconee 1, 2

The Spent Fuel Pool common to Oconee Units 1 and 2 has been re-racked to increase the spent fuel storage capacity to 1312 fuel assemblies through the use of neutron absorbing racks. This modification is pursuant to License Amendment Nos. 90, 90 and 87 for License Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station.

#### 9.1.2.1.1 Design Bases

The Spent Fuel Pool designed for Oconee 1 and 2 is an integral part of the Oconee 1 and 2 Auxiliary Buildings and conforms to Safety Guide 13, "Fuel Storage Design Basis." The fuel pools were designed for tornado wind and missiles, turbine generator missile, and seismic conditions as listed in Table 3-12. The Spent Fuel Pools were analyzed for the postulated cask drop accident as described in Section 3.8.4.4, "Design and Analysis Procedures" on page 3-128.

The spent fuel pool is constructed of reinforced concrete lined with stainless steel plate. The fuel pool concrete, reinforcing steel, liner plate and welds connecting the liner plate to the fuel pool floor concrete embedments are analyzed based on consideration of the new racks and additional fuel. Design criteria including loading combinations and allowable stresses are in compliance with Oconee FSAR Section 3.8.4, "Other Seismic Class I Structures" on page 3-127 for Class I structures. The determination of Ta (abnormal thermal load condition to be used in combination with E′) is based on the failure of one pump or cooler during normal operating conditions.

The function of the spent fuel storage racks is to provide for storage of spent fuel assemblies in a flooded pool, while maintaining a coolable geometry, preventing criticality, and protecting the fuel assemblies from excess mechanical or thermal loadings.

A list of design criteria is given below:

1. The racks are designed in accordance with the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," dated April 14, 1978 and revised January 18, 1979.
2. The racks are designed to meet the nuclear requirements of ANSI N210-1976. The effective multiplication factor, $K_{eff}$, in the spent fuel pool is less than or equal to 0.95, including all uncertainties and under all credible conditions.

# CHAPTER 9. AUXILIARY SYSTEMS

The Auxiliary Systems required to support the reactor during normal operations and servicing of the Oconee Nuclear Station are described in this section. Some of these systems have also been described and discussed in Chapter 6, "Engineered Safeguards" on page 6-1, since they serve as engineered safeguards. The information in this section deals primarily with the functions served by these systems during normal operation.

The design of the Auxiliary Systems has included consideration of system sharing, where feasible, between the three Oconee Nuclear Station units. This section describes the equipment for each unit and states where equipment is shared.

The majority of the components in these systems are located within the Auxiliary Building. Those systems connected by piping between the Reactor Building and the Auxiliary Building are equipped with Reactor Building isolation valves as described in Chapter 6, "Engineered Safeguards" on page 6-1.

# LIST OF FIGURES

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**Table 8-3 (Page 2 of 2). Single Failure Analysis for the Keowee Hydro Station**

| | Component | Malfunction | Comments & Consequences |
|---|---|---|---|
| 8. | Keowee Hydro Unit Emergency Startup and Switching Circuits from Oconee | Loss of one | No Consequence, since independent and redundant underground signal cables are provided. |

**Table 8-3 (Page 1 of 2). Single Failure Analysis for the Keowee Hydro Station**

| | Component | Malfunction | | Comments & Consequences |
|---|---|---|---|---|
| 1. | Keowee Hydro Units | Loss of one | (a) | One emergency power source would be lost; however, the other unit would supply 100% of emergency power load. |
| | | | (b) | If the 13.8 kV underground feeder were connected to the unit which was lost, it would also be lost; however, the other unit would supply power through the stepup transformer and the 230 kV switching station to the startup transformers and the underground feeder could be transferred by the Oconee operator to the running unit. |
| 2. | Generator Circuit Breakers and Buses | Loss of one | | Same as 1 above. |
| 3. | Stepup Transformer, Low Side Buses, 230 kV Overhead Line and PCB-9 | Loss of one | | Both hydro units would be separated from the 230 kV switching station; however, one hydro unit would supply emergency power through the 13.8 kV underground circuit. |
| 4. | 13.8 kV Underground Feeder Circuit Breaker, Cables, or Transformer | Loss of one | | One circuit of emergency power would be lost; however, both units could supply emergency power over the 230 kV overhead line. |
| 5. | Keowee Hydro Unit Automatic Startup and Unit Control Systems | Loss of one unit's system | | Same as 1 above. |
| 6. | Keowee Hydro Unit 125V DC Control Battery, Panelboard, Feeders, etc. | Loss of one | | Same as 1(a) above. |
| 7. | Keowee Hydro Unit Emergency Startup and Switching Logic | Loss of one | | Same as 1(a) above. |

## 8.4.3 REFERENCES

1. J. F. Stolz (NRC) to H. B. Tucker, Letter, Review of Adequacy of Station Electric Distribution System Voltages for Oconee Nuclear Station, Units Nos. 1, 2, and 3 (enclosing NRC SER and EG&G TER) Washington, D.C., March 1983.

2 **THIS IS THE LAST PAGE OF THE CHAPTER 8 TEXT PORTION.**

# 8.4 ADEQUACY OF STATION ELECTRIC DISTRIBUTION SYSTEM VOLTAGES

## 8.4.1 ANALYSIS

Each offsite power source was analyzed to the onsite distribution system under maximum and minimum load conditions with the offsite power sources at maximum and minimum anticipated voltages. The analysis included the transient effects on the Class 1E equipment from starting a large Class 1E and non-Class 1E load. The maximum voltage expected at the 4kV bus is slightly higher than the equipment rating. However, this voltage does not have detrimental effects on plant loads or motor feeder circuits. When voltage drops are accounted for, the maximum equipment terminal voltage is within the equipment rating. The minimum analyzed bus voltages shown in the analysis are high enough to account for feeder voltage drops that exist between the bus and the loads. The minimum equipment terminal voltage is within the equipment rating. It has been established that the 4160 volt, 600 volt and 208 volt emergency loads will operate within allowable voltage limits when supplied from the offsite power system.

Test were performed in accordance with NRC guidelines for verification of voltages and currents for the Unit 3 distribution system while the unit auxiliary transformer of that unit supplied 100% of the normal full power operating loads. The measured voltage values were compared with calculated voltage values, and in all cases, the measured values were acceptably close to the analyzed voltage values (0.21-0.28% for the 4 kV buses; within 0.33% for 600 volt buses; and within 1.05 to 1.73% for the 208 volt buses). This test verifies the accuracy of the analysis for the steady-state condition. The verification tests on Unit 3 are applicable to Units 1 and 2 also, since they employ identical equipment and distribution systems. Therefore, no separate tests are required on Units 1 and 2.

## 8.4.2 CONCLUSIONS

1. The voltages are within the operating limits of Class IE equipment for projected combinations of plant load and offsite power grid conditions provided one startup transformer is used for one unit.
2. Spurious separation from the offsite power system due to the operation of voltage protective relays will not occur (with the offsite grid voltage within its expected limits) as a result of starting safety loads.
3. It has been determined (by analysis) that no potential for either a simultaneous or consequential loss of both offsite power sources exists.

## 8.3.3 REFERENCES

1. Unistrut Corporation General Engineering Catalog No. 6, 1966, Page 11.
2. Unistrut Corporation Catalog KUR4P-2, Page 16.

3 5. Containment isolation valves fail closed on loss of air or power, can be manually closed, or have
3 diverse closure ability from the SSF as required in NUMARC 8700.

3 6. Restoration of power is accomplished from the unit control room by manual closure of switchgear
3 breakers.

3 Stripping the non-essential inverters from the 125VDC system allows operation of the TDEFWP and its
3 associated controls in the unit control room for 4 hours. However, the SSF ASW system remains the
3 licensing and design basis commitment for decay heat removal during the SBO event.

3 The 4 hour coping duration is derived from NUMARC 8700 based on meteorological data, grid stability,
3 switchyard features, and availability/reliability of emergency power sources. A program to control SSF
3 availability/reliability has been implemented to ensure at least a value of 95% as stated in the
3 Supplemental SER. The program is based on monitoring Unit 2 CCW intake piping supplying suction to
3 the SSF Auxiliary Service Water pump, the diesel engine cooling, and the SSF HVAC. Additionally,
3 controls are implemented so that planned maintenance on the SSF and Keowee does not occur
3 simultaneously.

#### 8.3.2.2.1 Single Failure Analysis of the 125 Volt DC Instrumentation and Control Power System

As shown in Table 8-5, the 125 Volt DC Instrumentation and Control Power System is arranged such that a single fault within either system does not preclude the Reactor Protective System, Engineered Safeguards Protective System, and the engineered safeguards equipment from performing their safety functions.

#### 8.3.2.2.2 Single Failure Analyses of the 125 Volt DC Keowee Station Power System

The 125 Volt DC Keowee Station Power System is arranged such that a single fault within either unit's system does not preclude the other unit from performing its intended function of supplying emergency power.

#### 8.3.2.2.3 Single Failure Analysis of the 120 Volt Vital Power Buses

The 120 Volt Vital Power System is arranged such that any type of single failure or fault will not preclude the Reactor Protective System, Engineered Safeguards Protective System, and engineered safeguards equipment from performing their safety functions. There are four independent buses available to each unit, and single failure within the system can involve only one bus. A single failure analysis is presented in Table 8-6.

3 #### 8.3.2.2.4 Station Blackout Analysis

3 Station Blackout (SBO) is the hypothetical case where all off-site power and both Keowee hydro-electric
3 units are lost. Electrical power is available immediately from the battery systems and within 10 minutes
3 from the SSF diesel generator. This event was originally included in FSAR section 15.8.3. As
3 documented in the NRC Safety Evaluation Report (SER) dated March 10, 1992 and the NRC
3 Supplemental SER dated December 3, 1992, Oconee Nuclear Station is in compliance with 10 CFR 50.63
3 and conforms to the guidance of NUMARC Report 8700 and Regulatory Guide 1.155. This regulation
3 requires that a licensed nuclear power plant demonstrate the ability to achieve safe shutdown from 100%
3 reactor power by ensuring containment integrity and adequate decay heat removal for a calculated
3 duration. The licensee must also demostrate that the required equipment be able to withstand the
3 resulting operating environment. The temperature of the control room and other areas where extensive
3 manual operations occur, shall not exceed habitability requirements of 120°F. Station blackout is not a
3 design basis event. Therefore, the SBO scenario is not concurrent with any design basis event or single
3 failures.

3 Oconee is capable of coping with a SBO by the following means:

3 1. The SBO duration is 4 hours by application of NUMARC 8700 guidance.

3 2. The SSF is the alternate AC (AAC) source.

3 3. The SSF Auxiliary Service Water system is the design basis source of decay heat removal. Actuation
3 of the Emergency CCW System is not required since the inventory in the CCW piping is sufficient for
3 4 hour operation of the SSF/ASW system.

3 4. The non-essential inverters (KI, KU, and KX) are manually stripped from the Vital 125VDC System
3 within 30 minutes to reduce the electrical heat loads of the unit control complex. Refer to FSAR
3 Selected Licensee Commitment 16.8.1. The resulting temperature in the unit control room does not
3 exceed the habitability requirement of 120°F. Therefore, command and control remain in the unit
3 control room to allow completion of restoration procedures as required in the Supplemental SER
3 dated December 3, 1992.

Feeder breaker open

Group 6, 7, 16, 17 for each of four vital inverters and panelboards

4 Inverter input voltage low

Inverter output voltage low

Bypass voltage low

Inverter bypassed

Panelboard voltage low (60%)

Group 8, 18, 19 for computer, ICS and auxiliary inverters and panelboards

4 Inverter input voltage low

Inverter output voltage low

Bypass voltage low

Inverter bypassed

Panelboard voltage low (60%)

### 8.3.2.2 Analysis

The 125 Volt DC Instrumentation and Control Power System and the 125 Volt AC Vital Power System are designed such that upon loss of power supplies no interactions exist between Reactor Protection Systems, Engineered Safeguards Protection Systems, and control systems that would preclude these systems from performing their respective functions. Also, any interactions between units as a result of the loss of power supplies does not preclude the safety and control systems of any unit from fulfilling their function. This is verified by safety analyses and is shown in Table 8-5, Table 8-6, and Table 8-7.

The ungrounded dc system has detectors to indicate when there is a ground existing on any leg of the system. A ground on one leg of the dc system will not cause any equipment to malfunction. Simultaneous grounds on two legs of the system may cause all energized equipment to drop out if the grounds are of sufficiently low resistance. This may be momentary if the grounded circuit is cleared by its
1 circuit breaker or sustained if the grounded circuit is not cleared by its circuit breaker.

Each battery is sized to carry the continuous emergency load for a period of one hour in addition to
2 supplying power for the operation of momentary loads during the one hour period. The Station Blackout
3 (Section 8.3.2.2.4, "Station Blackout Analysis" on page 8-25) coping strategy which manually strips
2 non-essential loads from the 125 Volt I&C Power System within 30 minutes into the event allows for the
2 operation of the equipment required during the scenario for four hours.

In normal operation the batteries are floated on the buses, and assume load without interruption on loss of a battery charger or ac power source.

The lead-acid batteries are tested to prove their ampere-hour capacity. Inservice periodic checks of the status of each cell is made through battery hydrometer log readings and cell voltage. Temperature readings are used to adjust hydrometer readings.

panelboard to test the operability of the system without affecting normal lighting. All associated lighting units are incandescent.

*8.3.2.1.7.2 Engineered Safeguards AC Lighting System*

The Engineered Safeguards AC Lighting System, which is normally de-energized, provides lighting in the following parts of the Auxiliary Building: control room, cable room, equipment room, stairs, exits, corridors, hot machine shop, spent fuel pool room, fuel unloading area, decontamination rooms, pump and tank room areas, fan and ventilation rooms of roof elevation, penetration rooms, and purge rooms. The stairs and platforms in the Reactor Building are also provided lighting to enable personnel to leave or enter the entire building. Power is provided from two engineered safeguards 600 volt ac control centers through two 600/208Y/120 volt ac dry type transformers which in turn feed each of two panelboards located in the equipment room area. The engineered safeguard lighting is energized automatically by undervoltage sensing relays monitoring the normal 600 volt ac feeder voltage.

**8.3.2.1.8 DC and AC Vital Power System Monitoring**

Failure and/or misoperation of all dc and ac vital power system equipment is being monitored on two local alarm annunciators located in the equipment room near most of the vital equipment. Several variables within each piece or redundant group of equipment are being monitored on one of the local
4 panels, with any alarm from each group being retransferred to a common group alarm in the control room. The control room alarms alert the operator if an alarm condition occurs on any piece or group of equipment, or if power is lost to the local alarm monitoring equipment.

The DC bus tie breakers, battery breakers and standby charger breaker position indication contacts; the standby charger trouble contact; and the computer, ICS and auxiliary inverter isolating diode trouble contacts are monitored directly in the control room.

The other vital alarms are divided into two separate and independent monitoring systems. Alarms for equipment which have battery 1CA for their primary source of power are maintained physically and electrically separate from battery 1CB powered equipment. For example, the distribution center, isolating diodes, breakers, panelboards, inverters and transfer switches associated with battery 1CA are alarmed on local and remote annunciators which are physically and electrically separated from the annunciators being used for monitoring battery 1CB associated systems.

Specifically, the variables being monitored locally with a composite alarm from each of the 17 groups being taken to the control room are as follows:

Group 1 and 11 for each of the two dc buses

System Ground

Charger trouble

Charger output breaker tripped

Bus voltage low (123 V dc)

Group 2, 3, 4, 5, 12, 13, 14, 15 for each of eight isolating diodes

Fuse blown

Diode failure

Input breaker open

Output breaker open

inverters. Each inverter has the synchronizing capability to permit synchronization with the regulated buses.

For each unit, each of the four redundant channels of the nuclear instrumentation and reactor protective system equipment is supplied from a separate bus of the four redundant buses. Also for each unit, each of the three redundant channels of the engineered safeguards protective system is supplied from a separate bus of the four redundant buses. The two engineered safeguards actuation power buses are supplied from separate vital power buses.

#### 8.3.2.1.5 240/120 Volt AC Uninterruptible Power System

For each unit, four uninterruptible power systems are provided to supply power.

They are:

1. The unit's Integrated Control System (ICS) power system, which is 120 volt ac, single phase.
2. The unit's Auxiliary Power System (APS) which is 120 volt ac, single phase.
3. The unit's original design Computer Power System (CPS), which is 240/120 volt ac, single phase.
4. The units' new Computer Power System (KOAC), which is 240/120 volt ac, single phase.

Each of these first three systems consist of a static inverter, with redundant 125 volt dc supplies from separate 125 volt dc buses, circuit breakers and distribution panelboards. The fourth system consists of a static inverter with a 250 volt dc supply from a single 250 volt dc bus, circuit breaker, and distribution panelboard. Also, a static transfer switch is provided in each system as a means for automatic transfer of system loads to the alternate ac regulated power system should the inverter become unavailable. The output of each inverter is synchronized with the ac regulated power system through the static switch in order to minimize transfer time from inverter to the regulated supply.

In addition, an automatic transfer switch is provided in the ICS power system as a means for automatic transfer of system loads to the alternate ac regulated power system should the static transfer switch become unavailable.

#### 8.3.2.1.6 240/125 Volt AC Regulated Power System

For each unit, a system is provided to supply instrumentation, control, and power loads requiring regulated ac power. It also serves as an alternate power source to both the vital power panelboards and to the uninterruptible power panel boards. The system consists of two distribution panels, two regulators, and two transformers fed from separate motor control centers. These systems are shown in Figure 8-8.

#### 8.3.2.1.7 Emergency Lighting System

For each unit, two separate emergency lighting systems are provided; namely, an Emergency 250 Volt DC Lighting System and a separate Engineered Safeguards 208Y/120 Volt AC Lighting System. These two systems are separate and distinct.

##### *8.3.2.1.7.1 Emergency 250 Volt DC Lighting System*

The 250 Volt DC Lighting System, which is normally de-energized, provides operating level lighting in the control room and lighting at selected stairs and corridors in the Auxiliary, Turbine, and Reactor Buildings. The emergency lighting is energized automatically by an undervoltage sensing relay mounted on the individual panelboards located in their associated areas. Control power for the undervoltage transfer circuit is provided from the 250 volt dc station batteries. A test button is also provided at each

failure with a back voltage of 50 volts continuous or 800 volts for 10 seconds. Since the battery under-voltage relay alarms are 123 vdc, only a voltage difference between battery voltages (back voltage) less than 9 volts can occur undetected, and, with one battery in the network completely discharged, the back voltage seen by an isolating diode assembly would be 25 volts. With a back voltage of 25 volts on a monitor assembly, the current flow from battery system to battery system would be less than 0.5 amps.

An alarm relay, connected to an individual control room annunciator point, is provided in each isolating diode assembly to advise the operator of diode trouble in the particular assembly in difficulty. The alarm system is designed to be void of nuisance tripping.

Test provisions are included in each isolating diode assembly to allow the in-service checking of the operability of individual diode monitors, and, in addition, to allow the out of service periodic checking of the peak inverse voltage capability of each individual diode. The latter test can be conducted on one isolating diode assembly with the other diode assembly in the network in operation. Breakers on the input and output of each isolating diode assembly are provided for complete isolation during maintenance and testing of an assembly.

#### 8.3.2.1.2 125/250 Volt DC Station Power System

For each unit a separate 125/250 volt dc power system is supplied. Each system consists of three 125/250 volt dc power supply battery chargers, a three conductor, metalclad distribution center assembly, and two 125 volt dc batteries. This arrangement provides 125 volts from "P" bus to "PN" bus, 125 volts dc from "PN" bus to "N" bus, and 250 volt dc from "P" bus to "N" bus. Loads on this system are basically the 250 volt dc power loads of units. Each 125 volt dc half of a bus section normally is supplied from one of the 125 volt dc power supply battery chargers with the associated 125 volt dc battery floating on the bus. The batteries supply the load without interruption should the battery charger or the ac source fail. A bus tie with normally open double breakers is provided between the three units' distribution center bus sections to backup a battery when it is removed for servicing. One standby 125 volt dc power supply battery charger is provided between each pair of the 125 volt dc batteries for servicing and to "backup" the normal power supply battery chargers.

#### 8.3.2.1.3 125 Volt DC Keowee Station Power System

For each Keowee hydro unit a separate 125 volt dc power system is supplied. Each system consists of one 125 volt dc power supply battery charger, one 125 volt dc, two conductor, metalclad distribution center assembly and one 125 volt dc battery. A bus tie with normally open double circuit breakers is provided between the two distribution center bus sections to "backup" a battery when it is removed for servicing. One standby 125 volt dc battery charger is also provided between the two 125 volt dc batteries for servicing. The batteries, battery charger and distribution center associated with one unit are physically separated in separate enclosures from those associated with the other unit.

#### 8.3.2.1.4 120 Volt AC Vital Power Buses

Figure 8-5 shows the arrangement of the 120 volt ac vital power buses. For each unit, four redundant 120 volt ac vital instrument power buses are provided to supply power in a predetermined arrangement to vital power, instrumentation, and control loads under all operating conditions. Each bus is supplied separately from a static inverter connected to one of the four 125 volt dc control power panelboards described in Section 8.3.2.1.1, "125 Volt DC Instrumentation and Control Power System" on page 8-20. Upon loss of power from 125 volt dc bus DCA or DCB, the affected inverter is supplied power from a 125 volt dc bus of another unit through dc control power panelboards and transfer diodes of the affected
2 125 volt dc panelboard. A tie with breakers is provided to each of the 120 volt vital ac buses from the alternate 120 volt ac regulated bus to provide backup for each vital bus and to permit servicing of the

excessive. Steps have also been taken to insure that no additional cables are routed through trays which are already overfilled.

The cable tray fill criterion for those trays containing power cables allows only one single layer of power cables to be routed in any tray, and, in general, separation of one-quarter the diameter of the larger cable is maintained between adjacent power cables within a tray. The cable spacing may vary between tiedown points due to cable snaking or cables entering/exiting a tray; however, if cables touch, the contact is limited to approximately two feet.

## 8.3.2 DC POWER SYSTEMS

### 8.3.2.1 System Descriptions

For each nuclear unit, two separate dc power systems are provided; namely, a 125 volt dc system provides a source of reliable continuous power for control and instrumentation for normal operation and orderly shutdown for each unit, and a separate 125/250 volt dc system is provided to supply large power loads for each unit. These systems are shown in Figure 8-5, Figure 8-8 and Figure 8-9. For each Keowee hydro unit, separate and independent dc power systems are provided to assure a source of reliable continuous power for normal and emergency operation. These systems are shown in Figure 8-6.

#### 8.3.2.1.1 125 Volt DC Instrumentation and Control Power System

For each unit, two independent and physically separated 125 volt dc batteries and dc buses are provided for the vital instrumentation and control power system. The dc buses are two conductor metalclad distribution center assemblies. Three battery chargers are also supplied, with two serving as normal supplies to the bus sections with the associated 125 volt dc battery floating on the bus. The batteries supply the load without interruption should the battery chargers or the ac source fail. Each of the three battery chargers are supplied from the redundant 600 volt ac engineered safeguards motor control centers of each unit. One of these three battery chargers serves as a standby battery charger and is provided for servicing and to backup the normal power supply chargers. A bus tie with normally open breakers is provided between each pair of dc bus sections to "backup" a battery when it is removed for servicing.

Four separate 125 volt dc instrumentation and control panelboards are also provided for each unit. Each panelboard receives its dc power through an auctioneering network of two isolating diode assemblies. One assembly is connected to the unit's 125 volt distribution system and the other assembly is connected to another unit's 125 volt distribution system. The functions of the diode assemblies are to discriminate between the voltage level of the two dc distribution systems, to pass current from the dc system of higher potential to the instrumentation and control panelboard connected on the output of the diode assemblies, and to block the flow of current from one dc distribution system to the other.

Each isolating diode assembly is composed of a series-parallel network of four diodes in each polarity leg of the dc supply to the panelboard it serves. With this series-parallel arrangement of diodes, either an open circuited or short circuited diode can be tolerated without affecting the operability of the diode assembly. The individual diodes are sized for a continuous current of 500 amperes with the maximum panelboard load current being 304 amps. Each diode is also rated for continuous operation with a peak inverse voltage of 800 volts.

Continuous monitoring of each diode is provided in the design of each isolating diode assembly to detect a shorted or open circuited diode. Since each individual monitor is connected across the diode it monitors, a complete failure analysis was conducted to assure that a failed component in the monitor does not prevent the detection of diode trouble. Factory tests are conducted to check monitor operability under varying voltage levels. The monitors are designed to operate continuously without component

These barriers will be attached to the bottom of the upper tray and fitted around cables which may pass through the barrier.

A minimum of five inches rail to rail separation will be maintained between all vertical trays on Oconee 2 and 3.

## 8.3.1.5 Cable Derating and Cable Tray Fill

### 8.3.1.5.1 Cable Derating

All cables are selected using conservative margins with respect to their current carrying capacities, insulation properties, and mechanical construction. Cable insulations in the Reactor Building are selected to minimize the effects of radiation, heat, and humidity. Appropriate instrumentation cables are shielded to minimize induced voltage and magnetic interference.

Power cables are derated based on IPCEA recommendations for interlocked armor power cables when installed with one-quarter cable diameter spacing in cable trays.

Studies of heating due to $I^2R$ loss in the cables were made. It was determined that the worst case was tray section 1ME8 which contained 322 cables. Cables were classed in three groups: control, control power and instrumentation. Losses were determined by conservative means and were found to be a total of 1.3 watts per lineal foot of tray. Assuming that one cable dissipates 36% of the total heat and that this cable is in the center of a nine inch pile of cable, its maximum temperature would be only 14°C above the ambient cable spreading room temperature, even though the insulation qualities of the cable pile were assumed to be almost perfect. No air flow was assumed through the cables; therefore, the addition of barriers does not alter the heating calculations. Due to the small amount of heat generated and since all cable used in this area is rated 90°C, these temperatures will have no detrimental affect on adjacent cables or on cables in other trays.

Temperature measurements have been made periodically at ten selected locations for the first-year of operation. These locations are where the tray over-fill is the most severe.

Overload protection for cables is very closely related to the basic power and control systems designs. The 4 kV power systems are protected by electro-mechanical overcurrent relays and solid state type ground relays. The relays are selected for the loads protected and the cables are sized based on the maximum currents which these relays should allow without tripping for the loads they are protecting. The 600 volt load centers are used to feed individual motor control centers. The feeder breakers used are furnished with long-time and instantaneous electromechanical or short-time trip elements. Cables to each breaker are sized in coordination with the trip elements selected for that particular breaker. Small motor loads at the 600 volt and 208 volt levels are generally handled through combination motor starters located in motor control centers. Short circuit protection for the load is provided by molded case circuit breakers with magnetic trip devices while overcurrent protection is provided by standard starter overload elements sized for the application. On small engineered safeguard motor loads two of the three overload elements are oversized for cable protection rather than motor protection and are wired in the contactor trip circuit. The third element is sized for motor protection but is wired to alarm only. This is based on the premise that the motor should operate even if motor damage does occur. Cable sizing is based on maximum service factor loading of the motor.

### 8.3.1.5.2 Cable Tray Fill

Early cable tray requirements were based on types of cable which had been used in the past which were primarily not armored. Armored cable was used at Oconee to achieve better mechanical protection and fire retardance. This caused the trays to fill faster than anticipated and in several locations the fill became

3 Calculations were made using existing loadings on one of the heaviest loaded HC-18 hangers. Detailed inventory lists of all cables in each tray section have been maintained, and from this list it was determined that the hanger was loaded as follows:

| | |
|---|---|
| Level H (top) | 112.6 pounds |
| G | 155.0 pounds |
| F | 282.0 pounds |
| E | 364.0 pounds |
| D | 194.5 pounds |
| C | 149.5 pounds |
| B | 97.0 pounds |
| A (bottom) | 196.0 pounds |

Calculations with these loads show that stresses reached were 373 psi under steady state conditions and 11,000 psi during an earthquake. These stresses are actually slightly lower than the original calculation. This is due to several factors. First, although some trays are loaded heavier than the assumed 200 pounds per foot, some of the trays are considerably under the 200 pounds per foot. Secondly, many of the tray sections which are volumewise overfilled are not overloaded from a weight standpoint because Oconee control cables have generally been randomly placed in the tray which has caused many voids to exist.

Overfilled trays were examined and it was determined that section 1ME8 contains 120.4 pounds of cable per linear foot. The tray manufacturers' safe load chart (Reference 2 on page 8-27) states that 24 inch tray with 9 inch rung spacing will support a load of 215 pounds per foot with a 2.2 safety factor. The tray used has an ultimate strength of 473 pounds (2.2 x 215). With an existing load of 120.4 pounds the minimum safety factor is 3.8. Therefore, the present tray system is capable of supporting the weight of the cable even with the existing overfilled conditions and the additional fire retardant barriers.

*8.3.1.4.6.2 Cable Separation*

Control, instrumentation, and power cables are applied and routed to minimize their vulnerability to damage from any source.

2 Our criteria for routing cables requires that mutually redundant safety related cables be run in separate trays. Trays are spaced vertically in the cable room a minimum of 10 inches apart and in some cases redundant cables are in vertically adjacent trays. It should be pointed out that the cable armors used provide excellent mechanical and fire protection which would not be provided with conventional, unarmored cable systems. An early warning fire detection system has also been provided in this area.

Wire and cables related to engineered safeguards and reactor protective systems are routed and installed to maintain the integrity of their respective redundant channels and protect them from physical damage. Power and control cables for redundant auxiliaries or services are run by different routes to reduce any probability of an accident disabling more than one piece of redundant equipment. Floor sleeves are filled with a fire retardant material.

It is our intent wherever physically possible to utilize metallically armored and protected cables systems. By this we mean the use of rigid and thin wall metal conduit, aluminum sheath cables, bronze armored control cables, steel interlocked armor power and control cables, and either interlocked armor or served wire armored instrumentation cables. With this type construction fire stops as such are not required.

Where overfill situations exist in Oconee 1 between vertically adjacent cable trays to the extent that the top cable in the lower tray is within three inches of the bottom cable in the tray immediately above, a one-eighth of an inch fire retardant fiberglass reinforced polyester barrier will be placed between the trays.

protection. All transformers are covered by automatic water spray systems. Transformers are well spaced to minimize their exposure to fire, water, and mechanical damage.

#### 8.3.1.4.2 Switchgear and Load Centers

The 6900 volt switchgear, 4160 volt switchgear, and 600 volt load centers are located in areas to minimize exposure to mechanical, fire, and water damage. This equipment is coordinated electrically to permit safe operation of the equipment under normal and short circuit conditions. Metalclad construction is used throughout for personnel and equipment protection.

The 4160 volt main feeder bus, switchgear sections, and standby power bus switchgear sections are located in a Class I enclosure. The redundant engineered safeguards 4160 volt switchgear bus sections and their associated 600 volt switchgear bus sections, motor control centers, etc. are located within the turbine building and auxiliary building below the operating floor level. They are located in areas with separation and protection to minimize exposure to mechanical, fire and water damage. This equipment is coordinated electrically to permit safe operation under normal and short circuit conditions. The engineered safeguards system is of Class I seismic design.

#### 8.3.1.4.3 Motor Control Centers

The 600 volt motor control centers are located in the areas of electrical load concentration. Those associated with the turbine-generator auxiliary system in general are located below the turbine-generator operating floor level. Those associated with the nuclear steam supply system are located in the auxiliary building. Motor control centers are located in areas with separation and protection to minimize their exposure to mechanical, fire and water damage.

#### 8.3.1.4.4 Batteries, Chargers, Inverters, and Panelboards

The 125 volt dc instrumentation and control power system batteries of a unit are physically separated in separate enclosures from batteries of another unit to minimize their exposure to any damage. The battery chargers and associated dc bus sections and switchgear of a unit are located in separate rooms from battery chargers and associated dc bus sections of another unit in the auxiliary building and physical separation is maintained between redundant equipment.

#### 8.3.1.4.5 Metal-Enclosed Bus

Metal-enclosed buses are used for all major bus runs where large blocks of current are to be carried. They are also routed to minimize exposure to mechanical, fire, and water damage.

#### 8.3.1.4.6 Cable Installation and Separation

##### *8.3.1.4.6.1 Cable Installation*

Loadings and stresses in the cable tray and hangers were examined under both the steady state and seismic conditions. Hanger type HC-18, which is one of the most heavily laden hangers, was checked. It
3 supports eleven to twelve trays vertically, some of which are overfilled.

Original hanger calculations were based on the assumption that all hangers would be loaded at 200 pounds per tray. Under those conditions maximum stresses reached in any hanger member are 382 psi during steady state conditions and 11,100 psi under seismic loadings. This stress occurred in the angle brace which was added due to lateral seismic forces. Since the material for hangers and braces used is rated at 25,000 psi allowable (Reference 1 on page 8-27), hangers are stressed at less than 50% of their allowable loading under the worst conditions.

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# CHAPTER 10. STEAM AND POWER CONVERSION SYSTEM

# 10.1 SUMMARY DESCRIPTION

The Steam and Power Conversion System (SPCS) is designed to convert the heat produced in the reactor to electrical energy.

The superheated steam produced by the steam generators is expanded through the high pressure turbine and then exhausted to the moisture separator reheaters. The moisture separator section removes the moisture from the steam and the two stage reheaters superheat the steam before it enters the low pressure turbines. The steam then expands through the low pressure turbines and exhausts into the main condenser where it is condensed and returned to the cycle as condensate. The heat rejected in the main condenser is removed by the Condenser Circulating Water System.

The first stage reheaters are supplied with steam from the A bleed steam line and the condensed steam is cascaded to the B feedwater heaters. The second stage reheaters are supplied with main steam and the condensed steam cascades to the A feedwater heaters. Heat for the feedwater heating cycle is supplied by the moisture separator reheater drains and by steam from the turbine extraction points.

The hotwell pumps take suction from the condenser hotwell and discharge to the condensate polishing demineralizers. Downstream of the polishers, the condensate flows through the condensate coolers, generator water coolers, hydrogen coolers, condenser steam air ejectors and the S.P.E. steam seal condenser before discharging to the suction of the condensate booster pumps. After the condensate booster pumps, the condensate passes through three stages of low and intermediate pressure feedwater heaters (F, E, and D). The flow passes through the C feedwater heater, then it divides to the suction of the steam generator feedwater pumps. The steam turbine driven main feedwater pumps deliver feedwater through two stages of high pressure feedwater heaters (B and A), to a single feedwater distribution header where the feedwater flow is divided into two lines to the steam generators.

The safety-related features of the SPCS include the main steam piping from the steam generators up to and including the main turbine stop valves. The steam lines supplying the emergency feedwater pump
0 turbine are also safety-related. The feedwater piping from the feedwater control valves to the steam generator and the Emergency Feedwater System (EFWS) is also safety-related.

4 SPCS safety-related instrumentation includes the steam generator level instruments which input to the
4 EFWS steam generator level control and steam generator dryout protection circuits. Another QA control
4 circuit monitors UST level and closes the UST to Hotwell makeup valves regardless of hotwell level in
4 order to maintain a minimum 6 foot level in the UST for an EFWS suction source. Other UST level
4 indication is used for post-accident monitoring. The only additional safety-related instrumentation
5 associated with the SPCS is the steam generator outlet pressure used for post-accident monitoring and as
5 input to the Main Steam Line Break (MSLB) circuitry (NSM 1,22873).

# 10.2 TURBINE-GENERATOR

## 10.2.1 DESIGN BASES

The turbine-generator converts the thermal energy of steam produced in the steam generators into mechanical shaft power and then into electrical energy. Each unit is operated primarily as a base loaded unit with an output of 866 MW net, but may be used for load following when required.

A maximum rate of turbine load change of 10 percent full load per minute is permitted by the Turbine Electro-Hydraulic Control (EHC) System without restriction if the minimum load involved in the change is 46 percent full load or greater. Below 46 percent full load, the maximum rate of change is still 10 percent full load per minute, but the total load change may be restricted by turbine metal temperature considerations.

The rate of change of reactor power is limited to values consistent with the characteristics of the Reactor Coolant System and its control systems. These limitations are imposed by the Integrated Control System on the Steam and Power Conversion System. See Section 7.2, "Reactor Protective System" on page 7-7 and Table 7-1.

Turbine-generator functions under normal, upset, emergency, and faulted conditions are monitored and controlled automatically by the Turbine Control System (TCS). The TCS includes redundant mechanical and electrical trip devices to prevent excessive overspeed of the turbine-generator. Additional external trips are provided to ensure operation within conditions that preclude damage to the turbine-generator. A standby manual control system is also provided in the event that the automatic control system is not available.

## 10.2.2 DESCRIPTION

Each unit's turbine-generator consists of a tandem (single shaft) arrangement of a double-flow high-pressure turbine, and three identical double-flow low pressure turbines driving a direct-coupled generator at 1800 rpm. The turbine is operated in a closed feedwater cycle which condenses the steam, and the heated feedwater is returned to the steam generators. The system is designed to utilize the entire output from the Nuclear Steam Supply System. The turbine generator is manufactured by the General Electric Company of Schenectady, New York.

The flow of main steam is from the steam generators to the high-pressure turbine through four stop valves and four control valves. After expanding through the high-pressure turbine, exhaust steam passes through
1 external moisture separators and two stage steam-to-steam, shell and tube type reheaters. 'A' bleed
1 extraction steam from the high-pressure turbine is supplied to the first reheater stage tube bundle in each reheater. Main steam is supplied to the second reheater stage tube bundle in each reheater. Reheated steam is admitted to the three low pressure turbines and expands through the low-pressure turbines to the main condensers.

Bleed steam for the six stages of feedwater heating is provided from the following sources:

| Heater | Extraction Source |
|---|---|
| A | H-P turbine |
| B | H-P turbine |
| C | H-P turbine exhaust |
| D | L-P turbines |
| E | L-P turbines |
| F | L-P turbines |

Each main generator is a 1038 MVA, 1800 rpm, direct connected, 3 phase, 60 cycle, 22,000 volt conductor cooled synchronous generator rated at 0.90 P.F., and 0.50 SCR at a maximum hydrogen pressure of 60 psig. Generator rating, temperature rise, and class of insulation are in accordance with IEEE standards. Excitation is provided by a shaft driven alternator with its output rectified.

## 10.2.3 TURBINE DISK INTEGRITY

### 10.2.3.1 Materials Selection

Turbine wheels and rotors are made from vacuum melted or vacuum degassed Ni-Cr-Mo-V alloy steel by processes which minimize flaw occurrence and provide adequate fracture toughness. Tramp elements are controlled to the lowest practical concentrations consistent with good scrap selection and melting practices, and consistent with obtaining adequate initial and long life fracture toughness for the environment in which the parts operate. The turbine wheel and rotor materials have the lowest Fracture Appearance Transition Temperatures (FATT) and highest Charpy V-notch energies obtainable, on a consistent basis from water quenched Ni-Cr-Mo-V material at the sizes and strength levels used. Since actual levels of FATT and Charpy V-notch energy vary depending upon the size of the part and the location within the part, etc., these variations are taken into account in accepting specific forgings for use in turbines for nuclear application. Charpy tests essentially in accordance with Specification ASTM A-370 are included.

### 10.2.3.2 Fracture Toughness

Suitable material toughness is obtained through the use of materials described in Section 10.2, "Turbine-Generator" on page 10-5 to produce a balance of adequate material strength and toughness to ensure safety while simultaneously providing high reliability, availability, and efficiency during operation. Bore stress calculations include components due to centrifugal loads, interference fit, and thermal gradients where applicable. The ratio of material fracture toughness, $K_{IC}$ (as derived from material tests on each wheel or rotor) to the maximum tangential stress for wheels and rotors at speeds from normal to 115 percent of rated speed (the highest anticipated speed resulting from a loss of load is 110 percent) is at least $2\sqrt{in.}$

Turbine operating procedures are employed to preclude brittle fracture at start-up by ensuring that the metal temperature of wheels and rotors is adequately above the FATT and is sufficient to maintain the fracture toughness to tangential stress ratio at or above $2\sqrt{in.}$

### 10.2.3.3 Turbine Design

The turbine assembly is designed to withstand normal conditions and anticipated transients including those resulting in turbine trip without loss of structural integrity. The design of the turbine assembly meets the following criteria:

1. Turbine shaft bearings are designed to retain their structural integrity under normal operating loads and anticipated transients, including those leading to turbine trips.

2. The multitude of natural critical frequencies of the turbine shaft assemblies existing between zero speed and 20 percent overspeed is controlled in the design and operation so as to cause no distress to the unit during operation.
3. The maximum tangential stress in wheels and rotors resulting from centrifugal forces, interference fit and thermal gradients does not exceed 0.75 of the yield strength of the materials at 115 percent of rated speed.

### 10.2.3.4 Pre-service Inspection

The pre-service inspection program is as follows:

1. Wheel and rotor forgings are rough machined with minimum stock allowance prior to heat treatment.
2. Each finish machined wheel and rotor is subjected to 100 percent volumetric (ultrasonic), surface, and visual examinations using General Electric acceptance criteria. These criteria are more restrictive than those specified for Class 1 components in the ASME Boiler and Pressure Code, Sections III and V, and include the requirement that subsurface sonic indications are either removed or evaluated to assure that they will not grow to a size which compromises the integrity of the unit during the service life.
3. All finish machined surfaces are subjected to a magnetic particle test with no flaw indications permissible.
4. Each fully bucketed turbine rotor assembly is spin tested at or above the maximum speed anticipated following a turbine trip from full load.

## 10.2.4 SAFETY EVALUATION

The turbine-generator and all related steam handling equipment are of conventional proven design. This unit automatically follows the electrical load requirements from station auxiliary load to turbine full load. There is also a tie-in with Keowee Hydro Station which can carry auxiliary load upon turbine trip.

Under normal operating conditions, there are no radioactive contaminants present. It is possible for this system to become contaminated only through steam generator tube leaks. In this event, radioactivity in the Main Steam System is detected and measured by monitoring condenser air ejector off-gas which is released through the unit vent and by monitoring the steam generator blowdown samples.

No radiation shielding is required for the components of the turbine-generator and related steam handling equipment. Continuous access to the components of this system is possible during normal conditions.

The condensate polisher demineralizers are available to remove radioactive particulates from the condenser hotwell in the event of primary to secondary leakage.

The turbine-generator is designed and manufactured in accordance with General Electric Company design criteria and manufacturing practices, procedures, and processes, as well as its Quality Assurance Program.

# 10.3 MAIN STEAM SYSTEM

## 10.3.1 DESIGN BASES

The Main Steam System is designed to achieve the following:

1. Provide steam flow requirements at main turbine inlet design conditions.
2. Dissipate heat from the Reactor Coolant System following a turbine and/or reactor trip by dumping steam to the condenser and atmosphere.
3. Provide steam as required for:
   a. Main and emergency feedwater pump turbines
   b. Condenser air ejectors
   c. Main feedwater pump turbine seals
   d. Steam reheaters
   e. Miscellaneous auxiliary equipment
4. Conform to applicable design codes presented in Table 3-2.
5. Allow visual in-service inspection.
6. Protect adjacent equipment against heat damage.

The following portions of the system are designed to withstand seismic loading (criteria for seismic loading defined in Appendix 1C);

1. Main steam lines from steam generator through the turbine stop valves
2. Main steam line relief valves
3. The steam supply from the main steam lines to the emergency feedwater pump turbine including valve AS-38 and that portion of the auxiliary steam supply downstream from the valve
4. Through the first valve of all other lines leaving the main steam lines

## 10.3.2 DESCRIPTION

Main steam is generated in the two steam generators by feedwater absorbing heat from the Reactor Coolant System. Main steam is conveyed by two lines, one per steam generator, to the turbine inlet valves. A pressure equalization and steam distribution header is connected to each main steam line upstream of the turbine inlet valves. The Main Steam System from the steam generators through the turbine stop valves (including connected piping through the first isolating valve of connecting lines) is Duke Piping Class F. All other piping is Class G. Main Steam piping inside the Reactor Building is considered Reg. Guide 1.26 Quality Group B for purposes of Inservice Inspection. See Figure 10-1, Figure 10-2, and Figure 10-3.

Eight self-actuated safety valves are located on each main steam line (a total of sixteen) to prevent overpressurization of the Main Steam System under all conditions. The valves are designed to pass 105 percent of the Engineered Safeguard Design (ESD) steam flow at a pressure not exceeding 110 percent of the system design pressure (1050 psig). See Table 3-1 and Table 3-2 for applicable codes.

The main steam lines and the main and emergency feedwater lines are the only lines of the Steam and Power Conversion System which penetrate the Reactor Building. These lines can be isolated by the turbine stop valves and the main and emergency feedwater line valving. Each of the lines leaving the main steam lines before the turbine stop valves has motor operated valves to complete the isolation of a steam generator. These lines are:

1. Steam bypass to condenser and steam supply for auxiliary steam header (See Figure 10-1 for line to auxiliary steam header)
2. Supply to feedwater pump turbines and condenser air ejectors
3. Supply to steam reheaters
4. Supply to emergency feedwater pump turbine.

4 The arrangement of the valving and parallel piping shown schematically in Figure 10-1 minimizes
4 blowdown of both steam generators from a single leak in the system. For a majority of the Main Steam
4 system, a postulated piping break would only depressurize one steam generator. However, if the break
4 were to occur in either the steam supply to the auxiliary steam header or the emergency feedwater pump
4 turbine cross-connect, blowdown of both steam generators could result. The motor operated valves that
4 are used to isolate the leak require operator action to close and may not get closed until the steam
4 generators are considerably depressurized. This situation has been analyzed and shown to have
4 consequences that are bounded by the consequences of the accident in Section 15.13 (Reference 1).

The steam supply for the emergency feedwater pump turbine (Figure 10-1) will come from either of two sources (the main steam line or the auxiliary steam header) and exhaust to the atmosphere. The solenoid operated valve which controls the steam shutoff valve MS-93 is de-energized on loss of both main feedwater pumps, thus opening the steam shutoff valve. As the steam shutoff valve leaves the closed position, a limit switch starts the emergency feedwater pump turbine bearing oil pump.

## 10.3.3 EVALUATION

The Main Steam System delivers the generated steam from the outlet of the steam generators to the various system components throughout the Turbine Building without incurring excessive pressure losses. Steam is generated at approximately 50°F superheat conditions. Functional requirements of the system are as follows:

1. Achieve optimum pressure drop between the steam generators and the turbine steam stop valves.
2. Assure similar steam conditions between each steam stop valve and between each steam generator.
3. Achieve adequate piping flexibility for acceptable forces and moments at equipment interfaces.
4. Assure adequate draining provisions for startup and for operation with saturated steam.

The once-through nature of this recirculating steam condensate cycle is utilized in the removal of contaminants resulting from steam generator leaks, since it allows the flow through the steam generator to be subjected to purification. Radioactive contaminants will be removed by the Powdex polishing demineralizers as described for the control of impurities (Section 10.2, "Turbine-Generator" on page 10-5). Provision is made for transferring the backwashed resins, when they contain radioactive material, as radwaste.

Trips, automatic corrective actions, and alarms will be initiated by deviations of system variables within the Steam and Power Conversion System. In the case of automatic corrective action in the Steam and Power Conversion System, appropriate automatic corrective action will be taken to protect the Reactor Coolant System. The more significant malfunctions or faults which cause trips, automatic actions or

2 alarms in the Steam and Power Conversion System are listed in Section 10.4.6.5, "Instrumentation
2 Application" on page 10-20.

2

The analysis of the effect of loss of full load on the Reactor Coolant System is discussed in Section 14.1, "Organization of Test Program" on page 14-3. Analysis of the effects of partial loss of load on the Reactor Coolant System is discussed in Section 7.2, "Reactor Protective System" on page 7-7.

The effects of inadvertent steam relief or steam bypass are covered by the analysis of the steam line break given in Section 15.13, "Steam Line Break Accident" on page 15-49. The effects of an inadvertent rapid throttle valve closure are covered by the loss of full load discussion in Section 15.8, "Loss of Electric Load Accidents" on page 15-27.

Following a turbine trip, the reactor will trip. The safety valves will relieve excess steam until the output is reduced to the point at which the steam bypass to the condenser can handle all the steam generated. Steam may also be released to the atmosphere through a manually operated angle-body control valve on each main steam line.

Pressure relief is required at the system design pressure of 1050 psig, and the first safety valve bank will be set to relieve at this pressure. The design pressure is based on the operating pressure of 925 psia plus a 10 percent allowance for transients and a 4 percent allowance for blowdown. Additional safety valve banks will be set at pressures up to 1104 psig, as allowed by the ASME Code. Pressure relief is provided by eight safety valves on each main steam line, and the valve relief pressures are:

| Number of Valves | Relief Pressure (psig) |
|---|---|
| 1 | 1050 ± 11 |
| 1 | 1065 ± 11 |
| 1 | 1080 ± 11 |
| 1 | 1090 ± 11 |
| 2 | 1100 ± 11 |
| 2 | 1104 ± 11 |

The relief valve capacity (13,105,000 lb/hr total for 16 valves) is such that the energy generated at the reactor high power level trip setting can be dissipated through this system.

## 10.3.4 INSPECTION AND TESTING REQUIREMENTS

Steam from the steam generators is admitted to the turbine through four cast 24 inch main steam stop valves, arranged in parallel and located in the main steam lines upstream from the turbine control valves (See Figure 10-1). In the event of a steam line rupture accident, the stop valves serve to isolate the unaffected steam generator.

The main steam stop valve is designed for tight seating throughout its life. The valve stem extends through a guide bushing which centers the disc on the stem with some degree of freedom, permitting self alignment of the disc on its seat. The valve seat and disc have spherical seating surfaces so that perfect contact is made even if they are not in precise alignment. The use of stem sealing permits relatively large stem to bushing clearance, minimizing the possibility of stem sticking. The seating surfaces of the valve and the stem seal are hardened inlay contact areas which resist erosion and mechanical damage and assure tightness. A coarse-mesh internal screen strainer with removable fine mesh startup strainer is provided for each stop valve.

The main steam stop valves are fail-safe, requiring hydraulic pressure to open and closure is spring-assisted. Each stop valve has two positions: fully open and fully closed. Each stop valve will be tested periodically (while the turbine is in operation) and any tendency of the valve to remain open in opposition to a control signal will be detected. A stop valve will be disassembled, inspected, and required corrective action taken when a valve test warrants such action. Stop valves are also disassembled and inspected during turbine inspections.

The main steam stop valves are designed and tested to assure proper functioning. In the event of a steam line rupture accident, the two stop valves serving the unaffected steam generator will close in the presence of steam flow in the normal direction, thus precluding the possibility of reverse flow through the other two stop valves.

The motor operated valve on each of the lines connected to the main steam lines can be tested for
4 operability when the unit is shutdown. These valves, the main steam stop valves, and the check valves
4 that are provided in the two branch lines that cross-connect the main steam lines prevent uncontrolled
4 blowdown of the unaffected steam generator in the unlikely event of a main steam line break. Their ability to close will be verified at periodic intervals.

Proper operation of the emergency feedwater pump and turbine, the steam shutoff valve (Figure 10-1), and the valves in the emergency feedwater supply to the steam generators (Figure 10-8) can be demonstrated when the unit is shutdown. The emergency feedwater pump and turbine, and the steam shutoff valve can be tested anytime by utilizing the recirculation test line. Proper functioning of the emergency feedwater supply will be verified at periodic intervals.

## 10.3.5 WATER CHEMISTRY

### 10.3.5.1 Secondary Side Water Chemistry

4 Hydrazine and/or carbohydrazide is added to the feedwater downstream of the condensate polishing
4 demineralizers for oxygen control. An alternate addition point is directly to the condensor hotwell.

3 Ethanolamine or an alternate approved amine is used to increase pH to minimize formation of corrosion products.

The condensate polishing demineralizer utilizes the Powdex process, developed by Graver Water Conditioning Company as a unique, high quality water purification system. The Powdex units will function as a combination demineralizer and high purity filter, treating 100 percent of the feedwater flow to the steam generator under conditions of startup, reduced load, and 70% normal full-load operation.

The Powdex process uses extremely fine particle-size (60-400 mesh) ion exchange resins which are applied to the external surface of specially design filter elements. The rapid ion exchange rates of these fine resins allows the use of a thin coating (1/16 inch to 1/2 inch) on the elements and permits a greater utilization of the ultimate capacities of the resins than is the case of bead type resins.

The Powdex resins are not chemically regenerated for repeated use but are replaced with fresh resins upon exhaustion. This continued resin replacement allows complete flexibility in the selection of the most advantageous type of resin or combination of resins for the removal of specific impurities.

The resins are selected for the effective removal of dissolved metallic cations and also anions such as halides, silicates, and sulfates. In addition, the resin will also remove by filtration the suspended and colloidal trace impurities such as corrosion products.

Exhaustion of each batch of resins is monitored and is indicated by an increase in pressure drop or by a decrease in treated water quality. Exhausted resins are backwashed from the units and pumped to a disposal facility.

# 10.4 OTHER FEATURES OF STEAM AND POWER CONVERSION SYSTEM

## 10.4.1 MAIN CONDENSER

### 10.4.1.1 Design Bases

The main condenser is designed to condense turbine exhaust steam for reuse in the steam cycle. The main condenser also serves as a collecting point for various steam cycle vents and drains to conserve condensate which is stored in the condenser hotwell. The condenser also serves as a heat sink for the
4 Turbine Bypass System which is capable of handling 25 percent of rated main steam flow. Rejected heat is removed from the main condenser by the Condenser Circulating Water System.

### 10.4.1.2 System Description

The main condenser consists of three surface type deaerating condenser shells with each shell condensing the exhaust steam from one of the three low pressure turbines. The condenser shells are of conventional shell and tube design with steam on the shell side and circulating water in the tubes. One low pressure feedwater heater is mounted in the neck of each of the condenser shells. The combined hotwells of the three condenser shells have a water storage capability equivalent to approximately 10 minutes of full load operation (nominally 142,000 gallons). The internal condenser design provides for the effective condensing of steam, scavenging and removal of noncondensable gases, and the deaeration of the condensate. Impingement baffles are provided to protect the tubes from incoming drains and steam dumps.

The main condenser can accept a bypass steam flow of approximately 18 percent of rated main steam flow without exceeding the turbine high backpressure trip point with design inlet circulating water temperature. This bypass steam dump to the condenser is in addition to the normal duty expected.

### 10.4.1.3 Safety Evaluation

The main condenser is not assigned a safety class as it is not required for a safe reactor shutdown. The inventory of radioactive contaminants in the main condenser is a function of primary to secondary system leakage.

### 10.4.1.4 Tests and Inspections

The main condenser is tested in accordance with the Heat Exchange Institute Standards for Steam Surface Condensers. Manways in the condenser provide access to waterboxes, tube sheets, shell, and hotwell for inspection, repair and tube plugging. The pH, sodium content, and oxygen content of the condensate leaving the hotwell is continuously monitored. However, the condensate system's polishing demineralizer
0 will remove many of the contaminants and thus reduce the impact of leakage from the Condenser
0 Circulating Water upon final feedwater chemistry.

The effect of the leakage upon unit operation can range from no effect in the case of small leakage to unit shutdown for condenser repair in the case of severe leakage.

### 10.4.1.5 Instrumentation Application

The main condenser hotwell is equipped with level control devices for automatic control of condensate makeup and rejection. On low water level in the hotwell, control valves supply condensate from the
5 upper surge tanks to the hotwell by gravity. A QA-1 control circuit monitors UST level and closes the
4 UST to Hotwell valves regardless of Hotwell level in order to maintain a minimum 6 foot water level in
4 the UST for an EFWS suction source. A low hotwell level alarm is provided in the control room. Loss of condenser vacuum will trip the respective unit turbine. All instrumentation for this system is operating instrumentation, and none is required for safe shutdown of the reactor.

## 10.4.2 MAIN CONDENSER EVACUATION SYSTEM

### 10.4.2.1 Design Bases

The Main Condenser Evacuation System is designed to remove noncondensable gases and air inleakage from the steam space of the three shells of the main condenser. The Main Condenser Evacuation System consists of the Condenser Steam Air Ejector System and the Main Vacuum System which are shown on Figure 10-5 for Oconee 1, 2 and 3.

### 10.4.2.2 System Description

The Condenser Steam Air Ejector System consists of three condenser steam air ejectors (CSAE) per unit. Normally each CSAE draws the noncondensable gases and water vapor mixture from one of the three main condenser shells to the first air ejector stage. The mixture then flows to the intercondenser where it is cooled to condense the water vapor and motive steam. The second air ejector stage draws the uncondensed portion of the cooled mixture from the intercondenser and compresses it further. The compressed mixture then passes through the aftercondenser where it is cooled and more water vapor and motive steam are condensed. The intercondenser drains back to the main condenser and the aftercondenser drains to the condensate storage tank.

The Main Vacuum System consists of three main vacuum pumps connected to the condenser crossties on the Condenser Steam Air Ejector System to allow the main vacuum pumps to evacuate the main condenser, the main turbine casing, and the upper surge tanks during startup. These pumps are only used during startup since normal operation requires the use of the CSAE only.

### 10.4.2.3 Safety Evaluation

The Main Condenser Evacuation System is not assigned a safety class as it is not required for a safe reactor shutdown. Control functions of the Main Condenser Evacuation System indirectly influence Reactor Coolant System operation in that upon loss of vacuum the main condenser no longer provides a heat sink.

The noncondensable gases and water vapor mixture discharged to the atmosphere from the Main Condenser Evacuation System are not normally radioactive; however, in the event of primary to secondary system leakage due to a steam generator tube leak, it is possible for the mixture discharged to become radioactive. A full discussion of the radiological aspects of a primary to secondary leakage including radioactive discharge rates under postulated design conditions is discussed in Chapter 11, "Radioactive Waste Management" on page 11-1 and Chapter 15, "Accident Analyses" on page 15-1.

#### 10.4.2.4 Tests and Inspections

Proper operation of the Main Condenser Evacuation System is verified during unit startup, and is subject to periodic inspections by plant operating personnel. A flowmeter is provided in the discharge piping of each CSAE. Periodic readings of these flowmeters will indicate whether or not the air inleakage to the condenser is within acceptable limits. These readings will also indicate the operating effectiveness of the CSAE.

#### 10.4.2.5 Instrumentation Applications

A radiation monitor is provided in the exhaust line from the CSAE's with remote indicator, recorder, and alarm located in the Control Room. Local indicating devices for pressure, temperature, and flow are provided as required for monitoring system operation. All instrumentation for this system is operating instrumentation and none is required for safe shutdown of the reactor.

### 10.4.3 TURBINE GLAND SEALING SYSTEM

#### 10.4.3.1 Design Bases

The Turbine Gland Sealing System (TGS) is designed to seal the annular openings around the rotor shafts of the high pressure (HP) and low pressure (LP) main turbines and the feedwater pump (FDWP) turbines where the shafts emerge from the shell casings. All seals for the LP main turbines and the exhaust end seals for the FDWP turbines are designed to prevent the leakage of atmospheric air into the turbines since the turbine shell pressures at these seal locations are subatmospheric at all unit loads. All seals for the HP main turbine and the steam inlet end seals for the FDWP turbines are designed to prevent atmospheric air leakage into the turbines since the turbine shell pressures at these seal locations vary from subatmospheric to above atmospheric as these turbines progress from startup to normal operation.

### 10.4.4 TURBINE BYPASS SYSTEM

#### 10.4.4.1 Design Bases

The Turbine Bypass System (TBS) is designed to reduce the magnitude of nuclear system transients following large turbine load reductions by dumping main steam directly to the main condenser and/or to the atmosphere, thereby creating an artificial load on the reactor.

### 10.4.5 CONDENSATE CLEANUP SYSTEM

#### 10.4.5.1 Design Bases

(See Section 10.3.5.1, "Secondary Side Water Chemistry" on page 10-12)

#### 10.4.5.2 System Description

The CCS for each unit consists of five powdered resin condensate polishing demineralizer vessels. Normally, all five vessels will be in service. There is also a separate regeneration skid for each unit consisting of a recirculation/resin feed tank and a precoat pump.

The current revision of the SGOG PWR Secondary Water Chemistry Guidelines (Chapter 3, "Design of Structures, Components, Equipment, and Systems" on page 3-1) and vendor recommendations are used to derive the operating specifications which are addressed in the Chemistry Section Manual.

The condensate polishing demineralizers are designed for automatic operation following mode initiation. This means that the operator is required to initiate each Phase of operation but, having once done so the polishers will operate automatically through that mode (i.e., backwash, precoat, filter, and hold). A polisher cycle continues until the effluent water quality deteriorates or until a predetermined differential pressure drop is reached across the polisher. When either of these conditions occur, the polisher will be backwashed.

Each polisher vessel normally requires backwashing of spent resin approximately every 20 days. The vessels are backwashed to the Powdex sump. Each backwash takes about 15,000 gallons of water and contains roughly 17 cubic feet of spent resin. The resin water mixture is pumped to the Radwaste Facility Powdex Backwash Tank or to the chemical treatment ponds.

The handling of polisher backwash during and after a steam generator primary to secondary leak is discussed in Chapter 11, "Radioactive Waste Management" on page 11-1.

### 10.4.5.3 Safety Evaluation

The Condensate Cleanup System is not assigned a safety class as it is not required for a safe reactor shutdown. The condensate polishing demineralizer vessels and all regeneration equipment are located in the Turbine Building. The spent resin and water mixture discharged to the backwash sump from the polisher vessels is not normally radioactive; however, disposal of the mixture in the event of a primary to secondary leakage is discussed in Chapter 11, "Radioactive Waste Management" on page 11-1.

### 10.4.5.4 Tests and Inspections

Proper operation of the Condensate Cleaning System is verified during unit startup, and is subject to periodic inspections by plant operating personnel.

## 10.4.6 CONDENSATE AND MAIN FEEDWATER SYSTEMS

### 10.4.6.1 Design Bases

The Steam and Power Conversion System for each unit is designed to remove heat energy from the reactor coolant in the two steam generators and convert it to electrical energy. The closed feedwater cycle condenses the steam and the heated feedwater is returned to the steam generators. The system is designed to utilize the entire output from the Nuclear Steam Supply System.

A maximum rate of turbine load change of 10 percent full load per minute is permitted by the turbine Electro-Hydraulic Control (EHC) system without restriction if the minimum load involved in the change is 46 percent full load or greater.

Below 46 percent full load, the maximum rate of change is still 10 percent full load per minute, but the total load change may be restricted by turbine metal temperature considerations.

The rate of change of reactor power is limited to values consistent with the characteristics of the Reactor Coolant System and its control systems. These limitations are imposed by the Integrated Control System on the Steam and Power Conversion System.

The Condensate and Main Feedwater Systems are shown in Figure 10-6 and Figure 10-7.

### 10.4.6.2 System Description

The closed cycle feedwater heaters are half-size units (two parallel strings), with the exception of "F" heater. There are three "F" heaters, one in each condenser neck. Deaeration is accomplished in the condenser.

All three hotwell pumps, two of the three one-half capacity condensate booster pumps and both of the main feedwater pumps are in normal use. Each of two main feedwater pumps is more than one-half capacity.

3

The main steam lines and the main and emergency feedwater lines are the only lines of the Steam and Power Conversion System which penetrate the Reactor Building. These lines can be isolated by the turbine stop valves and the normal and emergency feedwater line valving.

Feedwater supply to the steam generators following a reactor shutdown is assured by one of the following methods:

1. Either of the two main feedwater pumps is capable of supplying both steam generators at full secondary system pressure.
2. The hotwell and condensate booster pump combination has discharge shutoff head of approximately 550 psia. Three sets of half-size pumps are provided. If required, the Turbine Bypass System can be used to reduce secondary system pressure to the point where one of the hotwell and condensate booster pump combinations can supply feedwater to both steam generators.
3. A separate Emergency Feedwater System for each unit will supply feedwater at full system pressure (see Section 10.4.7, "Emergency Feedwater System" on page 10-22).
4. Alternate auxiliary feedwater supplies are available from the Emergency Feedwater System of each of the other units.
5. The Auxiliary Service Water System may be used to maintain steam generator water inventory following steam generator depressurization to remove decay heat in the long term.
6. The SSF Auxiliary Service Water System is capable of supplying both steam generators of all three units at full secondary system pressure.

### 10.4.6.3 Safety Evaluation

The turbine-generator equipment conforms to the applicable ASA, ASME, and IEEE standards.

The design, material, and details of construction of the feedwater heaters are in accordance with the ASME Code, Section VIII, Unfired Pressure Vessels.

The following portions of the system are designed to withstand seismic loading (criteria for seismic loading defined in Chapter 3, "Design of Structures, Components, Equipment, and Systems" on page 3-1):

1. Main steam lines from steam generator through the turbine stop valves.
2. Main steam line relief valves.
3. The steam supply from the main steam lines to the emergency feedwater pump turbine including valve AS-38 and that portion of the auxiliary steam supply downstream from the valve.
4. Through the first valve of all other lines leaving the main steam lines.

5. Both supply lines from the upper surge tank to the emergency feedwater pumps, including piping through the first valve of any connections to these lines.
6. Both discharge lines from the emergency feedwater pumps to the steam generators, including piping through the first valve of any connections to these lines.

The Feedwater System has been reviewed to determine the potential for "water hammer" during anticipated operational occurrences. It has been concluded that the existing Oconee Feedwater System is adequate to prevent flow instabilities. Because design features of the feedwater system preclude the probability of destructive "water hammer" forcing functions resulting from uncovering feedwater lines, no analyses have been performed nor test program conducted regarding this occurrence. The following considerations support this conclusion:

1. Neither the Main nor Emergency Feedwater Systems has horizontal or downward-sloping pipe runs adjacent to the steam generator. The auxiliary piping remains below the level of its junction with the steam generator. The main feedwater line rises above its steam generator connection only after downward and horizontal runs which effectively form a loop seal. Only in the unlikely event of steam generator shell pressure near the vapor pressure of the water in this pipe could a steam void occur.
2. The main and emergency feedwater distribution heads on the steam generator are designed to remain flooded regardless of steam generator water level, and would in any event be self-venting if steam were introduced. The main ring header is fed from the bottom, external to the steam generator, and empties upward through the vertical inlet lines. The auxiliary ring headers on Oconee 1 and 2 are similar in design to the main header. The original Oconee 3 auxiliary header was internal to the steam generator shell, the newly installed header is similar in design to Oconee 1 & 2. None of the feedwater headers can spontaneously drain into the steam generator.
3. Each steam generator has its auxiliary header separate from the main header. Therefore, there is no need to deliver the relatively cool auxiliary feedwater through the normal path for main feedwater. In addition, the QA-1 portions of Main FDW have been analyzed for pressure transient forces due to control valve closure and pump trip resulting from actuation of the Main Steam Line Break Detection and Feedwater isolation circuitry (NSM-1,22873).

### 10.4.6.4 Tests and Inspections

The operating characteristics of the hotwell, condensate booster, and main feedwater pumps are established throughout the operating range by factory tests. The main condensers, the hotwell pumps, the condensate polishing demineralizer vessels, the condenser steam air ejectors, the gland steam condenser, the condensate booster pumps, the feedwater heaters, and the main feedwater pumps are hydrostatically tested to the applicable code or standard.

Manways or removable heads are provided on all heat exchangers to provide access to the tube sheets for inspection and maintenance. A general routine visual surveillance of the system components and piping during operation and maintenance periods for signs of leakage or distress will be performed to verify system integrity.

### 10.4.6.5 Instrumentation Application

Sufficient instrumentation is provided to monitor system performance and to control the system automatically or manually under all operating conditions.

Trips, automatic corrective actions, and alarms will be initiated by deviations of system variables within the Steam and Power Conversion System. In the case of automatic corrective action in the Steam and Power Conversion System, appropriate automatic corrective action will be taken to protect the Reactor

Coolant System. The more significant malfunctions or faults which cause trips, automatic actions, or alarms in the Steam and Power Conversion System are:

#### 10.4.6.5.1 Turbine Trips

Any turbine trip provides an anticipatory reactor trip.

1. Loss of D-C supply to trip circuits
0 2. Low condenser vacuum
5 3. Thrust bearing wear (Units 1 and 3 only)
4. Loss of generator stator coolant (if runback fails)
5. Loss of both main feedwater pumps
6. Turbine overspeed
7. Reactor trip
8. Bearing oil low pressure
9. EHC Hydraulic Fluid low pressure
2 10. Moisture separator high level
5 11. Turbine exhaust hood high temperature (Units 1 and 3 only)
2 12. Manual trip

#### 10.4.6.5.2 Automatic Actions

4 (Also see Integrated Control System Description.)

4 1. Low Water level in Upper Surge Tank

#### 10.4.6.5.3 Principal Alarms

1. Low pressure at condensate booster pump suction
2. Low pressure at feedwater pump suction
3. Low vacuum in condenser
4. Low water level in condenser hotwell
5. High water level in condenser hotwell
6. High water level in steam generator
7. Low water level in steam generator
8. High pressure in steam generator
9. Low pressure in steam generator
10. Low feedwater temperature
11. Electrical malfunctions in the EHC
4 12. Low water level in Upper Surge Tank

### 10.4.6.6 Interactions with Reactor Coolant System

The effects of inadvertent steam relief or steam bypass are covered by the analysis of the steam line break given in Section 15.13, "Steam Line Break Accident" on page 15-49. The effects of an inadvertent rapid throttle valve closure are covered by the loss of full load discussion in Section 15.8, "Loss of Electric Load Accidents" on page 15-27.

Following a turbine trip, the reactor will trip automatically due to anticipatory trip logic. The safety valves will relieve excess steam until the output is reduced to the point at which the steam bypass to the condenser can handle all the steam generated.

In the event of failure of a main feedwater pump, there will be an automatic runback of the power demand. The one main feedwater pump remaining in service will carry approximately 60 percent of full load feedwater flow. If both main feedwater pumps fail, the turbine and reactor will be tripped, and the emergency feedwater pumps started.

1 On failure of a condensate booster pump, the spare condensate booster pump is automatically started.

## 10.4.7 EMERGENCY FEEDWATER SYSTEM

### 10.4.7.1 Design Bases

The Emergency Feedwater (EFW) System assures sufficient feedwater supply to the steam generators of each unit, in the event of loss of the Condensate/Main Feedwater System, to remove energy stored in the core and primary coolant. The EFW System is designed to provide sufficient secondary side steam generator heat sink to enable cooldown from reactor trip at power operation down to cold shutdown conditions. The EFW System may also be required in some other circumstances such as cooldown following a loss-of-coolant accident for a small break. The EFW System is shown in Figure 10-8.

2 The EFW System is designed to start automatically in the event of loss of both main feedwater pumps as
2 indicated by Main Feedwater Pump low hydraulic oil pressure or low feedwater pump discharge header
2 pressure. In addition, low water level in either steam generator, after a 30 second delay to prevent
2 spurious actuations, will start the Motor Driven Emergency Feedwater Pumps. The EFW System will supply sufficient feedwater at a flowrate as much as 450 gal/min to enable the Reactor Coolant System to cool down to conditions at which the Decay Heat Removal System may be operated.

Three EFW pumps are provided, powered from diverse power sources. Two full capacity motor-driven pumps are powered by the emergency A.C. Power System, each supplying feedwater to one steam generator. One turbine-driven pump, supplying feedwater to both steam generators, may be driven by any of three separate steam sources; A Main Steam, B Main Steam, or plant start-up steam (also called the Auxiliary Steam System). Although the total rated capacity of all three EFW pumps is 1780 gal/min, the flow capacity of any one of the pumps is sufficient to enable safe and orderly cooldown of the Reactor Coolant System. Sufficient redundancy and valving are provided in the design of the EFW piping system with isolation and cross-connections allowing the system to perform its safety-related function in the event of a single failure coincident with a secondary pipe break and the loss of normal station auxiliary A.C. power. All automatic initiation logic and control functions are independent from the Integrated Control System (ICS).

The three units are provided with separate EFW Systems. The discharge header of each EFW System is cross connected making each system capable of supplying either unit.

Automatic initiation of the turbine-driven EFW pump is independent of AC power. Based on the required emergency feedwater flow, sufficient inventory of EFW is available for maintaining hot shutdown
4 for at least 67 minutes from both upper surge tanks. The inventory in the upper surge tanks is assured by
4 auto closure of the hotwell makeup control valves on a low upper surge tank level signal. The upper surge tanks and the associated piping from them to the EFW pump suctions are seismically qualified. The condenser hotwell is also seismically qualified with a nominal capacity of 120,000 gallons. However, the condenser hotwell is seismically qualified without any piping connected to it, and not all of the piping from the hotwell to the EFW pump suctions has been seismically qualified.

In the event of a postulated break in the Main Steam or Main Feedwater System inside or outside containment coupled with a single active failure, the EFW System provides sufficient flow to ensure adequate core cooling.

The plant transient which requires the highest Emergency Feedwater System flow, and as such constitutes
2 the Emergency Feedwater design basis transient, is the loss of main feedwater transient. This transient combines the highest heat load, decay heat plus reactor coolant pump heat, with the minimum heat sink due to the instantaneous loss of both main feedwater pumps. A discussion of the demand on the EFW
5 system for each transient follows. The following, with the exception of Steam Line Break (Section
5 10.4.7.1.8, "Steam Line Break" on page 10-25) and Small Break LOCA (Section 10.4.7.1.9, "Small Break
5 LOCA" on page 10-25), should not be considered Design Basis Transients for the entire plant, but for
2 Emergency Feedwater only.

#### 10.4.7.1.1 Loss of Main Feedwater (LMFW)

Those transients which result in losing feedwater delivery from the Main Feedwater/Condensate System are classified as a loss of main feedwater. This initiating event causes a turbine and reactor trip and automatically starts the EFW pumps. Since the reactor coolant pumps remain on, the control valves
0 modulate to control steam generator level at 30 inches. The transient requires feedwater to be delivered at a rate sufficient to remove decay heat and reactor coolant pump heat. One motor driven emergency
4 feedwater pump delivering 400 gal/min. at a steam generator pressure of 1060.5 psia and an EFW temperature of ≤ 130°F will provide adequate heat removal capacity.

3 #### 10.4.7.1.2 LMFW with Loss of Offsite AC Power (LOOP)

3 The loss of offsite AC power causes the reactor to trip, the turbine to trip, and the condensate booster pumps and hotwell pumps to trip causing a loss of main feedwater. The emergency feedwater pumps are actuated on the main feedwater pump trip. Since the reactor coolant pumps have tripped, steam
3 generator level control increases the level setpoint to 240 inches on the extended startup range to promote the natural circulation mode of heat removal. The emergency feedwater control valves open to allow full system flow until the controlling level is attained. Feedwater requirements are determined by core decay heat removal demand. One motor driven EFW pump can deliver sufficient feedwater to meet the demand.

3 #### 10.4.7.1.3 LMFW with Loss of Onsite and Offsite AC Power (Station Blackout)

3 This transient is the result of a station blackout condition. This transient is similar to Case 2 with the additional assumption that the onsite emergency AC power sources have been lost. This results in the
3 loss of the motor driven emergency feedwater pumps. This transient is not a design basis event. The
3 turbine-driven emergency feedwater pump should be available for this event because of its AC power
3 independence; however, the SSF ASW is required to remove the decay heat in this transient. The
3 transient is described in Section 8.3.2.2.4, "Station Blackout Analysis" on page 8-25.

#### 10.4.7.1.4 Plant Cooldown

In addition to providing sufficient heat removal capacity immediately following a transient, the requirements for plant cooldown from full power operation to RCS temperatures where switchover to the Decay Heat Removal System can be accomplished has been analyzed. All heat sources have been included. The average hourly EFW flowrate to meet cooldown rates of 100°F/hr and 50°F/hr down to the switchover temperature of 250°F are given below.

| Time | Cooldown Rate 100°F/hr. | 50°F/hr. |
|---|---|---|
| 0-1 hr | 547 gpm | 480 gpm |
| 1-2 hr | 464 | 390 |
| 2-3.3 hr | 430 | - |
| 2-3 hr | - | 354 |
| 3-4 hr | - | 344 |
| 4-5 hr | - | 331 |
| 5-6 hr | - | 325 |
| 6-6.6 hr | - | 320 |

Cooldown of the RCS is a manual function controlled by the operator such that the EFW flow is throttled to obtain the cooldown rate desired and within Technical Specification and administrative limits.

#### 10.4.7.1.5 Turbine Trip

A turbine trip transient causes a reactor trip. The reactor trip initiates the ICS to control steam generator level at the minimum level so that the main feedwater Pumps are runback. With the main feedwater pumps in an untripped condition, there is no requirement for the EFW system to function.

#### 10.4.7.1.6 Main Steam Isolation Valve Closure

This transient, similar to the turbine trip, does not trip the main feedwater pumps so that the EFW system is not required.

#### 10.4.7.1.7 Main Feedwater Line Break

For a main feedwater line break upstream of the isolation check valve, the transient would have the same response as a loss of main feedwater. A break downstream of the check valve will cause the steam generator to blow down, but will be less severe than a steam line break transient due to less feedwater being delivered to the steam generators. The demand on the EFW system would be for decay heat and reactor coolant pump heat removal via the unaffected steam generator. One motor driven EFW pump has sufficient capacity to perform this function.

#### 10.4.7.1.8 Steam Line Break

A steam line break transient is primarily an overcooling transient. Only after the overcooling has been turned around and after isolation of the affected SG, does the need for heat removal by the intact SG arise. Since the EFW system is capable of delivering to either steam generator, the heat removal demand on the EFW system can be met by one motor driven EFW pump or the turbine driven EFW pump in the event the MFW system is unavailable.

#### 10.4.7.1.9 Small Break LOCA

For certain small break loss of coolant accidents (break sizes less than 0.1 $ft^2$), feedwater is required to remove the decay heat and reactor coolant pump heat which is not relieved through the break. A flow rate of 300 gal/min is adequate to provide this heat removal (Reference 1 on page 10-32). One motor-driven EFW pump has the necessary capacity.

#### 10.4.7.1.10 Summary of Transients

The above transients bound the EFW system performance requirements for all transients.

| Conditions of Transient | Criteria |
|---|---|
| Loss of Main Feedwater<br>3 Loss of Offsite Power<br>Turbine Trip | Peak RCS Pressure<br>≤ 2750 psig<br>No fuel failures |
| Steam Line Break<br>Feedwater Line Break | 10CFR 100 dose limits |
| Small Break LOCA | 10CFR 100 dose limits<br>10CFR 50.56 PCT limit |
| 3 Station Blackout | Not a design basis event |
| Plant Cooldown | 100°F/hr |

As discussed above, the requirements for EFW system performance are determined by the heat removal demand for the loss of main feedwater transient, and the successful cooldown of the RCS to decay heat removal mode. The assumptions utilized in the analysis of the plant response are consistent with those typically assumed in an FSAR analysis and allow for margin to realistic system performance for conservatism.

System initial conditions are consistent with an assumed initial 102 percent power level. Steam generator level is 60 percent corresponding to 38,000 lbs inventory per steam generator. The Turbine Bypass System is not available so that steam relief is by the main steam safety valves at a maximum pressure of 1065 psia. The EFW system is limited to one motor driven EFW pump delivering to one steam generator. The maximum allowable feedwater temperature for the above conditions is 130°F.

4 A loss of main feedwater is initiated assuming no pump coastdown. It is assumed that the turbine and
4 reactor do not trip for one second following the loss of main feedwater pumps. Reactor coolant pumps are left on to maximize the heat input. Decay heat power is based on infinite burnup with a 1.0 multiplier. The EFW system is assumed available at 40 seconds, at which time the steam generator inventory has decreased to 14,000 lbs/SG. For the cooldown part of the transient, all heat sources (decay heat, pump heat, fuel, structural steel, and coolant sensible heat) were included. The feedwater inventory
3 required for a 100°F/hr cooldown to decay heat removal switchover is 94,000 gallons, or 145,000 gallons for a 50°F/hr cooldown. These requirements are well within the available hotwell and upper surge tank
3 capacity (214,000 gallons combined). For cooldown in the recirculation mode, the minimum amount of water in the upper surge tank, condensate storage tank and hotwell is the amount needed for 11 hours of operation per unit. This is based on the conservative estimate of normal makeup being 0.5 percent of throttle flow. Throttle flow at full load, 11,200,000 lbs/hr, was used to calculate the operation time. For decay heat removal, the operation time with the volume of water specified would be considerably increased due to the reduced throttle flow.

### 10.4.7.2 System Description

Each reactor unit is provided with a separate EFW System, as shown in Figure 10-8. Controls for each system are located on the main control room panels. Each EFW System is provided with two full capacity motor driven pumps and one full capacity turbine driven pump. Each of the motor driven pumps normally serves a separate steam generator; the turbine driven pump serves both steam generators.
4 When supplying only one steam generator, a minimum of 400 gpm total EFW flow is required; when
2 supplying both steam generators simultaneously, minimum total EFW flow required is 450 gpm. The
2 EFW pumps will start automatically as outlined below:

2 <u>Motor Driven EFW Pumps (MDEFWP's)</u>:

2 Automatic starting of the MDEFWP's is determined by the position of the control room selector switch
2 for each pump. The MDEFWP's are provided with a four position selector switch which allows the
2 operator to select between Off, Auto 1, Auto 2 and Run. When the selector switch is in the Auto 1
3 position, LOW STEAM GENERATOR WATER LEVEL in either steam generator (OTSG) will start
2 the pump after a 30 second time delay to prevent spurious actuations. When the selector switch is in the
2 Auto 2 position, LOW STEAM GENERATOR WATER LEVEL or LOSS OF BOTH MAIN
2 FEEDWATER PUMPS will start the pump. Loss of both main feedwater pumps is sensed by pressure
2 switches which monitor feedwater pump turbine control oil pressure and feedwater pump discharge header
2 pressure. Loss of both Main Feedwater Pumps actuation may be by either the control oil pressure
2 switches sensing loss of feedwater pumps or low discharge header pressure. This design allows the
2 operator to select the Auto 1 switch position during startup and shutdown conditions when the operating
2 main feedwater pump(s) approach low pump discharge actuation pressures and still have automatic
2 initiation of EFW prior to OTSG dry out.

2 Turbine Driven EFW Pump (TDEFWP):

2 Automatic starting of the TDEFWP is determined by the position of the control room selector switch for
2 the pump. The TDEFWP is provided with a three position-pull to lock selector switch which requires
2 that the control room operator manually take the switch to the OFF position through a deliberate action.
2 The operator can select between Off, Auto and Run. When the selector switch is in the Auto position,
2 LOSS OF BOTH MAIN FEEDWATER PUMPS will start the pump. Loss of both main feedwater
2 pumps is sensed by pressure switches which monitor feedwater pump turbine control oil pressure and
5 feedwater pump discharge header pressure. If a main steam line break signal is present and the selector
5 switch is in AUTO, the TDEFWP will automatically stop and prevent an auto start. The operator can
5 manually start the TDEFWP by placing the selector switch to RUN (NSM-1, 22873).

Once automatically started, the EFW pumps will continue to operate until manually secured by the operator. Each emergency feedwater discharge line to each steam generator is provided with a control
5 valve and check valve. The control valves are normally closed due to steam generator level > 30". The valves are arranged to fail to the automatic control mode upon loss of DC control power to the manual/auto select solenoid. If the selected train of automatic control fails, then the valve would fail open. Also, upon loss of station air, the valves will maintain their position with $N_2$ backup. If $N_2$ backup fails then the valve would fail open. These modes of operation show that emergency feedwater
2 isolation is not possible with valve control circuitry or motive force failure. Open/Closed valve position
2 indication is provided for each control valve in the main control room at the valve manual loader.

5 In automatic, a solenoid valve on each control valve is de-energized, allowing the valve to receive a control air signal for valve modulation in response to steam generator level, independent from the ICS.

The EFW pumps normally discharge into separate lines feeding a separate steam generator through the auxiliary feedwater header.

A flow path is also provided to the upper surge tank dome (connected to the condenser) for minimum
0 recirculation flow and testing purposes. A continuous recirculation flow is provided for the turbine driven pump, limited by fixed orifices. A self-contained automatic recirculation valve is provided for each motor
0 driven pump to assure individual pump minimum flow when needed during operation. A flow path is
0 provided from the discharge of each motor driven pump to the upper surge tank for full flow testing. Power for the motor driven pumps is normally provided by the normal station auxiliary A.C. Power
3 System. During loss of offsite power operation, these pumps are aligned to the Emergency A.C. Power System. Motive steam for the turbine driven pump is provided from either of the two steam generators by main steam lines upstream of the stop valves, and is exhausted to the atmosphere. Either steam supply will provide sufficient steam for turbine operation. Either steam supply may be isolated if

necessary. A check valve is provided in each steam supply line to prevent uncontrolled blowdown of more than one steam generator.

The condensate/feedwater reserves for each unit are normally aligned to the EFW pump suctions. The condensate/feedwater reserve for each unit is maintained among the sources in Table 10-1.

Each of the EFW pumps is supplied with its own independent starting circuit. The independent control circuits are powered by the 125 VDC station batteries. These circuits are actuated by trip of both main feedwater pumps or low discharge pressure of both main feedwater pumps. Feedwater pump trip is detected by low feedwater pump turbine hydraulic control oil pressure for turbine driven and motor driven EFW pumps. Low feedwater pump discharge pressure is determined by pressure switch actuation on decreasing pressure. Each pump is provided with a control switch with which the operator may start the pump manually.

Sufficient indication is provided in the control room to allow the operator to monitor unit parameters during a cooldown. Specific indication provided for the EFW System are listed in Table 10-2.

Discharge flow from the EFW pumps is normally aligned and controlled by control valves FDW-315 and FDW-316. These valves are controlled independently of the Integrated Control System and arranged to fail to the automatic control mode upon loss of DC control power to the manual/auto select solenoid. If the selected train of automatic control fails, then the valve would fail open. Also, upon loss of all station air, the valves will maintain their position with $N_2$ backup. If $N_2$ backup fails, then the valve would fail open. In automatic, the control valve manual/auto select solenoid valves are de-energized, thereby aligning the valve to automatic control and positioning the valve per the automatic setting. Control valves FDW-315 and FDW-316 are modulated by separate control air signals. These valves may be automatically controlled, or manually controlled by the operator to limit or increase feedwater as necessary to maintain feedwater inventory and cooldown rate. A pushbutton is provided for each control valve to allow the individual valve to be placed in either an automatic level control mode or in a manual mode of operation. In automatic, the valves are positioned and controlled by the automatic level control. Independent level transmitters are utilized in the automatic control system. Upon loss of all four reactor coolant pumps, such as during blackout conditions, the level control setpoint is automatically raised to promote natural circulation in the Reactor Coolant System.

Although not normally aligned or utilized in the safety related function of the EFW System, a redundant, separate path of EFW to the steam generators and means of controlling EFW pump discharge flow is provided by startup control valves FDW-35 and FDW-44. This additional flow path is not required for normal EFW System function, but may be aligned manually if necessary or desirable during normal startup or cooldown. Normally closed motor operated valves FDW-38, FDW-47, FDW-374, and FDW-384 can be opened from the control room to provide this additional flow path if required. Control valves FDW-35 and FDW-44 are modulated by control signals based on steam generator water levels by the ICS. As in the case of control valves FDW-315 and FDW-316, the level control setpoint is automatically raised upon loss of all four reactor coolant pumps to promote natural circulation in the Reactor Coolant System.

The steam supply for the EFW pump turbine is provided from either main steam line. Valve MS-93 in the common supply to the turbine will fail open upon loss of station air or power to the normally energized solenoid valve. Upon receipt of a manual or automatic start signal, the solenoid valve will de-energize and immediately start the turbine.

Sufficient valving is provided to allow isolation and cross-connection as required to select and isolate water sources and assure system function in the event of various failures. During normal orderly shutdown as a result of blackout or loss of feedwater, no valve re-alignments or isolation is necessary. All

necessary valves are maintained in normal standby alignment to assure an open flow path for each pump,
2 and to assure piping separation and independence. All manually-operated valves in the piping from the
2 Upper Surge Tanks (UST) to the suction of the EFW pumps are locked open. (Reference 2 on
2 page 10-32)

The motor driven EFW pumps require cooling water for continuous operation. Sufficient cooling water is initiated automatically, upon manual or automatic start of motor driven EFW pumps.

Sufficient alarms are provided to alert the operator of conditions exceeding normal limits. Essential plant parameters are annunciated or alarmed by the process computer in addition to specific EFW System alarms as listed below:

1. Motor driven EFW pumps low suction pressure
2. Steam generator low level alarms
3. Hotwell low level alarms
4. UST low level alarms
5. Low motor driven EFW pump cooling water flow
6. Motor driven EFW pump stator winding high temperature
7. Motor driven EFW pump motor bearing high temperature
8. Motor driven EFW pump bearing high temperature
9. Motor cooler excessive leakage
10. Motor driven EFW pump A auto start blocked
11. Motor driven EFW pump B auto start blocked
12. Turbine driven EFW pump auto start blocked
2 13. Motor driven EFW pump A low level start
2 14. Motor driven EFW pump B low level start

### 10.4.7.3 Safety Evaluation

Feedwater inventory is maintained in the steam generators following reactor shutdown by one of the following methods listed:

1. Either of the two main feedwater pumps is capable of supplying both steam generators at full secondary system pressure.
2. The two EFW motor driven pumps are capable of supplying both steam generators at full secondary system pressure.
3. The single EFW turbine driven pump is capable of supplying both steam generators at full secondary system pressure.
4. Alternate EFW supplies may be available from the EFW Systems of the other Units, capable of supplying both steam generators at full secondary system pressure.
5. The hotwell and condensate booster pump combination has discharge shutoff head of approximately 550 psia. Three pairs of pumps are provided. If required, the Turbine Bypass System can be used to reduce secondary system pressure to the point where one hotwell and condensate booster pump combination can supply feedwater to both steam generators.

6. The Auxiliary Service Water System may be used to maintain steam generator water inventory following steam generator depressurization to remove decay heat in the long term.
7. The SSF Auxiliary Service Water System is capable of supplying both steam generators of all three units at full secondary system pressure.

A sufficient depth of backup measures is provided to allow steam generator water inventory to be maintained by any of the diverse methods listed above. Although redundancy and diversity is provided in the listed measures, the EFW System has been designed with special considerations to enable it to function when conventional means of feedwater makeup may be unavailable.

Redundancy is provided with separate, full capacity, motor and turbine driven pump subsystems. Failure of either the motor driven pumps or the turbine driven pump will not reduce the EFW System below minimum required capacity. Pump controls, instrumentation, and motive power are separate in design. Separate piping subsystems include redundant hotwell, upper surge tank, and condensate supply piping, aligned individually to the separate pump trains. Cross-connection is provided, however, to allow a subsystem to supply all pumps in the event of single failure of a suction piping subsystem. The same design philosophy is included in the discharge piping subsystems.

In order to provide sufficient EFW flow to the intact steam generator to ensure adequate core cooling, and under a main steam or main feedwater break in OTSG A with a single active failure of motor driven emergency feedwater pump B train, the operator must manually close the motor operated isolation valve (FDW-372) or the flow control valve FDW 315 on OTSG A. He is able to do this from the Control Room. The same is true for OTSG B and motor driven emergency feedwater Pump A. The operator has sufficient Control Room indication of steam generator level and pressure and would immediately be aware of such a situation.

Concurrently, the operator would monitor the intact steam generator to assure adequate inventory and secondary heat removal via either Main Feedwater or Emergency Feedwater Systems.

In the event of a postulated break in the Main Steam or Main Feed System, coupled with a single active failure of either one of the three emergency feed water pumps, sufficient flow will occur to provide adequate core cooling.

5 With a postulated break associated with the 'A' OTSG and a failure of the 'B' motor driven emergency
5 feedwater pump, the normal feedwater system will be isolated to both steam generators and the TDEFWP
5 will be inhibited from automatically starting. The TDEFWP can be manually started by placing its
5 control switch to RUN. (Units 1 and 2)

5 With a postulated break associated with the 'A' OTSG and an active failure occurs with the flow control
5 valve (FDW-316), the Main Steam Line Break Circuitry must be disabled by the operator to allow
5 emergency feedwater flow alignment through the main feedwater startup control valves to either the main
5 or auxiliary nozzles. (Unit 1 and 2)

5 In the unlikely event that FDW-315, 316 fail open (on a loss of compressed air and nitrogen), an operator
5 could manually adjust either one of the valves as they are located in the Penetration Rooms which are
5 adjacent to the Control Room.

The spectrum of transients which require EFW system performance for post trip heat removal have been evaluated assuming only one motor driven emergency feedwater pump is available to deliver the necessary feedwater. Any single failure in the three pump-two flowpath EFW system design will not result in only one motor driven EFW pump available, so that this assumption is overly conservative. These analyses verify the acceptability of the Emergency Feedwater System design.

### 10.4.7.4 Inspection and Testing Requirements

A comprehensive test program is followed for the EFW System. The program consists of performance tests in the manufacturers' shops, preoperational tests of the system, and periodic tests of the activation logic and mechanical components to assure reliable performance during the life of the unit.

During unit operation, the EFW System is tested by utilizing the recirculation test line to the upper surge tank dome. Pump head and flow is verified utilizing this method.

### 10.4.7.5 Instrumentation Requirements

Sufficient instrumentation and controls are provided to adequately monitor and control the EFW System. The safety related instrumentation and controls which monitor steam generator level and pressure, automatically start the EFW pumps, and automatically align the supply meet the system requirements for redundancy, diversity and separation. All nonsafety related instrumentation and controls are designed such that any failure will not cause degradation of any safety related equipment function.

## 10.4.8 REFERENCES

1. R. B. Davis (Babcock & Wilcox), letter to B&W 177 FA Owners Group TMI-2 Subcommittee, May 23, 1979.

2 2. W. O. Parker (Duke) letter to H. R. Denton (NRC), April 3, 1981, page 32.

**THIS IS THE LAST PAGE OF THE CHAPTER 10 TEXT PORTION.**

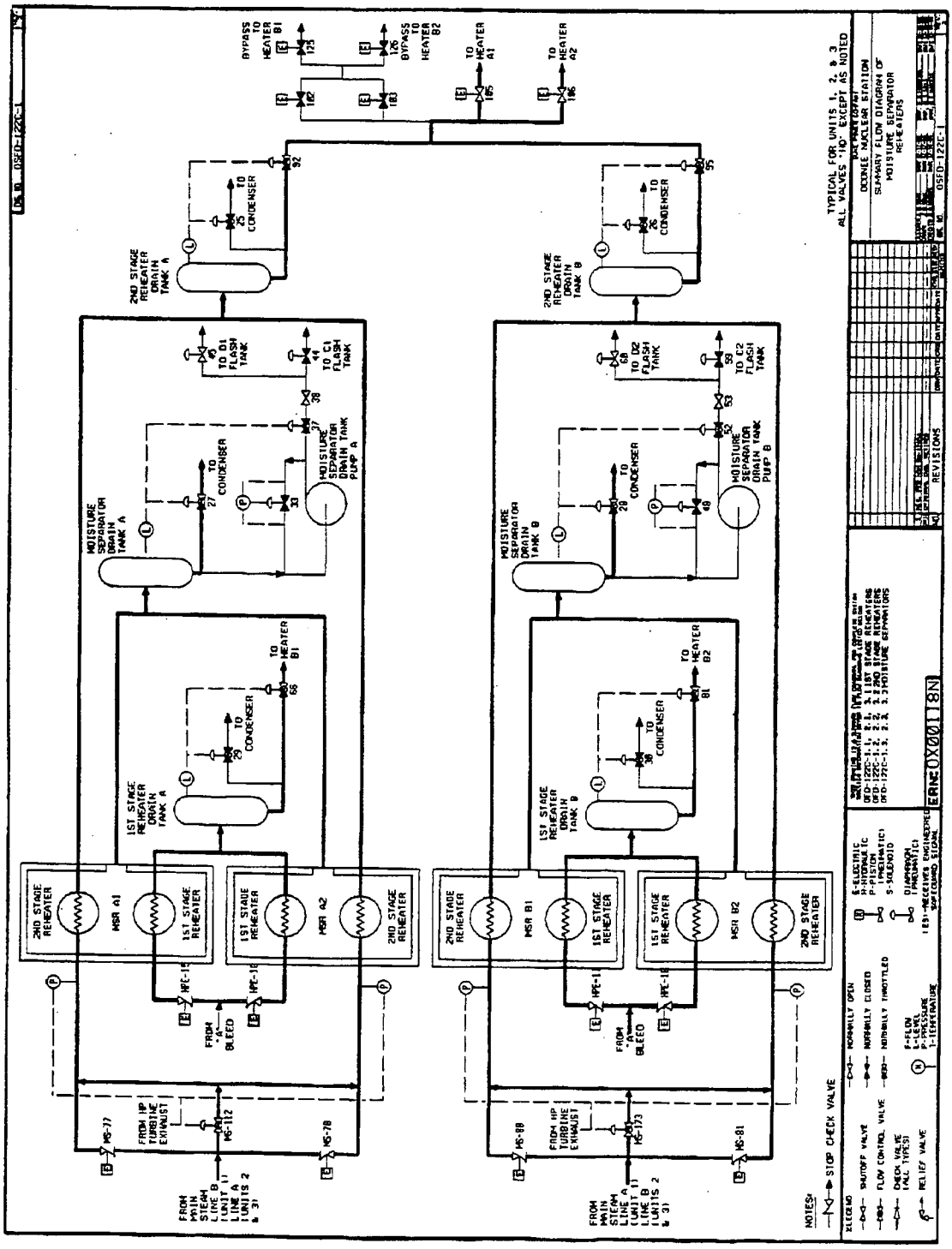


Figure 10-4.
Moisture Separator and Reheater Heater and Drain System

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# CHAPTER 11. RADIOACTIVE WASTE MANAGEMENT

# 11.1 DESIGN BASIS

The liquid and gaseous radioactive waste management systems will be utilized to reduce radioactive liquid and gaseous effluents such that compliance with the dose limitations of the Selected Licensee Commitments is assured. These dose limitations require that:

1. the concentration of radioactive liquid effluents released from the site to the unrestricted area will be limited to 10 times the effluent concentration (EC) levels of 10CFR 20, Appendix B, Table 2;
2. the exposures to any individual member of the public from radioactive liquid effluents will not result in doses greater than the design objectives of 10CFR 50, Appendix I;
3. the dose rate at any time at the site boundary from radioactive gaseous effluents will be limited to: for noble gases; less than or equal to 500 mrem/yr to the whole body and less than or equal to 3000 mrem/yr to the skin; and for iodine-131 and 133, for tritium, and for all radioactive materials in particulate form with half-lives greater than 8 days; less than or equal to 1500 mrem/yr to any organ;
4. the exposure to any individual member of the public from radioactive gaseous effluents will not result in doses greater than the design objectives of 10CFR 50, Appendix I; and
5. the dose to any individual member of the public from the nuclear fuel cycle will not exceed the limits of 40CFR 190 and 10CFR 20.

# 11.2 LIQUID WASTE MANAGEMENT SYSTEMS

2

## 2 11.2.1 DISPOSAL METHODS AND LIMITS

Liquid wastes from the station are disposed of, under continuous radiation monitoring and control, in one of the following three ways depending on the concentration of radioactivity and quantities involved:

1. Collected, sampled and analyzed, and discharged directly to the tailrace of the Keowee Hydroelectric Plant.
2. Collected, sampled and analyzed, processed by evaporation and/or demineralization; the evaporator bottoms and/or spent resins packaged and shipped off site to an NRC or approved agreement state licensed burial ground.
3. Collected, sampled and analyzed, processed by evaporation and/or demineralization.

Liquid waste effluent is diluted, as necessary in the hydroelectric plant tailrace to permissible concentration
2 limits in accordance with Selected Licensee Commitments.

## 11.2.2 DISPOSAL SYSTEM DESIGN

### 11.2.2.1 General Description

Liquid wastes are accumulated in storage tanks according to the waste source and expected process train. The Auxiliary Building coolant treatment header has been redesigned to facilitate the processing of liquid wastes from the high activity waste tanks, low activity waste tanks, and the miscellaneous waste holdup tanks in one of the Interim Radwaste Building Liquid Waste Disposal Systems and/or the Radwaste
0 Facility. The liquid wastes are directed to the Interim Radwaste Building (IRB) or Radwaste Facility for
0 processing by an evaporator package and/or demineralization to segregate impurities for ultimate disposal
0 as per Section 11.4.2, "System Design and Evaluation" on page 11-13. Based on the analysis, water is
0 either reclaimed to the coolant system or released as per Section 11.2.2.2, "Operation" on page 11-6.

0 In addition, vendor supplied equipment may be utilized to process water and reduce waste volumes.

The IRB floor drains and equipment drains are collected in two sumps. The floor and low activity drains sump collects floor drains and low activity degassed equipment drains. This sump normally discharges to the evaporator feed tanks; however, the contents may be pumped to the Oconee 3 low activity waste tank in the Auxiliary Building if necessary. The floor and low activity sump is vented to the Oconee 3 vent stack.

High activity equipment drains are collected in high activity equipment drains sump. Two sump pumps normally transfer the sump contents to the miscellaneous waste tanks. Sump pump discharge may be aligned to the Oconee 3 high activity waste tank in the Auxiliary Building if necessary. The high activity equipment drains sump is vented to the suction of the Oconee 3 waste gas compressors via the Oconee 3 waste gas vent header.

All piping and equipment in contact with reactor coolant are constructed of corrosion-resistant material. This equipment is arranged and located to permit detection and collection of system losses and to prevent

escape of any unmonitored radioactive liquid to the environment. Component data are shown in Table 11-6.

The liquid waste discharge header to the Keowee Hydro tailrace is shown in Figure 11-1.

Waste tanks in the IRB and the Auxiliary Building are vented as necessary to the gaseous waste vent header to provide for filling and emptying without overpressurization or creating a vacuum. In addition, each waste tank is equipped with a relief valve and vacuum breaker. Nitrogen is supplied to each waste collection tank for purging to the Gaseous Waste Disposal System as needed.

Flush water is provided at appropriate locations in the system for flushing of piping and components.

### 11.2.2.2 Operation

Liquid wastes are collected in the Auxiliary Building and are transferred to the IRB and/or the Radwaste Facility for processing.

#### 11.2.2.2.1 IRB Operation

Liquid wastes requiring processing can be pumped through the coolant treatment header to two 17,000 gallon interim evaporator feed tanks (Figure 11-2) located in the Interim Radwaste Building. To prevent solids accumulation in these tanks, an interim evaporator feed filter is located in the waste feed line upstream of these tanks. One of these feed tanks is normally aligned for collection and storage while the other is aligned to discharge its contents for processing through the evaporator or through portable equipment.

Evaporator condensate (distillate) is normally discharged directly to one of the interim evaporator condensate monitor tanks. If necessary, evaporator condensate (Figure 11-2) may first be processed through the interim evaporator condensate demineralizer for removal of ionic contaminants. The interim evaporator condensate filter is located downstream of the demineralizer to retain resin fines. The contents of a condensate monitor tank are mixed by recirculation and/or agitation and are transferred to the Radwaste Facility for further processing or release. A release rate consistent with dilution flow from the Keowee Hydro Station is determined and the radiation monitor alarm set points adjusted to comply with the limits specified in 10CFR20 and 10CFR50 Appendix I. Normally the wastes are then discharged via the plant discharge monitor to the Keowee tailrace. If reactor grade water quality is attained, the tank contents may be directed to the Auxiliary Building reactor coolant bleed holdup tanks for recycling. Condensate may also be returned to the evaporator feed tanks or to the evaporator condensate demineralizer for further processing.

Instrumentation and controls necessary for the operation of the IRB Liquid Waste Disposal System for Oconee 1 and 2, and for Oconee 3 are located on an auxiliary control board near the equipment. The instrumentation and controls for the low activity waste tanks are located on the auxiliary control board and are duplicated in the respective Control Rooms. However, the only controls for the release of liquid waste are in the Radwaste Facility Control Room.

Liquid wastes are released from the Condensate Monitor tanks or from the Waste Monitor Tanks in the Radwaste Facility. After the liquid is mixed, sampled, and analyzed, a release rate consistent with dilution flow from the Keowee Hydro Station is determined and the radiation monitor alarm set points adjusted to
2 comply with limits specified in Selected Licensee Commitments. The release is controlled from the Radwaste Facility control room and monitored by 1RIA 33. The RIA will terminate a release on a high alarm setpoint by closing LW-131. The release activity in CPM is recorded in the Radwaste Facility Control Room.

### 11.2.2.3 Liquid Waste Holdup Capacity

The estimates of liquid waste holdup times are based on the following assumptions:

1. The liquid waste generation rates are as follows (See Table 11-1):

| | |
|---|---|
| (a) Primary System | 161,019 $ft^3$ per year for 3 units |
| (b) Spent Fuel Pool | 26,349 $ft^3$ per year for 3 units |
| (c) Cask Decontamination | 17,566 $ft^3$ per year for 3 units |
| (d) Component Coolant | 17,566 $ft^3$ per year for 3 units |
| (e) Service Water | 58,553 $ft^3$ per year for 3 units |
| (f) Decontamination Room | 87,828 $ft^3$ per year for 3 units |
| (g) Resin Sluice | 23,421 $ft^3$ per year for 3 units |
| (h) Miscellaneous System Leakage | 351,312 $ft^3$ per year for 3 units |
| (i) OTSG Tube Leaks | 40,140 $ft^3$ per year for 3 units |
| (j) LHST | 161,019 $ft^3$ per year for 3 units |
| TOTAL | 944,773 $ft^3$ per year for 3 units |

2. Design holdup capacity equals the contents of the miscellaneous waste holdup tanks, interim evaporator feed tanks, and condensate monitor tanks A and B which is 83,793 gallons for Oconee 1, 2 and 3.
3. The time for filling and discharging the tanks is 6 hours or less.
4. The tanks fill at a linear rate and the contents are discharged when the tanks become full and are sampled.

From the assumptions above the holdup times are:

Oconee 1 and 2 Holdup Time = 5.25 days

Oconee 3 Holdup Time = 11.46 days

The Radwaste Facility provides an additional 140,000 gallons of storage capacity.

2

# 11.3 GASEOUS WASTE MANAGEMENT SYSTEMS

2

## 2 11.3.1 DISPOSAL METHODS AND LIMITS

2 Gaseous activity is generated by the evolution of radioactive gases from liquids stored in tanks throughout
2 the station. Gaseous wastes are disposed of, at a permissible rate, under continuous radiation monitoring
2 or periodic sampling and control, by any of the following methods depending on the concentration of radioactivity, quantities, and source of the material involved:

1. Release of Auxiliary Building ventilation air and Reactor Building purges to the unit vents.
2. Release of Reactor Building purges through high efficiency particulate and charcoal iodine filters to the unit vents.
3. Release of waste gas directly or through high efficiency particulate and charcoal iodine filters to the unit vents.
4. Diversion to waste gas tanks with controlled release after sampling and analysis through the waste gas system high efficiency particulate and charcoal iodine filters to the unit vents.
5. Release of Radwaste Facility HVAC and process exhaust.

The tank vent system and incinerator exhaust are processed through carbon and high efficiency particulate filters.

Gaseous wastes are released from the station at a controlled rate so that permissible concentration limits for Unrestricted Areas will not be exceeded at the Exclusion Area boundary, when averaged over a year in
2 accordance with the requirements of the Selected Licensee Commitments. The concentrations at the boundary are determined after applying appropriate dilution factors derived from on-site meteorological
2 studies (Section 2.3, "Meteorology" on page 2-11).

Waste releases from the three units are integrated and controlled by process radiation monitors, interlocks, and by the operator so as not to exceed the appropriate station release limits. Where effluents can be released from more than one location, administrative controls are also provided to insure that station limits are not exceeded.

## 11.3.2 DISPOSAL SYSTEM DESIGN

### 11.3.2.1 General Description

All components in the Auxiliary Building and IRB that can contain potentially radioactive gases are vented to a vent header. The vent gases are subsequently drawn from this vent header by one of two waste gas compressors or a waste gas exhauster. The waste gas compressor discharges through a waste gas separator to one of two waste gas tanks. The waste gas tanks and the waste gas exhauster discharge to the unit vent after passing through a filter bank consisting of a prefilter, an absolute filter, and a charcoal filter. A flow diagram of this system with the necessary instrumentation and controls for operation is shown in Figure 11-3. Component data are shown in Table 11-6.

Oconee 1 and 2 share a Gaseous Waste Disposal System. Oconee 3 has a separate Waste Gas Disposal System, which can be interconnected to the Gaseous Waste Disposal System for Oconee 1 and 2 through double isolation valves between the vent headers. These are normally operated separately, but may be tied together to facilitate maintenance of either of the systems.

The purpose of the Gaseous Waste Disposal System is to:

1. Maintain a non-oxidizing cover gas of nitrogen in tanks and equipment that contain potentially radioactive gas.
2. Hold up radioactive gas for decay.
3. Release gases (radioactive or non-radioactive) to the atmosphere under controlled conditions.

### 11.3.2.2 Operation

One waste gas compressor is normally in continuous operation with the other compressor in a standby condition. The waste gas compressor takes suction on the vent header and normally discharges into waste gas tank "A" which is used as a surge tank. The vent header pressure control operates a bleedback valve (GWD-1) allowing a continuous circulation of gas through the vent header. As liquid storage tanks connected to the systems are filled, the excess gas is stored in the waste gas tank. As liquid storage tanks are emptied, gas flows from the waste gas tank back into the vent header. As waste gas tank "A" is filled, the inlet valve on waste gas tank "B" (GWD-3) is opened and waste gas tank "A" inlet valve (GWD-2) is closed. The gas in waste gas tank "A" is allowed to bleed back into the vent header and is directed into waste gas tank "B" by the waste gas compressor until the pressure in waste gas tank "A" is at the desired operating pressure. The valves are then repositioned to utilize waste gas tank "A" as a surge tank and waste gas tank "B" for radioactive decay. Gas in waste gas tank "B" is sampled for laboratory analysis to determine the permissible release rate or need for holdup for radioactive decay.

Release of gas from the waste gas tanks to the unit vent is controlled by the waste gas tank outlet valves
2 GWD-4 and GWD-5. The volume of gas discharged to the unit vent is recorded in the Control Room
2 and is documented on the Gaseous Waste Release (GWR) permit governing the release. Monitoring of the gas discharged to the unit vent for radioactivity is provided by a radiation monitor which, on a high
2 radiation signal, will close the valves through which the gas is being discharged. In the event that the
2 applicable radiation monitor is not available for service, two independent samples of the gas to be released
2 are collected. The two samples independently verify the gas activity and serve as the basis for determining
2 the gaseous waste release rate.

The waste gas exhauster is used when large volumes of gas containing little or no radioactivity are available for release to the unit vent. The waste gas exhauster and its isolation valves are interlocked to trip the exhauster and close the isolation valves in case of a high radiation level in the line going to the unit vent. The waste gas exhauster does not normally operate and is normally valved off by the manual valve upstream of GWD-6. Therefore, no unintentional release of significant activity is possible through this line.

3 All indication and controls for this system are located in the Control Room.

### 11.3.2.3 Gaseous Waste Holdup Capacity

The estimates of gaseous waste holdup times are based on the following assumptions:

1. An annual waste gas generation rate of 131,400 $ft^3$ is evolved from three units (Table 11-1). Oconee 1, 2, and 3 contribute 43,800 $ft^3$ each per year.

0 2. Four waste gas tanks located in the Auxiliary Building and three waste gas tanks located in the Interin
0 Radwaste Building provide holdup capacity for Oconee 1, 2, and 3.

3. Holdup capacity is as follows:

| Auxiliary Building | Oconee 1 & 2 | Oconee 3 |
|---|---|---|
| Auxiliary Building Tanks ($ft^3$) | 2200 | 2200 |
| 0 Interin Radwaste Building Tanks ($ft^3$) | 2104 | 1052 |
| Total Storage Volume | 4304 | 3252 |

4. The times for filling and venting the waste gas tanks are negligible.

5. The waste gas tanks are initially filled with nitrogen at 10 psig and 100°F. The tanks may be filled to
2 approximately 85 psig and 100°F.

2

## 11.3.3 TESTS AND INSPECTIONS

Each process radiation monitoring channel will be functionally tested and calibrated periodically to verify proper operation of components and to insure that the desired detector sensitivities are maintained.

A signal generator located within the process monitor panel will be used to check the alignment of electronic modules. After the electronic alignment is completed, a remote operated calibration source is actuated to determine proper functioning of the detector.

The flow measuring instrument and controls associated with the gaseous waste effluent lines will be calibrated periodically to insure proper accuracy, measurement, and control of radioactivity releases from the station.

Efficiency of the particulate filters is determined in the factory, as well as in-place, in accordance with USA DOP (Dioctyl Phtholate) test method and UL standard 586. DOP smoke is introduced upstream of the filter and the quantity detected downstream of the filter is measured. This test is conducted at full rated flow capacity, minimum acceptable test efficiency for the particulate filter is 99.97 percent. The difference between factory and in-place tests is the test duration time: 30 seconds and 2 minutes, respectively.

Efficiency of the iodine filter is determined by two different methods in the factory. One employs $I_2$-131, $I_2$, and $CH_3I$; the other uses refrigerant-11.

In place testing of both the particulate and iodine filters are done in accordance with ANSI N-510.

### 11.3.3.1 Test with Iodine

The filter shall remove at least 99.9 percent of molecular iodine-131 ($I_2$-131) in the presence of a gaseous concentration of 50 mg per $m^3$ of non-radioactive molecular iodine ($I_2$) plus 5 mg per $m^3$ of non-radioactive methyl iodide ($CH_3I$). This performance level is maintained until the amount of non-radioactive $I_2$ having reached the test unit is equivalent to 200 gm in the full scale system. Following this loading, feeding of non-radioactive $I_2$ and $CH_3I$ is halted, and air at 70 percent relative humidity and

150°F is drawn through the test unit at its rated flow for two hours. The integrated $I_2$-131 removal efficiency for the test unit, including both iodine feed and elution periods, shall be no less than 90.0 percent. The $I_2$-131 feed periods is between 10 and 100 microcuries per gram of non-radioactive $I_2$ feed.

The filter is required to remove at least 99.0 percent of methyl iodide-131 ($CH_3I$-131) in the presence of a gaseous concentration of 50 mg per $m^3$ of non-radioactive methyl iodide ($CH_3I$). This performance level is maintained until the amount of $CH_3I$ having reached the test unit is equivalent to 200 gm in the full scale system. Following this loading, feeding of $I_2$ and $CH_3I$ is halted, and air at 70 percent relative humidity and 150°F is drawn through the test unit at its rated flow for two hours. The integrated efficiency in the removal of $CH_3I$-131 by the test unit, including both feed and elution periods, is required to be no less than 65 percent. The $CH_3I$-131 activity during $CH_3I$-131 feed periods is between 10 and 100 microcuries per gram of non-radioactive $CH_3I$ feed.

### 11.3.3.2 Test with Refrigerant-11

Refrigerant-11 is injected into an air flow of 333 cfm upstream of the filter until the concentration is 50 ppm. After 2 minutes, the refrigerant-11 concentration downstream of the filter is required to be less than 0.1 ppm.

Field tests for efficiency will be performed using refrigerant-11 only. The system will be operating at rated flow. Refrigerant-11 is introduced upstream of the filter to produce an R-11 concentration of 50 ppm. With an upstream concentration of 50 ppm and a test of 2 minutes, the maximum allowable downstream concentration is 0.1 ppm.

# 11.4 SOLID WASTE MANAGEMENT SYSTEM

## 11.4.1 DESIGN BASES

### 11.4.1.1 Solid Waste Activities

Solid radioactive wastes consist of Dry Active Waste (DAW), dewatered or solidified demineralizer resins, solidified evaporator concentrates, and miscellaneous solids. This activity is not released to the environment and influences only the shielding required to meet criteria stated in Section 12.3.1, "Facility Design Features" on page 12-9.

### 11.4.1.2 Disposal Methods and Limits

Solid wastes will be packaged to meet applicable regulations and shipped in accordance with DOT regulations to an NRC licensed burial ground. The solid radwaste Process Control Program has been reviewed by NRC (Reference 1 on page 11-15).

## 11.4.2 SYSTEM DESIGN AND EVALUATION

The Solid Waste Disposal System provides the capability to package solid wastes for shipment to an offsite NRC or approved agreement state licensed burial facility.

Evaporator concentrates can be pumped from the evaporator concentrates storage tanks in the Radwaste Facility, and can be transferred to a vendor supplied solidification system located at the Interim Radwaste Building.

The disposal of the powdered resins may be accomplished by backwashing the resins from the filter elements to a sump in the Turbine Building and then to the Resin Recovery System or the Powdex Backwash Tank (PBT; Figure 11-4) for processing. The resin is allowed to settle to the bottom of the Backwash Receiving Tanks (BRT) in the Radwaste Facility or the PBT at the Chemical Treatment Ponds (CTP). The excess water in the BRT is decanted to the Decant Monitor Tank for sampling and release to the environment. The excess water in the PBT is also sampled prior to decanting to CTPs.

Should the powdered resins be radioactively contaminated the resins can be incinerated by the VR system and packaged for disposal or transferred to vender dewatering equipment and then to an approved shipping container. If the PBT is used, then the resins are transferred to an approved shipping container and dewatered prior to disposal.

Various other bead resins can be sluiced to the Resin Batch Tank (RBT) where the water content can be adjusted. The resins are then transferred to the VR system where they are prepared for shipment to an NRC or approved agreement state licensed burial site. Bead resins can also be sluiced to an approved shipping container where they are prepared for shipment to an appropriate burial site.

Low level trash such as contaminated clothing, rags, paper, gloves, and shoe coverings are incinerated or compressed in DOT, NRC approved containers by a compactor or shipped to a vendor for compaction or processing. Containers loaded with compacted waste are monitored for surface radiation/contamination in a predesignated packaging area until they are shipped offsite for burial. Spent letdown filters are placed in station supplied casks and disposed of in DOT, NRC approved containers.

Disposal of slightly contaminated materials within the Company controlled area has been approved by the State of South Carolina and NRC. Prior to disposal onsite, each waste stream is analyzed and confirmed to have acceptably low radionuclide concentrations, permission is then obtained from the proper agencies
2 per 10CFR 20.2002 requirements. Each application for disposal is evaluated and approved on a case by case basis as determined by material quantities, material type, disposal methods and radionuclide concentrations.

## 11.4.3 REFERENCES

1. B. J. Youngblood (NRC) letter to H. B. Tucker (Duke) dated May 2, 1986.

# 11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

## 11.5.1 DESIGN BASES AND EVALUATION

Radiation monitoring of process systems provides early warning of equipment, component, or system malfunctions, or potential radiological hazards. The Process Radiation Monitoring System includes alarms, indications, and recording of data in the Control Rooms. In some cases automatic action is taken upon an alarm condition; in others the alarm serves as a warning to the operator so that manual corrective action can be taken. Radioactive liquid and gaseous waste effluents, particularly, are monitored, coordinated between Control Rooms, and controlled to assure that radioactivity released does not exceed 10CFR 20 and 10CFR 50 Appendix I limits for the station as a whole.

The sensitivity and the ranges of the detectors have been coordinated with system and environmental dilution factors to assure that releases due to normal, transient, and accident conditions will be monitored and that normal releases will not exceed permissible concentrations. The release of radioactive waste will generally be on a batch basis. Waste releases will also be integrated and recorded. Interlocks are provided to terminate any release of liquid or gaseous waste if a pre-set radiation level is reached. The monitoring and controls exerted by the Process Radiation Monitoring System and the operator during the release will also be supplemented by manual sampling, laboratory analysis, and counting prior to release.

Various detectors are also shielded against ambient background radiation levels that would exist in their location due to normal, transient, or accident conditions, so that accurate readings of radioactivity will be obtained.

The process monitors have been given a primary calibration with the particular radionuclides that they are expected to monitor. Their energy response has been determined as an aid in measurement of other
0 radionuclides that may also be encountered. A calibration source, related to primary calibration at
0 factory, is supplied with the system. The sources are held by Radiation Protection or I & E and used
0 periodically to calibrate the detector. A check source is used only to verify that the detector is functional. Spectrometer grade amplifiers have been supplied with all of the sodium iodide scintillation (NaI) detectors so that they can be used with a gamma analyzer for the identification of the specific radionuclides being monitored.

Monitors are also provided on various non-radioactive cooling water systems to detect leakage from normally radioactive systems due to any component failures and thus prevent their accidental release to the environment. In addition to the manual sampling of waste prior to release, mentioned above, the measurement of radioactivity in other process fluids is also supplemented by manual sampling, laboratory
2 analysis, and counting. This is particularly necessary for beta-emitting radionuclides such as tritium.

## 11.5.2 DESCRIPTION

The radiation monitoring equipment indications and alarms are located in the Control Rooms from which the systems being monitored are operated. Radiation monitor indications for liquid waste disposal and the Radwaste Facility vent effluents are displayed in the Radwaste Facility Control Room. Indications
0 for unit vent effluents can be displayed in both Control Rooms. Outputs from all process monitor
0 channels are recorded in the RIA computer system or on multipoint recorders except for the main steam line monitors which alarm only. Control Room annunciation of high radiation level is provided for each
0 channel. Most detector assemblies are equipped with a Control Room operated check source.

Table 11-7 lists the process radiation monitors and gives the following information:

1. Channel Number and Function - A Radiation Indicating Alarm (RIA) number has been assigned to each detector. Monitors serving the same function have the same number. Prefix numbers indicate the unit on which the detector is used. No prefix number indicates that the detector is utilized for Oconee 1 & 2. The function shows the system in which the monitor is employed.

2. Type of Detector - The standard detector type identification is given followed by the size of the crystal or the length of the detector. The lead shield thickness which has been applied to obtain the sensitivities indicated is also given.

3. Sensitivity - Monitor sensitivities are indicated in terms of background equivalent concentrations and count rate for the radionuclides listed. Background equivalent information shown in the table defines the ability of the monitor to detect the indicated radionuclide concentrations inside the sampler at a count rate that is equal to that resulting from a gamma field outside the sampler. The lead shielding is designed to reduce the count rate resulting from 1.5 MeV gammas in order to obtain the sensitivities shown.

4. Range - Readout range of monitoring instrumentation, upper range limits, and range overlap between different detectors monitoring the same sample are indicated.

The following is a description of the various applications of these monitors as they are applied to systems:

4 1. 1,2 and 3RIA16 and 17 detectors monitor the A and B Main Steam line piping respectively for the
4 presence of radioactivity in the process steam. The primary purpose for these monitors is to aid in
4 the detection of a steam generator primary to secondary leakage fault. Readout and alarms for these
4 monitors are located in the associated control rooms.

4 2. RIA-31 monitors gross gamma from the outlets of the A and B Low Pressure Injection Decay Heat
4 Coolers of Units 1, 2 and 3. Samples from the cooler outlets are sequentially automatically valved
4 and monitored. Sample valve scan rate is adjustable from the Unit 1 SCADA terminal. Unit 1
4 control room contains the main control terminal for the monitor. The output from the radiation
4 monitor is indicated in all three control rooms. Alarms are also provided in the control rooms. The
4 monitor is located inside the turbine building and is shielded to function during a loss of coolant
4 accident, including 100 percent release of fission gases inside the Reactor Building. The monitor is
4 provided to supplement indications from 1, 2 and 3RIA-35.

4 3. RIA-32 can monitor air from up to 12 locations and 3RIA-32 can monitor air from up to 6 locations,
each within the Auxiliary Building for early detection and location of equipment malfunctions. They
also are designed to warn personnel of the presence of radiological hazards. Each monitor
incorporates a sample pump that continuously draws samples through a three-way valve manifold at
2 the detector. Sample valves are sequenced by the RIA computer system to direct individual samples
2 to shielded beta sensitive detectors. Detector outputs are logged by the RIA computer system. Loss
2 of sample flow is annunciated in the Control Rooms as a fault alarm dectector.

2 Additionally, RIA-32 and 3RIA-32 are designed to monitor the discharge from the respective units
2 penetration room fans. Manually-selectable sample points permit detection of gaseous activity in the
2 Penetration Room resulting from Reactor Building design leakage following a Reactor Coolant
2 System failure and subsequent release of fission gases into the Reactor Building.

4 4. 1RIA-33 is used to monitor total liquid waste effluent from the station. Loss of sample flow is
0 annunciated in the Radwaste Facility Control Room. Interlocks from this monitor automatically
2 terminate a release at preset levels.

4 5. 1RIA-35, 2RIA-35, and 3RIA-35 continuously monitor samples of Low Pressure Service Water
2 (LPSW) for gross gamma in each LPSW effluent header from the Auxiliary Building. These monitors
2 are provided to monitor for any leakage of contaminated radioactive water into the LPSW system,

2 thereby allowing isolation and repair of components. The detectors are located inside the Turbine
3 Building. They are shielded to function in the presence of increased background from Loss of
3 Coolant Accident. Loss of sample flow is annunciated in the appropriate Control Room.

6. RIA-37 and RIA-38 monitor waste gas effluent from Oconee 1 and 2. One instrument channel using a plastic beta scintillation detector and one instrument channel using a Geiger-Mueller (G-M) tube provide the dynamic range indicated on Table 11-7. This range covers normal and abnormal operating conditions with overlap as indicated. Interlocks from these monitors automatically terminate release at preset levels. 3RIA-37 and 3RIA-38 are functionally identical and serve the same purpose for Oconee 3. These monitors are shown on Figure 11-3.

7. RIA-39 and 3RIA-39 monitor Control Room ventilation using beta sensitive detectors (Section 9.4.1.1, "Design Bases" on page 9-53). Samples of Control Room air are continuously pumped through shielded samplers. Loss of sample flow is annunciated in the appropriate Control Room.

8. 1RIA-40, 2RIA-40, and 3RIA-40 monitor condenser air ejector off gas effluent to each unit vent (Section 10.3.2.7) to detect activity in the steam system resulting from a steam generator tube leak. In
2 addition to this protection, 1RIA-16 and 1RIA-17 are located adjacent to the main steam headers. For Oconee 2 and 3, this monitoring function is served by 2RIA-16, 2RIA-17, 3RIA-16, and
2 3RIA-17, respectively.

9. RIA-41 and 3RIA-41 monitor ventilation air in both Spent Fuel Buildings using beta sensitive detectors (Section 9.4.2.1, "Design Bases" on page 9-55). Samples of Spent Fuel Building air are continuously pumped through shielded detectors. Loss of sample flow is annunciated in the appropriate Control Room.

10. RIA-42 and 3RIA-42 monitor recirculated cooling water return from Auxiliary Building for gross gamma activity.

11. 1RIA-43, 1RIA-44, 1RIA-45, and 1RIA-46 monitor Oconee 1 vent for radioactive air particulates,
0 gas, and iodine. A vent monitor incorporates a sample nozzle, a pumping system, and four detector
0 channels. The pump supplies samples to an air particulate monitor (moving filter paper), a fixed
0 charcoal filter that is monitored for iodine, and to two gas monitors. The pump also draws a portion
0 of the sample through an Iodine cartridge and filter paper for effluent analysis. Air particulates are detected by monitoring a moving filter paper with a plastic beta scintillator. Iodine is monitored with
2 a NaI scintillator monitoring a selected gamma energy range. Gaseous activity is detected by a plastic
0 beta scintillator for normal ranges. A cadmium telluride solid state detector is used in a separate
0 instrument gas channel to extend the dynamic range of the system. Sensitivity and overlap of the gaseous monitoring ranges are indicated in Table 11-7. Collection efficiency for the air particulate filter is 99 percent for particles 0.5 micron and larger. The activated charcoal cartridge type filter has a rated collection efficiency of at least 90 percent for radioiodine in forms anticipated.

Malfunctions involving loss of sample flow and depleted, torn, or clogged filter paper are alarmed in the Control Room.

For Oconee 2 and 3, this monitoring function is served by 2RIA-43, -44, -45, -46, and 3RIA-43, -44,
2 -45, -46, respectively.

Interlocks from the gas monitors automatically terminate a Reactor Building purge and close the purge isolation valves on high radiation level. These monitors are shown on Figure 11-6.

4RIA-45 and 4RIA-46 monitor the Radwaste Facility HVAC for noble gas. Particulate and radioiodine activity are continuously sampled by a filter paper and charcoal cartridge sampling
0 arrangement. The sampling filter paper and charcoal cartridge are periodically replaced and analyzed
0 to quantify and qualify radioactivity present in the HVAC system. Noble gas activity is detected by a plastic beta scintillator for normal ranges. A G-M tube is used in a separate instrument channel to

extend the dynamic range of the system. Sensitivity and overlap of the gaseous monitoring ranges are indicated in Table 11-7.

0 a. 1RIA-47, 1RIA-48, 1RIA-49, 1RIA-49A and associated equipment make up the Reactor Building Airborne Activity Monitoring System for Oconee 1. The equipment provided is
2 functionally identical to that described for the vent monitors except that a separate Iodine
2 cartridge and filter paper are not available for effluent analysis. For Oconee 2 and 3, this
2 monitoring function is performed by 2RIA-47, -48, -49, 49A, and 3RIA-47, -48, -49, 49A,
2 respectively. On high radiation level, interlocks from the gas monitors automatically close the Reactor Building sump line isolation valves.

12. 1RIA-50 monitors Oconee 1 Component Cooling System for gross gamma using a NaI scintillator (Section 9.2.1.7, "Leakage Considerations" on page 9-26). Sample flow loss is alarmed in the Control
2 Room. For Oconee 2 and 3, this monitoring function is performed by 2RIA-50 and 3RIA-50,
2 respectively.

3 13. 3RIA-53 is designed to monitor airborne effluent from the Interim Radwaste Building. One instrument channel using a plastic beta-scintillation detector and one channel using a G-M tube provide the dynamic range indicated in Table 11-7. This range covers normal and operating
2 conditions with overlap as indicated. Interim Radwaste Building particulate and radioactive gas
2 constituents are continuously sampled by a filter paper and charcoal cartridge sampling arrangement
2 contained on the 3RIA-53 skid. The particulate and iodine sampling media are periodically replaced
2 and analyzed to quality and quantify radioactivity present on the media.

3 14. RIA-54 monitors the Unit 1 and 2 Turbine Building sump and stops pumps during loss of power or
2 high activity. 3RIA-54 monitors the Unit 3 Turbine Building sump and stops pumps when high
2 radioactivity levels are detected.

3 15. 1RIA-56, 2RIA-56 and 3RIA-56 are designed to monitor gross gamma activity in each unit vent stack. The detector is an ion chamber located on the vent stack with the readout in the control room. The monitor provides very high range monitoring capabilities for gaseous effluents exiting the unit vent under accident conditions.

3 16. 1, 2, 3RIA-57 and 58 are designed to monitor gross gamma activity in each unit containment
2 building. These post-accident monitors are coaxial ion chambers with readouts in each control room.
2 The monitors are located in the east and west penetration room associated with each unit. 1, 2,
3 3RIA-58 have recorders in the Control Rooms.

3 17. The Hot Machine Shop Vent particulate and radioiodine constituents are continuously sampled by a filter paper and charcoal cartridge sampling arrangement. The sampling arrangement is periodically replaced and analyzed to quantify and qualify radioactivity present on the filter paper and/or cartridge. Because of the type of work conducted in the Hot Machine Shop, and because of the location of the Shop to the Auxiliary Building (and its associated ventilation system), noble gas activity is not released via the Hot Machine Shop vent. Therefore, noble gas monitoring capability is not required in the Hot Machine Shop.

4

# 11.6 RADWASTE FACILITY

## 11.6.1 GENERAL DESCRIPTION

### 11.6.1.1 Safety Evaluation

The radwaste facility was evaluated under a 10CFR 50.59 safety evaluation and was found not to involve
2 an unreviewed safety question. In accordance with 10CFR 20.2004 pursuant to 10CFR 20.2002 Duke applied for (Reference 1 on page 11-32) and obtained NRC approval to operate the incinerator under the ONS operating license and Technical Specifications. The NRC transmitted their safety analysis (Reference 2 on page 11-32) which concluded that operation of the incinerator would not diminish the safe operation of ONS nor present an undue hazard to public health and safety.

### 11.6.1.2 Site Characteristics

The site is located south of the Unit 3 Turbine and Auxiliary Buildings. The yard grade elevation in this area is about 796 feet (MSL). Approximately 80 ft. southeast of the proposed building the yard fill slopes downward at 2 to 1 (horizontal to vertical) to original ground about 55 ft. below.

The test borings encountered a profile of materials consisting from the ground surface of fill residual soil, partially weathered rock and finally rock or refusal materials. The thickness of fill varied from 18 to just over 70 feet within the proposed facility. The fill soils classify primarily as micaceous silty sands with included clayey layers of low to moderate plasticity.

The fill consistency based on the standard penetration test is loose to dense. The fill appears to be relatively well compacted overall based on penetration resistances. The standard penetration resistances range from less than 5 to greater than 40 blows per foot with values predominantly between 21 and 30 blows per foot.

Below the fill soils, the residual materials weathered from the parent bedrock were encountered. The residual profile consists of a variable thickness of soil underlaid by partially weathered rock. The residual soils primarily are silty sands or sandy silts. The standard penetration test values range from 4 to over 100 blows per foot.

Beneath the fill and residual soils, the test borings encountered refusal materials at depths of 30 to 85 feet below the present surface. The nature of the refusal materials was investigated by rock coring procedures. The rock classified as mica-gneiss.

### 11.6.1.3 Facility Description

The Radwaste Facility is designed to process liquid and solid radioactive wastes. The wastes will be separated into clean water and concentrated contaminants. The concentrated contaminants will be prepared for disposal and the clean water will be discarded or recycled for use in the station. The wastes will consist of miscellaneous liquid waste (radioactive equipment drains and floor drains, etc.) reactor coolant, powdered resin, and miscellaneous radioactive trash (gloves, paper, etc.)

Liquid wastes will be processed by an appropriate combination of equipment (filter, demineralizer, and/or evaporator) in the Liquid Waste and Recycle System. Contaminants collected by the demineralizers and

filters will be sent to the Solidification System. Boric acid concentrated from reactor coolant by the evaporator will be reused or sent to the Solidification System as are the waste concentrates.

Powdered resin used in the Condensate Polishing Demineralizers will be collected and monitored in the Resin Recovery System. Noncontaminated resin will be pumped to the Chemical Treatment Ponds. Excess water will be removed from contaminated resin and the resin sent to the Volume Reduction System or vendor supplied liners for dewatering.

The Volume Reduction System incinerates combustible wastes. The dried product (ash & salts) and wet wastes will be packaged to meet Federal and State regulations.

## 11.6.1.4 QA Condition Classifications and Inspection Program

### 11.6.1.4.1 Perspective

Duke Power Company's Quality Assurance program covers four QA conditions. Quality Assurance Condition 2 (QA 2) applies to radwaste systems and follows the guidance of Regulatory Guide 1.143. Regulatory Guide 1.143 lists systems to which it applies but does not contain criteria for determining applicability.

The criteria herein adopted for the application of QA 2 are based on the "as low as reasonably achievable" (ALARA) concept of radiation protection and generally relate to routinely expected occurrences. The criteria generally result in determinations which are consistent with Regulatory Guide 1.143.

### 11.6.1.4.2 General Criteria

An item or activity is ALARA related and a QA program is applied if:

a. Functional unavailability, lack of effectiveness, or non-catastrophic failures impair the ability to meet the ALARA objective for effluent releases.

b. Require routine maintenance or repair of anticipated failures would cause excessive or easily avoidable occupational exposure.

### 11.6.1.4.3 Implementation

a. Eliminating pressure boundary leakage of ALARA related piping systems (delineated on flow diagrams as Class E) is an ALARA related function, but pipe hangers and supports do not perform an ALARA related function because they are provided to prevent gross failure rather than leakage. Experience has shown that conventional power piping has a very low rate of gross failure but leakage is not unusual. Therefore, pipe hangers and supports are not QA Condition 2.

b. The pressure boundary of piping systems with only occasional radioactivity, very low radioactivity, and drains are not ALARA related. Generally, this applies to closed loop cooling and process steam, streams normally releasable without treatment and floor drains.

c. Equipment, parts, and components not part of an ALARA pressure boundary are functionally ALARA related if their failure would prevent the system from performing its intended function greater than 10% of a calendar quarter (about 10 days). Since most electrical equipment and small mechanical equipment can be repaired in this time, they are generally excluded.

d. Only the containment of leaks and spills within the structure is an ALARA related function which requires a QA 2 program by these criteria. Therefore, a QA 2 program will be applied to the "Bathtub Portion" of the radwaste facility structure.

## 11.6.2 STRUCTURES

### 11.6.2.1 Description of Building

The Oconee Radwaste Facility will consist of two separate adjoining structures, separated by a 3 inch expansion joint, both supported by poured in place reinforced concrete mats. One structure will be primarily of reinforced concrete construction with structural walls serving also as shielding for radioactive components or materials. The other structure will be primarily of braced structural steel construction with floors of reinforced concrete on metal deck and conventionally formed reinforced concrete columns and floors supporting large tanks. Exterior walls will be insulated metal siding on steel girts. Interior walls will by gypsum wallboard on metal studs and concrete masonry.

### 11.6.2.2 Design Bases

The structures are modeled as space frames using the McDonald Douglas version of ICES STRUDL, a structural design language computer program. The two dimensional finite element capabilities of STRUDL are used to represent walls and slabs while one dimensional beam elements are used for beams and columns. The supported points of the model have spring stiffnesses representing the force-deflection relationship of the underlying soil, thus differential settlement is accounted for. A modal and shock spectrum analysis was performed using the capabilities of the STRUDL DYNAL feature of the STRUDL program up to Elevation 799 + 6 as a minimum.

Both portions of the Radwaste Facility are designed and erected so that all liquid inventory will be contained within the structures in the event of pipe or tank ruptures caused by a seismic event or from other causes. Therefore, the reinforced concrete mats and a concrete wall of sufficient height to contain the entire liquid inventory are designed to withstand the effects of seismic loads as well as conventional loads. Loadings due to failure of the upper structure portions during the seismic event were not considered. Design, procurement and erection meet the requirements of the Duke Power Company Quality Assurance Condition 2 (QA2) program up to Elevation 799 + 6. A wall erected to Elevation 799 + 6 (bathtub) can contain the entire liquid inventory of the building.

For the east side of the facility, between column lines B and F, the framing is primarily of structural steel, and the structural design includes the effects of seismic and conventional loads. Design, procurement, and shop fabrication of the structural steel meet the requirements of the Duke Power Company QA 2 Program. Structural steel erection meets AISC requirements, but has no formal Quality Assurance requirements. The south-east portion of this area is reinforced concrete up to the floor at Elevation 819 + 0. The floor, supporting large tanks, is not designed to seismic requirements; the concrete columns are designed for seismic loadings except that the tie bars are reduced in size and number from the requirements for seismic forces, to permit ease in construction.

The west side of the facility, between column lines G and K, is a reinforced concrete structure, and the analysis and design will include the effects of seismic and conventional loads up to the bottom of the floor slab at Elevation 819 + 0. Design, procurement and construction of these parts will meet the requirements of the Duke Power Company QA program. The floor slab at Elevation 819 + 0 and all reinforced concrete elements above this floor, except for load bearing walls will be analyzed and designed for conventional loads only, with good engineering practice applied to design, procurement and construction. The design of load bearing walls above Elevation 819 + 0 will include seismic loads with no Quality Assurance requirements applied to design, procurement or construction.

Independent loads will be calculated on the following bases:

#### 11.6.2.2.1 Wind Loadings

The design wind velocity is 95 mph at 30 ft. above the nominal ground elevation. According to ASCE Paper 3269, "Wind Forces on Structures," this represents the greatest wind velocity with a recurrence interval of 100 years. ANSI A58.1-1972, "Building Code Requirements for Minimum Design Loads in Building and Other Structures," recommends that buildings with a height-to-minimum horizontal dimension ratio exceeding five should be dynamically analyzed to determine the effect of gust factors. However, since this structure will have a height-to-width ratio less than five, a gust factor of unity is used in determining wind forces. Tornado and tornado missiles are not included as a design load.

#### 11.6.2.2.2 Water Level Design

The yard grade is at elevation 796 + 0. All openings into the structure will be no lower than 797 + 0. A 2′-6″ minimum height curb is provided to contain any accidental spillage within the facility. The yard is provided with a surface water drainage system.

#### 11.6.2.2.3 Dead Loads and Equipment Loads

A density of 150 lb/ft$^3$ will be used for reinforced concrete dead weight computations. Structural steel weights will be based on their nominal weight per foot as given in the AISC "Manual of Steel Construction," eighth edition. Weights of metal decking and siding will be taken from supplier's catalogs. Weights of equipment, tanks, etc., weighing more than 1000 lbs will be taken from information supplied by the manufacturer. An additional load of 150 lb/ft$^2$ will be applied to floors, except for the drum storage area, and roofs in the reinforced concrete structure to account for suspended piping, electrical cable tray and small miscellaneous equipment weighing less than 1000 lbs. In the drum storage area, the additional load is 2250 lb/ft$^2$.

Additional loads of 50 lb/ft$^2$ on floors and 30 lb/ft$^2$ on roofs will be applied in the structural steel portion, for the same reason. Where cable tray is banked, the cable tray loading will be calculated and applied as additional equipment load. A dead load of 20 lb/ft$^2$ is applied to areas covered by grating.

#### 11.6.2.2.4 Live Loads

In the concrete portion, a live load of 125 lb/ft$^2$ is applied to floors and roof. In the structural steel portion, a live load of 150 lb/ft$^2$ is applied to floors and 20 lb/ft$^2$ is applied to roofs. A live load of 100 lb/ft$^2$ is applied to areas covered by grating.

#### 11.6.2.2.5 Seismic Design

A nonlinear finite element soil-structure analysis (FLUSH) is used to generate seismic response at the ground surface due to bedrock motion. The rock motion input is a synthetic 5%g time history developed so that response spectra derived from that motion envelope the NRC Regulatory Guide 1.60 curves. The design response spectra are developed using procedures set forth in NRC Regulatory Guide 1.60, with maximum ground acceleration in both horizontal and vertical directions obtained from the soil-structure interaction analysis. Response spectra analyses will be performed for both horizontal directions. Vertical earthquake loads will be obtained by applying the maximum vertical acceleration to static loads.

### 11.6.2.3 Loads and Loading Combinations

The loads and combinations thereof to be used in the analysis and design of the Radwaste Facility are described below:

1. Normal Loads

Normal loads are those loads to be encountered during normal facility operation.

They include the following:

D - Dead loads, including permanent equipment loads and hydrostatic loads.

L - Live loads, including any movable equipment loads and other loads which vary with intensity and occurrence, such as soil pressure.

2. Severe Environmental Loads

Severe environmental loads are those loads that could infrequently be encountered during the facility life.

Included in this category are:

E - Loads generated by the Operating Basis Earthquake

W - Loads generated by the design wind specified for the facility.

#### 11.6.2.3.1 Load Combinations for Concrete Structures

U designates the section strength required to resist design loads and is based on methods described in ACI 318-77. The following load combinations will be satisfied:

1. $U = 1.4D + 1.7L$
2. $U = .75(1.4D + 1.7L + 1.7W)$
3. $U = .75(1.4D + 1.7W)$
4. $U = .9D + 1.3W$
5. $U = .75(1.4D + 1.7L + 1.87E)$
6. $U = .75(1.4D + 1.87E)$
7. $U = .9D + 1.43E$

#### 11.6.2.3.2 Load Combinations for Steel Structures

S designates the section strength required to resist design loads and is based on the elastic design methods and the allowable stresses defined in Part I, Sections 1.5.1, 1.5.2, 1.5.3, 1.5.4 and 1.5.5 of the AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," seventh edition.

Y designates the section strength required to resist design loads and is based on plastic design methods described in Part 2 of AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," seventh edition.

The following load combinations will be used for the elastic working stress method:

1. $S = D + L$
2. $1.33S = D + L + E$
3. $1.33S = D + L + W$

In load combinations 2 and 3, S is increased by one-third in accordance with Section 1.5.6 in the AISC specification.

The following load combinations will be used for the plastic design method:

1. Y = 1.7D + 1.7L
2. Y = 1.3D + 1.3L + 1.3E
3. Y = 1.3D + 1.3L + 1.3W

**Note:** Loadings that include seismic factors will be used as a design basis to design the "bathtub."

## 11.6.3 MECHANICAL SYSTEMS

### 11.6.3.1 Liquid Waste and Recycle System

#### 11.6.3.1.1 Design Bases

The Liquid Waste and Recycle System (LW) is designed to appropriately process of all excess radioactive water generated at the station. Decontaminated water will be reused by the station as make up or released to the environment as appropriate. Generally, chemistry limits control recycle and radioactivity limits control discharge. Contamination removed from processed water will be transferred to the Volume Reduction and Solidification System.

#### 11.6.3.1.2 System Description

Four 10,000 gallon Feed Tanks are provided for batching reactor coolant and miscellaneous waste. One of the four is normally reserved for reactor coolant. One is always available for unscheduled receipt of water.

Feed pumps, process filters, demineralizers, and demineralizer fines filters are provided in pairs, each designed for ≤ 50 gpm. One 30 gpm evaporator is provided to be used either for concentration of boric acid from reactor coolant or, if necessary, for use with a filter and demineralizer to provide the greatest available decontamination for waste. An additional train of six demineralizers is available to process liquid waste. Sufficient crossconnection is provided so that two independent streams can be processed simultaneously. One such lineup would be: 1) a feed pump, filter, demineralizer, demineralizer fines filter, evaporator processing reactor coolant at 30 gpm and 2) a feed pump, filter, demineralizer, and demineralizer fines filter processing miscellaneous floor drains at ≤ 50 gpm. Other "normal" situations exist with total process rates from 30 to 100 gpm.

Six 10,000 gallon monitor tanks are provided for checking processed water quality and scheduling transfers. Two of the six tanks are normally reserved for reactor coolant recycle. One is always available for unscheduled receipt of water. Water may be released to the environment through two paths.

If dilution is required, it is sent through a monitor with a setpoint coordinated with a flow meter. The monitor will terminate discharge if it detects activity in excess of its setpoint. Since processed water is not released without analysis, the monitors are only provided to guard against administrative error.

### 11.6.3.2 Resin Recovery System

#### 11.6.3.2.1 Design Bases

The Powdered Resin Recovery System is designed to collect and sample each backwash from the Condensate Polishing Demineralizer and to separate non-radioactive water from contaminated spent resin if present.

#### 11.6.3.2.2 System Description

An expected contaminated backwash is directed to the Contaminated Backwash Receiving Tank (CBRT). Otherwise the backwash is sent to one of the two Backwash Receiving Tanks, BRT-A, or BRT-B. There the resin is sampled to determine the appropriate process path.

Backwashes which have been determined upon sampling to be sufficiently radioactive to prevent release to the environment are allowed to settle. After sufficient settling has occurred, the excess water is decanted. The decanted water is directed through Resin Fines Filters which are provided to insure that sufficient settling has occurred into the Decant Monitor Tank (DMT). Here the water is sampled and directed to one of three locations; 1) Liquid Waste System, 2) Condensate Storage Tank, or 3) Chemical Treatment Pond. The contaminated resin is transferred to the Volume Reduction System.

Backwashes which have been determined upon sampling to be noncontaminated are directed to the Chemical Treatment Pond.

2 The resin in the backwash receiving tanks may also be used to process laundry and hot shower water and
2 to process/reprocess miscellaneous waste. This is accomplished by agitating the water and resin, then
2 proceeding with the decanting as described above.

### 11.6.3.3 Volume Reduction and Solidification System

#### 11.6.3.3.1 Design Bases

The Volume Reduction and Solidification System (VR) is designed to prepare radioactive wastes for shipment and disposal, and to minimize the volume of waste shipped.

**Note:** The VR system (incinerator and dry product handling and drumming portions) has been placed in a layup condition until operating economics can justify its use.

#### 11.6.3.3.2 System Description

In order to prepare wastes for shipment and minimize the volume of waste, wet wastes (e.g., contaminated oil, powdered resins) and dry trash are incinerated and the scrub liquor produced is completely dried. The results of both fluid bed processes are a dry, free-flowing mixture of salt granules and ash. This sand-like material is then packaged to meet Federal and State regulations. Resin which is too radioactive to incinerate will be solidified and/or packaged to meet Federal and State regulations.

The incinerator may be fed resin slurries, contaminated oil or shredded trash. Fluidizing air is electrically heated for startup and thereafter maintained by the combustion process. Liquid sprays (resin slurry or condensate) are provided to control temperature.

All normal operations of the Volume Reduction and Solidification System involving radioactive material are carried out remotely from the Radwaste Control Room. A remote control crane moves new drums from the clean fill stations to the waste drumming stations, stores or retrieves drums in the storage pit, and loads truck-mounted shielded casks used to ship solidified waste off site for disposal.

### 11.6.3.4 Instrument and Breathing Air Systems

2 These systems are described in Section 9.5.2, "Instrument and Breathing Air Systems" on page 9-79.

2

## 11.6.3.5 Equipment Cooling System

### 11.6.3.5.1 Design Bases

The Equipment Cooling System is designed to remove heat from the components of the Liquid Waste Processing System and Radioactive Waste Solidification System. This system also supplies cooling water to the Radwaste Facility air compressors and HVAC coolers, and supplies service water for the facility.

### 11.6.3.5.2 System Description

The generating plants Condenser Circulating Water System serves as the suction source for the Equipment Cooling System. Two duplex basket-type strainers reduce particulate size to 1/16″ and two 100% capacity EC Supply Pumps rated at 2400 gpm, @ 160 ft. deliver flow to the secondary side of two plate-type heat exchangers. The primary side flow is circulated by two 100% capacity EC Circulating Pumps rated at 1600 GPM @ 85 ft. This flow provides cooling for the Liquid Waste Evaporator and the Volume Reduction System. An auxiliary supply is taken off the EC Supply Pump discharge for miscellaneous service water use.

## 11.6.3.6 Heating Ventilation and Air Conditioning

### 11.6.3.6.1 Design Bases

The Radwaste Facility HVAC consists of a Ventilation System and an Air Conditioning System. The principal objectives of the HVAC System are to supply sufficient filtered fresh air to maintain an aseptic condition, control the temperature for effective operation of process equipment, meet the "ALARA" related consideration with air flow by supplying air to clean areas and exhausting air from high radiation areas and to sample the exhaust air to monitor the release of airborne radioactive material from the building.

### 11.6.3.6.2 System Description

#### *11.6.3.6.2.1 Ventilation System*

The Ventilation System will supply filtered and tempered air to each area in sufficient quantity to reduce the heat build up and keep the temperature below 104 degrees in the process areas. A positive exhaust system will be used to exhaust a quantity of air from each area which is sufficiently larger than the supply air to maintain a directed flow of air in the building. The exhaust air quality will be monitored. A filter train including rough, HEPA and charcoal filters will be used for the exhaust air from tank vents and fume hoods to minimize the emission of contamination from the building. There will be no recirculation of air to any process area.

#### *11.6.3.6.2.2 Air Conditioning System*

The Air Conditioning System will supply tempered and dehumidified air including fresh air to each area. The areas to be air conditioned include the control room, the counting room, the RP Lab, the Supervisors Office and the clean maintenance shop. The Hot Instrument Shop and the personnel areas will be air conditioned with 100% fresh air.

## 11.6.3.7 Drains

0 Roof drains and clean floor drains are piped to the station storm drain system.

0 Personnel area drains that are potentially contaminated are pumped to the facility sumps.

0 Sanitary drains are piped to the station sewage treatment plant.

0 Contaminated process and floor drains are piped to the facility sump.

## 11.6.4 REMOTE CONTROL SYSTEM

### 11.6.4.1 Design Bases

The Radwaste Remote Control System is designed to provide a means for operating the various mechanical and electrical systems in the Radwaste Facility from a centralized control area. This design will minimize the requirements for manning the facility, and will minimize the radiation exposure to the operator. While it is impractical to control all functions from a centralized location, remote control is employed in a practical manner where possible, particularly in situations involving radiation exposure to the operator.

### 11.6.4.2 System Description

The Radwaste Control Room (RCR) will be located in the clean portion of the building where there will be no radiation shielding requirements. A cable spreading room will be provided behind the RCR to allow for control board and relay cabinet cable access.

Control boards designed by several different vendors as well as Duke-designed boards will be located in the RCR. The electrical project engineer will coordinate between all parties to insure as much compatibility between boards as is reasonably achievable. Human factors aspects of the control room and control board designs including color coding, control board enhancement, process mimics, operator/control interfaces, and RCR personnel traffic patterns will be taken into consideration.

Since the RCR is the primary area of personnel activity for this facility, the Fire Detection System central alarm station as well as any other "Facility protective" monitors will be located there. Annunciators, instrumentation, and control devices will be installed as necessary to satisfy the intent of the Remote Control System purpose.

## 11.6.5 FIRE DETECTION SYSTEM

### 11.6.5.1 Design Bases

The Radwaste Fire Detection System is designed to provide early warning at a central location in the event of a fire or conditions preceding the break out of a fire.

### 11.6.5.2 System Description

The Fire Detection System central alarm station will be located in the Radwaste control room. Individual strings of various types of detectors will emanate from the central alarm station to provide detection in selected areas of the facility. Detector locations and types (ionization, fixed temperature, rate-of-rise, etc.) will be determined by the fire protection engineer.

The detection system installed will be of the two-wire type which will allow trouble alarm indication. This design approach should minimize personnel radiation exposure encountered in maintaining the system. An alarm will be provided in the Oconee plant (e.g., Unit 3 control room) to notify the plant operations personnel of a fire in the Radwaste Facility.

## 11.6.6 RADIATION MONITORING SYSTEM

### 11.6.6.1 Design Bases

The Radiation Monitoring System is designed to accurately monitor process, area and noble gas radiation within the facility. Particulate and iodine collection samplers are also installed in the exhaust system.

### 11.6.6.2 System Description

The Radiation Monitoring System will consist of the components with their respective parameters as listed in Table 11-7.

## 11.6.7 RADIATION PROTECTION

### 11.6.7.1 Facility Design Features

The mechanical and electrical equipment is separated into clean, nonradioactive areas, curbed areas and shielded areas. Radioactive components are separated from each other to allow maintenance without subsequent exposure from nearby components. Radioactive equipment with valves is provided a valve gallery containing the valves and remote valve operators in an intermediate radiation area. Separation of system piping is also stressed to eliminate exposure in these galleries. Air regulators and other instrumentation associated with valve and system operation are located outside of the valve gallery, inside of the labyrinth entrance in a lower zone.

Feed tank exposure is minimized by using stainless steel lined rooms. Mixer motors for these tanks are located above the shielded tank room.

Process particulate filters are the backflushable type to eliminate exposure with filter replacement and are remotely operated.

Process resin demineralizers are used for ion removal.

All equipment with the potential for an accumulation of crud are flushed prior to maintenance. Periodic piping review insures minimum piping crud traps.

The Volume Reduction System layout utilizes several individually shielded cubicles to separate components containing the majority of the radioactive material from the mechanical components such as the pumps and blowers which contain small amounts of radioactive material and which are expected to require periodic maintenance. In addition, the components containing the majority of the radioactive material are all fitted with decontamination nozzles so that the radioactive salts can be flushed from the system and the components readily decontaminated prior to required maintenance.

### 11.6.7.2 Shielding

#### 11.6.7.2.1 Source Terms

Radiation source terms for the Radwaste Facility are separated into two systems; the LW Liquid Waste System and the VR Volume Reduction and Solidification System. The liquid waste source fluid derived from, "Oconee 1-3, Fluid Source Terms ," is processed according to assumptions of ANSI N237, "Source Term Specification," Final Draft, 1977 and the Oconee Radwaste Facility flow diagrams. Source terms for the Volume Reduction System are derived from concentrates activities, and from the system operation (see Section 11.6.3, "Mechanical Systems" on page 11-26).

#### 11.6.7.2.2 Radiation Zone Designations

The Radwaste Facility is divided into radiation zones based upon source term analyses, Regulatory Guide 8.8 and personnel radiation exposure limits; station general arrangement drawings and system diagrammatics are marked to denote applicable radiation zones. These radiation zones are as follows:

Zone I: Designation for areas adjacent to the station site where Duke Power Company does not normally
2 exercise authority to control access. In accordance with applicable regulations (10CFR 20.1301(a)(1)), the
2 dose rate in these areas does not exceed 0.1 rem/yr.

2 Zone II: Areas within the station site where the station staff is expected to work continuously. For conservatism, the limiting dose rate is selected as 0.5 mrem/hr. This is comparable to the criteria given in
2 10CFR 20.1302.

Zone III: Areas within the station where staff occupancy is expected to be periodic rather than
2 continuous. An employee could, however, remain in these areas and not exceed 5.0 mrem/hr.

Zone IV: Includes infrequently occupied work locations where the dose rate exceeds continuous occupational levels but access need not be physically restricted. The limit dose rate for this zone is
3 designated as 50 mrem/hr. The precautions given in 10CFR 20.1601, 1602, and 1901 through 1905 for Radiation Areas are employed where local dose rate levels in Zone IV warrant.

Zone V: Encompasses all areas of the station where the dose rate exceeds that of Zone IV. Access to these areas is physically restricted, and Radiation Protection surveillance is required for occupancy, if any.
3 The precautions given in 10CFR 20.1601, 1602, and 1901 through 1905 for High Radiation Areas are employed where local dose rate levels in Zone V warrant.

#### 11.6.7.2.3 Shield Wall Thickness

The KAP VI computer code is used to determine the shield wall thickness for each component. KAP VI utilizes the point kernel technique to calculate radiation levels at detector points located within or outside a complex radiation source geometry.

## 11.6.8 REFERENCES

1. H. B. Tucker (Duke) letter to H. R. Denton (NRC) dated June 10, 1985.
2. J. F. Stolz (NRC) letter to H. B. Tucker (Duke) dated October 30, 1986.

**THIS THE LAST PAGE OF THE CHAPTER 11 TEXT PORTION.**

**Table 11-6 (Page 1 of 9). Waste Disposal System Component Data (Component Quantities for Three Units)**

| | |
|---|---|
| Low Activity Waste Tank | |
| Quantity | 2 |
| Volume each, cu. ft. | 398 |
| Material | Concrete with Stainless Steel Liner |
| High Activity Waste Tank | |
| Quantity | 2 |
| Volume each, cu. ft. | 262 |
| Material | Concrete with Stainless Steel Liner |
| Misc. Waste Holdup Tank | |
| Quantity | 2 |
| Volume each, cu. ft. | 2,700 for Units 1 and 2 shared<br>1,550 for Unit 3 |
| Material | Carbon Steel with Stainless Clad |
| Design Pressure | Vessel Full Plus 10 ft. Hydro Head |
| Spent Resin Storage Tank | |
| Quantity | 2 |
| Volume each, cu. ft. | 450 for Units 1 and 2 shared<br>380 for Unit 3 |
| Material | Stainless Steel |
| High Activity Spent Resin Storage Tank | |
| Quantity | 1 for Unit 3 |
| Volume, cu. ft. | 380 |
| Material | Stainless Steel |
| Reactor Building Normal Sump | |
| Quantity | 3 |
| Volume each, cu. ft. | 45 |
| Material | Concrete |
| Reactor Building Emergency Sump | |
| Quantity | 3 |
| Volume each, cu. ft. | 540 |
| Material | Concrete |
| Waste Gas Tank | |
| Quantity | 4 |
| Volume each, cu. ft. | 1,100 |

**Table 11-6 (Page 2 of 9). Waste Disposal System Component Data (Component Quantities for Three Units)**

| | |
|---|---|
| Material | Carbon Steel |
| Design Pressure, psig | 100 |
| Misc. Waste Evaporator Feed Tank | |
| Quantity | 1 |
| Volume, cu. ft. | 400 |
| Material | Stainless Steel |
| Design Pressure | Vessel Full Plus 10 ft. Hydro Head |
| Waste Evaporator | |
| Quantity | 1 |
| Process Rates, lb/hr | 5,060 |
| Material | Stainless Steel |
| Design Pressure, psig | 15 |
| Low Activity Waste Tank Pump | |
| Quantity | 4 |
| Capacity each, gal/min | 50 |
| Diff. Head, ft. | 100 |
| High Activity Waste Tank Pump | |
| Quantity | 4 |
| Capacity each, gal/min | 50 |
| Diff. Head, ft. | 40 |
| Waste Transfer Pump | |
| Quantity | 4 |
| Capacity each, gal/min | 50 |
| Diff. Head, ft. | 50 |
| Spent Resin Sluicing Pump | |
| Quantity | 2 |
| Capacity each, gal/min | 50 |
| Diff. Head, ft. | 50 |
| Spent Resin Transfer Pump | |
| Quantity | 2 |
| Capacity each, gal/min | 10 |
| Diff. Head, ft. | 100 |
| Reactor Building Normal Sump Pump | |

**Table 11-6 (Page 3 of 9). Waste Disposal System Component Data (Component Quantities for Three Units)**

| | |
|---|---|
| Quantity | 6 |
| Capacity each, gal/min | 25 |
| Diff. Head, ft. | 28 |
| **Waste Evaporator Feed Pump** | |
| Quantity | 1 |
| Capacity, gal/min | 7-1/2 |
| Diff. Head, ft. | 60 |
| **Waste Evaporator Recirculating Pump** | |
| Quantity | 1 |
| Capacity, gal/min | 160 |
| Diff. Head, ft. | 53 |
| **Waste Evaporator Distillate Pump** | |
| Quantity | 1 |
| Capacity, gal/min | 9-1/2 |
| Diff. Head, ft. | 62 |
| **Waste Gas Filter** | |
| Quantity | 2 |
| Rating, scfm | 200 |
| Type | Prefilter, Absolute and Charcoal |
| Material | 11 Gauge Galvanized Steel |
| **Waste Gas Exhauster** | |
| Quantity | 2 |
| Rating, scfm | 200 at 6 in. Water Gauge External Static Pressure |
| Type | Backward Curved - Centrifugal |
| **Waste Gas Compressor** | |
| Quantity | 4 |
| Capacity each, cfm | 48 at 85 psig |
| Type | Centrifugal Displacement |
| **Waste Evaporator Feed Tanks** | |
| Quantity | 2 |
| Volume, gal | 17,000 |
| Design Pressure | Static head plus 5 psig |
| Design Temperature, °F | 200 |
| Material | 304 stainless steel |

**Table 11-6 (Page 4 of 9). Waste Disposal System Component Data (Component Quantities for Three Units)**

Waste Evaporator Condensate Monitor Tanks

| | |
|---|---|
| Quantity | 2 |
| Volume, gal | 9,000 |
| Design Pressure | Static head plus 5 psig |
| Design Temperature, °F | 200 |
| Material | 304 stainless steel |

Waste Evaporator Concentrates Storage Tank

| | |
|---|---|
| Quantity | 1 |
| Volume, gal | 3,000 |
| Design Pressure | Static head plus 5 psig |
| Design Temperature, °F | 200 |
| Material | 304 stainless steel |

Waste Evaporator Condensate Return Tank

| | |
|---|---|
| Quantity | 1 |
| Receiver volume, gal | 100 |
| Design Pressure | Atmospheric |
| Design Temperature, °F | 212 |
| No. of Pumps | 2 |
| Design Flow, gal/min | 25 |
| Design Head, ft | 65 |

Waste Evaporator Feed Filter

| | |
|---|---|
| Quantity | 1 |
| Type | Cage Assembly (disposable synthetic cartridged) |
| Design Pressure, psig | 200 |
| Design Temperature, °F | 250 |
| Design Flow Rate, gal/min | 35 |
| Pressure Drop at Design Flow, psi | Clean - 5 |
| Retention of 25 Microns Particles | Fouled - 20 |
| Material | 98% |
| | Stainless steel |

Waste Evaporator Condensate Filter

| | |
|---|---|
| Quantity | 1 |
| Type | Cage Assembly (disposable synthetic cartridge) |
| Design Pressure, psig | 300 |
| Design Temperature, °F | 250 |
| Design Flow Rate, gal/min | 150 |
| Pressure Drop at Design Flow, psi | Clean - 5 |

**Table 11-6 (Page 5 of 9). Waste Disposal System Component Data (Component Quantities for Three Units)**

| | |
|---|---|
| Retention of 25 Micron Particles | Fouled - 20 |
| Material | 98% |
| | Stainless steel |
| **Waste Evaporator Condensate Demineralizer** | |
| Quantity | 1 |
| Type | Non-regenerable |
| Design Temperature, °F | 200 |
| Design Pressure, psig | 150 |
| Vessel Volume, $ft^3$ | 55 |
| Resin Volume, $ft^3$ | 50 |
| Design Flow, gal/min | 310 |
| Material | Stainless Steel |
| Resin Type | Mixed bed |
| **Waste Evaporator Feed Pump** | |
| Quantity | 1 |
| Type | Canned centrifugal |
| Design Flow, gal/min | 35 |
| Design Head, ft | 250 |
| Design Pressure, psig | 150 |
| Design Temperature, °F | 200 |
| Operating Temperature, °F | 120 |
| Material | Stainless Steel |
| **Waste Condensate Monitor Tank Pumps** | |
| Quantity | 2 |
| Type | Canned centrifugal |
| Design Flow, gal/min | 100 |
| Design Head, ft | 250 |
| Design Pressure, psig | 150 |
| Design Temperature, °F | 200 |
| Operating Temperature, °F | 120 |
| Material | Stainless Steel |
| **Waste Evaporator Concentrates Transfer Pump** | |
| Quantity | 1 |
| Type | Canned centrifugal |
| Design Flow, gal/min | 35 |
| Design Head, ft | 250 |
| Design Pressure, psig | 150 |
| Design Temperature, °F | 200 |
| Operating Temperature, °F | 170 |
| Material | Stainless Steel |

**Table 11-6 (Page 6 of 9). Waste Disposal System Component Data (Component Quantities for Three Units)**

Floor and Low Activity Drains Sump Pumps

| | |
|---|---|
| Quantity | 2 |
| Type | Vertical |
| Design Flow, gal/min | 50 |
| Design Head, ft | 100 |
| Material | Stainless Steel |

High Activity Equipment Drains Sump Pumps

| | |
|---|---|
| Quantity | 2 |
| Type | Vertical |
| Design Flow, gal/min | 50 |
| Design Head, ft | 100 |
| Material | Stainless Steel |

Waste Evaporator Distillate Pump

| | |
|---|---|
| Quantity | 1 |
| Type | Canned Centrifugal |
| Design Flow, gal/min | 15.6 |
| Design Head, ft | 208 |
| Design Pressure, psig | 150 |
| Design Temperature, °F | 220 |
| Operating Temperature, °F | 80-110 |
| Material | Stainless Steel |

Westinghouse Waste Evaporator Package

| | |
|---|---|
| Quantity | 1 |
| Nominal Capacity, gal/min | 15 |
| Steam Supply Pressure, psig | 50 |
| Steam Flow, lb/hr | 10,500 |
| Cooling Water Flow, gal/min | 780 |
| Concentrates Batch Volume, gal | 500 |
| Max. Boron Concentration, ppm | 21,000 |
| Liquid DF* | $10^6$ |
| Gaseous DF** | $10^5$ |

Gaseous Waste Disposal System Waste Gas Decay Tanks

| | |
|---|---|
| Quantity | 3 |
| Volume, $ft^3$ | 1052 |
| Design Pressure, psig | 100 |
| Design Temperature, °F | Material |
| 200 | Carbon steel |

Interim Evaporator Feed Tanks

**Table 11-6 (Page 7 of 9). Waste Disposal System Component Data (Component Quantities for Three Units)**

| | |
|---|---|
| Quantity | 2 |
| Design Pressure | Static Head Plus 5 psig |
| Volume, cu. ft. | 2272 |
| Material | 304 Stainless Steel |
| **Interim Evaporator Condensate Monitor Tanks** | |
| Quantity | 2 |
| Design Pressure | Static Head Plus 5 psig |
| Volume, cu. ft. | 1203 |
| Material | 304 Stainless Steel |
| **Interim Evaporator Concentrates Storage Tank** | |
| Quantity | 1 |
| Design Pressure | Static Head Plus 5 psig |
| Volume, cu. ft. | 401 |
| Material | 304 Stainless Steel |
| **Interim Evaporator Condensate Return Unit** | |
| Quantity | 1 |
| Number of Pumps | 2 |
| Capacity, gpm | 25 |
| Diff. Head, ft. | 65 |
| **Interim Evaporator Feed Filter** | |
| Quantity | 1 |
| Type | Cage Assembly (disposable cartridge) |
| Design Pressure, psig | 200 |
| Design Temperature, °F | 250 |
| Design Flow Rate, gpm | 35 |
| Pressure Drop at Design Flow, psi | Clean - 5 |
| | Fouled - 20 |
| Retention of 25 Micron Particles | 98% |
| Shell Material | Stainless Steel |
| **Interim Evaporator Condensate Filter** | |
| Quantity | 1 |
| Type | Cage Assembly (disposable cartridge) |
| Design Pressure, psig | 300 |
| Design Temperature, °F | 250 |
| Design Flow Rate, gpm | 150 |
| Pressure Drop at Design Flow, psi | Clean - 5 |
| | Fouled - 20 |
| Retention of 25 Micron Particles | 98% |
| Shell Material | Stainless Steel |

**Table 11-6 (Page 8 of 9). Waste Disposal System Component Data (Component Quantities for Three Units)**

Interim Evaporator Condensate Demineralizer

| | |
|---|---|
| Quantity | 1 |
| Type | Non-regenerable |
| Design Pressure, psig | 150 |
| Design Temperature, °F | 200 |
| Design Flow Rate, gpm | 310 |
| Resin Volume, cu. ft. | 50 |
| Vessel Volume, cu. ft. | 55 |
| Resin Type | Mixed Bed |
| Material | Stainless Steel |

Interim Evaporator Feed Pump

| | |
|---|---|
| Quantity | 1 |
| Type | Canned Centrifugal |
| Design Pressure, psig | 150 |
| Design Temperature, °F | 200 |
| Design Flow Rate, gpm | 35 |
| Diff. Head, ft. | 250 |
| Material | Stainless Steel |

Interim Evaporator Condensate Monitor Tank Pumps

| | |
|---|---|
| Quantity | 2 |
| Type | Canned Rotor, Centrifugal |
| Design Temperature, °F | 200 |
| Design Flow Rate, gpm | 100 |
| Diff. Head, ft. | 250 |
| Material | Stainless Steel |

Interim Evaporator Concentrate Transfer Pump

| | |
|---|---|
| Quantity | 1 |
| Type | Canned Rotor, Centrifugal |
| Design Temperature, °F | 200 |
| Design Flow Rate, gpm | 35 |
| Diff. Head, ft. | 250 |
| Material | Stainless Steel |

Interim Evaporator Distillate Pump

| | |
|---|---|
| Quantity | 1 |
| Type | Canned Rotor, Centrifugal |
| Design Temperature, °F | 220 |
| Design Flow Rate, gpm | 15.6 |
| Diff. Head, ft. | 208 |
| Material | Stainless Steel |

**Table 11-6 (Page 9 of 9). Waste Disposal System Component Data (Component Quantities for Three Units)**

**Note:**

* DF for liquid = $\frac{\text{activity in concentrates}}{\text{activity in distillate}}$

** DF for gas = $\frac{\text{activity in feed}}{\text{activity in distillate}}$

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# CHAPTER 12. RADIATION PROTECTION

# 0 12.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES 0 ARE AS LOW AS IS REASONABLY ACHIEVABLE (ALARA)

## 12.1.1 POLICY CONSIDERATIONS

0 Duke Power Company management is firmly committed to the "As Low As Is Reasonably Achievable" (ALARA) philosophy for all nuclear operations. This commitment is stated in the DPC ALARA Manual. A formal ALARA program has been established in order to convey and enforce Duke management's commitment to ALARA. This program was established in conformance with the requirements of Regulatory Guide 8.8 to ensure that occupational exposures are maintained ALARA, within the regulatory limits. It consists of:

1. a published ALARA Manual;
2. continued surveillance and evaluation of in-plant radiation and contamination conditions, as well as the monitoring and control of the exposure of personnel, by the station and General Office Radiation Protection staff;
3. an ALARA Committee at each station consisting of management and representatives from applicable
2 groups, whose purpose is to refine aspects of the ALARA program at the nuclear facility;

The committee members have extensive background in nuclear plant radiation and exposure control, including such areas as layout, shielding, personnel access, ventilation, waste management, monitoring systems, operations, and maintenance.

Although upper level management is vested with the primary responsibility and authority for administering the Duke ALARA program, the responsibility for ALARA is extended through lower management to the individual employee. The specific responsibilities of the General Office and Station Radiation Protection staffs are to ensure that:

1. An effective ALARA program is established at each Duke nuclear station that appropriately integrates Duke management philosophy and NRC regulatory requirements and guidance.
0 2. A periodic written review of the on-site radiation control program is performed to assure that objectives of the ALARA program are attained.
3. Pertinent information concerning radiation exposure of personnel from other operating LWR power stations within and outside of the Duke system, are reflected in the design and operation of Duke stations.
4. Appropriate experience gained during the operation of nuclear power stations relative to in-plant radiation control is factored into revisions of procedures to assure that the procedures continually meet the objectives of the ALARA program.
5. Necessary assistance is provided to ensure that operations, maintenance, and decommissioning activities are planned and accomplished in accordance with ALARA objectives.
6. Trends in station personnel and job exposures are analyzed in order to permit corrective actions to be taken with respect to adverse trends.

Reports of the findings of the General Office and Station Radiation Protection staffs are also effectively conveyed to management.

Specific responsibilities of station personnel are to ensure that:

1. Activities are planned and accomplished in accordance with the objectives of the ALARA program.
2. Procedures and their revisions are implemented in accordance with the objectives of the ALARA program.
2 3. The General Office Radiation Protection staff and the Engineering staff are consulted as necessary for assistance in meeting ALARA program objectives.

Other group and individual responsibilities to the ALARA program are outlined in Section II of the DPC ALARA Manual.

## 12.1.2 DESIGN CONSIDERATIONS

ALARA is a major design consideration which is carried out in accordance with section C.1 of Regulatory Guide 8.8. Consideration was given to such factors as projected component dose rates, space, mobility, accessibility, etc., during the initial design and construction phases of Oconee Nuclear Station. There is a large degree of component separation between high and low radiation levels. Several components where the potential of exposure from CRUD exists are provided with flushing capability.
2 Engineering evaluations supplement a formal operational feedback program which is used to identify specific and/or generic problems and implement design improvements.

ALARA exposures receive further attention through the training of designers and in equipment selection. Piping designers attend training sessions where topics, such as methods of minimizing crud build-up in piping are covered.

These sessions provide designers with a working knowledge of radiation protection. Closely working with equipment vendors results in the purchase of low maintenance equipment with material properties suitable for minimizing corrosion.

## 12.1.3 ALARA OPERATIONAL CONSIDERATIONS

Consistent with Duke Power Company's overall commitment to keep occupational radiation exposures as
0 low as is reasonably achievable, (ALARA), specific plans and procedures are followed by station personnel to assure that ALARA goals are achieved. Operational ALARA policy statements are
2 formulated at the corporate staff level in the Nuclear Generation Department through the issuance of the
2 System Radiation Protection Manual, ALARA Manual and station procedures. These statements and procedures are consistent with the intent of Section C.1 of Regulatory Guides 8.8 and 8.10. Personnel and job exposure trends are reviewed by management at the station and in the general office, and appropriate action is taken. Summary reports of occupational exposure are provided that describe problem areas and jobs where high radiation doses are encountered and that identify which work group is accumulating the highest doses. Recommendations are then made for changes in operating, maintenance, and inspection procedures or for modifications to the station as appropriate to reduce doses.

Maintenance activities that could involve significant radiation exposure of personnel are carefully planned. They utilize any previous operating experience, and are carried out using well trained personnel and proper equipment. Radiation Work Permits (RWP's) for non-routine operations, or Standing Radiation Work Permits (SRWP's) for routine operations are issued for each job, listing Radiation Protection requirements that shall be followed by all personnel working in the Radiation Control Area
2 (RCA)/Radiation Control Zone (RCZ). Where applicable, specific radiation exposure reduction techniques, such as those set out in Regulatory Guide 8.8, are evaluated and used. Applicable procedures for such radiation exposure related operations as maintenance, inservice inspection, radwaste handling, and

refueling, are well planned and developed by cognizant groups, and are reviewed by the station radiation protection staff to ensure that exposures will be ALARA. Careful personnel radiation and contamination monitoring are integral parts of such maintenance activities. During and upon completion of major maintenance jobs, personnel radiation exposures are evaluated and assessed relative to estimated exposures so that appropriate changes can be made in techniques or procedures as soon as practicable for future jobs. The General Office Radiation Protection staff also conducts reviews of radiation exposure related activities to assure that procedures are adequate, that they are being followed properly, and that deficiencies are corrected as soon as practicable to ensure that exposures will be ALARA.

The station ALARA Committee carefully reviews operations and maintenance activities involving the major plant systems to further assure that occupational exposures are kept ALARA.

Information from these ongoing ALARA reviews is evaluated with regard to its relevance for the design of new plants, particularly those features and components dealing with radiation exposure and control. When problems are identified such as shielding or penetration leakage, requests are initiated by the
2 Nuclear Generation Department to Engineering so that appropriate changes may be made in the design of this, and all future stations as appropriate.

## 12.2 RADIATION SOURCES

System activity levels are based on the Reactor Coolant System design activity levels defined in Table 11-5. Operation of each unit at rated power is assumed. Other parameters employed in shielding analysis are listed in Table 12-1.

# 12.3 RADIATION PROTECTION DESIGN FEATURES

## 12.3.1 FACILITY DESIGN FEATURES

The shielding is designed to perform two primary functions: (1) to ensure that, during normal operation, the radiation dose to operating personnel and to the general public is within the limits set forth in 10CFR 20 and is ALARA; and (2) to ensure that operating personnel are adequately protected in the event of a reactor accident so that the accident can be terminated without undue hazard to the general public.

Each area in the station is classified according to the dose rate allowable in the area, based on the expected frequency and duration of occupancy. These radiation zones are summarized below.

| | Location | Dose Rate, mrem/hr |
|---|---|---|
| | Exclusion area boundary | 0.05 |
| 2 | Offices in Controlled Area (00B, Admin Bldg) | 0.25 |
| | Offices, control room Turbine Building | 0.5* |
| | Normally accessible areas in Auxiliary Building | 2.0 |
| | Above fuel storage pool with normal complement of fuel assemblies | 2.5-10 |
| | Above reactor vessel and over fuel storage pool when handling fuel assemblies | 10-20 |
| | Normally accessible areas in Reactor Building during full power operation | 25 |
| | Inside control room following maximum hypothetical accident | 3 rem whole body total dose: integrated over first 90 days after accident, assuming 8 hours per day per shift |

* Certain areas of the turbine buildings are controlled and have higher dose rates resulting from primary to secondary leakage.

Piping and equipment components are shielded by concrete walls and floors of varying thickness, depending on the magnitude of the sources in each pipe section and component, and on the access requirements in a particular area. In some areas local shielding in the form of removable lead or concrete blocks are utilized to facilitate maintenance or repair operations.

## 12.3.2 SHIELDING

The material used for the primary, secondary, and Reactor Building shields is ordinary concrete with a density of approximately 140 lbs./ft$^3$. Since the primary and secondary shielding walls serve as the refueling structure, give support for the reactor coolant components under pipe rupture conditions, and provide missile shielding, they are reinforced and designed to be self-supporting. Descriptions of areas requiring shielding are presented below.

### 12.3.2.1 Reactor Building Shielding

#### 12.3.2.1.1 Primary Shield

The primary shield consists of reinforced concrete which surrounds the reactor vessel and extends upward from the Reactor Building floor to form the walls of the fuel transfer canal. The shield thickness is 5 ft. up to the height of the reactor vessel flange, where the thickness is reduced to 4.5 ft. The primary shield is designed to meet the following objectives:

1. To attenuate the neutron flux in order to limit the activation of component and structural materials.
2. To limit the radiation level after shutdown so that access to the Reactor Coolant System equipment is permissible.
3. To reduce, in conjunction with the secondary shield, the radiation level from sources within the reactor vessel to allow limited access to the Reactor Building during normal full power operation.

#### 12.3.2.1.2 Secondary Shield

The secondary shield is a 4 ft. thick reinforced concrete structure which surrounds the reactor coolant equipment, including the piping, pumps, and steam generators. The shielding is designed to reduce radiation levels from activity in the reactor coolant and to supplement the primary shield in the attenuation of neutrons and secondary gamma rays to permit limited access to the Reactor Building during full power operation.

#### 12.3.2.1.3 Reactor Building Shield

The Reactor Building shield is a reinforced, prestressed concrete structure with 3.75 ft. thick cylindrical walls and a 3.25 ft. thick dome. In conjunction with the primary and secondary shields, it limits the radiation level outside the Reactor Building from all sources inside the Reactor Building to no more than 0.5 mrem/hr. at full power operation. The shielding is also designed to protect station personnel from radiation sources inside the Reactor Building following the Maximum Hypothetical Accident (gross release of fission products).

Other significant shielding inside the Reactor Building is listed in Table 12-2.

### 12.3.2.2 Auxiliary Building Shielding

The major radiation sources are piping and equipment components handling potentially contaminated fluid, practically all of which are located on the 758′-0″, 771′-0″, and 783′-9″ levels. Groups of equipment or individual equipment items are separated by shielding walls such that systems and equipment can be isolated for maintenance with no significant radiation interference from other systems or equipment. During normal operation, there is no need to occupy these potentially radioactive equipment areas. Potential radiation sources and associated shielding are listed in Table 12-2. Additional shielding is also provided around the control room to ensure that exposure to operating personnel in the control room is within the design limits following a Design Basis Accident (DBA).

### 12.3.2.3 Post LOCA Shielding Review

A post LOCA Shielding review of the Oconee Nuclear Station was conducted pursuant to the requirements of NUREG-0578. Shielding review identified a potential for exceeding personnel exposures in GDC-19 for the control room due to its proximity to the mechanical penetration room. The low pressure recirculation piping routed through the mechanical penetration room could potentially contain highly radioactive water post LOCA. Permanently installed lead shielding was provided along the control room walls adjacent to the mechanical penetration rooms to ensure that the personnel exposures in the control rooms do not exceed the limits specified in GDC-19 (NSM-1393) for all units. Caustic addition

valves were relocated and provided with remote operators to assure operability and access. The Shielding review verified that the required personnel access to all vital areas was feasible without exceeding the radiation exposure limits following a LOCA accident.

## 12.3.3 AREA RADIATION MONITORING SYSTEM

### 12.3.3.1 Design Bases

0 The Area Radiation Monitoring System, consists of coaxial ion chambers, G-M detectors, and beta scintillation detectors. It is designed to indicate existing radiation levels and to alarm when levels exceed setpoints in various remote locations throughout the station where personnel are most likely to be exposed. Indications from the monitors are used in conjunction with station operating procedures to assure that radiation exposure of personnel does not exceed 10CFR 20 limits.

### 12.3.3.2 Description

Numbers and locations of the Area Radiation Monitors are shown in Table 12-3.

2 Control room indication is provided for each monitor indicating R/hr, mrad/hr, or cpm. Indication for
2 Oconee 1 and 2 monitors are located in Oconee 1 and 2 control room. Indication modules for Oconee 3 monitors are located in Oconee 3 control room.

1 Each detector assembly (except for the high range area detectors, and the beta scintillation detector assemblies) is equipped with a check source that is automatically actuated on a periodic basis. The failure
0 of any applicable channel to respond to the source will initiate an alarm in the control room. Radiation levels exceeding the alarm setpoint for any detector will cause an alarm at that detector location and in the control room.

### 12.3.3.3 Evaluation

The Area Radiation Monitoring System detectors are located throughout the station in locations where significant radiation levels may exist, and change with time and the operation being performed. They are designed primarily for the protection of personnel performing such operations as routine coolant sampling, refueling, Reactor Building entry, radioactive waste disposal operations, and for certain other operating and maintenance work. The system has sufficient range and flexibility to permit readout during routine operations and during any transient or emergency conditions that may exist. The equipment is
4 self-checking for proper operation, and alarms both in the local area and in the respective control room.
1 Where necessary or desirable, readout is also provided locally in certain locations.

Several channels of the Area Radiation Monitoring System will be utilized for primary indication and
4 backup in evaluating the extent of fission product release involved in both the LOCA and DBA.

## 12.4 RADIATION PROTECTION PROGRAM

The administrative organization of the Radiation Protection program and the qualifications of the personnel responsible for the program and for handling and surveying radioactive material are discussed in Section 13.1, "Organizational Structure" on page 13-3. The administrative organization is responsible for and has appropriate authority for assuring that the three basic objectives of the Radiation Protection program at Oconee Nuclear Station are achieved. These objectives are to:

1. Protect personnel
2. Protect the public
3. Protect the station

Protection of Personnel, includes surveillance and control over internal and external radiation exposure and maintaining the exposure of all personnel within permissible limits and as low as is reasonably achievable (ALARA).

Protection of the public, includes surveillance and control over all station conditions and operations that may affect the health and safety of the public. Included are such activities as radioactive gas, liquid and solid waste disposal, shipment of radioactive materials, an environmental radioactivity monitoring plan and maintaining portions of the station emergency plan.

Protection of the station, includes the continuous determination and evaluation of the radiological status of the station for operational safety and radiation exposure control purposes. This work is performed in order to warn of possible detrimental changes and exposure hazards, to determine changes or improvement needed, and to note trends for planning future maintenance work.

This administrative organization is also responsible for and has appropriate authority for maintaining occupational exposures as far below the specified limits as reasonably achievable by assuring that:

1. Station personnel are made aware of management's commitment to keep occupational exposures as low as is reasonably achievable;
2. Formal reviews are performed periodically to determine how exposures might be lowered;
3. There is a well-supervised radiation protection capability with specific defined responsibilities;
4. Station workers receive sufficient training;
5. Sufficient authority to enforce safe station operation is provided;
6. Modification to operating and maintenance procedures and to station equipment and facilities are made where they should substantially reduce exposures at a reasonable cost;
7. The radiation protection staff understand the origins of radiation exposures in the station and seeks ways to reduce exposures;
8. Adequate equipment and supplies for radiation protection work are provided.

The Station Manager is responsible for the protection of all persons against radiation and for compliance with NRC regulations and license conditions. This responsibility is in turn shared by all supervisors. Furthermore, all personnel are required to work safely and to follow the regulations, rules, and procedures that have been established for their protection.

5 The Duke Power Company, General Office Technical System Manager, Radiation Protection, establishes the Radiation Protection Program including the program for handling and monitoring radioactive material for Oconee that is designed to assure compliance with applicable regulations, technical specifications, and
5 regulatory guides. The General Office Technical System Manager also provides technical guidance and support for conducting this program, reviews the results of the program to determine its effectiveness and modifies it as required based on experience and regulatory changes, to assure that occupational radiation exposure and exposure to the general public are maintained as low as is reasonably achievable.

2 This individual also provides technical assistance to the Vice President, Nuclear Generation, who has management authority to implement the "as low as is reasonably achievable" (ALARA) occupational exposure policy, to which Duke Power Company is committed.

The Station Radiation Protection Manager at Oconee is responsible for conducting the Radiation Protection Program that has been established for the station. The Station Radiation Protection Manager has the duty and the authority to measure and control the radiation exposure of personnel; to continuously evaluate and review the radiological status of the station; to make recommendations for control or elimination of radiation hazards; to assure that all personnel are trained in radiation protection; to assist all personnel in carrying out their radiation protection responsibilities; and to protect the health and safety of the public both on-site and in the surrounding area.

In order to achieve the goals of the Radiation Protection Program and fulfill these responsibilities for radiation protection; radiological monitoring, survey and personnel exposure control work are performed on a continuing basis for station operations and maintenance.

The Radiation Protection Section performs the major portion of the radiation protection work for the
0 station. Personnel in the Radiation Protection Section normally work on the day shift during periods of
0 routine operation; and deploy onto the other shifts for major maintenance, shutdown, and refueling work. A supervisor and several Radiation Protection Technicians are also assigned to each operating shift. The Radiation Protection Section is organized into major areas, such as surveillance and control, support functions, staff and shift.

## 12.4.1 PERSONNEL MONITORING SYSTEMS

Monitoring instruments are located at exits from the Radiation Control Area. These instruments are intended for use to prevent any contamination on personnel, materials, or equipment from being spread into the unrestricted/secondary systems areas of the station. Appropriate monitoring instruments are also
2 used at various locations throughout the station for contamination control purposes. Portal monitors are utilized as appropriate, to monitor personnel leaving the station.

3 Personnel monitoring equipment consists of thermoluminescent dosimeters (TLD's), electronic
3 dosimeters, or "self-reading" dosimeters which are worn by those persons who ordinarily work in the
4 Radiation Control Area or RCZ. In addition, monitoring devices are readily available for use for
3 measurement of extremity dose. This personnel monitoring equipment is issued by Radiation Protection. Personnel monitoring equipment is also available on a day-to-day basis for those persons, employees or visitors, not assigned to the station who have occasion to enter the Radiation Control Area or to perform work involving possible exposure to radiation.

The use of personnel monitoring equipment mentioned above refers specifically to compliance with
2 10CFR 20.1502. The Station Radiation Protection Manager may require additional equipment to be worn based on the actual or anticipated dose rates and other radiological problems encountered on the job.

2 Personnel monitoring badges are supplied by a centralized in-house personnel dosimetry service which meets all applicable requirements for sensitivity, range, and accuracy of measurement. This service is NVLAP approved. Conformance with appropriate standards is also required. This service has the
2 response capability for both routine and emergency purposes.

A body burden analyzer for routine screening of personnel for internal exposure is provided in the low background counting area in the Administration Building. Outside services for radiobioassay and whole body counting are utilized as required for backup and support of this program. The station equipment is
2 sufficiently sensitive to detect in thyroid, lungs or whole body a few percent of the allowable limit of
2 intake for those gamma emitting radionuclides encountered.

## 12.4.2 PERSONNEL PROTECTIVE EQUIPMENT

Special "protective" or "anti-contamination" clothing is furnished and worn as necessary to protect personnel against contact with radioactive contamination.

This consists of coveralls, lab coats, hoods, gloves, and shoe covers. Change rooms are conveniently located in the Radiation Control Area of the station for proper utilization of this protective clothing. Approved respiratory protective equipment is also available to supplement process containment and
2 ventilation controls, for the protection of personnel against airborne radioactive contamination. This equipment consists of compressed air systems, air-supplied respirators, air-purifying (filter) respirators and Self-Contained Breathing Apparatus (SCBA).

Maintenance of the respiratory protective equipment is in accordance with the manufacturer's
3 recommendations and NUREG 0041. The use and maintenance of protective clothing and radiological respiratory protective equipment is under the direct control of the Radiation Protection Section and personnel are trained in the use of this equipment before using them in the performance of their work. The use of respiratory protective equipment is in accordance with appropriate regulations (10CFR
3 20.1202, 1204 and 1701-1704) Regulatory Guides and ANSI Standards.

## 12.4.3 FACILITIES AND ACCESS PROVISIONS

Change room facilities are provided where personnel obtain clean protective clothing and other equipment required for station work. The change rooms serve the Reactor Buildings, the Auxiliary Building, the Spent Fuel Pools, and the Hot Machine Shop. A change room is also provided for female employees. These facilities are divided into clean and contaminated sections. The contaminated section of the change rooms is used for the removal and handling of contaminated protective clothing after use.

Showers, sinks, and radiation monitoring equipment are provided in all of the change rooms to aid in the decontamination of personnel.

Personnel who are required to utilize protective clothing obtain these items in the change rooms. They first enter the change room on the "clean" side, don the required protective clothing, and then proceed to the job location. After completing work, they remove outer contaminated protective clothing at the exit of the Radiation Control Zone set up about the work area. They then proceed to the "contaminated" side of the change room, where they remove inner protective clothing items, monitor themselves; if contaminated contact RP, if clean, proceed to the "clean" side, where they put on their personal clothing before leaving.

2 The personnel entrance/exit points to/from the Auxiliary Building (RCA) are provided with contamination control checkpoints that are equipped with appropriate monitoring instrumentation. All other personnel-access points into the RCA in the Auxiliary Building are protected by restricted-in/free

out doors in case of emergency. Contamination control check-points are strategically placed throughout the RCA to prevent the spread of contamination within this area.

Before leaving the Radiation Control Area, personnel are required to monitor themselves with the appropriate equipment, positioned near each control point exit door, to make sure that they are free of significant contamination.

3

In order to protect personnel from radiation and radioactive materials, the Radiation Control Area of the station is divided into areas of increasingly controlled access depending on radiation levels. Protection of
2 personnel from access to radiation areas, high radiation areas, extra high radiation areas, and very high
2 radiation areas that exist temporarily or permanently as a result of station operations and maintenance is by means of appropriate radiation warning signs, barricades, locked doors, audible and visual indicators and alarms, etc., as required by 10CFR 20.

All work on systems or in locations where radioactive contamination or external radiation is present requires a specific Radiation Work Permit (RWP) for nonroutine operations, or a Standing Radiation
2 Work Permit (SRWP) for routine operations, prepared under the direction of the Station Radiation
2 Protection Manager before work may begin. The radiological hazards associated with the job are determined and evaluated prior to issuing the permit whenever practical, and historical data will be used when this is not practical.

Keeping exposures ALARA is a major consideration. The Radiation Work Permit lists the precautions to be taken including, as appropriate, working time limits (for external and internal exposure), protective clothing to be worn, and any radiation monitoring that may be required during the performance of the work. The permit is issued for personnel use. A working copy is maintained by the Radiation Protection Section.

All persons performing radiological work are required to read and understand the instructions on the
0 appropriate RWP/SRWP and to respond to the prompts provided by the Electronic Dose Capture
0 System (EDC), or fill out the required information on their Daily Exposure Time Record dose card
0 before entering and after leaving the RCZ and/or Radiation Control Area if the EDC system is
0 unavailable for use. The information from the EDC system or the dose card is entered into the Radiation
2 Monitoring and Control (RM&C) System computer programs and serves, in part, as a personnel monitoring record for the individuals involved.

An equipment decontamination facility is provided at the station for large and small items of station equipment, components and tools. In addition, a cask decontamination area is provided adjacent to each spent fuel pool. A decontamination laundry and a respiratory protective equipment cleaning and repair facility are also provided.

Decontamination of work areas throughout the station is facilitated by the provision of janitor's sinks in the reactor containments and on each floor level in the Auxiliary Building.

Drains from all of these facilities go to appropriate radioactive liquid waste drain tanks. Written procedures govern the proper use of protective clothing, the change rooms, and the decontamination facilities.

## 12.4.4 RADIATION PROTECTION AND CHEMISTRY FACILITIES

The major Radiation Protection facilities including a shielded counting room are centrally located at the Oconee 1 and 2 Auxiliary Building interface for efficiency of operation. These facilities are equipped for
4 detecting, measuring, and analyzing radiation(s) of primary concern and for evaluating radiological
4 problems that may be reasonably expected. Portable equipment calibration and respirator maintenance
4 facilities are located at the Oconee 3 Auxiliary Building.

The chemistry facilities located in the auxiliary building include a primary lab and office area located at the Oconee 1 and 2 Auxiliary Building interface and a secondary lab and office area located in Oconee 3's Auxiliary Building. The primary lab is used to analyze primary system (reactor coolant, pressurizer, BWST, etc.) samples while the secondary lab is used to analyze secondary system (feedwater, hotwell,
3 etc.) samples.

The chemistry facilities located outside the auxiliary building include a chemistry laboratory in the Radwaste Facility. The laboratory is used to perform chemical analyses on radwaste samples and to prepare samples for gamma spectra and beta counting.

Body burden analysis measurements for personnel internal dosimetry purposes is performed in the administration building. Environmental samples are collected and sent to a Duke Power Company environmental facility for analysis.

## 12.4.5 RADIATION PROTECTION INSTRUMENTATION

### 12.4.5.1 Laboratory and Portable Instruments

The various types of portable and laboratory instruments used in the Radiation Protection program measure alpha, beta, gamma, or neutron radiation. These instruments are required for measurements to
2 provide protection against radiation for station personnel through surveys required by 10CFR 20.1501; to analyze and measure radioactivity prior to the release of effluents for the protection of the health and safety of the public; and to provide for all other radioactivity and radiation measurements and analyses necessary for personnel and public safety and for protection of property. They were selected to provide the appropriate detection capabilities, ranges, sensitivities, and accuracies for the anticipated levels of radiation at Oconee Nuclear Station during normal operation, anticipated transients and emergency conditions. Sufficient quantities are maintained for use, calibration, maintenance and repair.

Portable radiation survey and monitoring instruments for daily routine use are maintained with nominal operational characteristics as indicated below:

2 Beta-gamma survey meters (Geiger counters, nominal 0-50 mrad/hr) are used for detection of radioactive contamination on surfaces and for low level dose rate measurements.

2 Low and high range beta/gamma ionization chamber survey meters nominal 0-1000 Rad/hr are used to cover the range of dose rate measurements necessary for radiation protection purposes.

5 The above mentioned portable instruments are subject to preoperational response checks to low activity
0 Cs-137 sources. Calibrations are performed at least semiannually. The Cs-137 Shepard calibration sources and the variable pulse generator are also calibrated annually using National Institute of Standards and Technology (NIST) traceable secondary standards.

Neutron REM survey instruments (nominal 0-5 rem/hr) are used to measure the sum of thermal, intermediate, and fast neutron dose rates for radiation protection purposes. These instruments are
2 calibrated at least semiannually with a variable pulse generator and source checked using a Pu-Be source.

The laboratory equipment is maintained as indicated below:

Multi-channel analyzers are utilized in conjunction with solid state detectors, for identification and measurement of gamma emitting radionuclides in samples of reactor primary coolant, liquid and gaseous waste, airborne contaminants, etc.

Dual channel liquid scintillation counters are used for counting tritium, as well as gross beta activity, in reactor primary coolant and other radioactive liquids and wastes.

Smears for alpha and beta/gamma contamination are counted utilizing proportional, and GM counter-scalers.

3 A shielded body-burden analyzer having adequate sensitivity to detect radionuclides of interest is located in the Administration Building and is used for personnel bioassay purposes.

The counting room equipment is subject to annual calibration/calibration check by NIST traceable
3 sources in addition to daily response checks and routine inter-laboratory cross checks when equipment is
3 in service.

Various portable airborne gaseous, particulates, and iodine samplers are available for routine use to
0 evaluate air contamination. Samplers are calibrated at least semiannually. Magnehelix gauges used for calibration of these samplers are calibrated annually by NIST traceable instruments.

2 Respiratory protective equipment includes air purifying full-face masks, air supplied respirators. Chemical cartridge particulate respirators are also available. All are maintained according to applicable regulations such as those contained in 10CFR Part 20. Respiratory protective equipment is stored in the respirator issue facility, the Control Room(s), the Operations Support Center, and other emergency locations.

Portable instrumentation for use in emergency situations is stored in emergency kits which are located in the Control Room(s) lobby and in the respirator issue facility. The kits are examined periodically for maintenance and calibration.

### 12.4.5.2 Inplant Radiation Monitoring

Inplant Radiation Monitoring Systems provide station personnel with capabilities to assess the radiological situation in various areas of significance during normal operation as well as during off-normal and emergency situations. The monitoring systems include the Area Radiation Monitoring Systems and the Process Radiation Monitoring System. Portable radiation and air monitoring equipment is also used to supplement these systems.

The Area Radiation Monitoring System is provided to monitor radiation levels in various plant locations that are potential personnel exposure areas. This system consists of gamma sensitive detectors, signal conditioning and readout instrumentation, radiation level alarm sensing logic, audible and visible alarm devices and outputs available for recording. A complete description of the location, sensitivity, and accuracy of this system is presented in Section 12.3.3.2, "Description" on page 12-11.

The Process Radiation Monitoring System is provided in part to monitor station effluents that are potential sources of radioactivity. Also, gases, particulates, and liquid and iodine levels are monitored in

primary and secondary systems during normal operation, anticipated operational occurrences and emergencies. This system provides an indication of the radioactivity in the process line monitored and provides alarms in the control room at a preset level to ensure that concentrations are maintained within
0 the limits specified in the DPC Oconee Nuclear Station Selected Licensee Commitments Manual. In addition some of the monitors perform control functions during postulated accident conditions. A complete description of the Process Radiation Monitoring System, including its range, sensitivity, setpoint, and detector type is presented in Section 11.5, "Process and Effluent Radiological Monitoring and Sampling Systems" on page 11-17.

The process and area radiation monitoring systems are supplemented by periodic surveys and by periodic grab air samples, which are collected and analyzed by Radiation Protection and Chemistry, during normal and abnormal operations and maintenance. Either charcoal or silver zeolite sample cartridges are used for sampling air when the presence of iodine is suspected.

## 12.4.6 RADIO-BIOASSAY AND MEDICAL PROGRAMS

2 Duke employees and contract service employees issued a personnel monitoring badge and who plan on
3 entering the RCA/RCZ are given a body-burden analysis when the badge is initially issued and when employment is terminated or alternatively, when the person is transferred to a non-radiological
2 assignment. Visitors who plan on entering the RCA/RCZ are generally given a body-burden analysis each time a monitoring badge is issued and at the termination of the station visit. In addition, badged station personnel and appropriate other Duke system personnel participate in a routine body-burden analysis program which provides for at least one body-burden analysis per year for each participant. The program is performed randomly with respect to organization group and job function; furthermore, personnel who experience significant exposure to airborne contamination receive additional body burden analyses as appropriate. The Station Radiation Protection Manager may waive the requirement for any analysis on a case by case basis if in his judgement, the analysis is inappropriate or impracticable. No special medical examination is considered to be necessary for radiation workers whose exposure is maintained within permissible dose limits. However, a pre-employment physical is required of prospective radiation workers to determine their health status and their ability to perform the job. Also, personnel are also examined or screened by a physician to ensure that they are medically able to use respiratory
0 equipment. Personnel using respiratory equipment are given the appropriate training for respiratory use and fit tested as required for the respirator(s) to be used.

Anyone onsite, whether badged or not, who is involved in a radiological accident where internal exposure is likely, is given a body-burden analysis as soon as practicable thereafter.

2 Dose commitments are calculated by the Site or General Office Radiation Protection Staff.

Medical observation and treatment are available in case of over-exposure or excessive contamination. Physicians, a medical clinic, and hospital facilities are available for the treatment of injuries. A local physician has been retained, and trained in the care and treatment of radiation injuries, and facilities have been established in a local hospital for the handling and treatment of possibly contaminated injured or irradiated patients. Back-up support is also available through the Oak Ridge Radiation Emergency Assistance Center/Training Site, REAC/TS. Radiation Protection personnel are responsible for the radio-bioassay program and are available to assist the physicians and the hospital in maintaining medical control of over-exposed or contaminated personnel.

These programs are designed to monitor and protect the health of all employees concerned, to confirm the adequacy of the radiation control methods employed at the station and to provide for the treatment of injuries.

## 12.4.7 TESTS AND INSPECTIONS

Routine radiological monitoring to detect radiation, radioactive contamination, and airborne radioactivity is performed throughout the plant on periodic schedules. Monitoring frequencies are determined by the Station Radiation Protection Manager based upon the actual or potential radiological conditions. Schedules of routine monitoring are issued to the technicians who initial the schedule when the routine is completed. As plant conditions change, the schedule is updated. Radiological surveys are performed before personnel enter potential or actual radiation areas where there is any doubt as to the existing conditions. Radiological surveys are also performed as a backup to routine monitoring when conditions change. All survey and routine monitoring data is recorded and filed in the Radiation Protection files.
2 Retention of survey and monitoring records follows the requirements of 10CFR 20.2103, and appropriate Technical Specifications.

The Radiation Protection Section also performs essentially all of the work necessary to maintain (other than repair) the Counting Room instruments and the portable radiation monitoring instruments. Periodic NIST traceable calibrations, instrument checks and evaluations, and other manual checks are performed. Duke Power Company participates in NRC approved performance testing programs. "Self-reading" pocket dosimeters and related instruments are subjected to periodic leak test and calibration.

Personnel monitoring instrumentation is subjected to a continuing Quality Control Program. The Quality
2 Control Program includes the use of a computer program that compares TLD values and Electronic/ "self-reading" dosimeter totals covering the same monitoring period and lists those correlations that are
2 unacceptable so that effective problem resolution can be performed as necessary, thus helping to maintain a high level of personnel monitoring equipment performance.

Duties concerning radioactive gaseous and solid waste disposal are performed by the Radiation Protection section. The detailed analyses and records required to characterize the nature of radioactive gaseous waste releases and solid waste disposal are under the control of the Radiation Protection section.

Duties concerning radioactive liquid waste disposal are performed by the Chemistry section. While the analyses of radioactive liquid waste releases are under the control of the Radiation Protection section, the records required to characterize the nature of liquid waste releases, both qualitatively and quantitatively, are under the control of the Chemistry section.

Training and qualification of personnel in Radiation Protection are the responsibility of the Station
2 Radiation Protection Manager and are performed by the Radiation Protection Section, or by Nuclear
2 Generation Department Training personnel, under his direction.

2 The Radiation Protection Section maintains the Offsite Radiological Monitoring Program for the station
2 in conjunction with the Chemistry Section.

**THIS IS THE LAST PAGE OF THE CHAPTER 12 TEXT PORTION.**

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# CHAPTER 13. CONDUCT OF OPERATIONS

(THETA1-B only). Post-CHF heat transfer does not permit use of nucleate boiling heat transfer until that correlation gives a lower heat flux than transition boiling.

An error was discovered in the evaluation model: the model allowed return to nucleate boiling heat transfer during the post-CHF period. This error was corrected (Reference 13 on page 15-66).

For transition boiling the correlation of McDonough, Milich, and King is used. Dougall-Rohsenow is used for flow film boiling and the Morgan correlation for pool film boiling.

#### 15.14.3.3.4 Post-Blowdown Model

The evaluation of the LOCA during refill and reflood is conservatively conducted assuming the minimum containment backpressure consistent with the Reactor Building Cooling Systems performance, the ECCS injection with the design single failure, and conservative containment initial conditions, volume, and heat sink data. The REFLOD3 code calculates the heat transfer and hydraulic response with containment pressure input from CONTEMPT. During the refill period the core undergoes an adiabatic heatup. Steam venting and steam-water interaction, liquid entrainment, hot wall effects, and refill-reflood heat transfer are accounted for.

#### 15.14.3.3.5 Availability of Reactor Coolant Pumps

Sensitivity studies have shown that for the large break LOCA the highest PCT results for the case with reactor coolant pumps running. Therefore, for large break LOCA the pumps are left running and appropriate models for the degradation of pump performance with increasing void fraction are incorporated.

The SBLOCA has been analyzed assuming that the reactor coolant pumps trip and coast down coincident with reactor trip. This results in the coolant inventory change due to loss out the break and HPIS injection being reflected by the reactor vessel mixture level. The break size which resulted in the highest PCT was determined by a break spectrum analysis. This scenario was expected to represent the worst case SBLOCA, since if the reactor coolant pumps were running, the core would be cooled by pumping a two-phase mixture through the core, and no heatup would occur. Recent studies (Reference 14 on page 15-66) have shown that for certain SBLOCAs characterized by a limited range of break sizes and break locations, that a delayed reactor coolant pump trip at high system void fractions can result in extended core uncovery and consequences in excess of the 10CFR50.46 criteria. This constituted a new worst case scenario. This situation resulted in the implementation of operating procedures which instruct the operator to trip the reactor coolant pumps upon loss of subcooled margin (Reference 15 on page 15-66).

#### 15.14.3.3.6 ECCS Performance and Single Failure Assumption

The ECCS is comprised of two passive core flood tanks (CFT), each of which injects through its associated core flood line into the reactor vessel downcomer; three low pressure injection pumps separated into two trains which inject into separate core flood lines; and three high pressure injection pumps separated into two trains which split and inject into each cold leg. The ECCS configuration was analyzed to determine the worst single failure in addition to the assumption of the loss of offsite power for each LOCA. Historically, the worst single failure for a LOCA is the loss of one bus of emergency power which results in the loss of one train of HPI and one train of LPI. With the assumed loss of offsite power, this single failure results in a delay of 35 seconds until ECCS fluid is delivered to the RCS. The failure of transformer CT-4 has been identified as a more limiting single failure for the large break LOCA (Reference 33 on page 15-67). With the assumed loss of offsite power, this single failure results in a 48-second delay until ECCS fluid is delivered to the RCS (Reference 33 on page 15-67). Both ECCS trains would be available at 48 seconds. Reference 33 on page 15-67 demonstrates that having two ECCS

representation which allows for the prediction of counter-current flow. A change to the evaluation model was made in order to yield better agreement between the CRAFT2 and FOAM2 predicted steam escape rates. A bubble velocity multiplier of 2.38 is applied in the core and a multiplier of 2.0 in the remaining vessel volumes was incorporated (Reference 9 on page 15-66).

As a result of NUREG-0737, Section II.K.3.30, the SBLOCA evaluation model has been modified (Reference 32 on page 15-67). The modifications include a non-equilibrium pressurizer model, revised emergency feedwater model, two-phase RC pump model, and a more mechanistic steam generator model. The more mechanistic steam generator model includes a more detailed nodalization than is given in Figure 15-49.

## 15.14.3.3 Thermal Hydraulic Assumptions

Thermal hydraulic conditions and parameters are assumed in accordance with Appendix K.

### 15.14.3.3.1 Sources of Heat

The reactor is initially operating at 102 percent of 2,772 MWt, the maximum rated power for an Oconee class plant. Core peaking factors are obtained from the analysis based on the criteria of 10CFR50.46.
1 Core stored energy and fuel temperatures are calculated using the TACO2 or TACO3 code (References
1 10 on page 15-66 and 35 on page 15-67). Fission product decay heat is given by 1.2 times the ANS standard and decay of actinides is also assumed greater than the ANS decay curve. Direct moderator heating accounts for 2.7 percent of the fission energy released during the blowdown. Metal-water reaction is calculated using the Baker-Just equation without steam limiting. Heat transfer from non-fuel sources is accounted for, as is primary to secondary heat transfer.

### 15.14.3.3.2 Fuel Mechanical and Thermal Response

The detailed fuel response throughout the duration of the transient is predicted by the CRAFT2 and THETA1-B codes. Thermal expansion, elastic and plastic deformation, and the events leading to possible clad rupture are considered. Approved models for heat capacity and conductivity in the fuel, and gap conductance and heat transfer are used.

As a result of ongoing research programs, the NRC developed new models for fuel clad swelling and rupture which indicated that vendor evaluation models might be nonconservative (Reference 11 on page 15-66). These new models for cladding, swelling and rupture are described in NUREG-0630. In response to this NRC concern, a bounding assessment of the impact of NUREG-0630 on the LOCA linear heat rate limits has been performed (Reference 12 on page 15-66). The results of this assessment are discussed in Section 15.14.4.2, "Limiting Linear Heat Rate Analysis (LOCA Limits)" on page 15-60.

### 15.14.3.3.3 Blowdown Model

Break flow is calculated using the orifice equation for qualities up to 0.0 at which time a switch to the Moody correlation occurs. Discharge coefficients of 0.6, 0.8, and 1.0 are applied to each break. ECCS bypass is predicted to occur as long as the flow velocity is calculated to be sufficient to carry the ECCS fluid away from the core. The end of blowdown is considered either when zero leak flow occurs or when ECCS water starts entering the core. Friction and form loss factors account for system pressure drops and compare well with measured plant data. Single-phase and two-phase pump models are derived from homologous relationships. Core flow including cross flow is based on a correlation of experimental data.
0 The critical heat flux (CHF) correlations used are the B&W-2, BWC, Barnett, and modified Barnett. Pre-CHF heat transfer uses the Dittus-Boelter correlation for forced convection, the Thom correlation for nucleate boiling, and the Schrock and Grossman correlation for forced convection vaporization

### 15.14.2.5 Long-Term Cooling

After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

Conformance with the acceptance criteria must be demonstrated in a LOCA analysis which is conducted within the guidelines of 10CFR50 Appendix K, "ECCS Evaluation Models." Appendix K outlines the assumptions and analytical methods which have been accepted by the Nuclear Regulatory Commission (NRC) for evaluating the consequences of LOCA. The ECCS evaluation model applicable to Oconee is detailed in the following section.

## 15.14.3 ECCS EVALUATION MODEL

### 15.14.3.1 Methodology and Computer Code Description

The evaluation model which has been approved by the NRC for simulating the response of Oconee type plants to a LOCA is detailed in the B&W topical report, "B&W's ECCS Evaluation Model" (Reference 1 on page 15-66). The evaluation model consists of an integrated application of computer codes that calculate the system response from blowdown to peak clad temperature (PCT). Each code consists of models and assumptions which have been shown to be in accordance with Appendix K.

The CRAFT2 code (Reference 2 on page 15-66), which is a modified version of the FLASH-2 code (Reference 3 on page 15-66), solves the evolution of system hydrodynamics and core power generation during blowdown. The REFLOD3 code (Reference 4 on page 15-66) is used to determine the length of the refill period and the flooding rates during reflood. The CONTEMPT code (Reference 5 on page 15-66) calculates the Reactor Building pressure response. The THETAl-B code (Reference 6 on page 15-66) is used with the output from CRAFT2, REFLOD3, and CONTEMPT to determine the fuel thermal and mechanical response and the PCT. For the small break LOCA (SBLOCA) which results in core uncovery, the core mixture level is calculated by the FOAM2 code (Reference 7 on page 15-66). Clad metal-water reaction and hydrogen generation is calculated by the QUENCH code (Reference 8 on page 15-66). The code interfaces for the LOCA are shown in Figure 15-44, and for SBLOCA in Figure 15-45.

### 15.14.3.2 Simulation Model

The large break LOCA CRAFT2 simulation model is shown in Figure 15-47 and Figure 15-48. A noding scheme of 53 volumes and 86 junctions has been justified by sensitivity studies. All nodes except the pressurizer and the secondary side of the steam generators are treated as homogeneous. For break locations other than the pump discharge, the nodalization is appropriately modified. The core is divided into three radial regions - one for the hot fuel assembly, one for the eight assemblies surrounding the hot assembly, and one for the remainder of the core. Each radial region is divided into six axial levels in order to represent various axial flux shapes. The calculated flow, inlet enthalpy, power, and pressure transient from the CRAFT2 hot assembly is used as input to the THETA1-B code for the hot pin thermal response.

The SBLOCA CRAFT2 nodalization scheme is shown in Figure 15-49. This nodalization includes a revision to the nodalization given in Reference 1 on page 15-66 which separated the original single volume representing the core, core bypass, upper plenum, and upper head, into two volumes representing the core - core bypass, and the upper plenum - upper head (Reference 9 on page 15-66). Since the SBLOCA is a moderate transient, a less detailed model than that used for a large break can be utilized. All nodes use a bubble rise model and those flowpaths associated with the reactor vessel include a dual

# 15.14 LOSS OF COOLANT ACCIDENTS

## 15.14.1 IDENTIFICATION OF ACCIDENTS

A failure of the RCS pressure boundary will result in a loss of primary coolant inventory and the potential for the core to uncover. These hypothetical failures are considered to occur in all piping and components up to and including a double-ended rupture of the largest pipe in the system. If the core is not rapidly reflooded and long term heat removal established, decay heat will cause the fuel cladding to fail and release the fission product inventory. The ECCS is designed to deliver sufficient coolant to provide the necessary core decay heat removal for all credible LOCA's.

## 15.14.2 ACCEPTANCE CRITERIA

In order to judge the acceptability of the performance of the ECCS in mitigating a LOCA, the Final Acceptance Criteria specified in 10CFR50.46 require that the results of the LOCA analysis meet the following criteria.

### 15.14.2.1 Peak Cladding Temperature

The calculated maximum fuel element cladding temperature shall not exceed 2200°F.

### 15.14.2.2 Maximum Cladding Oxidation

The calculated total oxidation of the cladding shall nowhere exceed 0.17 times the total cladding thickness before oxidation. As used in this subparagraph total oxidation means the total thickness of cladding metal that would be locally converted to oxide if all the oxygen absorbed by and reacted with the cladding locally were converted to stoichiometric zirconium dioxide. If cladding rupture is calculated to occur, the inside surfaces of the cladding shall be included in the oxidation, beginning at the calculated time of rupture. Cladding thickness before oxidation means the radial distance from inside to outside the cladding, after any calculated rupture or swelling has occurred but before significant oxidation. Where the calculated conditions of transient pressure and temperature lead to a predication of cladding swelling, with or without cladding rupture, the unoxidized cladding thickness shall be defined as the cladding cross-sectional area, taken at a horizontal plane at the elevation of the rupture, if it occurs, or at the elevation of the highest cladding temperature if no rupture is calculated to occur, divided by the average circumference at that elevation. For ruptured cladding the circumference does not include the rupture opening.

### 15.14.2.3 Maximum Hydrogen Generation

The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

### 15.14.2.4 Coolable Geometry

Calculated changes in core geometry shall be such that the core remains amenable to cooling.

## 15.13.6 REFERENCES

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2. Parker, W. O. Jr., Letter to Case, E. G. (NRC), December 30, 1977.
3. Thies, A. C., Letter to Giambusso, A. (NRC), April 25, 1973.
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5. Parker, W. O. Jr., Letter to Denton, H. R. (NRC), October 5, 1979.
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7. Parker, W. O. Jr., Letter to O'Reilly, J. P. (NRC), May 7, 1980.
8. Parker, W. O. Jr., Letter to O'Reilly, J. P. (NRC), September 14, 1979.
9. Stolz, J. F. (NRC) to Tucker, H. B., October 14, 1982. Subject: Safety and Technical Evaluation Report of Duke's Response to IE Bulletin 80-04.
10. Stolz, J. F. (NRC), letter to Tucker, H. B. (Duke), transmitting the Safety Evaluation by the Office of Nuclear Reactor Regulation on Oconee LER 269/86-006 (More Negative Moderator Temperature Coefficient Than Assumed in Safety Analysis), November 26, 1986.
11. NPGD-TM-89, SECRUP An Analog Computer Program for Transient Analysis of Steam Line Rupture, B&W, March, 1970.

Changes to operating procedures now require the operator to trip the reactor coolant pumps in the event of actuation of the ESFS on low RCS pressure. This was ordered to aid in the mitigation of small break LOCA. Since steam line breaks can actuate this signal, the consequences were evaluated (Reference 8 on page 15-54). Similar to the earlier comments regarding loss of offsite power, the transient response including operator action to trip the reactor coolant pumps was less severe.

The reactor coolant pump trip criterion was subsequently changed from ESFS actuation to loss of cubcooled margin in the RCS. For the steam line break transient, the two events occur almost simultaneously. Therefore the evaluation of Reference 8 on page 15-54 remains valid.

The assumption that the tube wall thickness is reduced to one-half the original value is used to demonstrate that the chance of rupture is slight even when extreme effects of erosion, corrosion, vibration or leakage are considered. Results of tests conducted by B&W show that the tubes are not degraded to such an extent by those effects.

The steam line blowdown loads have been analyzed empirically and by simulation, and the results indicate these loads will not cause tube failure.

## 15.13.5 CONCLUSIONS

The steam line break accident has been analyzed with several assumptions regarding ICS and operator action to isolate feedwater. The results show that the unit can successfully mitigate the transient without taking credit for the ICS or operator action, although normal ICS and operator action will significantly moderate the plant response. The maximum core thermal power before reactor trip is only 106 percent of rated power. The peak return to power is not great enough to cause fuel damage.

The environmental consequences from this accident are calculated by assuming for the case analyzed in 15.13.3.1, "With ICS and Operator Action" on page 15-49 that:

1. The unit has been operating with a 1 gal/min steam generator tube leak.
2. The unit has been operating with 1 percent defective fuel rods.
3. The steam line break occurs between the Reactor Building and a turbine stop valve.
4. Reactor coolant leakage into the steam generator continues unabated for 3 hr before the RCS can be cooled down and the leakage terminated.

With these assumptions, the iodine inventory in the steam generator containing the 1 gal/min tube leak is 0.018 dose equivalent curie I-131. The steam line break is assumed to result in the release of the activity contained in the steam generator inventory, the activity contained in 12,800 lb feedwater, the activity contained in 35,500 lb of feedwater in the feedwater line, and the activity contained in 180 gal of reactor coolant. The iodine, a total of 3.11 dose equivalent curies I-131, primarily resulting from continued steam generator leakage in the 3 hr following the accident, is assumed to be released directly to atmosphere where it mixes in the wake of the Reactor Building. Atmospheric dilution is calculated using the 2-hr. ground release dispersion factor of $1.16 \times 10^{-4}$ sec/m$^3$. With these assumptions, the total integrated dose to the thyroid is 0.20 rem at the exclusion distance. The corresponding whole body dose is 0.002 rem.

An evaluation has shown that the peak stresses in the steam generator tubes as a result of the steam line break accident will not result in tube rupture. However, the radiological dose consequences assuming 1, 3, and 10 ruptured tubes concurrent with a steam line break accident have been calculated (References 1 on page 15-54, 2 on page 15-54). The resulting 2-hr. thyroid doses were 13.3, 13.7, and 7.3 rem, respectively.

The potential for a steam line break to be adversely impacted due to the consequences of pipe whip and jet impingement was addressed and resulted in the implementation of several station modifications (References 3 on page 15-54, 4 on page 15-54). Similarly, the potential for the resulting Reactor Building environment to cause an interaction with non-safety-grade control systems has been addressed (Reference 5 on page 15-54).

The steam line break analyses have been reviewed to determine the impact of the upgraded EFWS. An evaluation showed that the performance of the EFWS does not affect the FSAR analyses (References 6 on page 15-54, 7 on page 15-54).

| | |
|---|---|
| Secondary Pressure | 0 psi |
| Mean Shell Temperature | 553°F |
| Mean Tube Temperature | 453°F |

Power Level - 100 percent prior to rupture.

The shell temperature is not assumed to change as rapidly as tube temperature during the steam line rupture transient.

The pressure and temperature conditions produce axial tensile tube stresses as follows:

| | | |
|---|---|---|
| 1. Design size tube (no degradation) | tube bundle @ edge | 15.2 ksi |
| | tube bundle @ center | 7.6 ksi |
| 2. Degraded tube with assumed 1/2 original wall thickness | | |
| | tube bundle @ edge | 31.5 ksi |
| | tube bundle @ center | 15.3 ksi |

For the assumed worst case (2) the maximum stress intensity is calculated as follows:

<u>Stress Intensities</u>

For Edge Tube (Worst Case)

$\sigma_L = 31.5$ ksi

$$\sigma_c = \frac{2185(.574)}{0.017(2)} = 37.0 \text{ ksi}$$

$\sigma_R = -2.2$ ksi

Mean Diam $= 0.557 + 0.017 = .574$

$\sigma_C = 37.0$ ksi

$$\sigma_C = \frac{P\ r_M}{t}$$

$$r_M = \frac{.574}{2}$$

$\sigma_{C-R} = 39.2$ ksi

$\sigma_{L-R} = 33.7$ ksi

$\sigma_{L-C} = 5.5$ ksi

As shown above, the circumferential-radial stress combination yield the maximum stress intensity which is with the allowable stress limits, as calculated below:

$\sigma$ circumferential - $\sigma$ radial $= 39.2$ ksi $< 1.5\ (1.2)\ S_m$

39.2 ksi < 42.0 ksi

The degraded tube primary plus secondary stress intensity is less than the Oconee Nuclear Station FSAR Case III stress limit for primary stresses resulting from design loads plus pipe rupture loads.

The axial stress in the presumed degraded tube is at the yield point. However, the nature of thermal restraint stresses limits the amount of tube deformation during the steam line rupture transient.

unaffected steam generator increases and the turbine bypass opens at 9 sec after isolation, for about 6 sec, until the primary system temperature decreases below 550°F. The unaffected steam generator then remains in a reverse heat transfer mode until the transient stabilizes. The ICS closes all feedwater valves on both steam generators in order to reduce level to the normal minimum level control. The operator then places the feedwater controller on the affected steam generator in manual to prevent further feedwater addition. The affected steam generator blows dry in approximately 50 sec, effectively terminating the transient. The unaffected steam generator will provide for decay heat removal. The minimum subcritical margin of 0.4 percent Δk/k occurs at 50 sec.

### 15.13.3.2 Without ICS and Operator Action

The analysis in Section 15.13.3.1, "With ICS and Operator Action" on page 15-49 is reanalyzed assuming no operator action to assure feedwater isolation to the affected steam generator, and also assuming no operator action and no ICS action.

For the case in Section 15.13.3.1, "With ICS and Operator Action" on page 15-49 modified to assume no operator action to isolate feedwater, the initial 50 sec are not affected. However, as the level in the affected steam generator decreases below the minimum level setpoint, the ICS would open the feedwater control valves in order to maintain the minimum level. This results in a continued cooldown of the RCS. The HPI pump is actuated at 23 sec and the core flood tanks begin injecting at 75 sec. The boron injected by these systems keeps the core subcritical until returning to about 1 percent power just before the LPIS is actuated at 170 sec. LPI injection assures core subcriticality for the duration of the transient. The results of this analysis are shown in Figure 15-41 through Figure 15-43. If the ICS is also assumed to initiate the turbine-driven emergency feedwater pump in order to maintain minimum level, the resulting cooling effect returns the reactor to 35 percent of rated power in 65 sec. Assuming a nominal tripped rod worth of 5.66 percent Δk/k with the maximum rod stuck out, the reactor will return to less than 25 percent of rated power in about 350 sec. If the nominal tripped rod worth is used without consideration for the stuck rod, the reactor will remain subcritical.

The worst case assumptions for the steam line break accident include, in addition to no operator action, that the ICS does not perform its design function of controlling feedwater on minimum steam generator level. 135 percent flow from one main feedwater pump is assumed to be delivered to the affected steam generator following reactor trip on low pressure at 6 sec. The HPIS actuates at 30 sec, core flood tanks initiate injection at approximately 60 sec, and the LPIS actuates at 100 sec. The return to power peaks at 8 percent rated power and returns subcritical at 166 sec.

## 15.13.4 CONCURRENT STEAM GENERATOR TUBE RUPTURE ASSESSMENT

Based upon the results of testing and analysis done by B&W, steam generator tubes will not rupture concurrently with rupture of the main steam line as a result of blowdown loadings. Data taken from a steam line rupture test on a 37-tube model OTSG show that during the transient, the tubes become 100°F cooler than the shell. The steam line rupture test has no feedwater flow after approximately 80 sec. even though the worst case accident assumed continuous feedwater flow controlling on minimum level. The lack of feedwater flow causes the steam pressure to drop rapidly with an associated drop in the saturation temperature. This rapid drop in steam temperature causes the 100°F tube to shell temperature difference.

The calculations used to arrive at the tube stress intensity for an assumed worst case degraded tube are based on pressure and temperature conditions as follows:

| | |
|---|---|
| Primary Pressure | 2185 psi |

# 15.13 STEAM LINE BREAK ACCIDENT

## 15.13.1 IDENTIFICATION OF ACCIDENT

The worst case overcooling accident is the double-ended rupture of a 34 in. main steam line from rated power conditions with offsite power available. At rated power the steam generator inventory is at its maximum, so that the blowdown will result in the greatest heat removal. If a loss of offsite power was assumed, the heat transfer in the steam generator would be less due to the loss of forced flow and the loss of the Main Feedwater System. Therefore, these initial conditions result in the most rapid cooldown of the RCS.

The criteria for unit protection and the release of fission products to the environment for the steam line break accident are as follows:

1. The core will remain intact for effective core cooling, assuming minimum tripped rod worth with a stuck rod.
2. No steam generator tube loss of primary boundary integrity will occur due to the loss of secondary side pressure and resultant temperature gradients.
3. Doses will be within 10CFR100 limits.

## 15.13.2 METHODS OF ANALYSIS

The steam line break accident is analyzed using a hybrid computer simulation which includes primary and secondary loop models, a pressurizer model, point kinetics, and all necessary interfacing systems, control systems, and trip logic. A summary of the parameters utilized in the analyses is given in Table 15-5. The largest possible break size is selected in order to maximize the resulting cooldown. The kinetics parameters for EOC are assumed since the most negative moderator coefficient occurs at EOC. In conjunction with the cooldown there is therefore the potential for the reactor to return to power. Feedwater delivery to the steam generators following a steam line break is a major factor in the system response. The impact of ICS and operator action to control feedwater is analyzed in detail. Although two HPI and two LPI pumps are available, only one of each delivering the design flow at the initiation pressure is assumed. The boron injected by the ECCS is assumed to mix homogenously with the RCS coolant inventory. Structural metal heat capacity was not included in the analysis.

## 15.13.3 ANALYSIS AND RESULTS

### 15.13.3.1 With ICS and Operator Action

The steam line break accident is analyzed assuming normal ICS control of feedwater and prompt operator action to isolate feedwater from the affected steam generator. The results of the analysis are shown in Figure 15-40.

The failure of a main steam line initiates a simultaneous blowdown of both steam generators. The blowdown causes a rapid increase in the heat removal from the primary system which results in a depressurization and cooldown of the RCS. With the assumed EOC moderator coefficient, the reduction in moderator temperature will cause an increase in reactor power. A reactor trip on high power or low pressure will occur at approximately 6 sec. Reactor trip trips the turbine which closes the turbine stop valves and isolates the unaffected steam generator from the affected steam generator. The pressure in the

## 15.12.6 REFERENCES


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effect and terminated by the RPS with no serious core damage or additional loss of the coolant system integrity. Furthermore, it has been shown that an ejected rod worth greater than 1.52 percent Δk/k would be required to cause a pressure pulse, due to prompt dispersal of fragmented fuel and zirconium-water reaction, of sufficient magnitude to cause rupture of the pressure vessel.

As a result of the postulated pressure housing failure associated with the accident, reactor coolant is lost from the system. The rate of mass and energy input to the Reactor Building is considerably lower than that reported in the LOCA analysis and results in a much lower Reactor Building pressure. The maximum hole size resulting from a rod ejection is approximately 1.75 in.

The environmental consequences of this accident are calculated assuming that all fuel rods undergoing DNB release all of their gap activity to the reactor coolant. Subsequently, this gap activity and the activity in the reactor coolant from operation with 1 percent defective fuel pins is released to the Reactor Building. For the case of a 0.65 percent Δk/k rod ejection from rated power at BOC, 28 percent of the fuel rods are asumed to fail, releasing that fraction of the nuclide inventory in the gap given in Table 15-4.

Using environmental models and dose rate calculational methods discussed in Section 15.14.7, "Environmental Evaluation" on page 15-64, the total integrated 2-hr. dose at the 1-mi exclusion distance is 1.44 rem thyroid and 0.004 rem whole body. The total integrated thyroid dose at the 6-mi. low population zone distance is 1.57 rem for 30-day exposure. These doses are well below the guideline values of 10CFR100.

## 15.12.4 REACTOR VESSEL INTEGRITY ASSESSMENT

The reactor vessel has been analyzed to estimate the margin that exists between the rod worths assumed for the calculated rod ejection accident transients and those worths that could initiate RCS failure. The pressure vessel material is SA-533 Grade B steel. Table 15-3 lists the values used in this analysis. The radial deformation which is assumed to represent failure of the vessel is 50 percent of the total elongation, or 0.13 in./in. To calculate the weight of an explosive charge required to reach 50 percent elongation, the vessel is simulated by a single cylinder with the same outer diameter as the actual vessel, but with an increased thickness to account for the thermal shield and core barrel.

The expression used for the weight of explosive required to strain the vessel a given amount is (Reference 10 on page 15-47):

$$W = \left[ \frac{1.407\, E_s(3.41 + 0.117\, R_i/t)\,(R_e^2 - R_i^2)^{1.85}}{10^5 w^{-0.85}(1.47 + 0.0373\, R_i/t)^{0.15}(R_i)^{0.15}} \right]^{0.811}$$

where

$W$ = charge weight (TNTor Pentolite), lb

$w$ = weight density of vessel material, lb/ft$^3$

$R_i$ = initial internal radius of vessel, ft

$R_e$ = initial external radius of vessel, ft

$t$ = initial wall thickness of vessel wall, ft

$E_s$ = wall strain energy, in.-lb/in.$^3$

Using this formula on the equivalent vessel, the required weight of explosive charge is calculated. The results of this calculation indicate that 1,410 pounds of TNT would strain the mid-meridian ring up to the 50 percent $\varepsilon_u$, i.e., 0.13 in./in. The 1,410 lb of TNT has an energy equivalent of 6.74 x $10^8$ cal.

An analysis of ejected rod worths higher than those reported in the preceding sections has been made to estimate the transient required to generate the deformation energy equivalent to 1,410 lb of TNT. These cases are evaluated to find the amounts of fuel melting and zirconium-water reaction. Using the conservative assumption that all the fuel that exceeds the melting threshold is fragmented, dispersed into the coolant, and quenched to the coolant average temperature, a total thermal energy release can be determined. The conversion of this energy release to an equivalent deformation energy is dependent upon the duration of the release. TNT has an energy release in micro-sec, and a deformation conversion efficiency of about 50 percent. The energy generated during a reactor transient from the zirconium-water reaction and a molten fuel dispersal is in the range from millisec to sec. Thus, the conversion efficiency to deformation energy would be considerably less, and is assumed to be 1/5 that of TNT (Reference 11 on page 15-47). Using these figures, the reactor vessel capability is 3.37 x $10^8$ cal, and, under the foregoing assumptions, a reactivity addition of 1.52 percent $\Delta k/k$ is required to release this much energy to vessel deformation.

## 15.12.5 CONCLUSIONS

The hypothetical rod ejection accident has been investigated in detail at different initial reactor power levels, rated power and zero power, and both BOC and EOC conditions were considered. The results of the analysis prove that the reactivity transient resulting from this accident will be limited by the Doppler

ejection transient is assumed to occur whenever the peak thermal power of a given fuel rod exceeds the peak at steady-state conditions which could result in DNB, which in turn is assumed to occur for a DNBR of 1.3 using the W-3 correlation.

In determining the environmental consequences from this accident, an even more conservative approach is taken in computing the extent of DNB experienced in the core. All fuel rods that undergo DNB to any extent are assumed to experience cladding failure with subsequent release of all the gap activity.

Actually, most of the fuel rods will recover from DNB and no fission product release will occur. The fuel rods that experience DNB at BOC are assumed to have EOC gap activities.

## 15.12.3 ANALYSIS AND RESULTS

The rod ejection accident is analyzed over a wide range of initial conditions in order to identify the worst case. The initial conditions that result in the most severe core thermal response are the maximum rod worth, 0.65 percent $\Delta k/k$, from rated power at BOC. The results of this analysis are shown in Figure 15-29 and Figure 15-30. Figure 15-29 shows that the neutron power peaks at 710 percent prior to reactor trip which occurs at 0.6 sec. System pressure increases to 2450 psig due to the increased heat transfer to the coolant. Figure 15-30 shows that the DNBR reaches the 1.3 limit at 0.4 sec, at which time the heat transfer mode switches from nucleate to film boiling. The heat transfer coefficient drops to 450 Btu/hr-ft$^2$-°F initiating a cladding temperature increase to a maximum hot spot temperature of 1560 °F. The fuel temperature response indicates that fuel melting occurs at the hot spot between 0.8 and 1.9 sec. A parameter study was performed to determine the percentage of fuel pins that would experience a DNBR less than or equal to 1.3. It was determined that for the ejected rod worth analyzed about 28 percent of the pins would undergo DNB. The maximum hot spot fuel enthalpy is calculated to be 147 cal/g.

Sensitivity studies on rod worth, Doppler coefficient, moderator coefficient, and trip delay time were performed for rated power and $10^{-3}$ rated power for both BOC and EOC conditions. These analyses were performed for undensified fuel, however the trends of the results are applicable for densified fuel. The results of these analyses are shown in Figure 15-31 through Figure 15-39.

Figure 15-31 and Figure 15-32 shows the effect of rod worth on the resulting neutron and thermal powers. The EOC cases appear to be more rapid excursions than BOC. This is a result of the burnup dependence of the $\beta$-eff parameter, since the kinetic worth of the rod is determined by the ratio $\rho/\beta$-eff rather than $\rho$. For the zero power cases the impact of $\beta$-eff in relation to prompt critical is clearly evident in the slope change of the curves. All rated power cases trip on high flux, whereas the rod worth determines whether high flux or high pressure provides the trip for the zero power cases. Comparing the neutron and thermal power responses for Figure 15-29 and Figure 15-30 with Figure 15-31 and Figure 15-32, it is apparent that with the assumption of fuel densification the results of the rod ejection accident are less severe.

Figure 15-33 shows that the peak fuel enthalpy occurs for the rated power BOC case. The zero power cases do not exceed 75 cal/g. Again comparing with the results of the fuel densification analysis, it is apparent that the results are less severe for densified fuel. Figure 15-34 through Figure 15-37 indicate that the rated power cases are not sensitive to variations in the Doppler coefficient and moderator coefficient. The zero power cases are more sensitive but remain less severe than the rated power cases.

Oconee 3 has the Type C control rod drive mechanism with an additional 0.1 sec trip delay for both high flux and high pressure trips. The sensitivity to the trip delay time is demonstrated to be small in Figure 15-38 and Figure 15-39.

internal pressures developed in the fuel rod are insufficient to cause cladding rupture, but subsequent heat transfer from fuel to cladding raises the temperature of the cladding and weakens it until local failure occurs. DNB usually accompanies and contributes to this mode of failure, and little or no fuel melting would be expected. In this mode of failure, fuel fragmentation is usually only minor, and any dispersal of fuel to the coolant would occur very gradually, with system contamination being the worst probable consequence.

The second failure mode occurs when significant fuel melting causes a rapid increase in internal fuel rod pressure which, combined with clad loss of strength at higher temperatures, causes the fuel rod clad to rupture. The increase in volume associated with the melting of $UO_2$ is 9.6 percent (Reference 6 on page 15-47). Some fuel vaporization may occur, contributing to the pressure buildup. Considerable fragmentation and dispersal of the fuel would be expected in this mode.

The third and most serious mode of fuel rod failure occurs when, as a result of a very large and rapid reactivity transient in which there is insufficient time for heat to be transferred from fuel to cladding, extensive fuel melting followed by vaporization occurs. Destructive internal pressures can be generated without increasing cladding temperatures significantly in this mode.

In evaluating the effects of the failure modes discussed above, two failure thresholds are considered. The first is associated with a gradual, and usually minor, cladding failure and may be approximately defined by the minimum heat flux for DNB at the cladding surface. The second failure threshold, defined as the enthalpy threshold for prompt fuel failure with significant fragmentation and dispersal of fuel and cladding into the coolant, is used to describe the energy required to cause failure by either the second or the third failure mode described above.

A correlation of the results of different experiments conducted on Zircaloy-2 clad $UO_2$ fuel rods at TREAT (Reference 7 on page 15-47) has been interpreted by the experiments to show a threshold at 280 cal/g of fission energy input. That is, below the value the fuel rod can be expected to remain intact, and above this value fragmentation can be expected. The enthalpy corresponding to the melting point of $UO_2$ is about 260 cal/g (Reference 8 on page 15-47), and the heat of fusion is at least 78 cal/g (Reference 9 on page 15-47). Thus the 280 cal/g represents a condition where only part of the fuel is molten. Also of interest as a probable indication of the degree and rapidity of fuel and cladding dispersal are the measurements of pressure rise rates in the autoclave in the TREAT experiments. Preliminary analysis indicates that there is only a modest pressure rise up to an energy input of 400 cal/g. Above 500 cal/g, however, there is a very definite pressure pulse. Thus between 400 and 500 cal/g there is a transition which probably corresponds to the change from the second to the third failure mode discussed previously. A fuel failure threshold of 280 cal/g, at the pellet radius corresponding to the average temperature of the hottest fuel pellet, has been used in this study to define the extent of fuel failure.

In computing the average enthalpy of the hottest fuel pellet during the excursion for the rated power cases, it is assumed that no heat is transferred from the fuel rod between the time the accident is initiated and the time when the neutron power returns to the rated power level. For the zero-power cases, the enthalpy increase was based on the peak value of the average fuel temperature. In all cases the average enthalpy rise, from the integrated energy or the fuel temperature traces, is multiplied by the maximum peaking factor to obtain the enthalpy increase in the hottest fuel pellet.

The latest correlation of the ANL TREAT data for the meltdown experiments on Zircaloy-2 clad $UO_2$ fuel rods shows the threshold for the zirconium-water reaction to be 210-220 cal/g energy input. A conservative threshold value of 200 cal/g is used in this study.

In calculating the volume of the core that experiences burnout in a given rod ejection accident, it is assumed that any DNB conditions result in burn out for each rod where the DNB occurs. DNB in a rod

Since most of the reactivity is added during the central 75 per cent of this travel, only this distance is used in the analysis, resulting in an ejection time of 0.15 sec. for the analysis.

As a check on the point kinetics calculation, the rod ejection accident was also analyzed for a limited number of cases using the exact, one-dimensional, space-and-time dependent WIGL2 digital computer program (Reference 5 on page 15-47). The point kinetics model assumes that the flux shape remains constant during a transient. This flux shape contains peaking factors which reflect unusual rod patterns such as the flux adjacent to a position where a high worth rod has been removed. Therefore, these point kinetics peaking factors are much higher than any that would actually occur in the core during normal operation. The purpose of using an exact space-time calculation is to find the flux shape during a transient. But to have a transient where a rod is ejected from the core, one must start with a flux shape that is necessarily depressed in the region of the ejected rod. In fact, the higher the worth of the rod, the more severe becomes the depression. This flux depression also causes a fuel temperature depression. When the rod is ejected from this position, the flux quickly assumes a shape that shows some local peaking.

Results from WIGL2 indicate that for rod worths greater than 0.2 percent $\Delta k/k$ this local peaking is in excess of the maximum peaking applied to the point kinetics results. However, when this "exact" peaking is applied to a region initially at depressed fuel temperatures, as it is in the case of the region adjacent to the ejected rod, the resultant energy deposited in this region causes a lower peak temperature and peak thermal power than does applying a lower maximum peaking factor to an undepressed peak power region. The result is that this local region simulated in the WIGL2 code actually undergoes a less severe transient than the hottest fuel rod assumed in the point kinetics model. This result is uniformly true for all rod worths up through 0.5 percent $\Delta k/k$.

Comparison of Space-Dependent and Point Kinetics Results
on the Fuel Enthalpy

| Rod Worth (%Δk/k) | BOC Rated Power Peak-to-Average Values | | Fuel Enthalpy (cal/g) | |
|---|---|---|---|---|
| | WIGL2 | Point Kinetics | WIGL2 | Point Kinetics |
| 0.2 | 4.2 | 3.24 | 72 | 141 |
| 0.3 | 4.8 | 3.24 | 86 | 149 |
| 0.4 | 5.4 | 3.24 | 109 | 159 |
| 0.5 | 6.0 | 3.24 | 143 | 172 |

Thus it can be seen that the space-time dependent code gives a less conservative treatment of the accident analysis than does the point kinetics code.

The consequences of a rod ejection accident are largely dependent upon the rate at which the thermal energy resulting from the nuclear excursion is released to the coolant. If the fuel rods remain intact while the excursion is being terminated by the negative Doppler coefficient and by reactor trip, then the energy release rate is limited by a relatively low surface-to-volume ratio for heat transfer. The energy stored in the fuel rods will then be gradually released to the coolant, over a period of several seconds, at a rate which poses no threat to the integrity of the RCS. However, if the magnitude of the nuclear excursion is such that the fuel rod cladding does not remain intact, then fuel and clad may be dispersed into the coolant to such an extent as to cause a significant increase in the heat transfer rate.

Power excursions caused by reactivity disturbances of the order of magnitude occurring in rod ejection accidents could lead to three potential modes of fuel rod failure. Failure by the first mode occurs when

# 15.12 ROD EJECTION ACCIDENT

## 15.12.1 IDENTIFICATION OF ACCIDENT

Reactivity excursions initiated by uncontrolled rod withdrawal are shown to be safely terminated without damage to the reactor core or RCS integrity. For reactivity to be added to the core at a more rapid rate, physical failure of a pressure barrier component in the control rod drive assembly must occur. Such a failure could cause a pressure differential to act on a control rod assembly and rapidly eject the assembly from the core region. The power excursion due to the rapid increase in reactivity is limited by the Doppler effect and terminated by the RPS.

Control rod assemblies are used to control load transients only, with soluble boron used to compensate for fuel depletion. Current reload core designs allow only partial insertion of one full length control rod group at rated power. The design criteria for the rods in that group require that each rod must be worth less than 1.0 percent $\Delta k/k$ at hot zero power and less than 0.65 percent $\Delta k/k$ at full power. An additional design criterion is that the boron concentration must be high enough to maintain a 1 percent $\Delta k/k$ shutdown margin at all times including the highest worth rod stuck in the fully withdrawn position. This criterion assures that a rod ejection from shutdown conditions will not cause a nuclear excursion.

The original rod ejection accident was reanalyzed to include the effects of fuel densification and an increase in the maximum ejected rod worth to 0.65 percent $\Delta k/k$ (References 1 on page 15-47, 2 on page 15-47, 3 on page 15-47). The results presented include these effects unless specified otherwise.

The criteria for the rod ejection accident are that the accident will not further damage the RCS, and that the offsite dose will be within the 10CFR100 limits. The first criterion is met if the reactivity excursion does not result in fuel vaporization and therefore the potential for an explosion in the core. This is shown by demonstrating that the peak fuel pellet enthalpy does not approach the threshold 280 cal/g.

## 15.12.2 METHODS OF ANALYSIS

A B&W digital computer program has been used to analyze the rod ejection accident. This program agrees to within a few percent in all cases with CHIC-KIN (Reference 4 on page 15-47). The B&W program is a point kinetics model with a primary loop and pressurizer model. The core heat transfer model allows for up to 30 radial mesh points in the fuel and clad, and the mesh size can be different in the two regions. The model accounts for the gap conductivity and film coefficient of heat transfer. Reactivity feedback is calculated in each mesh point and in the coolant and is weighted for inclusion in the kinetics simulation. The thermal properties are input separately for each mesh point but remain constant with time. The loop model includes a simulation of the steam generator which can have a nonlinear heat demand input on the secondary side. Trip action is initiated on high or low RCS pressure or on high neutron flux. Decay heat can be taken into account as well. This code was used to calculate the neutron and thermal power, integrated energy, reactivity components, pressure, and fuel rod and loop temperatures. Six delayed neutron and groups are considered. The control rod trip is represented by a 25-segment curve of reactivity insertion during trip versus time, obtained by combining the actual rod worth curve with a rod velocity curve. Nominal values for the various nuclear and physical parameters used as inputs are listed in Table 15-2. A rod must be fully inserted in the core to have the foregoing reactivity worth values. Assuming that the failure occurs so that the pressure barrier no longer offers any restriction to the ejection and that there is no viscous drag force limiting the rate of ejection, the control rod travel time to the top of the active region of the core is calculated to be 0.176 sec.

## 15.11.3 REFERENCES


5 1. DPC Engineering Calculation OSC-6070, "Offsite Dose Consequences From Fuel Handling
5 Accidents", dated March 21, 1996.

5 2. WCAP-7828, "Radiological Consequences of a Fuel Handling Accident", December 1971.

5 3. DPC Engineering Calculation OSC-6089, "Oconee Fission Product Source Terms", dated March 21,
5 1996.

5 4. Parker, W. O. Jr., Letter to Rusche, B. C. (NRC), March 10, 1977.

5 5. Parker, W. O. Jr., Letter to Rusche, B. C. (NRC), November 3, 1975.

5 6. Parker, W. O. Jr. (Duke), Letter to Denton, H. R. (NRC), July 25, 1980.

5 7. Tucker, H. B. (Duke), Letter to Denton, H. R. (NRC), November 19, 1985.

5 8. DPC Engineering Calculation OSC-2840, "Cask Drop Accidents - Radiological Consequences - ISFSI
5 Cask", dated Feb 26, 1988.

5 9. DPC Engineering Calculation OSC-3631 Rev 2, "Criticality Consequences of a Heavy Load Drop in
5 the Spent Fuel Pool", dated February 7, 1996.

5 10. Weins, L. A. (NRC), Letter (with attached Safety Evaluation) to Tucker, H. B. (DPC) dated
5 November 16, 1989.

5 11. Oconee Nuclear Station Independent Spent Fuel Storage Installation, Final Safety Analysis Report,
5 Chapter 8.

5 **15.11.2.5.2 Potential Damage to SFP Structures from Dry Storage Transfer Cask Drop**

3 The SFP is founded on rock, and the concrete floor slab is designed to withstand the 100 ton cask drop.
3 However, localized concrete could be crushed and the steel liner plate punctured in the area of dry storage
3 cask impact. For the purpose of analyzing the event, a gap of 1/64 inch for a perimeter of 308 inches in
3 the liner plate was assumed. The calculated leakage of pool water through the gap is 21.3 gallons per day.
3 This amount of water loss is within the capability of the SFP makeup sources. The cask is never carried
3 over any safety related equipment.

5 **15.11.2.5.3 Radiological Dose from Dry Storage Transfer Cask Drop**

3 The worst radiological consequences resulting from a dry storage cask drop accident into either the Unit
3 1&2 or the Unit 3 SFP were analyzed. The calculation assumes a total of 1024 SFAs would be damaged
3 in the Unit 1&2 SFP. Of this number, two full core inventories (354 SFAs) with worst case fission
3 product concentration and less than 1 year decay time are assumed to be present. For the Unit 3 SFP, all
3 825 fuel cell locations are assumed to contain SFAs that would be damaged by the cask drop. One full
3 core inventory (177 SFAs) with worst case fission product inventory and less than 1 year decay is
3 considered to be present in the Unit 3 pool. Thus, the analysis assumes 670 and 648 SFAs, for Unit 1&2
3 and Unit 3 SFPs respectively, have a minimum of 1 year decay time.

3 Oconee Technical Specification 3.8.13 requires that fuel stored in the first 64 rows closest to the cask
3 handling area be decayed a minimum of 65 days prior to movement of the dry storage transfer cask in the
3 Unit 1&2 SFP area. Likewise, all SFAs stored in the Unit 3 pool must be decayed a minimum of 57 days
3 before movement of the cask is permitted in that area. The maximum fission product inventories for the
3 iodine and noble gas nuclides of interest at times of 57 days, 65 days, and 1 year were calculated using the
3 N237 BURP computer program. This information, in conjunction with the assumed pool inventories,
3 was used to determine the curies of each nuclide released from the postulated cask drop accidents. The
3 total activity releases for each pool were used to determine the worst case offsite dose consequences
3 reported below:

| FUEL POOL | NUMBER OF SFAs DAMAGED | TOTAL BODY DOSE (REM) | THYROID DOSE (REM) |
|---|---|---|---|
| Units 1&2 | 1024 | 0.14 | 60 |
| Unit 3 | 825 | 0.12 | 60 |

3 The above doses, taken separately, are less than 25% of the 10 CFR Part 100 limits of 25 rem to the
3 whole body and 300 rem to the thyroid. Therefore, the accident dose criteria will not be exceeded for the
3 limiting postulated dry storage cask drop accident.

5 ## 15.11.2.5 Dry Storage Transfer Cask Drop Accident in Spent Fuel Pool Building

3 Dry storage transfer operations from the spent fuel pool (SFP) buildings to the Independent Spent Fuel
3 Storage Facility (ISFSI) are routinely performed at Oconee. The major steps in the process involve
3 transporting the transfer cask/dry storage canister (DSC) into the fuel building, placing into the SFP,
3 loading with 24 qualified fuel assemblies, drying/sealing, and removing to the ISFSI. The potential exists
3 for dropping the cask in the SFP area during transfer operations.

5 ### 15.11.2.5.1 Criticality Analyses for Dry Storage Transfer Cask Drop Scenarios

3 While the transfer cask is never carried directly over spent fuel, the potential always exists for failure of the
3 overhead crane or handling equipment. Thus, an analysis was performed assuming the cask, yoke, and
3 yoke block are deflected into the Unit 1&2 SFP. In such a case, it was postulated that 1024 spent fuel
3 assemblies (SFAs) would be damaged (the first 64 rows, each containing 16 SFAs). It was assumed that
3 220 fuel storage cells directly beneath the falling parts buckle and deflect into adjacent cells until all the
3 energy of the dropping cask is absorbed. For a cask drop in the smaller Unit 3 SFP, it was assumed all
3 825 fuel cell locations would be damaged.

5 The potential for criticality in the SFPs was analyzed using the methodology identified in NUREG-0612.
5 It was assumed the racks and fuel were deformed such that $k_{eff}$ was maximized. Credit was taken for pool
5 boron, boraflex, and stainless steel walls to determine the $k_{eff}$ under the assumed damage conditions. The
5 confirmatory calculations utilized a specific neutronic analysis for each SFP with the following
5 assumptions:

5 1. An infinite array of SFAs is crushed together into a geometry that optimizes $k_{eff}$.
5 2. The affected SFAs are unirradiated and have the maximum enrichment permitted for storage in the
5 Oconee SFPs.
5 3. The minimum technical specification for SFP boron concentration is maintained.

5 This is a Plant Condition IV event as defined in ANSI/ANS-57.2-1983. The acceptance criteria for this
5 accident, from Reference 10 on page 15-39, is that $k_{eff}$ will be less than or equal to 0.98 including all
5 uncertainties. A series of calculations involving cases of varied pin pitch modeling the crushed cells and
5 SFAs was performed. The maximum $k_{eff}$ value determined for the Unit 1&2 SFP was 0.97206. The
5 maximum $k_{eff}$ value calculated for the Unit 3 SFP was 0.94169. These analyses verify that subcriticality in
5 the SFP will be maintained after a dry storage cask drop accident.

3 The DSC internals are designed to prevent criticality during the wet loading and unloading process. As
5 long as a minimum SFP boron concentration of 2210 ppm is maintained, and the DSC is drained of
3 water within 50 hours of loading the SFAs, criticality is precluded. Strict administrative controls are in
3 place at Oconee to ensure the SFP boron concentration is maintained above the minimum required and
3 that the draining time limit not exceeded. Once the moderator (water) is removed, there is no criticality
3 concern during normal transportation of the DSC. However, the potential exists for the fully loaded cask
3 to fall a distance of 40-50 feet onto the transfer trailer and rupture, with damage possible to the DSC.
3 Even so, it is not possible for the 24 fuel assemblies contained in the DSC to be forced into a critical
3 configuration while surrounded by air or helium gas.

3 The consequences of dropping the dry storage transfer cask outside the fuel building are described in the
3 ISFSI FSAR (Reference 12).

energy in the storage racks. For additional conservatism it is assumed that the storage racks which are directly impacted by the falling load in turn buckle and deflect into adjacent racks until the total energy of the falling cask is absorbed. The Unit 1 and 2 spent fuel pool contains 154 fuel storage positions under the direct impact area, with a total of 576 spent fuel assemblies which can potentially suffer a loss of integrity during a cask drop accident. The Unit 3 pool contains 156 fuel storage positions under the projected impact area, with a total of 518 assemblies which can be damaged during the accident. These analyses are based on the TN-8 three element shipping cask.

Once the number of fuel assemblies which could be damaged is determined, dose analyses are performed which are consistent with Regulatory Guide 1.25, and NUREG-0612. The following assumptions apply:

1. Spent fuel stored in the first 36 rows of the Unit 1 and 2 spent fuel pool closest to the spent fuel cask handling area has decayed at least 55 days. This is consistent with Technical Specification 3.8.13.a.
2. All fuel assemblies assumed damaged in excess of two full cores (354 assemblies) in the Unit 1 and 2 spent fuel pool are assumed to have decayed at least one year.
3. Spent fuel stored in the first 33 rows of the Unit 3 spent fuel pool closest to the spent fuel cask handling area has decayed at least 70 days. This is consistent with Technical Specification 3.8.13.b.
4. All fuel assemblies assumed damaged in excess of one full core (177 assemblies) in the Unit 3 spent fuel pool are assumed to have decayed at least one year.
5. The affected assemblies have the maximum core activity corresponding to a radial peaking factor of 1.2.
6. All rods of the affected assemblies are ruptured.
7. The iodine decontamination factor in pool water is 89.
8. There is no removal of activity by the spent fuel pool ventilation system filters prior to release to the environs.
9. Activity is released at ground level with an assumed $\chi/Q$ factor of $2.2 \times 10^{-4}$ sec/m$^3$.
10. The fractions of noble gases and iodine in the gaps are

| | |
|---|---|
| Kr-85, I-129 | 30% |
| All other noble gases | 10% |
| All other iodines | 10% |

The results of the cask drop analyses are as follows:

| | Dose at Exclusion Area Boundary (Rem) | |
|---|---|---|
| | Whole Body | Thyroid |
| Unit 1 and 2 Cask Drop | 0.13 | 142 |
| Unit 3 Cask Drop | 0.07 | 20 |

The offsite radiological consequences of the postulated cask drop accident in the Unit 1 and 2 spent fuel pool is within the 10 CFR100 limits. The offsite radiological consequences of the postulated cask drop accident in the Unit 3 spent fuel pool is well within 10CFR100 limits.

5 If it is assumed that the spent fuel pool water only retains 90 of the iodine released from the damaged fuel
5 assembly (DF = 10), and that the Spent Fuel Pool Ventilation System captures the radioisotope release
5 from the water surface (elevated release: $X/Q = 3.35 \times 10^{-5}$ sec/m$^3$; but assuming no iodine filtration), the
5 calculated two-hour EAB doses are 71.1 Rem thyroid and 0.1 Rem whole body.

5 CASE C:

5 If it assumed that a fuel assembly is damaged by the Cask Platform in the Spent Fuel Pool (el. 825 ft.)
5 such that the iodine decontamination factor is only 46.2, the two-hour EAB doses for a ground release
5 ($X/Q = 2.2 \times 10^{-4}$ sec/m$^3$), are calculated to be 101.1 Rem thyroid and 0.24 Rem whole body.

5 **15.11.2.4 Fuel Shipping Cask Drop Accidents**

Fuel shipping casks are used to transport irradiated fuel assemblies from the site and also between the Oconee 1 and 2 spent fuel pool and the Oconee 3 spent fuel pool. Two hypothetical accident scenarios, the maximum free fall drop of a cask containing an irradiated assembly, and a cask drop onto the irradiated assemblies in the storage racks in the spent fuel pools are considered (References 5 on page 15-39, 6 on page 15-39, 7 on page 15-39).

The analysis of the maximum free fall drop of a cask containing an irradiated fuel assembly is made based on the worst position of the cask suspended by the crane above the cask handling area. A drop of 46 ft. is assumed to rupture the cask and release the activity from the contained fuel assembly. The assembly is assumed to have the maximum inventory from the core corresponding to a radial peaking factor of 1.65 and has undergone a 90 day decay period. The dose consequences were calculated in accordance with
5 Regulatory Guide 1.25. The results are a whole body dose of $5.2 \times 10^{-4}$ rem and 2.5 rem thyroid at the
5 site boundary, within the 10CFR100 limit.

The worst case fuel handling accident sequence in which the fuel shipping cask impacts on the irradiated fuel assemblies in a spent fuel pool is evaluated. At no time is the cask suspended above the spent fuel pool; however, it is credible that with failure of the cask hoist cable that the cask, yoke, hook, and load block could, as a result of an eccentric drop, deflect and fall into the spent fuel pool and impact on top of the assemblies in the pool. The analysis is performed separately for the shared Unit 1 and 2 spent fuel pool and the Unit 3 spent fuel pool. In the first part of the analysis, the number of fuel assemblies damaged as a result of the cask drop is found. Subsequently the radiological consequences of the damaged assemblies are determined.

The following conservative assumptions are employed for determining the number of fuel assemblies damaged.

1. The cask, lifting yoke and load block are free to fall from elevation 844 ft., the top of the spent fuel pool, to elevation 816 ft. 5 in., the top of the fuel storage racks.
2. The drag on the cask, lifting yoke and load block from falling through 25.5 ft. of water is neglected.
3. The ability of the fuel storage cells to absorb energy beyond the point of elastic buckling is neglected.
4. The energy which is expended in deformation of the rack interconnecting members is neglected.
5. A deformed fuel storage cell results in the total loss of integrity of one fuel assembly.
6. The projected areas of the cask, lifting yoke and load block are oriented to contact the maximum number of fuel assemblies.

Using the above assumptions, the falling cask, lifting yoke, and load block will have $2.093 \times 10^6$ ft-lbf of kinetic energy at the instant of impact with the storage racks. This energy must be absorbed by the strain

5 d = effective bubble diameter, cm

5 Since the minimum water depth over a dropped fuel assembly is less than 23 feet (21.34 feet), the assumed
5 iodine DF must be less than 100, according to Reg. Guide 1.25, and calculated with comparable
5 conservatism as done in Reg. Guide 1.25. Using the above relationship, with a water depth of 21.34 feet,
5 a comparable DF is equal to 89 (Reference OSC-6070).

5 The fuel assembly gap gas pressure, at a Spent Fuel Pool bulk temperature of 150 F, is calculated to be
5 less than 1200 psig, based on the present TACO2 computer code licensing limit of 2200 psia at operating
5 system conditions (Reference FSAR Section 4.2.3.1.3, "Fuel Thermal Analysis" on page 4-13).

5 The activity released from the water's surface is released within a two-hour period as a ground release.
5 The atmospheric dilution is calculated using the two-hour ground release dispersion factor of $2.2 \times 10^{-4}$
5 $sec/m^3$.

5 The total integrated dose (2-hr EAB) to the whole body at the 1-mile exclusion distance is 0.185 Rem and
5 the thyroid dose at the same distance is 52.45 Rem. These values are far below the limits given in
5 10CFR100 of 25 Rem whole body and 300 Rem thyroid.

### 5 15.11.2.2 Base Case Fuel Handling Accident Inside Containment

5 In 1977, the NRC asked Oconee to evaluate the offsite dose consequences for a fuel handling accident
5 inside containment, per the guidance given in Reg. Guide 1.25. Since the shallow end of the fuel transfer
5 canal is at an elevation of 816.5 feet, the same iodine decontamination factor used for the Fuel Handling
5 Accident in the Spent Fuel Pool is used for the Fuel Handling Accident inside Containment. The activity
5 released from the refueling water is released as a ground release, which has an atmospheric dispersion
5 factor of $2.2 \times 10^{-4}$ $sec/m^3$. There is no credit taken for any containment closure/integrity resulting in the
5 released activity from the refueling water going straight outside.

5 Using the fuel assembly gap inventory in Table 15-1, and assuming all 208 fuel pins are damaged, the
5 two-hour EAB dose is 0.185 Rem to the whole body and 52.45 Rem to the thyroid. These values are
5 appropriately within the guidelines given in 10CFR100 (appropriately within means 100 Rem to the
5 thyroid), and are identical to the base case Spent Fuel Pool Fuel Handling Accident described in Section
5 15.11.2.1, "Base Case Fuel Handling Accident in Spent Fuel Pool" on page 15-33.

### 5 15.11.2.3 Supplemental Cases of Fuel Handling Accidents

5 To provide additional information as to the sensitivity of various input assumptions into the offsite dose
5 consequences of the fuel handling accident, additional supplemental cases are described here.

5 <u>CASE A:</u>

5 If the radioisotope release from the spent fuel pool water's surface is assumed to be captured by the Spent
5 Fuel Pool Ventilation System, resulting in an elevated release, (atmospheric dispersion factor is equal to
5 $3.35 \times 10^{-5}$ $sec/m^3$) and assuming that the Spent Fuel Pool Filters are 90% efficient for the removal of
5 elemental and particulate iodine, and 70% efficient in the removal of organic iodine, the resultant
5 two-hour offsite dose is calculated to be 1.2 Rem thyroid and 0.021 Rem whole body at the exclusion
5 area boundary (EAB).

5 <u>CASE B:</u>

# 15.11 FUEL HANDLING ACCIDENTS

## 15.11.1 IDENTIFICATION OF ACCIDENT

Spent fuel assemblies are handled entirely under water. Before refueling, the reactor coolant and the fuel transfer canal water above the reactor are increased in boron concentration so that, with all control rods removed, the $k_{eff}$ of a core is no greater than 0.99. In the spent fuel storage pool, the fuel assemblies are
5 stored under water in storage racks with a minimum boron concentration of 2210 ppm in the pool water. Under these conditions, a criticality accident during refueling is not considered credible. Fuel handling
3 consists of all fuel assembly shuffling and transfer operations between the reactor, the spent fuel pool, the
3 fuel shipping casks, and dry storage transfer cask. Mechanical damage to the fuel assemblies during transfer operations is possible but improbable. The mechanical damage type of accident is considered the maximum potential source of activity release during refueling operations.

## 15.11.2 ANALYSIS AND RESULTS

5 **15.11.2.1 Base Case Fuel Handling Accident in Spent Fuel Pool**

5 During fuel handling operations, it is possible that a fuel assembly can be dropped, causing mechanical
5 damage with a subsequent release of fission products. To conservatively evaluate the offsite dose
5 consequences of such an accident, conservative assumptions are made. The following analysis assumes
5 the accident occurs within the spent fuel pool building.

5 The fuel assembly gap inventory is assumed to contain a fission product inventory from a maximum
5 burned fuel assembly at a radial peaking factor of 1.65. The gap fractions used are from Reg. Guide 1.25
5 and the reactor has been shutdown for 72 hours, which is the minimum time for RCS cooldown, reactor
5 closure head removal, and removal of the first fuel assembly. The actual isotopic curie contents are listed
5 in Table 15-1. It is also assumed that all 208 fuel pins are mechanically damaged such that the entire gap
5 inventory is released to the surrounding water. Since the fuel pellets are cold, only the gap inventory is
5 released.

5 The gases released from the damaged fuel assembly pass upward through the spent fuel pool water prior
5 to reaching the Auxiliary Building atmosphere. Noble gases are assumed to not be retained in the pool
5 water. According to Reg Guide 1.25, an iodine decontamination factor of 100 can be used for pin
5 pressures less than, or equal to, 1200 psig and water depths of 23 feet or greater. Since the spent fuel pool
5 racks are at an elevation of 816.5 feet and the minimum water level in the Spent Fuel Pool is equal to or
5 greater than 837.84 feet (-2.0 ft indicated level), there is a minimum of 21.34 feet of water over the fuel
5 storage racks, including instrument error. An experimental test program (Reference WCAP-7828)
5 evaluated the extent of removal of iodine released from a damaged irradiated fuel assembly. Iodine
5 removal from the released gas takes place as the gas rises through the water. The extent of iodine removal
5 is determined by mass transfer from the gas phase to the surrounding liquid and is controlled by the
5 bubble diameter and contact time of the bubble with the water. The following analytical expression is
5 given as a result of this experimental test program:

5 $$\text{Iodine Decontamination Factor (DF)} = 73\ e^{0.313\,(t/d)}$$

5

5 Where: t = bubble rise time, seconds

# 15.10 WASTE GAS TANK RUPTURE ACCIDENT

## 15.10.1 IDENTIFICATION OF ACCIDENT

Rupture of a waste gas tank would result in the release of the radioactive contents to the plant ventilation system and to the atmosphere through the unit vent. This accident is analyzed in order to evaluate the resultant dose at the site boundary.

## 15.10.2 ANALYSIS AND RESULTS

A tank is assumed to contain the maximum possible inventory of gas activity. The gaseous activity in the tank is:

| Isotope | Waste Gas Tank Inventory Activity (Ci) |
|---|---|
| Kr-85m | 501 |
| Kr-85 | 3,050 |
| Kr-87 | 284 |
| Kr-88 | 902 |
| Xe-131m | 735 |
| Xe-133m | 936 |
| Xe-133 | 80,900 |
| Xe-135m | 314 |
| Xe-135 | 2,210 |
| Xe-138 | 170 |
| I-131 | 10.7 |
| I-132 | 15.7 |
| I-133 | 12.7 |
| I-134 | 1.67 |
| I-135 | 6.35 |

The Auxiliary Building is ventilated and discharges to the unit vent. The activity is assumed to be released as a puff. Atmospheric dilution is calculated using the 2-hr elevated dispersion factor of 3.35 x $10^{-5}$ sec/$m^3$. The total integrated dose at the 1-mi exclusion distance is 0.17 rem whole body and 0.27 rem thyroid. These doses are well below the limits of the 10CFR100 guidelines.

## 15.9.3 REFERENCES

1. Parker, W. O. Jr., Letter to O'Reilly, J. P. (NRC), June 12, 1980.
2. Watson, L. C., Bancroft, A. R., and Hoelke, C. W., Iodine Containment by Dousing in NPD-11, *AECL-1130*.
3. Styrikovich, M. A., et al., Transfer of Iodine From Aqueous Solutions to Saturated Vapor, *Soviet Journal of Atomic Energy 17*, July 1964.
4. Duke Power Calculation OSC-6036, "Radiological Dose Evaluation for NSM ON-1, 2, 32903", dated April 3, 1996.

# 15.9 STEAM GENERATOR TUBE RUPTURE ACCIDENT

## 15.9.1 IDENTIFICATION OF ACCIDENT

The steam generator tube rupture accident assumes the severance of a steam generator tube which initiates a blowdown of primary coolant through the break and into the steam generator. The primary system response is similar to a small LOCA. The main safety concern is the release of fission products from the secondary system which are in high concentration due to the primary to secondary coolant leakage.

The criteria for the steam generator tube rupture accident are the dose limits specified in 10CFR100.

## 15.9.2 ANALYSIS AND RESULTS

The accident is initiated by a double ended rupture of one steam generator tube from full power conditions. The initial leak rate, approximately 435 gal/min, exceeds the normal makeup of 45 gal/min to the RCS, and the system pressure decreases. No operator action is assumed, and a low RCS pressure trip will occur in about 8 min.

Following reactor trip, system pressure continues to decrease until the HPIS is actuated at a pressure of 1,500 psig. The capacity of the HPIS is sufficient to compensate for the leakage and maintains both pressure and volume control of the RCS. Thereafter, the reactor is assumed to be cooled down and depressurized at the normal rate of 100°F per hour. The RCS is cooled and depressurized until the affected steam generator can be isolated by closing the steam bypass isolation valves. Cooldown continues with the unaffected steam generator until the temperature is reduced to 250°F. Thereafter, cooldown to ambient conditions is continued using the Low Pressure Injection System (LPIS).

Following reactor trip, the turbine stop valves will close causing secondary pressure to increase and lifting the main steam safety valves and opening the turbine bypasss to condenser dump (Reference 1 on page 15-30). Note: The three units currently have different turbine bypass system characteristics. This analysis and results section is based on the most conservative case. Approximately 62,200 lb of steam is vented through the main steam safety valves, of which 15,260 lb was primary coolant.

At the design cooling rate of 100°F per hour, depressurization of the RCS to the steam line safety valve setpoint requires approximately 34 min. During this time period, 1,980 $ft^3$ of reactor coolant leaks to the secondary system. This leakage corresponds to approximately 29,200 Ci (Dose equivalent Xe-133) of noble gases if the reactor has been operating for 400 days with 1 percent of the fuel pins in the core defective.

The radioactivity is released during this accident via three flowpaths. Noble gases are assumed to be released directly to the unit vent. The iodine content in the 15,260 lb of steam released directly to the atmosphere does not take credit for iodine separation between steam and liquid. A partition factor of $10^4$ is assumed for iodine bypassed to the condenser and then exiting the unit vent (References 2 on page 15-30, 3 on page 15-30).

Atmospheric dilution is calculated using the 2-hr. elevated release dispersion factor of $3.35 \times 10^{-5}$ sec/$m^3$. The total dose to the body from all the xenon and krypton released is only 0.007 rem at the 1-mi. exclusion distance. The corresponding dose to the thyroid at the same distance is 1.11 rem.

Unit operation with 1 percent defective fuel and 1 gal/min steam generator tube leakage is shown to be safe by the analysis presented. For the same conditions, the steam relief accompanying a loss of load transient would not change the whole body dose. The whole body dose is primarily due to the release of Xe and Kr. Release of these gases is not increased by the steam relief because, even without relief, all of these gases are released to the atmosphere through the condenser air ejector. The rate of release of iodine during the approximately two min. of relief would be increased by almost a factor of $10^4$ because the iodine is released directly to the atmosphere rather than through the condenser and station vent. However, the quantity released during this short time is small, 0.04 dose equivalent curies of I-131. Atmospheric dilution is calculated using the 2-hr elevated release dispersion factor of $3.35 \times 10^{-5}$ sec/m$^3$. The total integrated thyroid dose from this release is 0.0008 rem at the 1-mi exclusion distance.

## 3 15.8.3 RESULTS OF LOSS OF ALL STATION POWER ANALYSIS

3 The loss of all station power condition is not a design basis accident. However, the analysis of this
3 postulated event is bounded by the analysis performed for the LOCA accident analyzed in Oconee Fsar
3 Section 15.14.3.3.6, "ECCS Performance and Single Failure Assumption" on page 15-58 and the station
3 blackout event (10 CFR 50.63) described in Section 8.3.2.2.4, "Station Blackout Analysis" on page 8-25.

# 3 15.8 LOSS OF ELECTRIC LOAD ACCIDENTS

## 15.8.1 IDENTIFICATION OF CAUSE

3 Each unit is designed to withstand the effects of a loss of electric load caused by separation of the unit
3 from the transmission system.

3

3 The criteria for plant protection during a loss of electric load transient are that the minimum DNBR will not be less than 1.3 and the system pressure will not exceed code limits.

## 15.8.2 RESULTS OF LOSS OF LOAD CONDITION ANALYSIS

The effect of a loss of load condition on a unit would be that the unit generator breakers would open and thus disconnect the unit from the transmission system. When this occurs, a runback signal causes an automatic power reduction to 15 percent power. Depending on the initial power level at the time of the loss of load, the RPS may initiate a reactor trip on high reactor coolant temperature or pressure. Electrical power would then be supplied from any of the available sources as described in Section 8.3.1, "AC Power Systems" on page 8-9. Assuming that no RPS actuation is required, the following actions occur:

1. All vital electrical loads, including power to the reactor coolant pumps, condenser circulating water pumps, hot well and condensate booster pumps, and other auxiliary equipment, will continue to obtain power from the unit generator. Feedwater is supplied to the steam generators by the main feedwater pumps.
2. As the electrical load is dropped, the turbine generator accelerates and closes the governor valves and intercept valves. The unit frequency will peak at less than the overspeed trip point and decay back to set frequency in 40-50 sec.
3. Following closure of turbine governor valves and intercept valves, steam pressure increases to the turbine bypass valve setpoint, and may increase to steam system safety valve setpoint. Steam is relieved to the condenser and to the atmosphere. Steam venting to the atmosphere occurs for about two min. following loss of load from 100 percent initial power until the turbine bypass can handle all excess steam generated. About 148,000 pounds of steam will be relieved to the atmosphere. Steam relief permits energy removal from the RCS to prevent a high pressure reactor trip. The initial power runback is to 15 percent power, which is a higher power level than needed for the unit auxiliary load. This allows sufficient steam flow for regulating turbine speed control. Excess steam above unit auxiliary load requirements is rejected by the turbine bypass valve to the condenser.
4. During the short interval while the turbine speed is high, the vital electrical loads connected to the unit generator will undergo speed increase in proportion to the generator frequency increase. All motors and electrical gear so connected will withstand the increased frequency.
5. After the turbine generator has been stabilized at auxiliary load and set frequency, the station operator may reduce reactor power to the auxiliary load as desired.

The loss of load transient does not result in any fuel damage or excessive pressures on the RCS. There is no resultant radiological hazard to station operating personnel or to the public from this accident as only secondary system steam is discharged to the atmosphere.

## 15.7.5 REFERENCES

1. EPRI NP-1850-CCMA, RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, Volume 1, Revision 2, November, 1984.
2. EPRI-NODE-P, Advanced Recycle Methodology Program System Documentation, Electric Power Research Institute.
3. EPRI NP-2511-CCM, VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, Volume 1, Revision 2, July 1985.
4. Stolz, J. F. (NRC), letter to Tucker, H. B. (Duke), transmitting the Safety Evaluation by the Office of Nuclear Reactor Regulation on Oconee LER 269/86-006 (More Negative Moderator Temperature Coefficient Than Assumed in Safety Analysis), November 26, 1986.

Control rod malfunctions are accommodated by the core design without ICS action. Since the most severe case analyzed for the dropped rod does not result in reactor trip nor does the thermal power exceed its initial value, core and RCS boundary protection is assured. Additional protection for the dropped rod accident is provided through the ICS which detects a dropped rod and inhibits withdrawal of the control rods. The ICS is designed to run back the steam generator load demand upon receiving the dropped rod signal from the rod drop detection circuitry. The reactor thermal power will assume a lower value that matches the load demand and will provide additional margin toward not exceeding any design limit.

The reactor is assumed to be operating at 100 percent of rated power when the control rod is dropped. In order to achieve the most adverse response the most negative values of moderator coefficient [$-3.0 \times 10^{-4}$ ($\Delta k/k$)/°F] and Doppler coefficient [$-1.3 \times 10^{-5}$ ($\Delta k/k$)/°F] occurring at end of cycle are used. The maximum rod worths expected to occur during full operation are used to examine the effects of ICS protective action. These rod worths correspond to operation at full power without xenon (0.46 percent $\Delta k/k$) and with xenon (0.36 percent $\Delta k/k$). It is assumed that the steam generator load demand is reduced linearly to 60 percent at 30%/min. The effects of a dropped rod without ICS action were also examined for a very conservative rod worth of 0.65 percent $\Delta k/k$. The rod was assumed to drop to 2/3 insertion in 1.4 sec.

An analysis of the sensitivity to the assumed end-of-cycle moderator coefficient was also performed. The plant response to a dropped rod without ICS action is simulated using the RETRAN02-MOD003 computer code (Reference 1 on page 15-26). A conservatively bounding dropped rod worth of 0.4 percent $\Delta k/k$ is assumed, and moderator coefficients of $-3.0 \times 10^{-4}$ ($\Delta k/k$)/°F and $-3.5 \times 10^{-4}$ ($\Delta k/k$)/°F are analyzed. The power peaking resulting from the dropped rod is calculated using the EPRI NODE-P core simulation code (Reference 2 on page 15-26), and the minimum DNBR is analyzed using the VIPRE-01 computer code (Reference 3 on page 15-26).

## 15.7.4 RESULTS OF ANALYSIS

The results of the analysis with ICS action are presented in Figure 15-26 and Figure 15-27. Figure 15-26 shows the response to a 0.46 percent $\Delta k/k$ dropped rod. The neutron power decreases rapidly to about 55 percent of rated power. This causes rapid decreases in the core moderator temperature and fuel temperature. These temperature decreases overcompensate for the worth of the control rod, and the power rises until the reduced steam generator demand begins to increase the inlet temperature and decreases the power. The thermal power levels out briefly at about 78 percent of its initial value but soon begins to decrease in response to the decreased steam generator demand. The pressurizer pressure swing is about $\pm$ 60 psi before returning to equilibrium.

Figure 15-27 shows the results of the 0.36 percent $\Delta k/k$ rod drop. The initial neutron power decrease is slightly less in this case, resulting in the thermal power leveling off at 83 percent, a slightly higher value than in the 0.46 percent $\Delta k/k$ case. The pressurizer pressure peaks at a higher value due to this higher thermal power.

Figure 15-28 shows the results of a 0.65 percent $\Delta k/k$ dropped rod analysis conservatively based on no ICS action and operation at a power level of 2,772 MWt. The neutron power decreases causing a rapid decrease in both the core moderator temperature and the fuel temperature. These temperature decreases overcompensate for the worth of the control rod, and the neutron power rises slightly above the initial neutron power level. The neutron power then decreases to below the initial power level and eventually levels out at the initial power level. The thermal power response is similar to the neutron power, however, the thermal power level never exceeds the initial power level. Both the core moderator temperature and pressurizer pressure decrease during the transient and level out at a value lower than the initial values.

Several cases have been run for rod drops at beginning of cycle (BOC) conditions. These transients yield power levels that are lower than the end of cycle (EOC) conditions and may result in reactor trip. These are therefore not included in this discussion because they represent less severe conditions.

The results of the RETRAN sensitivity study on the end-of-cycle moderator coefficient show that the consequences of the dropped rod transient are insensitive to this parameter.

# 15.7 CONTROL ROD MISALIGNMENT ACCIDENTS

## 15.7.1 IDENTIFICATION OF CAUSE

Control rods are normally grouped into patterns which maintain a symmetric core power distribution. A mechanical or electrical failure can cause a control rod to become misaligned from its group, causing an asymmetric reactivity distribution and a reduction in the total available control rod worth for shutdown of the reactor.

Three modes of misalignment can occur. During withdrawal or insertion of a control rod group, one rod can become stuck at some position as the rod group continues in motion. This condition will affect the power distribution in the core and could lead to excessive power peaking. The second mode of misalignment can occur on reactor trip if one rod fails to insert and remains stuck in the fully withdrawn position. This condition requires an evaluation to determine that sufficient negative reactivity is available for tripping the reactor and maintaining a hot shutdown condition when considering the maximum worth stuck rod. The third mode can occur when one rod drops partially or fully into the core. The resulting transient causes a rapid reduction in power and moderator temperature which is followed by an increase in power due to the negative moderator coefficient. The magnitude of the return to power, in consideration of the asymmetric power distribution, could lead to excessive power peaking.

The criteria for plant protection during these transients are that the minimum DNBR will not be less than the acceptance criterion for the correlation used and the system pressure will not exceed code limits.

## 15.7.2 PROTECTIVE BASIS

The concern that a stuck rod could result in insufficient negative reactivity on reactor trip to maintain the reactor shutdown is prevented by the core design criteria. All cores are required to be capable of maintaining a 1 percent $\Delta k/k$ shutdown margin at hot shutdown conditions with the assumption of the maximum worth rod stuck in the fully withdrawn position.

For protective purposes a misaligned control rod is defined as the deviation of a control rod from its group reference position by more than a maximum of 9 inches. This definition then covers both the action of dropping a rod and sticking a rod while moving a group. The action taken by the ICS is:

1. Rod withdrawal is inhibited.
2. The steam generator load demand is run back to 60 percent of rated load at 30%/min.

Although these ICS actions are available to mitigate the consequences of the accident, they are not required functions for safe plant operation as the results of the accident analysis demonstrate.

## 15.7.3 METHODS OF ANALYSIS

For the three modes of control rod misalignment, the transient which results from the dropped rod is analyzed since it presents the greatest potential for exceeding the criteria.

The transient response to a dropped control rod is analyzed using a detailed B&W digital model. This program includes fuel pin, point kinetics, pressurizer, and loop models, including the steam generators.

## 15.6.4 REFERENCES

1. Oconee 1 Fuel Densification Report, Babcock & Wilcox, *BAW-1388*, February 1973.
2. Oconee 2 Fuel Densification Report, Babcock & Wilcox, *BAW-1398*, September 1973.
3. Oconee 3 Fuel Densification Report, Babcock & Wilcox, *BAW-1399*, November 1973.

The results of the sheared shaft accident showed that the flow response was within ±2 percent of the locked rotor analysis results. Therefore, the calculated power, temperatures, and minimum DNBR are essentially the same.

For both types of mechanical failures of the reactor coolant pumps which lead to a rapid partial loss of coolant flow, the minimum DNBR exceeds the 1.3 limit and a moderate increase in the peak cladding temperature occurs for a short period of time. No fuel melting is predicted to occur.

The RCS is capable of providing natural circulation flow after the pumps have stopped. The natural circulation characteristics of the RCS have been calculated using conservative values for all resistance and form loss factors. No voids are assumed to exist in the core or reactor outlet piping. The following tabulation shows the natural circulation flow capability as a function of the decay heat generation.

| Time After Loss of Power (sec) | Decay Heat Core Power (%) | Natural Circulation Core Flow Available (% Full Flow) | Flow Required For Heat Removal (% Full Flow) |
|---|---|---|---|
| $3.6 \times 10^1$ | 5.0 | 4.1 | 2.3 |
| $2.2 \times 10^2$ | 3.0 | 3.3 | 1.2 |
| $1.2 \times 10^4$ | 1.0 | 1.8 | 0.36 |
| $1.3 \times 10^5$ | 0.5 | 1.2 | 0.20 |

The flows above provide adequate heat transfer for core cooling and decay heat removal by natural circulation.

The reactor is protected against reactor coolant pump failure(s) by the RPS and the ICS. The ICS initiates a power reduction on pump failure to prevent reactor power from exceeding that permissible for the available flow. The reactor is tripped if insufficient reactor coolant flow exists for the power level.

An additional loss of coolant flow mechanism has been analyzed in the reactor design evaluation. This involves possible flow or leakage past the seat of a reactor internals vent valve. These valves are designed to be closed during all normal operations and all normal and accident transients except for those accident transients where reverse flow through the core would occur. The design provides for positive closure even with no flow, and several rotational clearances are provided to assure free motion and to prevent any tendency to stick. The valves also have a self alignment feature to prevent reactor coolant leakage. Hydrostatic and vibrational tests have been made to demonstrate that the valves will operate as designed. However, an analysis has been made for the case where a reactor internals vent valve remains open. This malfunction reduces the effective core flow by about 5 percent which results in a lower initial DNBR. The effect of a stuck open vent valve on the loss of coolant flow transient is also shown in Figure 15-23.

The impact of reduced grid frequency operation of the reactor coolant pumps has been evaluated assuming a minimum frequency of 57Hz. Below this frequency, automatic switchgear will remove the unit from the grid. At 57Hz the pumps will deliver approximately 95 percent of normal rated flow. The results of a loss of coolant flow transient initiated from this condition will be the same as the case for a stuck open vent valve.

The locked rotor accident results in a rapid decrease to the three pump flow condition within 3 sec. The reactor trips on the flux/flow ratio at 0.9 sec. The results of the analysis are shown in Figure 15-24 and Figure 15-25.

Figure 15-24 shows the flow transient and the neutron and thermal power response. Figure 15-25 shows the fuel and cladding temperature response and the minimum DNBR. Since the neutron power only increases slightly prior to the trip due to the positive moderator coefficient, the fuel temperature remains relatively constant until decreasing with reactor trip. The minimum DNBR reaches the 1.3 limit at 0.9 sec, at which time the cladding temperature increases in response to the assumed switch from nucleate to film boiling at the limit. The peak cladding temperature of 1390 °F occurs at 4.0 sec after the initiation of the transient.

The original loss of coolant flow accidents were reanalyzed to include the effects of fuel densification (References 1 on page 15-21, 2 on page 15-21, 3 on page 15-21). The results presented include these effects unless specified otherwise.

The loss of coolant flow analysis was performed from 102 percent rated power with the following assumptions:

1. Initial core inlet temperature is +2°F in error.
2. Initial system pressure is -65 psi in error.
3. Trip delay time, i.e., time from sensor detection of loss of power to the pumps until initial downward movement of contol rod, is 0.5 sec.
4. The percentage of neutron power at beginning of life as a function of time after initiation of control rod insertion is as shown in Figure 4-11. This figure also contains the shutdown characteristics for a minimum of 1.0 percent shutdown margin at the hot shutdown condition.
5. The pump inertia is 70,000 lb-ft$^2$
6. The fuel rod gap conductance is 669 Btu/hr-ft$^2$°F.
7. Axial flux shape in Figure 15-19.

The locked rotor analysis was performed from 102 percent rated power with the following assumptions:

1. Initial core inlet temperature is +2°F in error.
2. Constant system pressure of 2135 psig, -65 psi in error.
3. Trip delay time on flux/flow is 0.65 sec.
4. Film boiling occurs at DNBR = 1.3.
5. Pump inertia stepped to infinity.
6. Axial flux shape in Figure 15-19.
7. Moderator coefficient $+0.525 \times 10^{-4}$ ($\Delta k/k$)/°F.

The sheared shaft analysis was analyzed with the same assumptions as the locked rotor analysis with the exception that the pump inertia was stepped from 70,000 to 2,400 lb-ft$^2$.

## 15.6.3 RESULTS OF ANALYSIS

The loss of coolant flow accident is initiated by a loss of power to all four reactor coolant pumps which causes a flow coastdown as shown in Figure 15-20. The core thermal response to the transient is shown in Figure 15-21 and Figure 15-22. Figure 15-21 shows the neutron power and thermal power in relation to core flow. Figure 15-22 shows the film coefficient of heat transfer from the cladding to the coolant and the hot channel minimum DNBR. The minimum DNBR decreases from approximately 1.74 to 1.53 at 1.65 sec after loss of power to the pumps. Since the DNBR limit of 1.3 was not exceeded, no fuel failures are predicted for this transient.

Under normal conditions, the maximum indicated reactor power level from which a loss of coolant flow accident could occur is 102 percent rated power. This power level provides an allowance of +2 percent rated power for transient overshoot. The 102 percent rated power is an instrument indicated value and is subject to the following maximum errors: (a) ±2 percent heat balance and (b) ±4 percent nuclear instrumentation. The true power level could theoretically be as high as 108 percent at 102 percent indicated power. A sensitivity study on initial power level (undensified fuel) is shown in Figure 15-23.

# 15.6 LOSS OF COOLANT FLOW ACCIDENT

## 15.6.1 IDENTIFICATION OF CAUSE

A reduction in the reactor coolant flow rate occurs if one or more of the reactor coolant pumps should fail. A pumping failure can occur from mechanical failures or from a loss of electrical power. With four independent pumps available, a mechanical failure in one pump will not affect operation of the others. A mechanical failure results in the locked rotor accident or the sheared shaft accident.

Each reactor coolant pump receives electrical power from one of the two electrically separate buses of the 6,900 volt system. Loss of the unit auxiliary transformer to which the 6,900 volt buses are normally connected will initiate a rapid transfer to the start up transformer source without loss of coolant flow. Faults in an individual pump motor or its power supply could cause a reduction in flow, but a complete loss of forced flow is extremely unlikely. In spite of the low probability of this event, the nuclear unit has been designed so that such a failure would not lead to core damage.

The reactor protection criterion for loss of coolant flow conditions resulting from electrical malfunction of the reactor coolant pumps or their power supply is that the minimum departure from nucleate boiling ratio (DNBR) experienced by the core shall not be less than 1.3.

## 15.6.2 METHODS OF ANALYSIS

The loss of coolant flow accident is analyzed with a combination of B&W analog and digital computer programs. The analog simulation is used to determine the coolant flow coastdown at the core inlet. The analog model includes a description of the loop pressure drops and predicts the dynamic pump behavior based on the pump homologous curves. Flow-speed, flow-torque, and flow-head relationships are solved by affinity laws. The transient flow response is then input to a digital simulation model to calculate the core power, inlet temperature, and system pressure. The digital code includes a point kinetics model with six delayed neutron groups, a control rod scram reactivity insertion model, and a loop simulation with a pressurizer and steam generator models. The results of this code are then input to a digital core thermal analysis code which calculates fuel and cladding temperatures, and DNBR.

The core thermal analysis code simulates the reactor core through the use of a two channel model. Each channel consists of one fuel rod with its associated flow area and spacer grid geometry. Given the necessary input, as stated above, the code will calculate a pressure drop across a typical reactor channel (average channel) as a function of time. This pressure drop is then imposed on the second channel (usually a hot channel) to determine hot channel flow and DNBR, in addition to fuel and clad temperatures. Compared to the average channel, the hot channel has greater heat generation and reduced flow area, as well as statistical hot channel factors. The analytical fuel pin model contains a transient response calculation while the hydraulic model considers the steady state solutions of energy, mass, and momentum balances at each time step.

The transient response is obtained by applying the changing flow, power, inlet temperature, and system pressure to the initial conditions of the average channel. This calculation yields the average channel pressure drop as a function of time. This pressure drop as a function of time applied to the hot channel yields clad temperature, fuel temperature, and DNBR as a function of time.

## 15.5.5 REFERENCES

1. Swindlehurst, G. B., *Oconee Nuclear Station Restarting RCP at Power*, Letter to Smith, J. E., Duke File OS-802.10; Modification ON-1466, September 13, 1979.
2. EPRI NP-1850-CCMA, RETRAN-02 - A Program for Transient Thermal-Hydraulic Analysis of Complex Fluid Flow Systems, Revision 2, November 1984.
3. Stolz, J. F. (NRC), letter to Tucker, H. B. (Duke), transmitting the Safety Evaluation by the Office of Nuclear Reactor Regulation on Oconee LER 269/86-006 (More Negative Moderator Temperature Coefficient Than Assumed in Safety Analysis), November 26, 1986.

to the increased thermal power. The swing from maximum to minimum pressure is about 100 psi, which represents a change of pressurizer level of about 1.8 ft.

Based on the analysis, which conservatively assumes that the interlock does not function, it is concluded that the core is protected in the event that an idle pump is started. Additionally, the automatic control system could serve to limit the imbalance between the reactor and steam generator powers and reduce the severity of the accident. Since the thermal power does not exceed 65 percent of rated power at full flow and the pressure does not exceed 2300 psia, it is concluded that the protection criteria are met.

The results of the sensitivity study using a moderator coefficient of $-3.5 \times 10^{-4}$ $\Delta K/K/°F$ and an instantaneous reactor coolant pump start time are similar to the original FSAR analysis. The thermal power increases to 75% as compared to 62% for the original analysis. The results of the sensitivity study using the more negative moderator coefficient and an increased reactor coolant pump start time show an increase in peak thermal power. For a 5 second reactor coolant pump start time, which was determined to be the most limiting case, the peak thermal power reaches 96.5%. For both of the sensitivity studies performed the thermal power remains below 100%. Therefore, the protection criteria are met with a more negative moderator coefficient and a worst case reactor coolant pump start time.

# 15.5 COLD WATER ACCIDENT

## 15.5.1 IDENTIFICATION OF CAUSE

An evaluation of possible accident modes has established the worst case for the cold water accident. This case occurs when one loop is in operation at 50 percent power with the down steam generator operating at minimum level. For this case back flow through the down steam generator will enter the steam generator at the reactor inlet temperature (557°F) and is assumed to leave the steam generator at the saturation temperature of the secondary side (532°F). Therefore, if the two idle pumps are started under these conditions the potential for a cold water accident exists. The increase in core flow as a result of starting the two pumps causes the core average temperature to decrease before the cold water in the idle loop piping reaches the core. Both effects cause a net reduction in core moderator temperature. If the temperature coefficient is negative, reactivity will be introduced into the core and a power rise will occur.

## 15.5.2 PROTECTIVE PROCEDURE AND INTERLOCK

For the case where one or more pumps are idle, an administrative procedure is initiated whereby the operator must reduce power to less than 50 percent before starting a pump. This procedure assures that a trip will not occur before reaching the new equilibrium for the new pump combination. In addition, the pump control circuitry has an interlock to prevent startup of an idle pump if the power is above 50 percent full power. (Reference 1 on page 15-15).

## 15.5.3 METHODS OF ANALYSIS

A detailed B&W proprietary digital simulation of the plant is used to evaluate the transient response to this accident. The model includes point kinetics and a multi region fuel pin model connected through time delays to a pressurizer model. A steam generator model is included. The reactor is assumed to be operating normally at 50 percent power with two pumps in one loop not operating. It is assumed that the two idle pumps are started and accelerated instantaneously to their normal flow value. It is assumed that no additional heat is removed from the steam generators. Conservative values of the moderator
3 coefficient ($-3.0 \times 10^{-4}$ ($\Delta k/k/°F$) and Doppler coefficient ($-1.3 \times 10^{-5}$ ($\Delta k/k$)/°F) are used. The protection criteria for this accident are that the minimum DNBR be greater than the acceptance criterion for the correlation used and that system pressure limits not be exceeded.

An analysis of the sensitivity of the system response to a more negative moderator coefficient ($-3.5 \times 10^{-4}$ $\Delta K/K/°F$) and to the reactor coolant pump start time was performed using the RETRAN02-MOD003 computer code (Reference 3 on page 15-15).

## 15.5.4 RESULTS OF ANALYSIS

The results are presented in Figure 15-18. The curve clearly shows that the cold water that will enter the core approximately 5 sec. after pump startup has an irrelevant effect when compared to the effect of starting two pumps with the assumption of instantaneous pump acceleration. As the results show, the power increases rapidly in response to the decreased temperature that results from the step change in flow and reaches the trip setpoint about 0.5 sec. after the initiation of the accident. The maximum neutron power reached was about 130 percent, however, the thermal power reached a maximum value of only 62 percent. The pressure decreases initially in response to the system temperature decrease but recovers due

concentration. Both the moderator coefficient and boron concentration values used are conservative. The effects of the three dilution rates discussed above on the reactor are as follows:

| Dilution Water Flow (gal/min) | Reactivity Rate ((Δk/k)/sec) | Average RCS Temperature Change (°F/sec) |
|---|---|---|
| 70 | $+2.2 \times 10^{-6}$ | 0.007 |
| 100 | $+3.2 \times 10^{-6}$ | 0.010 |
| 500 | $+1.6 \times 10^{-5}$ | 0.049 |

The highest rate of dilution can be handled by the ICS, which would insert rods to maintain the power level and thus limit the RCS temperature rise. If an interlock failure occurred while the reactor was under manual control, these reactivity additions would cause a high reactor coolant temperature trip or a high presssure trip. In any event, the thermal power will not exceed 112 percent rated power, and the system pressure will not exceed code allowable limits. Therefore, moderator dilution accidents will not cause any damage to the plant.

During refueling when the reactor closure head has been removed, dilution water additions to the Reactor Coolant System, Low Pressure Injection System, fuel transfer canal, and spent fuel storage pool are administratively controlled. Surveillance of the source range nuclear instrument(s), periodic boron analysis, and the unexpected water level increases alert the operator to a possible moderator dilution accident. Moderator dilution via the letdown storage tank is not plausible until after the Reactor Control System has been closed and filled, since the high pressure injection pumps are not operating during refueling operations. At the beginning of core life when the boron concentration is highest, the reactor is about 9.5 percent Δk/k subcritical with the maximum worth rod stuck out. To demonstrate the ability of the reactor to accept moderator dilution during refueling, the consequences of accidentally filling the letdown storage tank with dilution water and starting the high pressure injection pumps have been evaluated. The entire water volume from the letdown storage tank could be pumped into the RCS assuming only the coolant in the reactor vessel is diluted, and the reactor would still be 4.9 percent Δk/k subcritical.

# 15.4 MODERATOR DILUTION ACCIDENT

## 15.4.1 IDENTIFICATION OF CAUSE

The reactor utilizes boron in the form of boric acid in the reactor coolant to control excess reactivity. The boron content of the reactor coolant is periodically reduced to compensate for fuel burnup. The dilution water is supplied to the RCS by the High Pressure Injection System (HPIS). This system is designed with several interlocks and alarms to prevent improper operation. These are as follows:

1. Flow of dilution water to the letdown storage tank must be initiated by the operator. The dilution water addition valve can be opened only when the Group 6 control rods have been withdrawn to a preset (95 percent withdrawn) position and the timing device to limit the integrated flow has been set. Dilution water is added at flow rates up to 70 gal/min at 2200 psia.
2. Flow of dilution water is automatically stopped when either the flow has integrated to a preset value
3 or when the Group 6 rods have been inserted to a preset position (75 percent withdrawn).
3. A continuous dilute permit light and feed and bleed valve position lights on the console are on whenever continuous dilution is in progress.

The HPIS normally has one pump in operation which supplies makeup to the RCS and the required seal flow to the reactor coolant pumps. The total makeup flow available is normally limited to 70 gal/min. When the makeup rate is greater than the letdown rate, the net water increase will cause the pressurizer level control to close the makeup valves. The nominal moderator dilution event considered is the pumping of water with zero boron concertration into the RCS.

It is possible, however, to have a slightly higher flow rate during transients when the system pressure is lower than the nominal value and the pressurizer level is below normal. This flow might temporarily be as high as 100 gal/min.

Furthermore, with a combination of multiple valve failures or maloperations, plus more than one high pressure injection pump operating with reduced RCS pressure, the resulting inflow rate could be as high as 500 gal/min. This constitutes the maximum dilution accident. A reactor trip would terminate unborated water addition to the letdown storage tank, and total flow into the coolant system would be terminated by a high pressurizer level.

The criteria for reactor protection in this accident are:

1. The reactor thermal power will be limited to less than the design overpower of 112 percent rated power.
2. The RCS pressure will be limited to less than the code allowable limit.
3. The reactor minimum shutdown margin of 1 percent $\Delta k/k$ subcritical will be maintained.

## 15.4.2 ANALYSIS AND RESULTS

The reactor is assumed to be operating at rated power with an initial boron concentration of 1,400 ppm in the RCS. The dilution water is uniformly distributed throughout the reactor coolant volume. Uniform distribution results from a discharge rate of 70 to 500 gal/min into the reactor coolant flow. A change in concentration of 75 ppm produces a 1 percent $\Delta k/k$ reactivity change. The analysis is based on $+0.94 \times 10^{-4}$ $(\Delta k/k)/°F$ moderator coefficient, $-1.17 \times 10^{-5}$ $(\Delta k/k)/°F$ Doppler coefficient, and 1,400 ppm boron

## 15.3.4 REFERENCES

1. Oconee 1 Fuel Densification Report, Babcock & Wilcox, *BAW-1388*, February 1973.

## 15.3.3 RESULTS OF ANALYSIS

Figure 15-11 shows the results of the nominal rod withdrawal from rated power using the 1.5 percent $\Delta k/k$ rod group at $1.09 \times 10^{-4}$ $(\Delta k/k)$/sec. The transient is terminated by a high neutron flux trip, and the reactor thermal power is limited to 108 percent, well below the design overpower of 112 percent of rated power. The changes in the parameters are all quite small. For example, the average reactor coolant temperature rise is only about 3 °F and system pressure increase is only 120 psi.

A sensitivity analysis of important parameters was performed around this nominal case, and the resultant RCS pressure responses are shown in Figure 15-12 through Figure 15-15.

Figure 15-12 shows the pressure variation for a very wide range of rod withdrawal rates, more than an order of magnitude smaller and greater than the nominal case. For very rapid rates, the high neutron flux trip is the primary protective function for the reactor core. It also protects the system against high pressure during fast rod withdrawal accidents. The high pressure trip is relied upon for the slower transients. In no case does the thermal power exceed 112 percent rated power.

Figure 15-13 through Figure 15-15 show the pressure response to variations in the trip delay time, Doppler coefficient, and moderator coefficient. Oconee 1 and 2 have the Type B control rod drive mechanism, whereas Oconee 3 has the Type C with the additional 0.1 sec trip delay.

An analysis has been performed extending the evaluation of the rod withdrawal accident for various fractional initial power levels up to rated power. This evaluation is performed assuming simulated withdrawal of all 61 control rods giving a maximum reactivity addition rate of $7.25 \times 10^{-4}$ $(\Delta k/k)$/sec.

This rate is a factor of seven higher than that used in the cases evaluated for withdrawal of a single group. The results of this analysis are shown in Figure 15-16 and Figure 15-17.

As seen in Figure 15-16, the peak thermal power occurs for the rated power case and is well below the maximum design power of 112 percent. The peak neutron power for all cases is approximately 119 percent of rated power and represents a slight overshoot above the assumed trip level of 112 percent. Figure 15-17 shows that the maximum fuel temperature reached in the average rod and the hot spot are well below melting. Even in the most severe case at rated power, the average fuel temperature increases by only 28°F. It is, therefore, readily concluded that no fuel damage would result from simultaneous withdrawal of all rods from any initial power level.

This analysis demonstrates that the high pressure trip and the high flux level trip adequately protect the reactor against any rod withdrawal accident from power operation.

# 15.3 ROD WITHDRAWAL ACCIDENT AT RATED POWER

## 15.3.1 IDENTIFICATION OF CAUSE

A rod withdrawal accident assumes an operator error or an equipment failure which results in accidental withdrawal of a control rod group while the reactor is at rated power. As a result of this assumed accident, the power level increases, the reactor coolant and fuel rod temperatures increase, and, if the withdrawal is not terminated by the operator or protection system, core damage would eventually occur.

The following provisions are made in the design to indicate and terminate this accident:

1. High reactor coolant outlet temperature alarms.
2. High reactor coolant system pressure alarms.
3. High pressurizer level alarms.
4. High reactor coolant outlet temperature trip.
5. High reactor coolant system pressure trip.
6. High power level (i.e., neutron flux level) trip.

The rod withdrawal accident analysis is performed with the criterion that the RPS will limit (a) the reactor thermal power to 112 percent of rated power, and (b) the RCS pressure to code allowable limits.

## 15.3.2 METHODS OF ANALYSIS

A B&W digital computer code is used to determine the characteristics of this accident. A complete kinetics model, pressurizer model, average fuel rod model, steam demand model with secondary coastdown to decay heat level, coolant transport model, and a simulation of the instrumentation for pressure and flux trip are included. The initial conditions are normal rated power operation without automatic control. Only the moderator and Doppler coefficients of reactivity are used as feedback. The nominal values used for the main parameters are as follows:

| | | |
|---|---|---|
| A. | Trip delay time (high pressure trip), sec | 0.5 |
| B. | Trip delay time (high flux trip), sec | 0.3 |
| C. | CRA insertion time (2/3 insertion), sec | 1.4 |
| D. | Doppler coefficient, $(\Delta k/k)/°F$ | $-1.17 \times 10^{-5}$ |
| E. | Moderator coefficient, $(\Delta k/k)/°F$ | $+0.5 \times 10^{-4}$ |
| F. | Control rod speed, in./min | 30 |
| G. | Control rod group worth, % $(\Delta k/k)$ | 1.5 |

The criterion for minimum movable control rod worth is that a shutdown margin of 1 percent $\Delta k/k$ at the hot standby condition is required. The rod withdrawal accident has been analyzed using the minimum tripped rod worth as part of the analysis. The effects of fuel densification on the rod withdrawal accident have been determined to be insignificant (Reference 1 on page 15-9).

Figure 15-2 shows the results of withdrawing all 61 control rod assemblies at the maximum speed (with a total worth of 10 percent $\Delta k/k$) from 1 percent subcritical and $10^{-9}$ rated power. This results in a reactivity addition rate of $7.25 \times 10^{-4}$ (k/k)/sec. The neutron power peaks at 275 percent, where the power rise is stopped by the negative Doppler effect. The high neutron flux trip takes effect 0.25 sec after the peak power is reached and terminates the transient. The peak thermal heat flux is only 37 percent of the rated power heat flux. Note that the time scale on Figure 15-1 and Figure 15-2 does not begin with initiation of rod withdrawal, but rather shows the time interval of greatest interest.

A sensitivity analysis was performed on both of these startup accidents to determine the effect of varying several key parameters. Figure 15-3 through Figure 15-6 show typical results for the single rod group, 1.5 percent $\Delta k/k$ startup accident. Oconee 3 has the Type C control rod drive mechanism which is characterized by an increased trip delay time of an additional 0.1 sec. The effect of the trip delay time on the peak thermal power is shown to be minimal in Figure 15-7.

Figure 15-3 and Figure 15-4 show the effect of varying the reactivity addition rate on the peak thermal power and peak neutron power. The reactivity rate is varied from more than an order of magnitude below the nominal single rod group rate used for analysis (i.e., the rate for the 1.5 percent $\Delta k/k$ group) to a rate above that for simultaneous withdrawal of all rods. The slower rates up to about $2.15 \times 10^{-4}$ ($\Delta k/k$)/sec will result in the pressure trip being actuated. Only the very fast rates actuate the high neutron flux level trip.

Figure 15-5 and Figure 15-6 show the peak thermal power variation as a function of a wide range of Doppler and moderator coefficients for the 1.5 percent $\Delta k/k$ rod group. The peak thermal power varied about 10 percent from the nominal case for the moderator coefficient variation, and also by about 10 percent from the nominal for the range of Doppler coefficients. Figure 15-8 and Figure 15-9 are the corresponding results from the withdrawal of all rods (10 percent $\Delta k/k$).

None of these postulated start-up accidents, except for reactivity addition rates greater than $2 \times 10^{-3}$ ($\Delta k/k$)/sec, which is three times greater than for withdrawal of all rods at once, causes a thermal power peak in excess of 100 percent rated power or a nominal fuel rod average temperature greater than 1,150°F. The nominal 1.5 percent $\Delta k/k$ rod group withdrawal causes a peak pressurizer pressure of 2,515 psia, the code safety valve setpoint.

Figure 15-10 shows the pressure response for a reactivity addition rate of $2.15 \times 10^{-4}$ ($\Delta k/k$)/sec, which is the maximum reactivity insertion rate which will not result in a high flux trip. The system expansion rate reaches its maximum value of 1000 lb/sec at 58 sec. The safety valves are capable of preventing pressure rises above 2515 psia up to surge flows of about 1000 lb/sec, therefore the pressure increases above 2515 psia and reaches a maximum value of 2575 psia at 62 sec. The pressure then decreases to the safety valve setpoint and is approximately constant for the rest of the calculation.

It is concluded that the reactor is completely protected against any startup accident involving the withdrawal of any or all control rods, since in no case does the thermal power approach 112 percent, and the peak pressure never exceeds code allowable limits.

# 15.2 STARTUP ACCIDENT

## 15.2.1 IDENTIFICATION OF CAUSE

The objective of a normal startup is to bring a subcritical reactor to the critical or slightly supercritical condition, and then to increase power in a controlled manner until the desired power level and system operating temperature are obtained. During a startup, an uncontrolled reactivity addition could cause a nuclear excursion. The excursion is terminated by the strong negative Doppler effect alone, if no other protective reactor trips are actuated. The following design provisions minimize the possibility of inadvertent continuous rod withdrawal and limit the potential power excursions:

1. The control system is designed so that only one control rod group can be withdrawn at a time, except that there is a 25 percent overlap in travel between two regulating rod groups successively withdrawn. This overlap occurs at the minimum worth positions for each group since one group is at the end of travel and the other is at the beginning of travel. The nominal calculated worth of any single control rod regulating group is 1.5 percent $\Delta k/k$ when the reactor is critical.
2. Control rod withdrawal rate is limited to 30 in./min.
4 3. A withdrawal stop and alarm are provided in the wide range.
4 4. A high flux level and a high pressure trip are provided.

The criteria for the analysis of this accident is that the RCS shall be designed to limit (a) the reactor thermal power to 112 percent of rated power, and (b) the RCS pressure so as not to exceed code pressure limits.

## 15.2.2 METHODS OF ANALYSIS

A B&W digital computer model of the reactor core and coolant system is used to determine the characteristics of this accident. This model uses full reactor coolant flow, but no heat transfer out of the system and no sprays in the pressurizer. The rated power Doppler coefficient is used although the Doppler is much larger than this for the principal part of the transient. The rods are assumed to be moving along the steepest part of the integral rod worth curve. The values of the principal parameters used are 0.5 sec trip delay for high pressure trip, 0.3 sec trip delay for high flux trip, $+0.5 \times 10^{-4}$ $(\Delta k/k)/°F$ moderator coefficient, and $-1.17 \times 10^{-5}$ $(\Delta k/k)/°F$ Doppler coefficient. In addition, the criterion for minimum movable control rod worth is that a shutdown margin of 1 percent $\Delta k/k$ at the hot shutdown condition is required. The startup accident is analyzed using the minimum tripped rod worth as part of the analysis.

The initial conditions for the startup accident are 1 percent $\Delta k/k$ subcritical at $10^{-9}$ rated power, a hot shutdown initial condition.

## 15.2.3 RESULTS OF ANALYSIS

Figure 15-1 shows the results of withdrawing a control rod group at a rod speed of 30 in./min from 1 percent subcritical and $10^{-9}$ rated power. The group is worth 1.5 percent $\Delta k/k$. The rod velocity and worth result in a reactivity addition rate of $1.09 \times 10^{-4}$ $(\Delta k/k)/sec$. The Doppler effect begins to slow the neutron power rise (total energy release from fission), but the heat transfer to the coolant increases RCS pressure past the trip setpoint, and the transient is terminated by the high pressure trip.

# 15.1 UNCOMPENSATED OPERATING REACTIVITY CHANGES

## 15.1.1 IDENTIFICATION OF CAUSE

During normal operation of the reactor, the overall reactivity of the core changes because of fuel depletion and changes in fission product poison concentration. These reactivity changes, if left uncompensated, can cause operating limits to be exceeded. In all cases, however, the RPS prevents safety limits from being exceeded. No damage occurs from these conditions.

## 15.1.2 ANALYSIS AND RESULTS

During normal operation, the Integrated Control (ICS) System senses any reactivity change in the reactor. Depending on the direction of the reactivity change, the reactor power increases or decreases. Correspondingly, the Reactor Coolant System (RCS) average temperature increases or decreases, and the ICS acts to restore reactor power to the power demand level and to reestablish average temperature at the setpoint. If manual corrective action is not taken or if the control system malfunctions, the RCS average temperature changes to compensate for the reactivity disturbance. It is assumed in the analysis that the secondary system follows the temperature changes in the RCS. The reactivity effect due to fuel depletion and xenon buildup is as follows:

| Cause | Maximum Reactivity Rate ($\Delta k/k$)/min) | Rate of Average Temperature Change (Uncorrected) (°F/min) |
|---|---|---|
| Fuel Depletion | $-2.9 \times 10^{-7}$ | -0.0007 |
| Xenon Buildup | $-2.2 \times 10^{-5}$ | -0.060 |

These results are based on $+0.5 \times 10^{-4}$ ($\Delta k/k$)/°F moderator coefficient and $-1.17 \times 10^{-5}$ ($\Delta k/k$)/°F Doppler coefficient, representative of beginning of core life for the first cycle. These reactivity changes are extremely slow and allow the operator to detect and compensate for the change.

# CHAPTER 15. ACCIDENT ANALYSES

This section details the expected response of the plant to the spectrum of transients and accidents which constitute the design basis events. The analyses presented show that the plant response is either inherently limited by the characteristics of the system or is terminated by the normal functions of the Reactor Protective System (RPS) and the Engineered Safeguards Protective System (ESPS). The analyses are evaluated for an initial core power of 2,568 MWt unless specified otherwise. The consequences of the worst case loss of coolant accident (LOCA) are demonstrated to be within the limits specified in 10CFR50.46. For all transients and accidents resulting in a release of fission products to the environment, the dose consequences using the dispersion model developed in Section 2.3, "Meteorology" on page 2-11 are shown to be within the limits specified in 10CFR100.

# LIST OF FIGURES

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## 14.6.1 REFERENCES


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**THIS IS THE LAST PAGE OF THE CHAPTER 14 TEXT PORTION.**

## 14.6 OPERATING RESTRICTIONS

During initial operations and associated testing, the normal plant safety procedures and technical specifications are in effect. In addition, special safety precautions and limitations are included in the test procedures and more restrictive operating limitations than those in the technical specifications are imposed, where required, from initial criticality through the power escalation program. The Reactor Protective System power level trip point was initially set at a low value and raised as the power escalation program progresses.

## 14.5 STARTUP PHYSICS TEST PROGRAM

3

# 14.4 POSTCRITICALITY TEST PROGRAM

2 The Postcriticality Test Program was performed to provide assurance that the plant is operating in a safe and efficient manner. Systems and components which cannot be operationally tested prior to initial
2 criticality were tested during the Postcriticality Test Program to verify reactor parameters and to obtain information required for plant operation. A list of the postcriticality tests is provided in Table 14-2. This
2 section summarizes the test program after each unit achieved initial criticality. The startup reports and supplements, References 1 on page 14-16 through 14 on page 14-16, provide the results of the startup test program for each unit.

## 14.4.1 ZERO POWER PHYSICS TESTS

2 Following initial criticality, a program of reactor physics measurements was undertaken to verify the
2 physics parameters. Measurements were made under zero power condition at sufficient temperature plateaus to verify calculated worths of individual control rods and control rod groups, moderator temperature coefficient, boron worth, and excess reactivity of the core. In addition, the response of the
2 source and intermediate range nuclear instrumentation were verified.

Detailed written procedures specifying the sequence of tests, parameters to be measured, and conditions under which each test is to be performed were followed. These tests involve a series of prescribed control rod configurations and boron concentrations with intervening measurements of control rod and/or boron worth during boron dilution or boron injection.

## 14.4.2 POWER ESCALATION TEST PROGRAM

Following determination of the operating characteristics and physics parameters of the reactor at zero power, a detailed power escalation test program was conducted. This program consists of specified incremental increases in power levels up to full power with appropriate testing conducted at each power level. An analysis of the significant parameters at each step was made prior to initiating an additional power escalation.

At selected power levels, the following tests were performed:

1. Unit heat balance test
2. Power coefficient measurement
3. Core power distribution measurement
4. Unit load steady state test
5. Unit transient test.

Other Power Escalation Tests were performed at one or more power levels in the test sequence.

A reactor coolant flow test and a reactor coolant flow coastdown test were conducted under cold reactor conditions to assure that the flow characteristics of the Reactor Coolant System had not materially changed as a result of the reactor core installation.

## 14.3.3 INITIAL CRITICALITY

A written procedure was followed during the approach to initial criticality. This procedure specified in detail the sequence to be followed, the limitations and precautions, the required plant status, and the prerequisite system conditions. (This procedure also specified the alignment of fluid systems to assure controlled boron dilution and core conditions under which the approach to criticality proceeded.)

Permissible rod group withdrawal and deboration are based on calculated reactivity effects. Two independent plots of inverse multiplication characteristics are maintained during rod group withdrawal and deboration. A predicted rod group position or boron concentration for criticality is determined before the next rod group withdrawal or deboration is started.

# 14.3 INITIAL CRITICALITY TEST PROGRAM

The Initial Criticality Test Program consists of the initial fuel loading followed by initial criticality.

## 14.3.1 INITIAL FUEL LOADING

Fuel was loaded into the reactor in accordance with a step-by-step written procedure. This procedure contains a number of safety precautions and operating limitations.

The fuel loading procedure includes:

1. A sequence of loading temporary detectors, sources, control rods, and fuel assemblies in order to maintain shutdown margin requirements.
2. The conditions under which fuel loading may continue after any step.
3. An identification of responsibility and authority.
4. During any reactivity changes, a minimum of two detectors will be operating and indicating neutron level after the source has been inserted. At all other times, at least one detector shall be indicating neutron level.
5. Two completely independent plots of reciprocal neutron multiplications as a function of the parameter causing reactivity change are maintained.
6. Reactivity effects for each fuel assembly addition are checked prior to the release of the fuel assembly by the fuel handling grapple.
7. An estimate of the reactivity effect for the next fuel addition is made prior to insertion of the next fuel assembly.
8. The boron concentration in the reactor vessel, spent fuel pool, and Reactor Coolant System is maintained at a value to assure the required subcritical margin at all times.
9. The valve alignment of the auxiliary systems connected to the Reactor Coolant System is checked periodically to prevent dilution of the reactor coolant boron concentration.
10. Chemical analysis and water level monitoring is used to assure that inadvertent dilution of the reactor coolant boron concentration has not occurred.
11. Communication between control room and fuel handling areas is maintained.
12. The Plant Radiation Monitoring Systems are in operation.
13. Radiation Protection and chemistry monitoring and services are provided.

## 14.3.2 PREPARATION FOR INITIAL CRITICALITY

Upon completion of the initial fuel loading, prestartup checks were completed prior to the approach to initial criticalty. The prestartup checks included:

1. Control rod trip test
2. Reactor coolant flow test
3. Reactor coolant flow coastdown test

1. Operational test of systems, components, and non-nuclear instrumentation and controls at no load operating pressure and temperature.
2. Operator training.
3. Verification of normal operating procedures.
4. Verification of emergency operating procedures.

Following the hot functional test, the reactor vessel intervals were removed and inspected for signs of distress, e.g., loose parts, cracking, or fretting.

# 14.2 TESTS PRIOR TO REACTOR FUEL LOADING

The tests prior to reactor fuel loading assure that systems are complete and operate in accordance with design. The test program was divided into two phases: Preheatup Test Phase and Hot Functional Test Phase. In many instances systems were tested during both the Preheatup Test Phase and the Hot Functional Test Phase. A list of the tests performed prior to fuel loading is provided in Table 14-1. This section summarizes the initial test program prior to fuel loading for Oconee 1, 2, and 3. The startup reports and supplements, References 1 on page 14-16 through 14 on page 14-16, provide the results of the startup test program for each unit.

The types of tests are classified as hydro/leak, operational, electrical, and functional with the following definitions for each classification:

| | | |
|---|---|---|
| Hydro/Leak Test | – | Structural integrity leak test of the various systems and components at the appropriate pressure. |
| Operational Test | – | Operation of systems and equipment under operating conditions. |
| Electrical Test | – | Consists of: grounding, megger, continuity, and phasing checks; circuit breaker operation and control checks; potential measurement and energizing of buses and equipment to ensure continuity, circuit integrity, and proper functioning of electrical apparatus. |
| Functional Test | – | Tests to verify that systems and equipment will function as intended. |

Instruments and controls of each system or component were also subjected to a preoperational instrumentation and controls calibration prior to the initial operation of that system or component to assure proper operation.

An Engineered Safeguard Actuation System test was performed to assure actuation and proper operation of the Engineered Safeguards System and to evaluate the test method and frequency for future testing.

## 14.2.1 PREHEATUP TEST PHASE

The objective of the Preheatup Test Phase was to assure that the equipment and systems perform as required for hot functional testing. This phase of the testing included certain preoperational calibration, hydro/leak, operational, electrical, and functional tests as required. The Reactor Building Containment System has undergone a structural integrity and integrated leakage rate test to verify the building design and to ensure that leakage is within the design limit.

## 14.2.2 HOT FUNCTIONAL TEST PHASE

The Hot Functional Test Phase was a period of hot operation of the Reactor Coolant System and the associated auxiliary systems prior to the initial fueling of the reactor. The Reactor Coolant System was heated up to no-load operating pressure and temperature.

The Hot Functional Test Phase continued the preparation toward the initial fuel loading. The objectives of this phase of the test program were:

Retests were performed on systems and components as necessary to verify the adequacy of the corrective action.

Prior to any revisions relating to the health and safety of the public or plant personnel, structural integrity of plant components and systems, and items covered by codes and nuclear standards, review and approval was necessary by the Duke Power Design Engineering Department with assistance from vendors or consultants as necessary.

During the initial criticality (including fuel loading) and post-criticality phases of the test program, the nuclear physics and thermal hydraulics aspects of the reactor operation were under the technical responsibility of the Nuclear Production Department Nuclear Engineer and the Oconee Technical Support Engineer with assistance from B&W Site Operations, B&W Nuclear Power Generation Division, and Duke Engineering Department nuclear engineers as needed. A very close coordination between these groups existed with the appropriate support available when needed.

## 14.1.2 RESPONSIBILITIES

### 14.1.2.1 Superintendent

The Superintendent or his authorized representative has final responsibility for the overall test program which included the approval of the test procedures, modification of test procedures, scheduling, completion of the tests, and approval of the test results. Approval of test procedures, modifications of test procedures, and approval of test results was not be made without giving proper consideration to recommendations of Babcock and Bechtel in their areas of interest.

### 14.1.2.2 Test Working Group

A Test Working Group (TWG) coordinated the activities of B&W, Duke Construction, and Nuclear Production Department during the preoperational test program. Representatives were from Oconee Nuclear Station and B&W (Site Operations Engineer). Duke Engineering; Construction; Steam Production General Office; and Electrical, Maintenance, and Construction Departments had representatives participate as required. The Oconee representative was chairman of the TWG. The TWG met at regular intervals; approximately every week during the most active phases of the program.

### 14.1.2.3 Station Test Coordinator

A station test coordinator was designated for each test. His responsibility was to develop the test procedure, coordinate the performance of the test, analyze results, identify discrepancies in test and acceptance criteria, initiate action to correct discrepancies, obtain approval of other parties when test had been completed satisfactorily, and file results in the master final documentation file.

### 14.1.2.4 Nuclear Test Engineer

A general office nuclear test engineer was designated for the testing program. His responsibility was to furnish technical guidance for the test program; to assist in the development of the approved procedures; and to assist the station personnel in conducting and evaluating the tests. Other members of the general office staff assisted in the test program as necessary.

### 14.1.2.5 Nuclear Safety Review Committee

An audit of safety related tests and their results was performed by the Nuclear Safety Review Committee.

## 14.1.3 RESOLUTION OF DISCREPANCIES

Any discrepancies in systems or equipment found during the Test Program was promptly reported by the station test coordinator to the Superintendent. A corrective action request was made to the appropriate departments by the Superintendent to initiate any revision or repair deemed necessary. After the corrective action had been completed the Superintendent or his authorized representative was notified.

# 14.1 ORGANIZATION OF TEST PROGRAM

## 14.1.1 GENERAL ORGANIZATION

The organization for development and execution of the test program had major participants from the Oconee Nuclear Station operating personnel, the Nuclear Production Department General Office staff, and Babcock & Wilcox (B&W) Site Operations. Additional participants were from the Duke Engineering Department; Construction Department; and Electrical, Maintenance, and Construction Department. Bechtel Corporation participated in the tests associated with the Reactor Building.

The Oconee Nuclear Station organization for the test program consisted of the Superintendent, Assistant Superintendent, Station Review Committee (SRC), and a station test coordinator assigned for each test.

The Nuclear Production Department General Office staff organization for the test program consisted of a Nuclear Production General Office test coordinator assigned for each test.

The B&W Site Operations organization for the test program consisted of the Site Operations Manager, Site Operations Engineer, and Site Service Engineers who worked in the specific areas of test procedures, testing, startup, operations, maintenance, fueling, field analysis, and reports. The test program had technical support from B&W Nuclear Power Generation Division engineers. This support included technical analysis of the test results of certain tests with the result analyses transmitted to the Nuclear Production Department through normal channels of communication for checking and final analyses prior to test completed approval. Special rapid channels of communication were utilized where results were needed as soon as possible for other operations to proceed. The qualifications for the B&W Site Operations organization are listed below:

1. The minimum qualification for the B&W Site Operations Manager are:
   a. Graduate in engineering, or related physical science, or equivalent experience. (2 years experience for one year of college).
   b. Four years of responsible power plant experience or two years of responsible nuclear reactor experience.
   c. One year engineering or test program preparation experience for this or similar nuclear plant.
2. The minimum qualifications for the B&W Site Operations Engineer are:
   a. Graduate in engineering, or related physical science or equivalent experience. (2 years experience for one year of college).
   b. Two years of responsible power plant experience or one year of responsible nuclear reactor experience.
   c. One year engineering or test program preparation experience for this or similar nuclear plant.

Various individuals from the Mechanical, Electrical, and Civil sections of the Duke Engineering Department furnished technical support as needed in specific areas. Similarly, individuals in the Duke Construction Department, Duke Electrical, Maintenance, and Construction Department; and Bechtel Corporation furnished technical support as needed. This support principally applied to the review of test procedures prior to approval, analysis of test results, and the development and installation of modifications to the equipment and systems as required and identified during the test program. Qualifications for Duke personnel are contained in Chapter 13, "Conduct of Operations" on page 13-1.

# CHAPTER 14. INITIAL TESTS AND OPERATION

A comprehensive initial testing and operating program was conducted at the Oconee Nuclear Station. The purpose of this program was (1) to assure that the equipment and systems perform in accordance with design criteria, (2) to effect initial fuel loading in a safe efficient manner, (3) to determine the nuclear parameters, and (4) to bring the unit to rated capacity.

The test program began as installation of individual components and systems was completed. The individual components and systems were tested and evaluated according to written test procedures. An analysis of the test results verified that each component and system performed satisfactorily.

The written procedures for the initial tests and operation included the purpose, conditions, precautions, limitations, prerequisites, and the acceptance criteria.

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## 2 13.6 NUCLEAR SECURITY

The Station Physical Security Plan, Safeguards Contingency Plan and the Training and Qualification Plan (T&Q Plan) were submitted and NRC accepted for the protection of Oconee nuclear station against potential acts of radiological sabotage. This information is to be withheld from public disclosure pursuant to 10CFR 2.790(d) and 10CFR 73.21. The general scope of Safeguard Activities encompassed by the
2 Safeguard Plans include: (1) the physical security organization; (2) selection and training of personnel for
2 security purposes; (3) communications systems for controls to the plant including physical barriers and
2 means of detecting unauthorized intrusions; and (4) arrangements with law enforcement authorities for
2 assistance in responding to any security threat.

The Safeguard Plans conform to the requirements of 10CFR 50.34(c) and (d) and 10CFR 73.55.

5 A vehicle barrier system (VBS) was installed at Oconee in accordance with NRC guidance provided in
5 NUREG/CR-6190 and 10CFR73.55 (Reference NSM-52973). The VBS prevents vehicle intrusion and is
5 required to comply with 10CFR73.55(c)(7)(8)(9)&(10). The vehicle barrier system is not nuclear safety
5 related.

**THIS IS THE LAST PAGE OF THE CHAPTER 13 TEXT PORTION.**

1 The Chemistry Manager has responsibility for preparation and implementation of chemistry procedures.

#### 13.5.2.2.5 Radioactive Waste Management Procedures

Radioactive waste management activities associated with the station's liquid, gaseous, and solid waste systems are performed in accordance with approved, written procedures.

Each procedure is sufficiently detailed that qualified workers can perform the required functions without direct supervision. Written procedures, however, cannot address all contingencies, and therefore contain a degree of flexibility appropriate to the activities for which each is applicable.

The station's Operations group, Chemistry, and Radiation Protection sections have responsibility for preparation and implementation of the radioactive waste management procedures.

#### 13.5.2.2.6 Radiation Protection Procedures

Information concerning these procedures is presented in Chapter 12, "Radiation Protection" on page 12-1.

#### 13.5.2.2.7 Plant Security Procedures

Station Security Procedures shall be developed to implement the scope of Safeguard Activities required by the safeguard plans addressed in Section 13.6, "Nuclear Security" on page 13-37 of the FSAR.

#### 13.5.2.2.8 Emergency Preparedness Procedures

2 Information concerning these procedures is presented in the Oconee Nuclear Site Emergency Plan which
2 is discussed in topic 13.3, "Emergency Planning" on page 13-23.

#### 13.5.2.2.9 Material Control Procedures

Information concerning these procedures is presented in the Duke Power Company Topical Report,
1 Quality Assurance Program, DUKE-1A.

#### 13.5.2.2.10 Modification Procedures

Information concerning these procedures is presented in the Duke Power Company Topical Report,
1 Quality Assurance Program, DUKE-1A.

#### 13.5.2.2.11 Fire Protection Procedures

Information concerning these procedures is presented in Section 13.5.1.3.11, "Fire Protection Procedures" on page 13-31.

5 The Maintenance Superintendent has responsibility for preparation and implementation of maintenance procedures.

The administrative control of maintenance is maintained as follows:

1. In order to assure safe, reliable, and efficient operation, a comprehensive maintenance program for the station's safety-related structures, systems, and components is established.
2. 5 5 The Maintenance Superintendent is responsible for directing the performance of station maintenance activities affecting instrumentation and electrical and mechanical equipment.
3. Personnel performing maintenance activities are qualified in accordance with applicable codes and standards, as appropriate.
4. Maintenance is performed in accordance with written procedures which conform to applicable codes, standards, specifications, criteria, etc.
5. Maintenance is scheduled so as not to jeopardize station operation or the safety of a reactor or reactors.
6. Maintenance histories are maintained on station safety-related structures, systems, and components.

1 The administrative control of modifications is discussed in "Quality Assurance Program," Topical Report, DUKE-1A.

#### 13.5.2.2.2 Instrument Procedures

Maintenance, testing, and calibration of station safety-related instruments is performed in accordance with written, approved procedures.

Instrument procedures are sufficiently detailed that qualified workers can perform the required functions without direct supervision. Written procedures, however, cannot address all contingencies, and therefore contain a degree of flexibility appropriate to the activities for which each is applicable.

5 The Maintenance Superintendent has responsibility for preparation and implementation of instrument procedures.

#### 13.5.2.2.3 Periodic Test Procedures

Testing conducted on a periodic basis to determine various station parameters and to verify the continuing capability of safety-related structures, systems, and components to meet performance requirements is conducted in accordance with approval, written procedures. Periodic test procedures are utilized to perform such testing, and are sufficiently detailed that qualified personnel can perform the required functions without direct supervision.

5 Periodic test procedures are performed by the station's Engineering, Operations, and Maintenance groups.

#### 13.5.2.2.4 Chemistry Procedures

Chemical and radiochemical activities associated with station safety-related structures, systems, and components are performed in accordance with approved, written procedures and the station chemistry manual.

Each procedure is sufficiently detailed that qualified workers can perform the required functions without direct supervision. Written procedures, however, cannot address all contingencies, and therefore contain a degree of flexibility appropriate to the activities for which each is applicable.

Duke Power Company has also in place a program for preparing and implementing emergency operating procedures. This program was developed in response to NUREG-0737 Item I.C.1, "Guidance for the Evaluation and Development of Procedures for Transients and Accidents." Duke Power Company's program for developing emergency operating procedures for Oconee Units 1, 2, and 3 has been reviewed and approved by NRC. (Letter from John F. Stolz (NRC) to Hal B. Tucker (Duke) date June 7, 1985. Subject: Safety Evaluation Report on "Procedures Generation Package").

#### 13.5.2.1.3 Temporary Operating Procedures

Temporary operating procedures are approved written procedures issued for operating activities which are of a nonrecurring nature. Examples of such uses are: (a) to direct operating activities during special testing or maintenance; (b) to provide guidance in unusual situations not within the scope of normal procedures; and (c) to assure orderly and uniform operations for short periods of time when the station, a unit, a structure, a system, or, a component is performing in a manner not addressed by existing procedures, or has been modified or extended in such a manner that portions of existing procedures do not apply.

The format of these procedures includes a purpose, limits and precautions, initial conditions, and step-by-step instructions for each mode of operation and necessary enclosures.

Temporary operating procedures are sufficiently detailed that qualified individuals can perform the required functions without direct supervision. Written procedures, however, cannot address all contingencies, and therefore contain a degree of flexibility appropriate to the activities for which each is applicable.

#### 13.5.2.1.4 Annunciator Response Procedures

Annunciator response procedures are written which specify operator actions necessary to respond to an off-normal condition as indicated by an alarm. The format for annunciator response procedures includes alarm setpoints, probable causes, automatic actions, immediate manual actions, supplementary actions, and applicable references.

In order to insure that annunciator response procedures are readily accessible for reference, a positive method is employed to allow their retrieval. Each annunciator panel is designated by a unique and obvious nameplate. All of the annunciator windows within a panel are designated by identifying names. The annunicator response procedures are grouped by panels, then subdivided by annunciator names so that the response procedure for any annunciator may be quickly located.

### 13.5.2.2 Other Procedures

#### 13.5.2.2.1 Maintenance Procedures

Maintenance of station safety-related structures, systems, and components is performed in accordance with written procedures, documented instructions, or drawings appropriate to the circumstances (for example, skills normally possessed by qualified maintenance personnel may not require detailed step-by-step delineation in a written procedure) which conform to applicable codes, standards, specifications, criteria, etc. Where appropriate sections of related vendor manuals, instructions, or approved drawings with acceptable tolerances do not provide adequate guidance to assure the required quality of work, an approved, written maintenance procedure is provided.

Maintenance procedures are sufficiently detailed that qualified workers can perform the required functions without direct supervision. Written procedures, however, cannot address all contingencies, and therefore, contain a degree of flexibility appropriate to the activities for which each is applicable.

Instrument Air System Operation
Low Pressure Service Water System Operation
Nitrogen System Operation
Nuclear Fuel Control and Accountability
Reactor Coolant Pump Operation
Receipt, Inspection and Storage of New Fuel
Recirculated Cooling Water System Operation
DHR Cooling System Operation
Injection System Operation
1 Spent Fuel Pool Cooling and Purification System Operation
Spent Fuel Handling and Shipping
5 Standby Shutdown Facility Systems Operation
Steam Generator Secondary Side Operation
Turbine-Generator Operation
Unit Operation at Power
Unit Shutdown
Unit Startup

#### 13.5.2.1.2 Emergency Procedures

Emergency procedures are written which specify steps to be taken during foreseeable emergency situations. These procedures are based on a sequence of observations and actions, with emphasis placed on operator responses to indications in the Control Room. When immediate operator actions are required to prevent or mitigate the consequences of an emergency situation, procedures require that those actions be implemented at the earliest possible time, even if full knowledge of the emergency situation is not yet available.

The actions outlined in emergency procedures are based on a conservative course of action to be followed by the operating crew. Written procedures, however, cannot address all contingencies, and emergency procedures, therefore, contain a degree of flexibility consistent with the fact that an emergency situation may not follow an anticipated sequence.

Typical situations addressed by emergency procedures are:

Abnormal Release of Radioactivity
Acts of Nature (Earthquake, Flood, Tornado, etc.)
Inoperable Control Element Assemblies
Loss of Component Cooling
Loss of Containment Integrity
Loss of Control Room
Loss of Electrical Power
1 Loss of Feedwater
Loss of Instrument Air
Loss of Reactor Coolant
Loss of Reactor Coolant Flow
Loss of Residual Heat Removal
Loss of Flux Indication
Reactor Trip
Spent Fuel Damage
Steam Generator Tube Failure
Steam Supply System Rupture
Turbine-Generator Trip

5 2. Operations Logbook - This document contains documentation of significant events occurring each
5 shift. Examples include reactivity changes, alarms received, abnormal conditions of operation due to
5 auxiliary equipment and all releases of radioactive waste. It contains a summary of unit operation for
5 each shift. Entries are made by Reactor Operators and/or Senior Reactor Operators.

#### 13.5.1.3.10 Temporary Procedures

The use of temporary procedures is discussed in Section 13.5.2.1.3, "Temporary Operating Procedures" on page 13-33.

#### 13.5.1.3.11 Fire Protection Procedures

Fire protection procedures are written to address such topics as training of the fire brigade, reporting of fires, and control of fire stops. The station's administrative services group has responsibility for fire protection procedures in general, with the station's maintenance section having responsibility for certain fire protection procedures, such as control of repairs to station fire stops.

## 13.5.2 OPERATING AND MAINTENANCE PROCEDURES

### 13.5.2.1 Operating Procedures

#### 13.5.2.1.1 System Procedures

Operating activities which affect the proper functioning of the station's safety-related systems and components are performed in accordance with approved, written procedures. These procedures are intended to provide a pre-planned method of conducting operations of systems, in order to eliminate errors due to on-the-spot analyses and judgements.

Operating procedures are sufficiently detailed that qualified individuals can perform the required functions without direct supervision. Written procedures, however, cannot address all contingencies, and operating procedures, therefore, contain a degree of flexibility appropriate to the activities for which each is applicable.

Typical activities addressed by operating procedures are:

Auxiliary Building Ventilation System Operation
Emergency Feedwater System Operation
Boron Recycle System Operation
Chemical and Volume Control System Operation
Component Cooling Water System Operation
Condensate and Feedwater Systems Operation
Condenser Circulating Water System Operation
Reactor Building Ventilation System Operation
Reactor Building Spray System Operation
Control Room Ventilation System Operation
Degasification of the Reactor Coolant System
Demineralizer Resin Removal and Replacement
Electrical Systems Operation
Failed Fuel Detection and Handling
Filling and Draining of the Refueling Canal
Filling, Venting and Draining of the Reactor Coolant System
Fire Protection Systems Operation

#### 13.5.1.3.3 Activities Affecting Station Operation or Operating Indications

All station personnel performing functions which may affect unit operation or control room indications are required to notify the Control Operator (licensed reactor operator) prior to initiating such action. Removal of an instrument or component from service requires the permission of the Operations Shift Manager or Shift Supervisor (licensed senior reactor operators). During refueling outages, the Work Control Center SRO may be the primary point of contact, acting as a representative of the Operations Shift Manager and/or Shift Supervisor.

#### 13.5.1.3.4 Manipulation of Facility Controls

No one is permitted to manipulate the facility controls who is not a licensed reactor operator or senior reactor operator, except for license trainees operating under the direction of a licensed operator. The licensed operators are required to comply with the requalification program as described in Section 13.2, "Training" on page 13-11.

Operations Management Procedures are written that delineate the responsibilities of the reactor operators on the control board and the responsibilities of the senior reactor operator in the Control Room. When Technical Specifications require one (1) man in the Control Room (at the controls) this is defined as: Must be in visible line of Nuclear Instrumentation. See cross hatched area on Figure 13-5 and Figure 13-6. When Technical Specifications require two (2) R.O.'s in the Control Room, one (1) R.O. will be "at the controls" as defined above and the second R.O. will be inside the CAD key doors that are used for entering and exiting the Control Room.

#### 13.5.1.3.5 Responsibility for Licensed Activities

Responsibility for directing the licensed activities of licensed operators is assigned to individuals with senior reactor operator licenses by virtue of their position within the station organization.

#### 13.5.1.3.6 Relief of Duties

This procedure provides a detailed checklist of applicable items for shift turnover.

#### 13.5.1.3.7 Equipment Control

Equipment control is maintained and documented through the use of tags, labels, stamps, status logs, or other suitable means.

#### 13.5.1.3.8 Master Surveillance Testing Schedule

This procedure establishes a master surveillance testing schedule to assure that required testing is performed and evaluated on a timely basis. Surveillance testing is scheduled such that the safety of the station is not dependent on the performance of a structure, system, or component which has not been tested within its specified testing interval. The master surveillance testing schedule identifies surveillance and testing requirements, applicable procedures, and required test frequency. Assignment of responsibility for these requirements is also indicated.

#### 13.5.1.3.9 Log Books

The following log books are maintained and reviewed by appropriate personnel:

1. Switchboard Record - This document contains data on station and unit electrical power generation, bus voltages, etc.

# 13.5 STATION PROCEDURES

## 13.5.1 ADMINISTRATIVE PROCEDURES

### 13.5.1.1 Conformance With Regulatory Guides

1 Regulatory Guide 1.33, "Quality Assurance Program Requirements," and ANSI N18.7-1976, "Standard for Administrative Controls for Nuclear Power Plants" shall be used for the preparation of administrative and plant procedures.

### 13.5.1.2 Preparation of Procedures

For operating, emergency, maintenance, instrument, periodic test, chemistry, radioactive waste management, radiation protection, emergency preparedness, and modification procedures, each procedure is assigned to a member of the station staff for development. Initial procedure drafts are reviewed by
1 members of the station staff, the Nuclear Generation Department General Office, and other departments within Duke, personnel from the NSSS supplier, and other vendors as appropriate. Following resolution of review comments, if any, a revised procedure is prepared and forwarded to a previously designated qualified reviewer for review and comment. This qualified reviewer also makes the determination whether or not any additional, cross-disciplinary review is required. After all required and appropriate reviews
1 have been completed a final version of a procedure is prepared. Upon approval by the responsible
1 implementing manager as previously designated, a procedure becomes available for use. Additional
1 discussion of procedure preparation control is contained in "Quality Assurance Program," Topical Report, DUKE-1A and in the Technical Specifications.

Administrative, annunciator response, security, and material control procedures are prepared by qualified
5 personnel, reviewed as necessary, and approved by the station Manager or his designee prior to use.

### 13.5.1.3 Administrative Procedures

2 Station administrative procedures are written as necessary to control station testing, maintenance, and operating activities. Listed below are several areas for which administrative procedures are written, including principle features:

#### 13.5.1.3.1 The Reactor Operator's Authority and Responsibility

The reactor operator is given the authority to manipulate controls which directly or indirectly affect core reactivity, including a reactor trip if he deems necessary. He is also assigned the responsibility for knowing the limits and setpoints associated with safety-related equipment and systems as specified in the Technical Specifications and designated in the operating procedures.

#### 13.5.1.3.2 The Senior Reactor Operator's Authority and Responsibility

The senior reactor operator, in addition to the authorities and responsibilities described for the reactor operator, is given the authority to direct the licensed activities of the reactor operator, and ultimately is held responsible for all licensed activities at the station within his control.

The on-site organization includes the Station Manager, who is responsible for the operation and safety of the plant; the Station Safety Review Group (SRG), which prepares reports on events and reviews the adequacy of corrective actions; and the supervisors of areas relevant to an event, who may perform the initial investigation and must implement corrective actions.

The off-site organization consists of principal engineering support groups, who interface with station personnel and other organizations in investigating events and developing remedial actions; company management, who holds overall responsibility for nuclear plant safety; and the Nuclear Safety Review Board (NSRB) which performs an independent review function.

5

c. The results of actions taken to correct deficiencies occurring in equipment, structures, systems, or methods of operation that affect nuclear safety.

d. The performance of activities required by the quality assurance program to meet the criteria of Appendix B to 10 CFR 50.

e. The station emergency plan and implementing procedures.

f. The station security plan and implementing procedures.

#### 13.4.2.2 Onsite

5 The Safety Review Group (SRG) is established for the purpose of independently examining and making recommendations to management on plant operating characteristics, NRC issuances, Licensing information Advisories, and other appropriate sources of plant design and operating experience
3 information that may indicate areas for improving plant safety. This is accomplished by performing
3 independent reviews of plant activities including maintenance, modifications, operational problems, and
3 operational analyses, and aiding in the establishment of program requirements for plant activities. The
3 SRG verifies and reports on plant operations and maintenance activities and that human errors are
3 reduced as far as practical. The SRG performs incident investigations and the required reports as assigned
3 by the station organization. Periodically the SRG advises the station and corporate management on the
3 overall quality and safety of plant operations. Other information on the conduct of operation of the SRG
3 is provided in SRG procedures.

5 The Safety Review Group (SRG) is an independent review group, located onsite, and reporting to the
1 Manager, Safety Assurance. It is composed of a permanent manager and a minimum of four members. Qualifications of all members are that they shall have at least six years of technical experience with a minimum of two years being experience in their field at a nuclear plant or directly related to a nuclear plant. A maximum of four years of the six years may be fulfilled by academic or related technical training.

### 13.4.3 AUDIT PROGRAM

4 Operational quality assurance activities are periodically audited by the NAID, Regulatory Audits Group. A detailed description of this audit program is contained in Topical Report, DUKE-1A.

The Nuclear Safety Review Board is also responsible for audits of certain operating activities as discussed in Section 13.4.2, "Independent Review" on page 13-26.

### 13.4.4 OPERATING EXPERIENCE PROGRAM

The purpose of the Operating Experience Program (OEP) is to confirm nuclear safety and to optimize reliability through systematic evaluation of events occurring at Duke Power Company nuclear units, as well as at other facilities. This evaluation serves to verify that plant response was as expected during events and transients, and assures that any unexpected behavior is investigated thoroughly and is well understood. The results of this evaluation are then used to identify procedural and/or design changes which may mitigate or preclude the recurrence of a similar event or transient. In order to assure that the program is truly effective, information gained from Duke Power Company experience is disseminated to other organizations, as appropriate.

In order to achieve the objectives of the OEP in an efficient manner, an offsite organization has been established, as well as an on-site organization at each of the nuclear stations.

## 13.4.2 INDEPENDENT REVIEW

### 13.4.2.1 Offsite

The Nuclear Safety Review Board (NSRB) is established to verify that the operation of a station is performed in a safe manner consistent with Company policy, approved operating procedures and license provisions; to review important proposed station modifications, tests, and procedures; to verify that reportable occurrences are promptly investigated and corrected in a manner which reduces the probability of occurrence; and to detect trends which may not be apparent to a day-to-day observer. The Board
1 reports its findings and recommendations to the Executive Vice President, Power Generation, and as
1 required by Technical Specifications.

The membership of the NSRB collectively has the competence required to review problems in the following areas: Nuclear power station operations, nuclear engineering, chemistry, radio-chemistry, metallurgy, instrumentation and control, radiological safety, mechanical engineering, electrical engineering, and administrative control and quality assurance practices. The NSRB is composed of no less than five persons, of whom no more than one is a member of the station organization. A quorum consists of three members and must include either the Director or his designated alternate.

Formal meetings are held at least semi-annually. More frequent meetings are held if necessary.

4 Minutes of meetings are prepared and distributed to the Senior Vice President, Nuclear Generation, the
4 Site Vice President, and as required by Technical Specifications. The NSRB has the following general responsibilities:

a. Review safety evaluations for (1) changes to procedures, equipment, or systems, and (2) tests or experiments completed under the provisions of 10 CFR 50.59 (a) (1), to verify that such actions did not constitute an unreviewed safety question.

b. Review proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in 10 CFR 50.59.

c. Review tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59.

d. Review proposed changes in Technical Specifications or Facility Operating Licenses.

e. Review violations of applicable statutes, codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Review significant operating abnormalities or deviations from normal, and expected performance of station equipment that affect nuclear safety.

g. Review incidents that are the subject of non-routine reports submitted to the Commission.

4 h. Review NAID, Regulatory Audits Group audits relating to station operations, and actions taken in response to these audits.

Audits of station activities are performed by and under the cognizance of the NSRB. These audits encompass such items as:

a. The conformance of station operation to provisions contained within the Technical Specifications and applicable Facility Operating License conditions.

b. The performance, training, and qualifications of the station staff.

# 13.4 REVIEW AND AUDIT

In matters of nuclear safety, both onsite and off-site review of station startup, operation, maintenance, and technical matters is performed. Offsite review is performed by the Nuclear Safety Review Board (NSRB), whereas onsite review is performed by the Safety Review Group (SRG) and by other designated, qualified individuals. This review process commences at least six months prior to the initial operation of a station, so as to include preoperational testing and checkout of the station. Guidance in the development of the review program for test and operation is derived from ANSI N18.7-1976, Administrative Controls for Nuclear Power Plants.

## 13.4.1 ONSITE REVIEW

Qualified individuals from the station supervisory staff are assigned to review procedures, procedure changes, Technical Specifications changes, and plant modifications involving nuclear safety. These individuals are previously designated to perform these reviews. The final approval of the above reviews is by the Station Manager or other senior station management. In addition, for each review conducted, a determination is made as to whether or not additional cross-disciplinary review is necessary. If concluded that it is necessary, the additional review would be performed by the appropriate designated station review personnel.

The Site Vice President appoints a Plant Operations Review Committee (PORC) to review selected nuclear safety related issues. The PORC is composed of specified senior members of the site management team most responsible for the safe and reliable operation of the station. The PORC also reviews the effectiveness of corrective actions taken for specified reportable events.

In general, any issue that has the potential to significantly impact safe and reliable nuclear operations, and may benefit from a cross disciplinary review at the site management level, is within the scope of the PORC. The PORC has the following general responsibilities:

1. Reviews Justifications for Continued Operation.
2. Reviews situations where station structures, systems, or components are determined to be operable, but degraded, and the resulting compensatory actions. On a selected basis, reviews operability determinations that have resulted in the conclusion that station structures, systems, or components are fully operable.
3. Reviews pre-job briefings and management oversight plans.
4. Reviews reactor restart decisions.
5. Reviews independent review team outage assessment results and resulting contingency plans.
6. Reviews proposed Technical Specifications/License amendments.
7. Reviews Technical Specifications interpretation proposals.
8. On a selected basis, reviews reportable events documentation such as Licensee Event Reports, NRC Violation Responses, Station Reports, and reports of unplanned onsite releases of radioactive material to the environs.
9. On a selected basis, reviews evaluations performed pursuant to 10CFR50.59.

Incident investigation including LERs and special reviews are performed by the SRG or other designated qualified individuals.

3 3. A source of pertinent information, forms, and data to ensure prompt actions are taken and that
3 proper notifications and communications are carried out.

3 4. A record of the completed actions.

3 5. The mechanism by which emergency preparedness will be maintained at all times.

## 13.3 EMERGENCY PLANNING

3 The Emergency Program for the Duke Power Company's Oconee Nuclear Site consists of the Oconee
3 Nuclear Site Emergency Plan and related implementing procedures. Also included are related radiological
3 emergency plans and procedures of state and local governments. The purpose of these plans is to provide
3 protection of plant personnel and the general public and to prevent or mitigate property damage that
3 could result from an emergency at the Oconee Nuclear Site. The combined emergency preparedness
3 programs have the following objectives:

3 1. Effective coordination of emergency activities among all organizations having a response role.

3 2. Early warning and clear instructions to the population-at-risk in the event of a serious radiological
3 emergency.

3 3. Continued assessment of actual or potential consequences both on-site and off-site.

3 4. Effective and timely implementation of emergency measures.

3 5. Continued maintenance of an adequate state of emergency preparedness.

3 The Emergency Plan has been prepared in accordance with Section 50.47 and Appendix E of 10CFR Part
3 50. The plan shall be implemented whenever an emergency situation is indicated. Radiological
3 emergencies can vary in severity from the occurrence of an abnormal event, such as a minor fire with no
3 radiological health consequences, to nuclear accidents having substantial onsite and/or offsite
3 consequences. In addition to emergencies involving a release of radioactive materials, events such as
3 security threats or breaches, fires, electrical system disturbances, and natural phenomena that have the
3 potential for involving radioactive materials are included in the plans. The plan contains adequate
3 flexibility for dealing with any type of emergency that might occur.

3 The activities and responsibilities of outside agencies providing an emergency response role are detailed in
3 the State of South Carolina emergency plans and the emergency plans for Oconee and Pickens Counties.

3 The emergency response resources available to respond to an emergency consist of the following: 1. ONS
3 Site Personnel, 2. Duke Power corporate headquarters personnel, 3. Other Duke Power nuclear station
3 personnel, and, in the longer term, federal emergency response organizations (e.g. NRC, DOE, FEMA).
3 The first line of defense in responding to an emergency lies with the normal operating shift on duty when
3 the emergency begins. Therefore, members of the Oconee staff are assigned emergency response roles that
3 are to be assumed whenever an emergency is declared. The overall management of the emergency is
3 initially performed by the Shift Operations Manager until he/she is relieved by the Station Manager. In
3 the event of an emergency, he serves as the Emergency Coordinator. Onsite personnel have preassigned
3 roles to support the Emergency Coordinator and to implement his directives.

3 Special provisions have been made to assure that ample space and proper equipment are available to
3 effectively respond to the full range of possible emergencies. The emergency facilities available include the
3 Oconee Control Room, Operational Support Center, Technical Support Center, Joint Information
3 Center, and the Emergency Operations Facility. These facilities are described in the site emergency plan.

3 Emergency plan implementing procedures define the specific actions to be followed in order to recognize,
3 assess, and correct an emergency condition and to mitigate its consequences. Procedures to implement the
3 Plan provide the following information:

3 1. Specific instructions to the plant operating staff for the implementation of the Plan.

3 2. Specific authorities and responsibilities of plant operating personnel.

The performance and competency of Licensed Reactor Operators and Senior Reactor Operators is evaluated as described in the Duke Power Company-Employee Training and Qualification System Standard No. 2306.0 "Periodic Training Licensed Operator Requalification" dated effective March 1, 1987.

## 13.2.5 TRAINING AND QUALIFICATIONS DOCUMENTATION

5 Records are maintained on each employee's participation in training activities. It is the Site Training
5 Manager's responsibility to ensure training records are accurate and retrievable.

1 Records shall be retained according to the requirements established by Duke Power Company Employee
1 Training and Qualification System Standard No. 1001, 'Training Records Retention", and No. 1002, "OJT
1 and Qualifications Records Management".

Documentation associated with Licensed Operator training and Requalification is maintained according to the requirements established by the Duke Power Company-Employee Training and Qualification System Standards No. 2303.0 "License Preparatory Reactor Operator Program," No. 2304.0 "License Preparatory Senior Reactor Operator Program," and No. 2306.0 "Periodic Training Licensed Operator
5 Requalification."

1

## 13.2.3 OPERATOR LICENSE TRAINING

Duke Power Company's reactor operator and senior reactor operator training and requalification programs are based upon "a systematic approach to training" as defined by 10CFR55.4. These training programs were initially accredited by the Institute of Nuclear Power Operations and the National Nuclear Accrediting Board on August 17, 1983. They received accreditation renewal January 28, 1988 and February, 1992. Pursuant to 10CFR55.31 (a)(4), 10CFR55.59 (c), and Generic Letter 87-07 certification of these training programs' accreditation has been made to the NRC.

### 13.2.3.1 Operations Initial Training

The initial training program for operators is outlined in the Duke Power Company-Employee Training and Qualifications System Standard 2301.0 "Operations Training and Qualifications Overview." This initial training program provides the trainee with the necessary concepts of nuclear power plant systems, plant operation, mathematics, physics, thermodynamics, fluid flow, nuclear physics, radiation protection and instrumentation and control.

### 13.2.3.2 Operator License Training

The training for reactor operator and senior reactor operator replacement is based upon a "systematic approach to training" and is described in the Duke Power Company - Employee Training and Qualifications System Standards 2303.0 "License Preparatory Reactor Operator Program" and 2304.0 "License Preparatory Senior Reactor Operator Program."

### 13.2.3.3 Licensed Operator Requalification Training

Licensed operator requalification training is designed based upon "a systematic approach to training" to maintain and demonstrate continued competence of all licensed operators. The training is described in the Duke Power Company - Employee Training and Qualifications System Standard 2306.0 "Periodic Training Licensed Operator Requalification."

## 13.2.4 TRAINING PROGRAM EVALUATION

Training and qualifications activities are monitored by the Site Training Division. Regulatory Audits, NAID, Division audits site Employee Training and Qualification System. Trainees and vendors may provide input concerning training program effectiveness. Methods utilized to obtain this information may be surveys, questionnaires, performance appraisals, staff evaluation, overall training program effectiveness evaluation instruments, etc. Frequently conducted classes are not evaluated each time; however, they are evaluated at a frequency sufficient to ensure program effectiveness. Evaluation information may be collected through:

- verification of program objectives as related to job duties for which intended;
- testing to determine student accomplishment of these objectives;
- student evaluation of the instruction;
- instructor evaluations of the students;
- supervisor's evaluation of trainee performance on the job, following the training;
- supervisor's evaluation of the instructor; or
- periodic working (review) group evaluation.

designed to supplement and compliment training received through formal classroom, laboratory, and/or simulator training. The objective of the program is to assure the trainee's ability to perform job tasks as described in the task descriptions and the Training and Qualification Guides.

#### 13.2.2.2.3 Continuing Training

1 Continuing Training is any training not provided as Initial Qualification and Basic Training or training which maintains and improves job-related knowledge and skills such as the following:

a. Plant Systems and Component Changes

b. OJT/Qualification Program Retraining/Requalification

c. Procedure and Directive Changes

d. Operating Experience Program Documents Review to include Industry and In-House Operating Experiences

e. Continuing Training required by Regulation (Emergency Plan Training, etc.)

f. General Employee, Special, Administrative, Vendor, and/or Advanced Training topics supporting tasks.

g. Training identified to resolve deficiencies (task-based) or to reinforce seldom used knowledge and skills

h. Refresher training on initial training topics

i. Structured pre-job instruction, mock-up training, walk-throughs, etc.

5 "Requalification" is a term used in the Operations Training Programs. While requalification training and
5 continuing training may share some similarity in definition, requalification-type training is more clearly
5 associated with the well defined and structured topical areas which are periodically re-taught to the
5 Operations crews to ensure they maintain operating proficiency.

Continuing Training may consist of formal and informal components. Each Section or Division's
5 Continuing Training Program is developed using a systematic approach that includes job performance
5 information from a job and task analysis, and safe operation, as the basis for determining the content of continuing training. Continuing training may be offered, as needed, on any of the topics or programs listed in Section 13.2.2.2.3, "Continuing Training."

Once the objectives for Continuing Training have been established, the methods for conducting the
2 training may vary. The method selected should provide clear evidence of objective accomplishment and consistency in delivery.

### 13.2.2.3 Employee Development and Management/Supervisory Training

Training that falls outside of the scope of Technical Training and General Employee Training is considered to be either Employee Development or Management/Supervisory Training.

Employee Development or Management/Supervisory Training may consist of various classes for different
1 management personnel levels. An individual's training and development will depend on his/her position description and nomination by management.

5. Chemistry

a. Power Plant Fundamentals

| | | Program Component | Setting | Content |
|---|---|---|---|---|
| 1 | b. | Station Familiarization | In-Plant | Orientation to plant layout |
| 1 | c. | Initial Plant Systems | In-Plant | Structured Orientation to plant systems, their function, and physical layout in plant |
| 1 | d. | Fundamentals (Chemistry) | Classroom - Laboratory | Topics may include:<br>-Administration<br>-Chemical Principles<br>-Systems<br>-Process Instrumentation<br>-Radioactive Waste System<br>-Laboratory |
| 1 | e. | On-The-Job Training | In-plant | Completion of Power and Radwaste Training and Qualification (T&Q) Guides |

1 6. Engineering Support Initial Training

This program is accredited by the Institute of Nuclear Power Operations.

1 a. Station Orientation enables the Engineering Support Personnel at ONS to become familiar with plant layout, and roles and responsibilities of each section in the plant. Orientation is conducted
4 using a Task List that identifies training requirements/objectives for each area.

2 b. Fundamentals Training provides a basic understanding of how electricity is generated in a power
2 plant, the conversion and transfer of energy into the ultimate product, basic reactor theory,
2 chemistry, process control systems, and components.

2 c. Systems Training covers normal and emergency purposes, components, and flowpaths of
1 site-specific systems. The course includes specific modules covering Core Damage Mitigation that
1 meet the intent of INPO Guidelines.

4 d. Position Specific Guides

#### 13.2.2.2.2 On-the-Job Training and Qualification

On-the-job training is a systematic method of providing the required job related skills and knowledge for a
4 position. The Qualification process consists of three steps: 1) Training conducted in the work
4 environment/simulated work environment by qualified OJT trainers; 2) an independent evaluation; and 3)
4 a signature by the trainee's supervisor or a member of management awarding qualification. Applicable tasks and related procedures make up the OJT/qualifications program for each technical area which is

4. Radiation Protection

a. Power Plant Fundamentals

| | | Program Component | Setting | Content |
|---|---|---|---|---|
| 1 | | | | |
| 1 | b. | Station Familiarization | In-Plant | Orientation to plant layout |
| 1 | c. | Initial Plant Systems | In-Plant | Structured Orientation to plant systems, their function, and physical layout in plant |
| 1 | d. | Fundamentals of Radiation Protection | Classroom - Laboratory | Topics may include:<br>-Administration<br>-RP Theory<br>-RP Applications |
| 1 | e. | On-The-Job Training | In-Plant | Completion of Training and Qualification (T&Q) Guides |

3. Performance

a. Power Plant Fundamentals

| | | Program Component | Setting | Content |
|---|---|---|---|---|
| 1 | | | | |
| 5 | b. | Station Familiarization/ Initial Plant Systems | In Plant | Orientation to plant systems function, layout |
| 1 | c. | Fundamentals (Basic Performance) | Classroom - Laboratory | Topics may include:<br>-Thermal Science<br>-Component Principles |
| 1 | | | | -Inservice Testing<br>-Leak Rate Testing<br>-Basic Electrical<br>-Pre-Lab<br>-Instrumentation and Controls Measuring Methods |
| 1 | | | | |
| 1 | d. | On-The-Job Training | In Plant | Completion of Training and Qualification (T&Q) Guides |

2. Instrument and Electrical

a. Power Plant Fundamentals

b. Plant Orientation Program

| | | Program Component | Setting | Content |
|---|---|---|---|---|
| 1 | | | | |
| 1 | 1) | Station Familiarization | In-Plant | Orientation to plant layout |
| 1 | 2) | Fundamentals (Basic Instrument/ Electrical) | Classroom - Laboratory | Topics may include:<br>-Basic Electricity<br>-Basic Electronics<br>-Operational Amplifiers<br>-Digital Electronics<br>-Micro-processors<br>-Soldering<br>-Instrumentation and Control Methods<br>-Process Control<br>-Process Control Applications<br>-Electrical<br>-Electrical Valve Actuator Maintenance<br>-Nuclear Instrumentation<br>-Nuclear Safety Systems<br>-Nuclear Non-Safety Systems |
| 1 | c. | On-The-Job Training | In Plant | Completion of Training and Qualification (T&Q) Guides |

1. Mechanical Maintenance Initial Training

a. Power Plant Fundamentals

b. Plant Orientation Program

| | | Program Component | Setting | Content |
|---|---|---|---|---|
| 1 | | | | |
| 1 | 1) | Station Familiarization | In Plant | Orientation to plant layout |
| 1 | 2) | Fundamentals (Shop) | Classroom - Laboratory | Topics may include:<br>-Administration<br>-Maintenance Management<br>-Safety<br>-Dimensional Metrology<br>-Basic Metallurgy<br>-Fasteners<br>-Hand/Portable Power Tools<br>-Basic Machine Shop Practices<br>-Basic Piping/ Maintenance<br>-Rigging/Weight Handling<br>-Antifriction Bearing Maint.<br>-Drawings<br>-Industrial Hydraulics<br>-Mechanical Drives<br>-Gaskets, Packing, and Seals<br>-Valve Maintenance<br>-Welding<br>-Machining<br>-Pumps |
| 2 | | | | |
| 1 | c. | On-The-Job Training | In Plant | Completion of Training and Qualification (T&Q) Guides |

#### 13.2.2.1.1 Fire Brigade Training

1 The primary purpose of the Fire Brigade Training Program is to develop a group of site employees skilled in fire prevention, fire fighting techniques, first aid procedures, and emergency response. They are trained
1 and equipped to function as a team for the fighting of fires. The site fire brigade organization is intended to be self-sufficient with respect to fire fighting activities.

The Fire Brigade Training program provides for initial training of all new fire brigade members, quarterly classroom training and drills, annual practical training, and leadership training for fire brigade leaders.

### 13.2.2.2 Technical Training

1 Technical training is designed, developed and implemented to assist site employees in gaining an understanding of applicable fundamentals, procedures, and practices; and in developing manipulative skills necessary to perform assigned work in a competent manner. Technical training may consist of three segments:

Initial Training

On-the-job Training and Qualification

Continuing Training

#### 13.2.2.2.1 Initial Job Training

Initial job training is designed to provide knowledge of the fundamentals, basic principles, and procedures involved in work to which an employee is assigned.

2 This training may consist of, but is not limited to, live lectures, taped and filmed lectures, computer based
2 training, guided self-study, demonstrations, laboratories and workshops, on-the-job training, and where applicable, simulator training and/or training on a research reactor.

2 New employees or employees transferred from other division locations may be partially qualified by reason of previous applicable training or experience. The extent of further training for these employees is determined by applicable regulations, performance in review sessions, comprehensive examinations, or other techniques designed to identify the employee's present level of ability.

1 Initial job training and qualification programs are developed for Operations, I&E, Mechanical
2 Maintenance, Radiation Protection, and Chemistry non-exempt classifications. Engineering Support
2 position-specific training for newly hired or transferred engineers and other selected technical staff
2 personnel is provided to guide and document development of knowledges and skills needed for activities
2 that could have a significant effect on safe and reliable plant operation.

2 The training programs for technicians include Non-licensed Operator, Mechanical Maintenance,
4 Instrument and Electrical, Operations Test, Radiation Protection and Chemistry. These training
2 programs are accredited by the Institute of Nuclear Power Operations. The Basic Training program is
2 typically divided into three modules: Power Plant Fundamentals, Station Familiarization/Initial Plant Systems, and Fundamentals (Section Specific).

2 All technician training program attendees share the Power Plant Fundamentals module. This module includes an introduction, mathematics, physical science, manuals and publications, systems, and components. A brief description of the rest of the modules in the Initial Training Program for
4 Mechanical Maintenance, Instrument and Electrical, Operations Test, Radiation Protection and
2 Chemistry is as follows:

- Regulatory Guide 8.8 "Information Relevant to maintaining Occupational Radiation Exposures as Low as Reasonably Achievable (Nuclear Power Reactor)"
- Regulatory Guide 8.13, "Instructions Concerning Prenatal Radiation Exposure"
- NUREG-0737
- 10 CFR Part 20, "Standards for Protection Against Radiation"

## 13.2.2 PROGRAM DESCRIPTION

Station assigned personnel may be trained and qualified through participation in prescribed parts of the Employee Training and Qualification System which consists of the following:

General Employee Training

Technical Training

Employee/Professional Development Training

### 13.2.2.1 General Employee Training

General Employee Training (GET) encompasses those general administrative, safety, emergency and control procedures established by site management and applicable regulations. A summary description of plant systems and equipment is provided. All persons under the supervision of site management must participate in General Employee Training; however, certain station support personnel, depending on their normal work assignment, may not participate in all topics. Certain portions of General Employee Training may be included in an employee orientation program. Temporary maintenance and service personnel shall receive General Employee Training to the extent necessary to assure safe execution of their duties.

All persons regularly employed at the nuclear power plant and under the supervision of site management receive training in the following areas commensurate with the level of knowledge required for their job duties.

a. General administrative control and quality assurance policies and procedures

b. Plant systems and equipment

c. Radiological safety including the use of protective clothing and equipment

d. Industrial health, safety and first aid

e. Emergency plan and procedures

f. Station security program and procedures

g. Fire protection program and procedures

h. New Employee Orientation

i. Environmental compliance overview

j. Fitness for Duty

k. Respiratory Protection and Fit Testing

Continuing training is conducted in these areas as necessary to maintain employee proficiency.

# 13.2 TRAINING

## 13.2.1 GENERAL PROGRAM DESCRIPTION

The principal objective of the Duke Power Company Employee Training and Qualification System
2 (ETQS) is to assure job proficiency of all station personnel involved in safety related work. An effective
2 training and qualification system is designed to accommodate future growth and meet commitments to and comply with applicable established regulations and accreditation standards.

2 Qualification is indicated by successful completion of prescribed training and demonstration of the ability
2 to perform assigned work or tasks competently. Where required, maintaining a current and valid license
2 issued by the regulating agency establishes the requirements.

5 The Oconee Site Training Manager has overall responsibility for the administration of the Employee
1 Training and Qualification System (ETQS). The Vice President, Oconee site, is responsible for the
1 quality of work performed by individuals at the nuclear site. Line Management is responsible for the timely and effective development of assigned personnel.

1 Training is analyzed, designed, developed, implemented, and evaluated according to a systematic approach to training. Employees are provided with formal training to establish the knowledge foundation and on-the-job training to develop work performance skills. Continuing training is provided, as required, to maintain proficiency in these knowledge and skill components and to provide further employee development.

The Employee Training and Qualification System is designed to prepare initial and replacement station personnel for safe, reliable and efficient operation of the nuclear facility. The program is intended to meet
2 or exceed INPO accreditation standards and Nuclear Regulatory Commission requirements.

Appropriate training for personnel of various training and experience backgrounds is provided. The level at which an employee initially enters the training and qualifications system for the particular area is determined by an evaluation of the employee's past experience and level of ability.

### 13.2.1.1 Regulatory Requirements

The applicable portions of the NRC regulations, regulatory guides, and reports listed below will be used in providing guidance in plant staffing and training.

1 • 10CFR PART 50 "Domestic Licensing of Production and Utilization Facilities"
- 10CFR PART 55 "Operators' Licenses" including Appendix A
- 10CFR PART 19 "Notices, Instructions and Reports to Workers; Inspections"
- Regulatory Guide 1.8 "Personnel Selection and Training"
- NRC "Operator Licensing Guide," NUREG-0094, July 1976
- "Utility Staffing and Training for Nuclear Power," WASH-1130, USAEC Revised 1973
- NUREG-0654
- Regulatory Guide 8.2 "Guide for Administrative Practices in Radiation Monitoring"

5 A Shift Work Manager shall have a minimum of a Bachelor's degree in an engineering or science discipline, or a Professional Engineer's license, and four years of nuclear power plant experience. A Shift
5 Work Manager shall hold a Senior Reactor Operator's license.

4 **(k) Other Supervisors Required to Hold an NRC License**

2 Members of the station supervisory staff other than those identified in Section 13.1.3.1(a) through
2 13.1.3.1(j) preceeding who are responsible for directing the actions of operators, technicians or repairmen (e.g., intermediate and first line supervisors), and who are required to hold an NRC license, shall have a high school diploma, or equivalent, and a minimum of four (4) years of responsible nuclear or fossil station experience, of which a minimum of one (1) year shall be nuclear station experience. A maximum of two (2) years of the remaining three (3) years of experience may be fulfilled by academic or related technical training on a one-for-one time basis.

4 **(l) Other Supervisors Not Required to Hold an NRC License**

2 Members of the station supervisory staff other than those identified in Section 13.1.3.1(a) through
2 13.1.3.1(j) preceeding who are responsible for directing the actions of operators, technicians or repairmen (e.g., intermediate and first line supervisors), and who are not required to hold an NRC license, shall have a high school diploma, or equivalent, and a minimum of four (4) years of experience in the craft or discipline supervised.

4 **(m) Operators**

Operators to be licensed by the Nuclear Regulatory Commission shall have a high school diploma, or equivalent, and two (2) years of nuclear or fossil station experience, of which a minimum of one (1) year shall be nuclear station experience. In order to be acceptable for full responsibility in a job, they shall hold a Reactor Operator license.

Operators, whether or not they are to be licensed by the Nuclear Regulatory Commission, should have a high school diploma, or equivalent, and should possess a high degree of manual dexterity and mature judgment. Selection interviews and examinations, bearing a significant relationship to job performance, should be used for operators to aid in determining an individual's ability to progress to high levels of responsibility and for eventual Nuclear Regulatory Commission licensing.

4 **(n) Technicians**

Technicians in responsible positions (i.e., individuals who direct the activities of others, but who are not supervisors) shall have a minimum of two years of experience in their specialty. These personnel should have a minimum of one year of related technical training in addition to their experience.

4 **(o) Maintenance Personnel**

Maintenance personnel in responsible positions (i.e., individuals who direct the activities of others, but who are not supervisors) shall have a minimum of three years of experience in one or more crafts. They should possess a high degree of manual dexterity and ability, and should be capable of learning and applying basis skills in maintenance operations.

1

1 **(e) Shift Operations Manager**

1 See Oconee Technical Specification 6.1.1.4.

1

1 **(f) Chemistry Manager**

The Chemistry Manager shall have a minimum of five years of experience in chemistry, of which a minimum of one year shall be in radiochemistry. A minimum of two years of this five years of experience should be related technical training. A maximum of four years of this five years of experience may be fulfilled by academic or related technical training.

1 **(g) Radiation Protection Manager**

See Section 13.1.3, "Qualifications of Station Personnel" on page 13-7.

2 **(h) Regulatory Compliance Manager**

2 The Regulatory Compliance Manager shall have a minimum of five years of technical experience, of which a minimum of one year shall be nuclear experience. A maximum of four years of this five years experience may be fulfilled by related technical or academic training.

1

4 **(i) Maintenance Superintendent**

4 The Mechanical Superintendent and the I&E Superintendent roles are combined under the title
4 Maintenance Superintendent as long as the Maintenance Superintendent meets the required qualifications
4 under both subtitles.

4 1. Mechanical Superintendent

The Mechanical Superintendent shall have a minimum of seven years of responsible nuclear or fossil station experience, or applicable industrial experience, of which a minimum of one year shall be nuclear station experience. A maximum of two years of the remaining six years of experience may be fulfilled by satisfactory completion of academic or related technical training on a one-for-one time basis. The Mechanical Superintendent should also have non-destructive testing familiarity, craft knowledge, and an understanding of electrical, pressure vessel and piping codes.

4 2. Instrument and Electrical Superintendent

The Instrument and Electrical Superintendent shall have a minimum of five years of experience in instrumentation and control of which a minimum of six months shall be in nuclear instrumentation and control. A minimum of two years of this five years of experience should be fulfilled by academic or related technical training. A maximum of four years of this five years of experience may be fulfilled by academic or related technical training.

1

5 **(j) Shift Work Manager**

An individual who temporarily replaces the RPM shall have a bachelor's degree in a science or engineering subject or the equivalent in experience and shall have at least two years experience, one of which shall be nuclear power plant experience. Six months experience shall be on site.

Replacement personnel for positions in the nuclear stations are fully trained and qualified to fill their
1 appointed positions. Qualifications of key site personnel are available for inspection on site.

### 13.1.3.1 Minimum Qualification Requirements

The minimum qualification requirements for station personnel are outlined in the succeeding paragraphs.

1 **(a) Station Manager**

The Station Manager shall have a minimum of ten years of responsible nuclear or fossil station experience, of which a minimum of three years shall be nuclear station experience. A maximum of four years of the remaining seven years of experience may be fulfilled by academic training on a one-for-one time basis. To be acceptable, this academic training shall be in an engineering or scientific field generally associated with power production. The Station Manager shall have acquired the experience and training normally required for examination by the NRC for a Senior Reactor Operator license, whether or not the examination is taken.

2 The qualification requirements described above may be reduced in accordance with ANSI/ANS-3.1-1978
2 which states:

2 "In an organization which includes one or more persons who are designated as principal alternates
2 for the plant manager and who meet the nuclear power plant experience and training requirements
2 established for the plant manager, the requirements of the plant manager may be reduced, such
2 that only one of his ten years of experience need to be nuclear power plant experience and he
2 need not be eligible for the NRC examination."

1 **(b) Operations Superintendent**

1 Refer to Oconee Technical Specification 6.1.1.4.

1 **(c) Safety Assurance Manager**

1 The Safety Assurance Manager should have a minimum of eight years of responsible nuclear or fossil station experience, of which a minimum of one year shall be nuclear station experience. A maximum of four years of the remaining seven years of experience should be fulfilled by satisfactory completion of academic training.

1

1 **(d) Work Control Superintendent**

1 The Work Control Superintendent shall have a minimum of seven (7) years of responsible nuclear or fossil station experience, or applicable industrial experience, of which a minimum of one (1) year shall be nuclear station experience. A maximum of two (2) years of the remaining six (6) years of experience may be fulfilled by satisfactory completion of academic or related technical training on a one-for-one time
1 basis. The Work Control Superintendent should also have a familiarity with the scheduling and project management techniques used at the Duke nuclear stations, management skills, and an understanding of the Duke administrative policies and procedures.

### 13.1.2.4 Nuclear Services Organization

The Nuclear Services organization provides corporate oversite and specific technical services to all Duke Nuclear sites. This organization is headed by the General Manager, Nuclear Services. The organization chart is shown on Figure 13-4. The function and responsibilities are described in the succeeding paragraphs.

1. Nuclear Engineering

   This organization has the responsibility for safety analysis, probabilistic risk assessments, reactor core design, out-of-core fuel management, core thermal hydraulic design, fuel fabrication, and failed fuel analysis.

2. Engineering/Maintenance Support

   This organization provides operating support to all Duke nuclear sites with an emphasis on generic programs and the promotion of consistancy. Support is provided for the areas of Civil and Electrical Engineering, Mechanical and Instrument & Electrical (I&E) Maintenance, Fire Protection, and Mechanical and Electrical Materials Procurement Engineering.

3. Safety Assurance

   This organization provides oversight and support to assure safe nuclear station operation and compliance with regulatory requirements. Work units include Nuclear Licensing Services, Operational Event Analysis, Emergency Planning, and Environmental Licensing Services.

4. Operations, Performance, and Automation Support

   This organization provides services and leadership which support, and supplement station personnel efforts as they relate to the operation of the station. Primary roles include support of long term development of new programs, policies and technology in the areas of reliability improvement, generic operating issues, system and component performance, and automation projects; as well as support for the development of projects that are not feasible or economical endeavors for the individual station.

5. Nuclear Technical Services

   This organization is responsible for providing oversight and technical support in the areas of chemistry, radiation protection, radwaste and radioactive materials control. It also determines and maintains all dosimetry records for Duke Power personnel.

## 13.1.3 QUALIFICATIONS OF STATION PERSONNEL

The qualifications of personnel in the site organization are in accordance with Section 4 of ANSI 3.1-1978, "Selection and Training of Nuclear Power Plant Personnel," and are in accordance with Regulatory Guide 1.8 (Rev. 1) with the exception of those for the Superintendent of Operations, the Shift Operating Manager, and the Radiation Protection Manager (RPM) in Part C of Regulatory Guide 1.8.

The Radiation Protection Manager shall have a bachelor's degree in a science or engineering subject or the equivalent in experience, including some formal training in radiation protection, and shall have at least five years of professional experience in applied radiation protection of which three years shall be in applied radiation protection work in one of Duke Power Company's nuclear stations. A qualified individual who does not meet the above requirements, but who has demonstrated the required radiation protection management capabilities and has professional experience in applied radiation protection work at one of Duke Power Company's multi-unit nuclear stations, may be appointed to the position of Radiation Protection Manager by the Station Manager, based on the recommendations of the Nuclear Services Technical Manager of Radiation Protection and as approved by the site Vice President.

1 The Work Control Superintendent manages the station's efforts to support Oconee Nuclear Station's
1 operational and outage activities through the coordination, development, shift and outage management of
1 a timely and effective integrated station schedule.

5 **(l) Shift Work Manager**

5 The Shift Work Manager is responsible for plant accident assessment functions during transients and
1 operations assessment functions during normal operations.

4 **(m) Safety Assurance Manager**

1 The Safety Assurance Manager is responsible for directing the activities of Regulatory and Environmental
5 Compliance, Safety Review, Emergency Preparedness, INPO Coordinator, and HPES.

4 **(n) Regulatory Compliance Manager**

2 The Regulatory Compliance Manager has responsibility for coordinating station interfaces with regulatory
1 agencies and for providing review of appropriate station technical matters.

5 **(o) Organizational Effectiveness Manager**

5 The Organizational Effectiveness Manager is responsible for coordination of site administrative functions
5 including clerical, personnel, safety, fire protection, security, and medical.

5 **(p) Training Manager**

5 The Site Training Manager is responsible for implementation and oversight of the training programs for
5 site personnel. The Site Training Division provides the analysis, design, development, implementation
5 and evaluation of Training and Qualifications programs in support of personnel performing work in the
5 nuclear station. Furthermore, the Site Training Division ensures station training programs meet or exceed
5 all facility licensing, FSAR, Nuclear Policy or regulatory requirements.

5 **(q) Commodities and Facilities Manager**

5 The Commodities and Facilities Manager is responsible for assuring that the work units provide quality
5 commodities, services, and facility support to their customers.

5 Commodities and Facilities provides acquisition, management, and maintenance services for parts,
5 supplies, tools, equipment, and commercial facilities required for the operation of the nuclear station.
5 Primary objectives are to provide these services and related commodities in a safe and economical manner.
5 Work units include Facilities and Equipment Maintenance, Regional Commodities and Facilities Support,
5 Facilities and Equipment Management, Inventory Management, Commodities/Services Management,
5 Customer Support, and Commodities/Facilities Technical Support.

1 ### 13.1.2.3 Shift Crew Composition

5 The operating shift crew consists of an Operations Shift Supervisor, a Shift Work Manager (who fulfills
5 the STA role), a Shift Supervisor for each unit, a Control Room Supervisor in each Control Room, and
5 appropriate licensed and nonlicensed operators. In addition, Radiation Protection, Chemistry,
Maintenance and I&E technicians are on site at all times when there is fuel in a reactor.

4 performed while he is on duty. The Operations Shift Manager on duty has both the authority and the
1 obligation to shut down a unit if, in his opinion, conditions warrant this action.

4 **(e) Shift Supervisor**

4 The Shift Supervisor assists the Operations Shift Manager in operation of the station on his assigned shift.
4 The Shift Supervisor on duty has both the authority and the obligation to shut down a unit if, in his
1 opinion, conditions warrant this action.

1 **(f) Reactor Operator**

1 A Reactor Operator is responsible for the actual operation of a Unit on his assigned shift. The Reactor
1 Operator has both the authority and obligation to shut down a unit if, in his opinion, conditions warrant
1 this action.

1 **(g) Nuclear Equipment Operator**

1 A Nuclear Equipment Operator is responsible for the operation of equipment outside of the Control
1 Room.

1 **(h) Radiation Protection Manager**

1 The Radiation Protection Manager has the responsibility for conducting the radiation protection program.
1 His duties include the training of personnel in use of equipment, control of radiation exposure of
1 personnel, continuous determination of the radiological status of the station, surveillance of radioactive
1 waste disposal operations, conducting the radiological environmental monitoring program and maintaining
1 all required records. He has direct access to the Station Manager in matters concerning any phase of
1 radiological protection. The Radiation Protection Manager also has direct support as required from the
5 Technical Manager of Radiation Protection in Nuclear Services and his staff.

1 **(i) Chemistry Manager**

1 The Chemistry Manager is responsible for overall chemistry and radiochemistry requirements, with special
1 emphasis on primary and secondary system water chemistry.

4 **(j) Maintenance Superintendent**

4 The Mechanical Superintendent and the I&E Superintendent roles are combined under the title
4 Maintenance Superintendent as long as the Maintenance Superintendent meets the required qualifications
4 under both subtitles.

4 1. Mechanical Superintendent

4 The Mechanical Superintendent has responsibility for maintenance of mechanical equipment.

4 2. I&E Superintendent

4 The I&E Superintendent has responsibility for maintenance of electrical equipment, instrumentation,
4 controls, and computers.

4 **(k) Work Control Superintendent**

1 ### 13.1.2.1 Nuclear Generation Department Organization

1 Duke's Nuclear Generation Department, headed by the Senior Vice President, Nuclear Generation, has
1 corporate responsibility for overall nuclear safety, as established by Technical Specifications. Reporting to
5 the Senior Vice President is a Vice President for each nuclear site, and Managers of the Nuclear
5 Engineering Division, Nuclear Assessment and Issues Division, Engineering Support Division, and Station
5 Support Division.

1 The Nuclear Generation Department Organization is shown on Figure 13-3.

1 ### 13.1.2.2 Nuclear Site

1 #### 13.1.2.2.1 Site Organization

1 The nuclear site organization centralizes the resources for safe and efficient nuclear plant operations under
1 a vice president at the nuclear site.

1 The Vice President of Oconee Nuclear site has the responsibility for overall plant nuclear safety as
1 established by Technical Specifications. The site staff is fully capable and equipped to handle all
1 situations involving safety of the station and public. The Nuclear site staff is shown on Figure 13-4.

1 As established by the Duke Quality Assurance Program Topical Report, Duke-1A, anyone involved in
1 quality activities in the Duke organization has the authority and responsibility to stop work if they
1 discover deficiencies in quality.

#### 13.1.2.2.2 Personnel Functions, Responsibilities and Authorities

The functions and responsibilities of key supervisory staff are described in the succeeding paragraphs.

**(a) Station Manager**

The Station Manager reports to the Vice President, Oconee Site and has direct responsibility for operating the station in a safe, reliable and efficient manner. He is responsible for protection of the station staff and the general public from radiation exposure and/or any other consequences of an accident at the station. He bears the responsibility for compliance with the facility operating license. The Station Manager or his designee has the authority to approve and issue Station Directives and procedures.

**(b) Operations Superintendent**

The Operations Superintendent has the responsibility for directing the actual day-to-day operation of the station. In the event of the absence of the Station Manager, the Operations Superintendent, if so designated, assumes the responsibilities and authority of the Station Manager.

1 **(c) Shift Operations Manager**

4 The Shift Operations Manager is responsible for the overall activities of all the on-shift licensed and
1 non-licensed operating personnel.

4 **(d) Operations Shift Manager**

4 An Operations Shift Manager is responsible for the overall operation of the station on his assigned shift.
4 He oversees the activities of the operators on his shift and is cognizant of all maintenance activity being

# 13.1 ORGANIZATIONAL STRUCTURE

## 13.1.1 CORPORATE ORGANIZATION

The corporate structure of Duke Power Company is shown in Figure 13-1 and Figure 13-2.

### 13.1.1.1 Corporate Functions, Responsibilities and Authorities

Duke Power Company has nearly 91 years of experience in the design, construction and operation of electric generating stations. As of 1994, Duke's total system capacity was approximately 18,000 MWe. Duke operated eight fossil stations with a 38% share of this total capacity, three nuclear steam-electric stations with a 60% share, and 27 hydroelectric stations, four pumped storage units, and combustion turbine and diesel peaking units accounting for the remaining 2% share.

Company involvement in nuclear power began in the early 1950's with various personnel receiving nuclear training. Selected personnel have been involved full time in nuclear projects since the mid-1950's. Duke participated in the Carolinas-Virginia Nuclear Power Associates (CVNPA), which resulted in a 17,000 kWe nuclear steam-electric unit at Parr, South Carolina. This unit, the Carolinas-Virginia Tube Reactor (CVTR), produced electricity over the period 1963 to 1967 as part of a five-year operating research program. Duke's three unit Oconee Nuclear Station began operation in 1973, the two unit McGuire Nuclear Station began operation in 1981, and two unit Catawba Nuclear Station began operation in 1984. As a result of these and other assignments, many personnel in the Duke organization have had prior nuclear experience as well as extensive experience in the power field.

Various departments within the Company have responsibility for design, construction, quality assurance and operation of each nuclear station. Duke contracts with a nuclear steam supply system (NSSS) vendor for the design and manufacture of the complete NSSS. The NSSS vendor also provides technical consultation in areas such as construction, testing, startup and initial fuel loading.

Duke's corporate functions, responsibilities and authorities for quality assurance are addressed in Topical Report DUKE-1A.

The Chairman of the Board and Chief Executive Officer has overall responsibility for corporate functions involving planning, design, construction and operation of the Company's generation, transmission, and distribution facilities, as well as other staff functions.

Line responsibilities relative to Nuclear Generation are delegated through the President and Chief Operating Officer, Power Generation Group, to the Senior Vice President, Nuclear Generation as shown in Figure 13-1 and Figure 13-2.

### 13.1.1.2 Organization for Design and Construction

Effective November 1, 1991, Duke reorganized to create the Power Generation Group, which includes the Nuclear Generation Department. Separate organizations for design and construction ceased to exist.

## 13.1.2 OPERATING ORGANIZATION

# CHAPTER 17. QUALITY ASSURANCE

The Duke Power Company Quality Assurance Program is presented in the Duke Power Company Topical Report, "Quality Assurance Program," DUKE-1A. The report is organized like and is generically used for Chapter 17, "Quality Assurance" - Quality Assurance of Duke's Safety Analysis Reports.

The Duke quality assurance program conforms to applicable regulatory requirements such as 10CFR 50,
1 Appendix B and to approved industry standards such as ANSI N45.2-1971 and ANSI N18.7-1976 and corresponding daughter standards, or to equivalent alternatives. The Duke Power quality assurance program also conforms to the regulatory position of the NRC Regulatory Guides listed in Table 17.0-1 of this report with the exception of the clarifications, modifications, and alternatives stated therein.

Section 17 describes the purpose of this report, provides definitions, and shows conformance to regulations, standards, and guides.

1

1 Section 17.3 describes the quality assurance program and organization for station operation.

1 The description in Section 17.3 follows the format of NUREG-0800, "Standard Review Plan for the
1 Review of Safety Analysis Reports for Nuclear Power Plants", except that the Duke Power Company
1 Quality Assurance Program is based on ANSI N18.7-1976 in lieu of ANSI/ASME NQA-1 and NQA-2. The topical is intended to be a comprehensive up-to-date description of Duke's quality assurance program for nuclear power plants.

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# TABLE OF CONTENTS

## 16.2 RELATION TO TECHNICAL SPECIFICATIONS

The Oconee Nuclear Station Selected Licensee Commitments Manual contains a listing of selected commitments for which Oconee Technical Specification 6.4 (Station Operating Procedures) requires written procedures to be established, implemented, and maintained.

**THIS IS THE LAST PAGE OF THE CHAPTER 16 TEXT PORTION.**

## 16.1 INTRODUCTION

The Oconee Nuclear Station selected licensee commitment program provides a single location where certain selected operational related commitments are controlled. While this program constitutes Chapter 16, "Selected Licensee Commitments" on page 16-1 of the Oconee FSAR, the contents of the program are maintained in a separate manual. This manual, "The Oconee Nuclear Station Selected Licensee Commitments Manual," facilitates the administration of this program, including more timely revision and addition of important commitments and user convenience.

The "Oconee Nuclear Station Selected Licensee Commitments Manual" contains commitments to control important plant equipment and operating conditions not controlled elsewhere. Issuance of the manual, as well as subsequent revisions, are done under the approval of the Station Manager.

# CHAPTER 16. SELECTED LICENSEE COMMITMENTS

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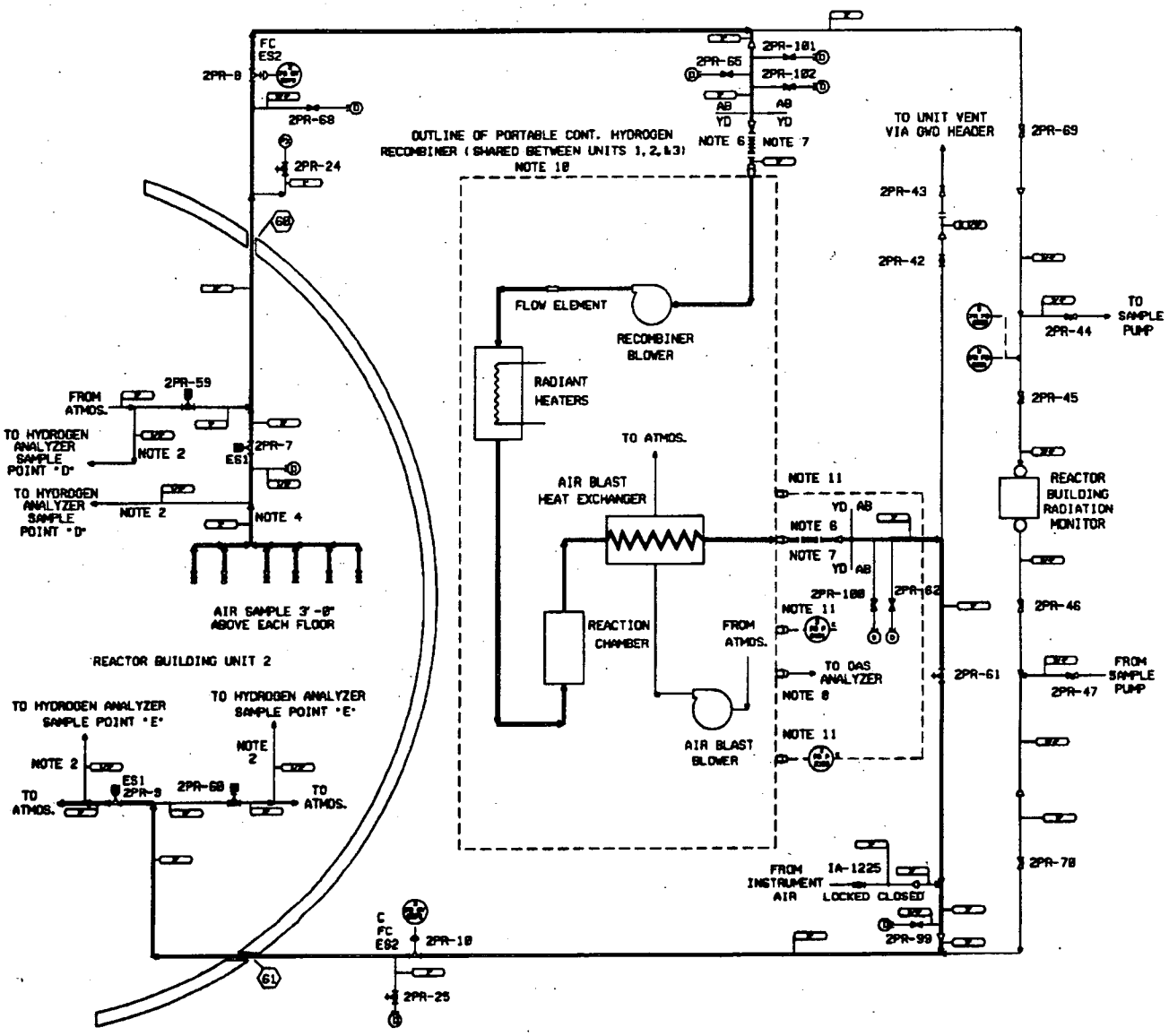


NOTES:

1. HYDROGEN RECOMBINER MODE REPRESENTED BY BOLD LINES.
2. REDUNDANT "SAFETY-RELATED" INSTRUMENT TAPS TO BE SEPARATED 10 INCHES MINIMUM.
3. FLOW ADJUSTMENT CONNECTION ON MONITORS.
4. CLASS BREAK 6" UPSTREAM OF BRANCH.
5. THE ORIGINAL ISSUE OF THIS DRAWING IS BASED ON OFD-101A-2.1, REV. 3.
6. PORTABLE PIPING, USED TO CONNECT RECOMBINER OR PURGE UNIT TO PIPING.
7. BLIND FLANGE NUTS ARE TACK WELDED TO PREVENT INADVERTANT UNMONITORED GASEOUS EFFLUENT RELEASE.
8. THIS CONNECTION IS NO LONGER USED FOR RB HYDROGEN PURGE SYSTEM.
10. FOR HYDROGEN PURGE UNIT SEE OFD-116C-1.1
11. THIS CONNECTION IS FOR THE HYDROGEN PURGE UNIT ONLY.

Figure 15-110.
Containment Hydrogen Recombiner System

5 Figure 15-69.
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5 Figure 15-70.
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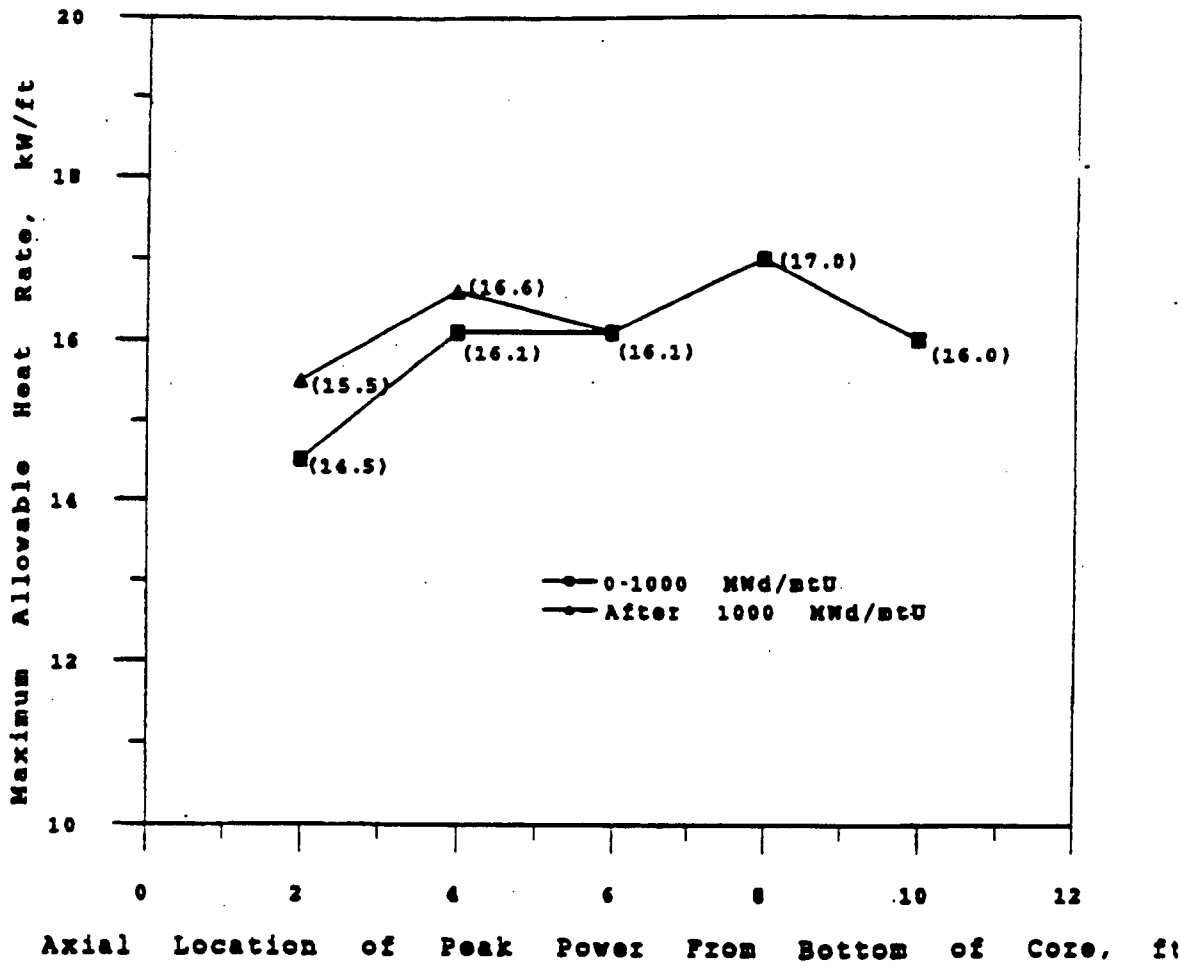


Figure 15-51.
LOCA - Allowable Linear Heat Rate vs Axial Core Elevation

Figure 15-52.
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Figure 15-53.
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Figure 15-54.
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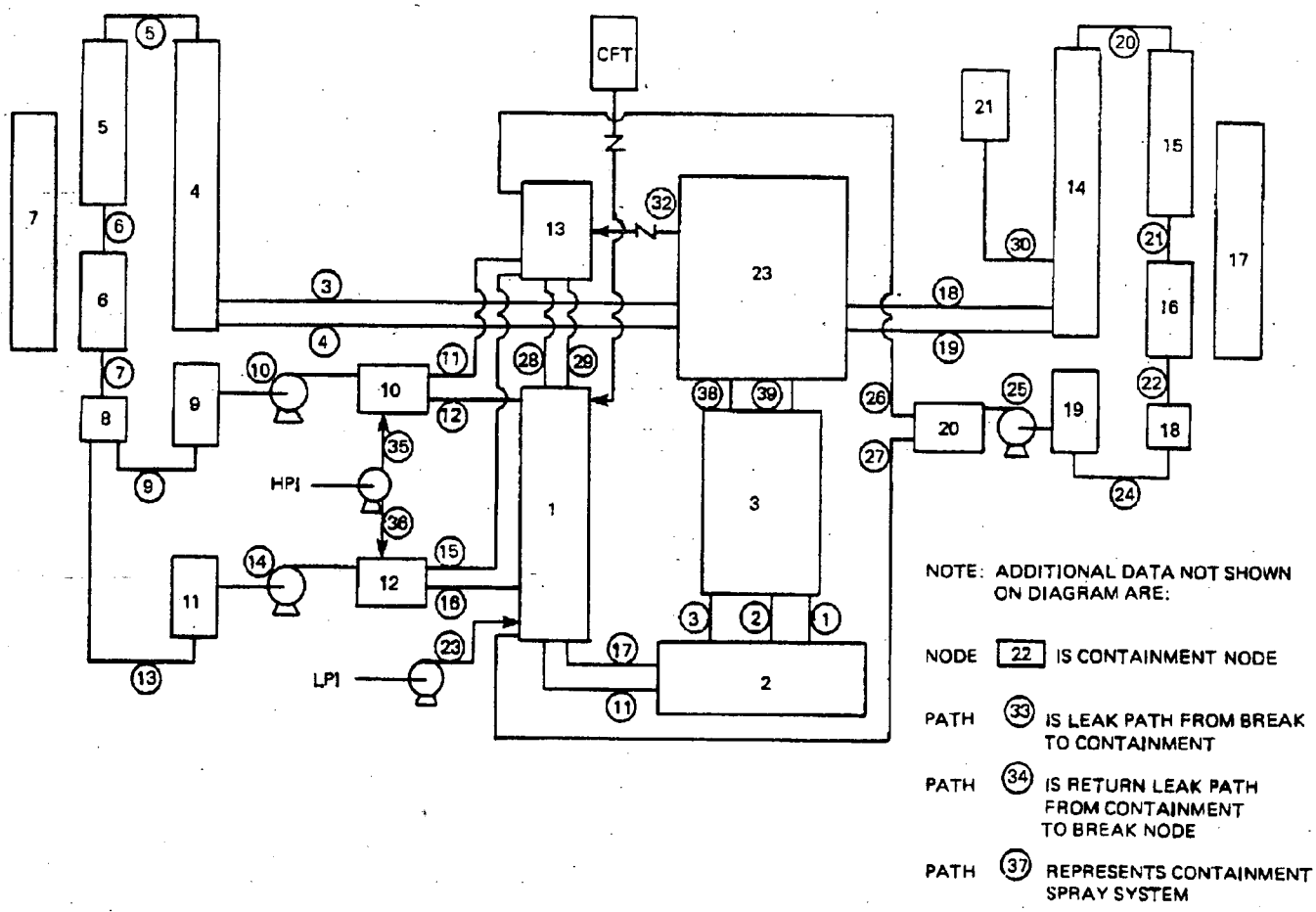


| NODE NO. | IDENTIFICATION |
|---|---|
| 1 | DOWNCOMER |
| 2 | LOWER PLENUM |
| 3 | CORE |
| 4,14 | HOT LEG PIPING |
| 5,15 | SG & UPPER HEAD |
| 6,16 | STEAM GENERATOR TUBES |
| 7,17 | SECONDARY SG |
| 8,18 | SG LOWER HEAD |
| 9,11,19 | COLD LEG PIPING |
| 10,12,20 | COLD LEG PIPING |
| 13 | UPPER DOWNCOMER |
| 21 | PRESSURIZER |
| 22 | CONTAINMENT |
| 23 | UPPER PLENUM |

| PATH NO. | IDENTIFICATION |
|---|---|
| 1,2 | CORE |
| 3,4,18,19 | HOT LEG PIPING |
| 5,20 | HOT LEG, UPPER |
| 6,21 | SG TUBES |
| 7,22 | SG LOWER HEAD |
| 8 | CORE BYPASS |
| 9,13,24 | COLD LEG PIPING |
| 10,14,25 | PUMPS |
| 11,12,15,16,26,27 | COLD LEG PIPING |
| 17,31 | DOWNCOMER |
| 23 | LPI |
| 28,29 | UPPER DOWNCOMER |
| 30 | PRESSURIZER |
| 32 | VENT VALVE |
| 33,34 | LEAK & RETURN PATH |
| 35,36 | HPI |
| 37 | CONTAINMENT SPRAYS |

**Figure 15-49.**
**LOCA - Craft2 Small Break System Nodalization**

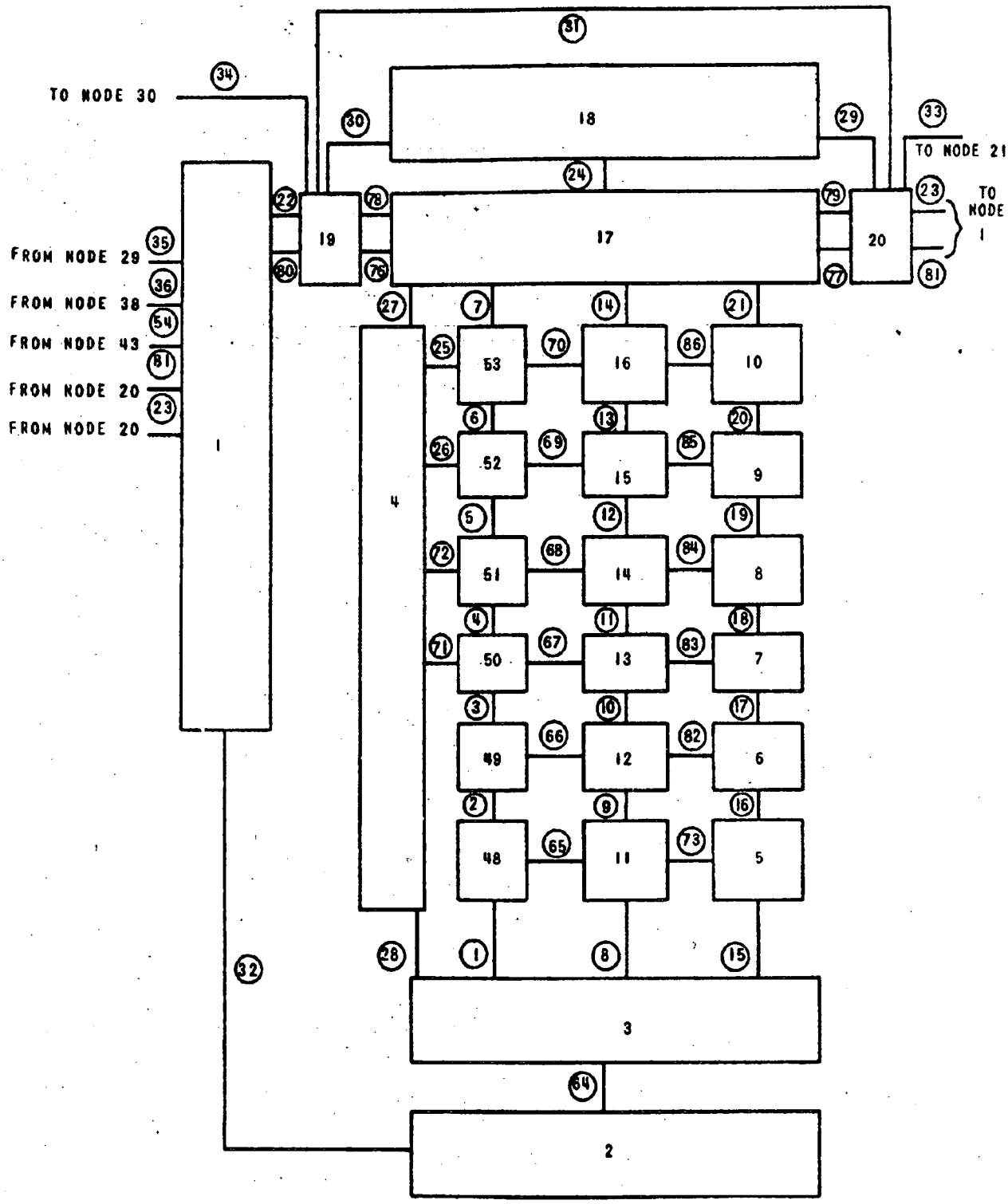


Figure 15-48.
LOCA - Craft2 Reactor Vessel Nodalization

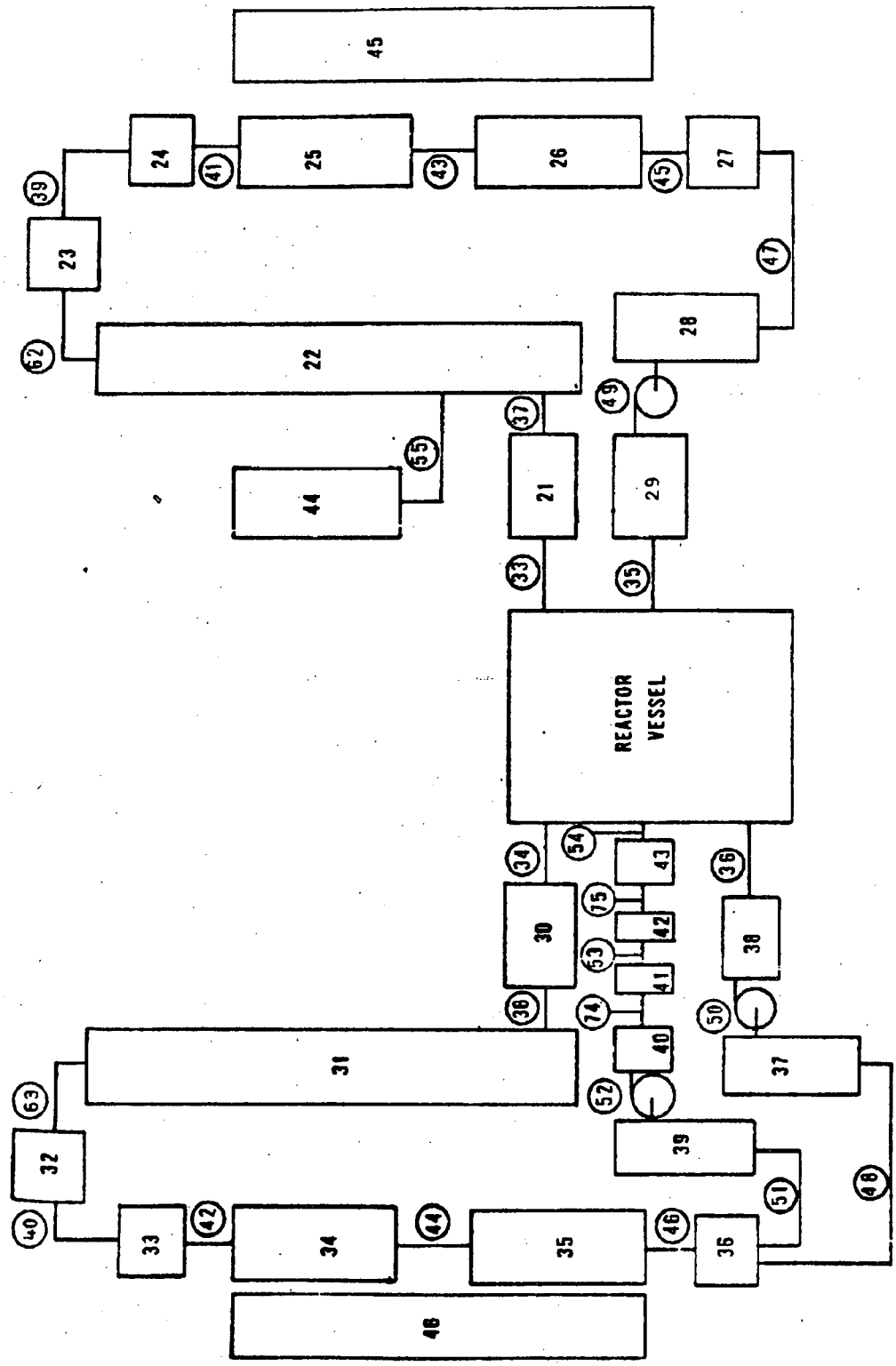


Figure 15-47.
LOCA - Craft2 System Nodalization

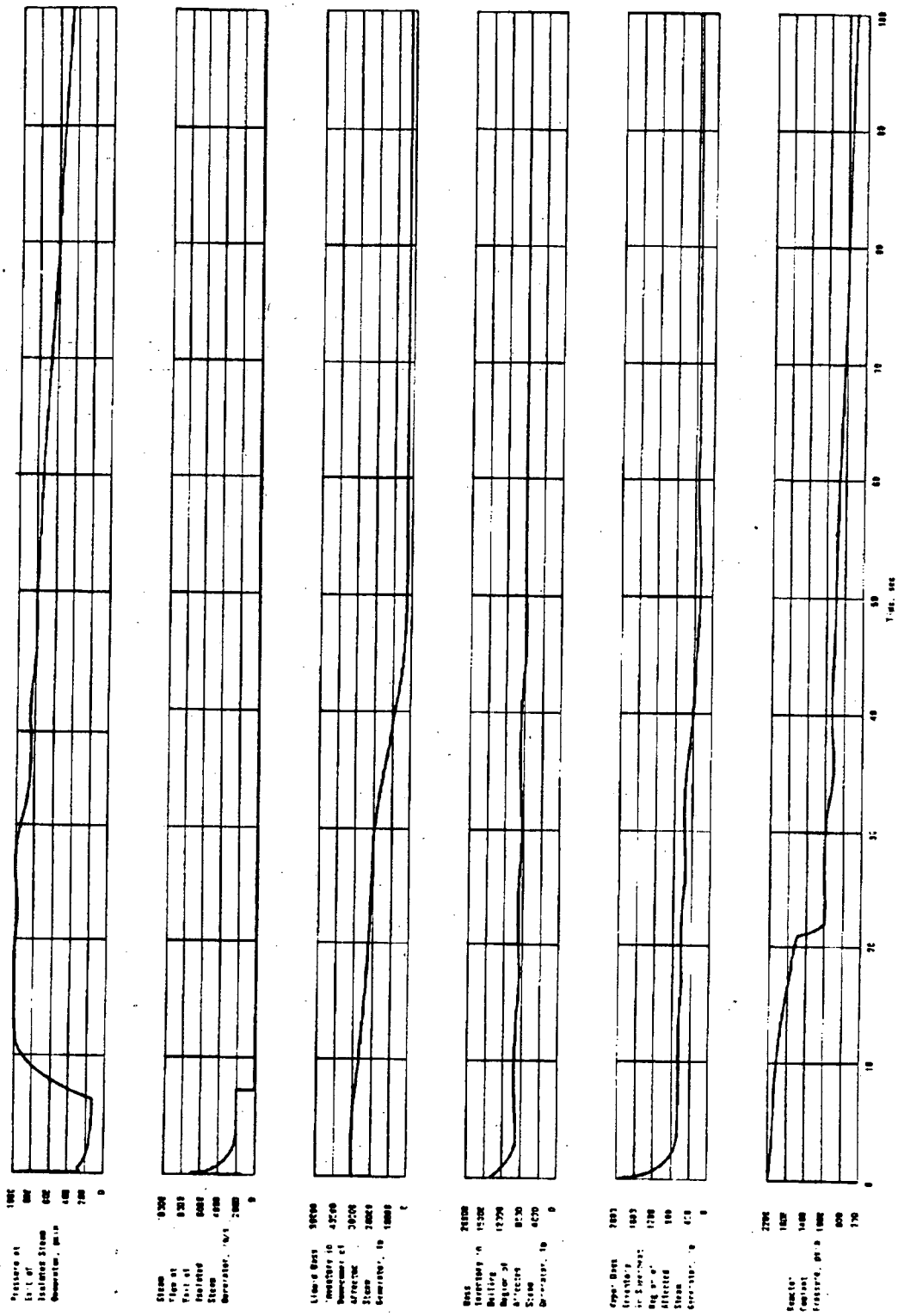

Figure 15-43.
Steam Line Break Accident - Without ICS and Operator Action

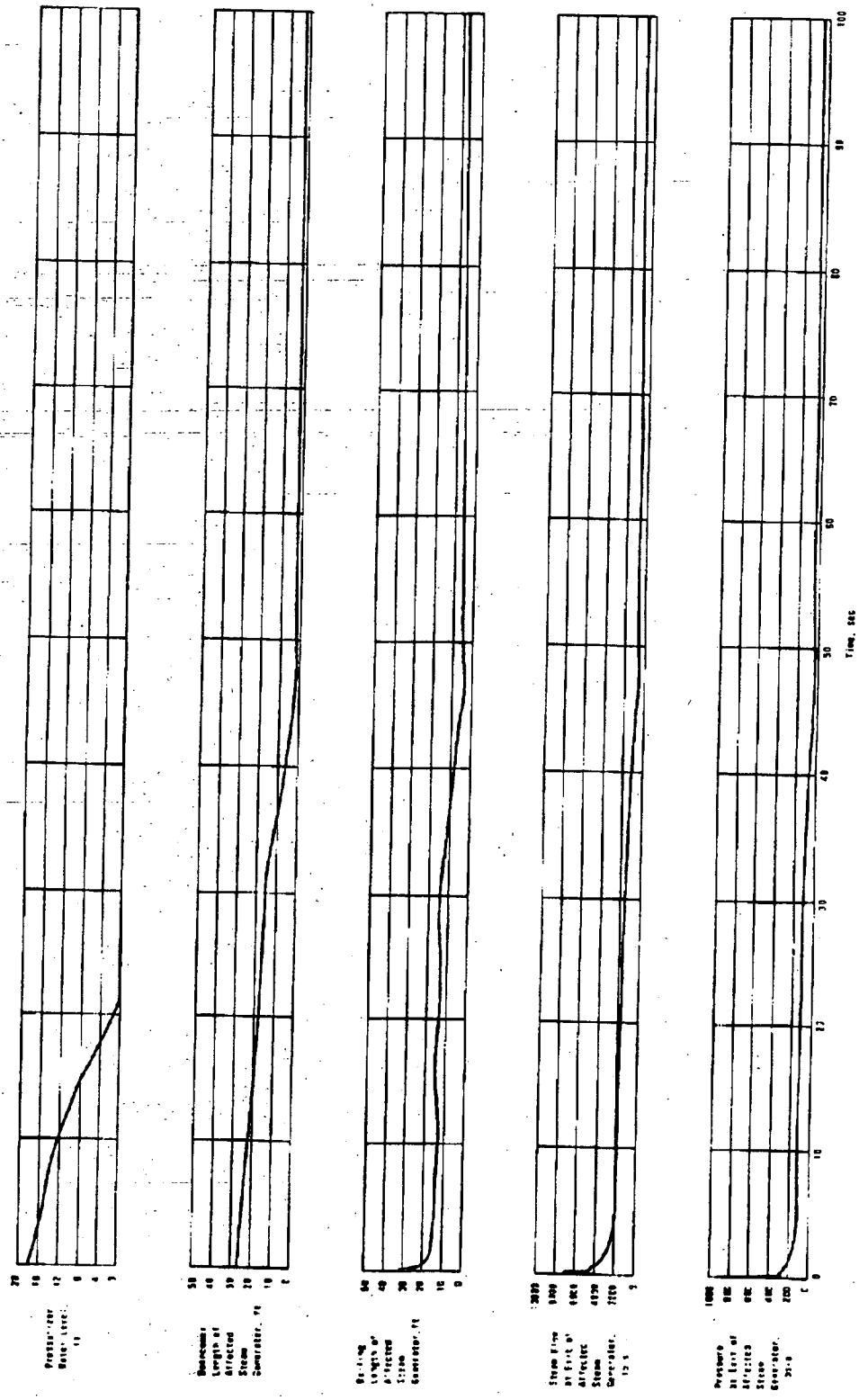

Figure 15-42.
Steam Line Break Accident - Without ICS and Operator Action

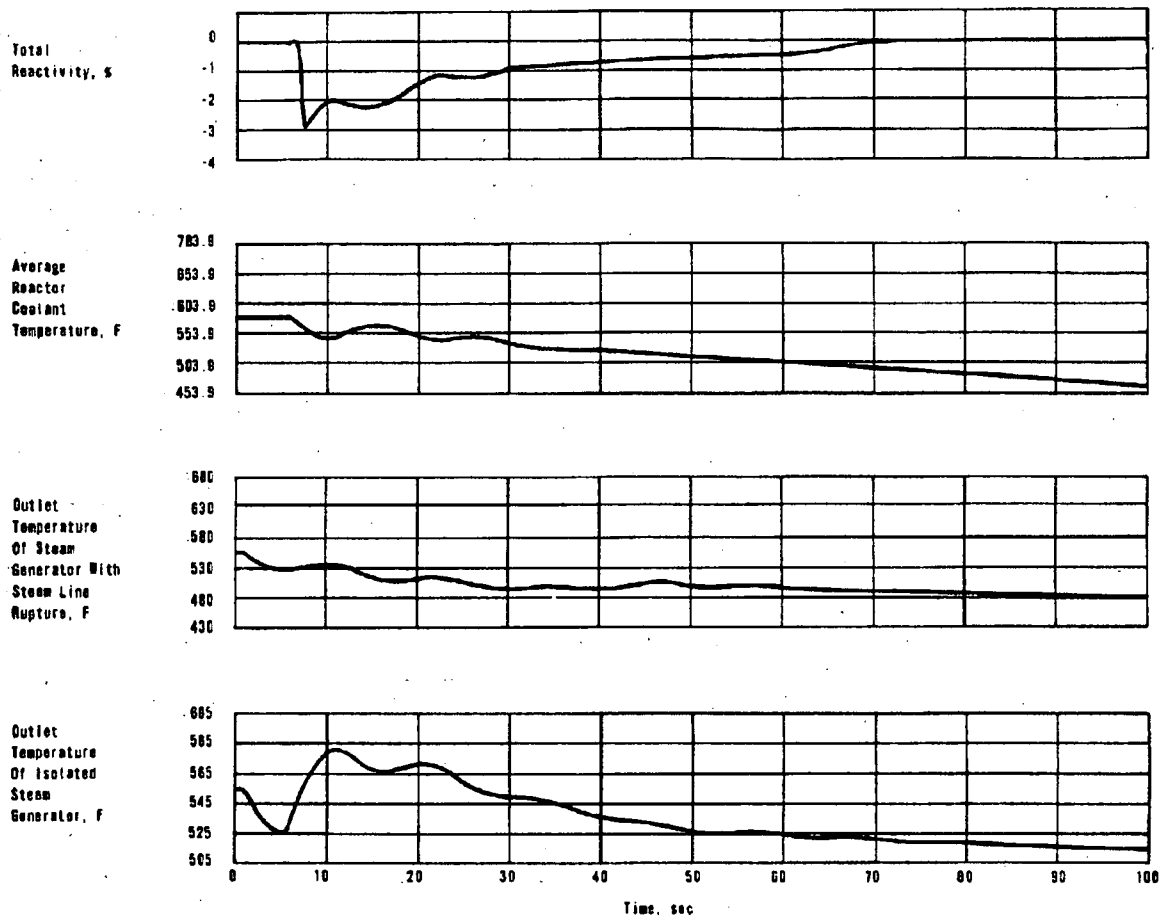


Figure 15-41.
Steam Line Break Accident - Without ICS and Operator Action

2 **Table 15-28. HPI Flow Assumed in Core Flood Line Small Break LOCA Analyses**

| RCS Pressure (psig) | HPI Flow Rate (gpm) |
|---|---|
| 0 | 465.5 |
| 600 | 440 |
| 1300 | 365 |
| 1500 | 340 |
| 1600 | 325 |
| 3000 | 325 |

2 **Table 15-29. HPI Flow Assumed in RCP Discharge Small Break LOCA Analyses**

Flow rates prior to credit for operator realignment of HPI at 10 minutes

| RCS Pressure (psig) | Total Pump Flow (gpm) | Broken Leg Flow (gpm) | Intact Leg Flow (gpm) |
|---|---|---|---|
| 0 | 515 | 257.5 | 257.5 |
| 600 | 450 | 225 | 225 |
| 1200 | 380 | 190 | 190 |
| 1500 | 342 | 171 | 171 |
| 1800 | 300 | 150 | 150 |
| 2400 | 260 | 130 | 130 |

Flow rates after credit for operator realignment of HPI at 10 minutes

| RCS Pressure (psig) | Total Pump Flow (gpm) | Broken Leg Flow (gpm) | Intact Leg Flow (gpm) |
|---|---|---|---|
| 0 | 548 | 164.4 | 383.6 |
| 600 | 500 | 150 | 350 |
| 1200 | 437.1 | 131.1 | 306 |
| 1500 | 404.3 | 121.3 | 283 |
| 1800 | 364.3 | 109.3 | 255 |
| 2400 | 260 | 78 | 182 |

5 **Table 15-26. Deleted Per 1995 Update**

2 **Table 15-27. Results of LOCA Limits Analysis.** MK-B9 Fuel Time Sequence of Events vs. Core Elevation, ft.

| | 2 | 4 | 6 | 8 | 10 |
|---|---|---|---|---|---|
| 2 2 Allowable peak linear heat rate, kW/ft | 16.7 | 17.5 | 17.0 | 17.0 | 17.0 |
| 3 3 ECCS Actuation Setpt. reached in hot leg, sec | 0.5 | 0.5 | 0.5 | 0.5 | 0.5 |
| 2 CFTs Begin Injecting, sec | 18.8 | 15.8 | 15.8 | 15.9 | 15.9 |
| 2 End of Blowdown, sec | 25.6 | 24.4 | 24.0 | 24.4 | 24.4 |
| 2 LPI Begins Injecting, sec | 48.5 | 48.5 | 48.5 | 48.5 | 48.5 |
| 2 2 Peak cladding temp of unruptured node/time, °F/sec | 1871/38.0 | 2034/90.8 | 1980/116.7 | 1917/122.3 | 1846/147.9 |
| 2 2 Peak cladding temp of ruptured node/time, °F/sec | 1931/38.8 | 1681/38.0 | 1596/33.0 | 1502/33.0 | 1476/36.9 |
| 2 Initial pin pressure, psia | 1045 | 1045 | 1045 | 1045 | 1045 |
| 2 Rupture time, sec | 24.8 | 25.8 | 26.7 | 31.9 | 36.9 |
| 2 2 Local metal-water reaction, % | 2.39 | 2.91 | 2.81 | 2.55 | 2.47 |

2 **Table 15-28. HPI Flow Assumed in Core Flood Line Small Break LOCA Analyses**

| | RCS Pressure (psig) | HPI Flow Rate (gpm) |
|---|---|---|
| 2 | 0 | 465.5 |
| 2 | 600 | 440 |
| 2 | 1300 | 365 |
| 2 | 1500 | 340 |
| 2 | 1600 | 325 |
| 2 | 3000 | 325 |

Table 15-7. Results of LOCA Limits Analysis. Core Elevation, ft.

| | 2 | 4 | 6 | 8 | 10 |
|---|---|---|---|---|---|
| Allowable peak linear heat rate, kW/ft | 14.0 | 16.1 | 16.5 | 17.0 | 16.0 |
| Peak cladding temp of unruptured node/time, °F/sec | 1813/37.0 | 1921/80.5 | 2108/115.7 | 1959/126.0 | 1757/215.9 |
| Peak cladding temp of ruptured node/time, °F/sec | 1865/37.7 | 1943/42.6 | 1847/42.0 | 1691/40.5 | 1490/39.5 |
| Initial pin pressure, psia | 1555 | 1555 | 1555 | 1555 | 1555 |
| Rupture time, sec | 18.0 | 22.4 | 22.7 | 25.8 | 27.0 |
| Local metal-water reaction, % | 1.83 | 3.48 | 2.96 | 1.67 | 0.57 |

5 Table 15-8. Deleted Per 1995 Update

5 Table 15-9. Deleted Per 1995 Update

5 Table 15-10. Deleted Per 1995 Update

5 Table 15-11. Deleted Per 1995 Update

5 Table 15-12. Deleted Per 1995 Update

5 Table 15-13. Deleted Per 1995 Update

# APPENDIX 15. CHAPTER 15 TABLES AND FIGURES

5 **Table 15-1. Reg. Guide 1.25 Fuel Handling Accident Source Term**

| Isotope | Gap Fraction | 72 Hour Gap Inventory (Ci's) |
|---|---|---|
| Kr-83m | 0.10 | 3.548-05 |
| Kr-85m | 0.10 | 2.920-01 |
| Kr-85 | 0.30 | 1.994+03 |
| Kr-87 | 0.10 | 3.539-13 |
| Kr-88 | 0.10 | 1.269-03 |
| Xe-131m | 0.10 | 7.644+02 |
| Xe-133m | 0.10 | 2.407+03 |
| Xe-133 | 0.10 | 1.061+05 |
| Xe-135m | 0.10 | 1.067+01 |
| Xe-135 | 0.10 | 1.347+03 |
| Xe-138 | 0.10 | 0.000+00 |
| I-129 | 0.30 | 8.973-03 |
| I-131 | 0.10 | 5.470+04 |
| I-132 | 0.10 | 5.241+04 |
| I-133 | 0.10 | 1.261+04 |
| I-134 | 0.10 | 0.000+00 |
| I-135 | 0.10 | 6.660+01 |

**Table 15-2 (Page 1 of 2). Rod Ejection Accident Parameters**

| | | |
|---|---|---|
| Worth of Ejected Rod | | |
| Rated Power, No Xenon, Densified | 0.65 %Δk/k | |
| Rated Power, No Xenon, Undensified | 0.46 %Δk/k | |
| Rated Power, Xenon, Undensified | 0.36 %Δk/k | |
| Hot Zero Power, Critical, Undensified | 0.56 %Δk/k | |
| Rated Power Level | 2,772 MWt | |
| Rod Ejection Time | 0.150 sec | |
| Reactor Trip Delay | | |
| High Flux Trip | 0.3 sec | |
| High Pressure Trip | 0.5 sec | |
| Time to 2/3 Insertion | 1.4 sec | |
| | BOL | EOL |
| Delayed Neutron Fraction, $\beta_{eff}$ | 0.0071 | 0.0053 |
| Neutron Lifetime, $\mu$s | 24.8 | 23.0 |
| Moderator Coefficient, (Δk/k)/°F | $+0.5 \times 10^{-4}$ | $-3.0 \times 10^{-4}$ |
| Doppler Coefficient, (Δk/k)/°F | $-1.17 \times 10^{-5}$ | $-1.33 \times 10^{-5}$ |

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13. H. B. Tucker (Duke) letter to H. R. Denton (NRC) dated October 20, 1986.
14. 5 5 Regulatory Guide 1.7 (Rev 2), "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident"
15. 5 OSC - 6191 (Rev. 0), "Reanalysis of Oconee Hydrogen Recombiner and Purge System Requirements"
16. 5 Wiens, L. A. (NRC) letter to J. W. Hampton (Duke) dated February 7, 1996.
17. 5 OSC - 123 (Rev. 1), "Activity on Filter RB Hydrogen Purge"
18. 5 OSC - 6534 (Rev. 0), "Hydrogen Purge Cart Operator Dose Rate"
19. 5 5 OSC - 3781 (Rev. 5), "Documentation of Maximum Hypothetical Accident(MHA) Dose Model For Oconee Nuclear Station"
20. 5 5 OSC - 6064 (Rev. 1), "Estimated Radiation Dose Rates in the Auxiliary Building Following a Large Break LOCA"


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available, however, for a temporary modification to be initiated and cables pulled from station power supplies that could supply the recombiner. This process could be performed in less than 24 hours. The temporary modification that would now be needed after an accident will also use non-safety power supplies that have the capability to be powered from Keowee. Multiple load centers exist in the plant that could supply power to a recombiner under the temporary modification assuring that no one single failure would prevent power from being supplied. Because the power supply for the recombiner is not as was described during the Technical Specification change process, the system is considered operable but degraded. Work is currently in progress to correct this deviation from the licensing basis.

Detailed design information is provided in Reference 13 on page 15-83.

## 15.16.5 CONTAINMENT HYDROGEN RECOMBINER SYSTEM OPERATION AND TESTING

Following a LOCA, the process of placing the CHRS in service is begun when the containment hydrogen concentration in containment reaches 0.5% by volume. The recombiner and control cabinet are placed on the appropriate pads of the affected Unit. Electrical and mechanical connections are made to connect the recombiner and temporary drain system to the affected Unit. Mechanical connections are leak rate tested and valve alignments are made to align the CHRS and temporary drain system for operation. Once system operating parameters are within allowable limits, the recombiner is placed in service.

Testing of the recombiner and flow path is performed each refueling outage. This testing includes:

- Visual inspection of the CHR Unit
- Calibration of all recombiner instrumentation and control circuits
- Operation of the CHRS in the post-LOCA configuration
- Verifying proper operation of heaters and controls
- Verifying acceptable flowrates through the recombiner
- Verifying acceptable flowrates through the hydrogen recombiner and hydrogen purge flowpaths
- Leak rate testing of the recombiner and piping
- Leak rate testing of the blind isolation flanges on the CHRS permanent piping (tested after each installation)

In addition to the above testing, the temporary drain system (while installed) shall be tested quarterly to insure integrity and readiness of the system.

## 15.16.6 CONCLUSIONS

Figure 15-86 shows that if no measures were taken to control hydrogen accumulation in the Reactor Building, the hydrogen concentration within the Reactor Building can be expected to reach the lower flammability limit of 4 volume percent at approximately 350 hours. Table 15-24 represents a summary of the generation rates for each of the hydrogen sources at various times following a DBA LOCA. It is apparent from this summary that the core radiolysis is the largest contributor to the total amount of hydrogen produced, and the corrosion of aluminum has the largest hydrogen generation rate long term.

The analyses also show that the hydrogen generated in the Reactor Building following a LOCA can be adequately controlled using the CHRS with a flowrate of greater than 50 SCFM (Figure 15-87).

5 1(2)(3)PR-9 would not prevent operation of the CHRS. Power is supplied to these valve from non-safety
5 related, non-load shed power. The flow configuration of the CHRS is shown on Figure 15-110.

5 Manual action is relied upon outside of containment to restore failed valves or other components that
5 may fail to operate post-LOCA. Sufficient time is available to correct any problems that may occur.

5 In February of 1996, a problem was discovered with the existing CHRS arrangement. Once the system is
5 put into operation to draw gases from the post-accident containment atmosphere, the moisture laden hot
5 air will have to travel through hundreds of feet of convoluted piping to the recombiner. Along the piping
5 pathway outside containment (through the relatively cooler Pen Room environment), it is likely that the
5 moisture entrained in the gases will condense and collect in the recombiner piping low points. For this
5 reason Temporary Modifications (TM) 1237, 1238 and 1239 have been installed on Units 2, 3 and 1
5 respectively. These TMs have installed a temporary collection system that collects condensate from the
5 low point traps in the system on each Unit into tanks. Two tanks are installed in each Units penetration
5 room and would collect drainage from piping in the penetration rooms post-LOCA. In the event of a
5 LOCA on a Unit, two tanks would be moved from a non-LOCA unit penetration room and be installed
5 in the LOCA Unit Cask Decon room. These tanks would collect drainage from piping in this room and
5 drainage from the recombiner flexible hosing outside of this room.

5 The temporary drain system pumps collected condensate to the Low Pressure Injection System through a
5 drain valve upstream of the affected Units Emergency Sump Isolation valve 1(2)(3)LP-20. If
5 1(2)(3)LP-20 failed to open post-LOCA, condensate with a reduced boron concentration would collect in
5 the Emergency Sump that could be later injected into the system. To ensure that the condensate
5 adequately mixes with the LPI system, a secondary path will be used to inject the condensate into the
5 LPI system should 1(2)(3)LP-20 fail to open. This pathway will also be used should the drain valve on
5 the upstream side of 1(2)(3)LP-20 fail in the closed position.

5 Due to the time necessary to install this temporary modification on all three Oconee Units, a one-time
5 extension of the CHRS piping flow path Technical Specification Limiting Condition for Operating time
5 was requested by Duke Power and granted by the NRC to allow an additional seven days with the flow
5 path inoperable. These modifications were completed and the CHRS for each Unit was returned to
5 service on February 10, 1996. Urgent Modifications are currently being developed to permanently install
5 a hard piped drain system that will allow gravity draining of the collected condensate to the Reactor
5 Building Normal Sump of the affected Unit.

5 All piping and equipment necessary for the function of the CHRS are designed to withstand a Safe
5 Shutdown Earthquake without a loss of function except CHRS power which, if interrupted, can be
5 manually restored using alternate power sources.

5 In January 1996, the CHRS was declared operable but degraded based on inconsistencies between the
5 power supplies for the Recombiners and the power supplies described to the NRC during the Technical
5 Specification Change Process that included the CHRS in the Oconee Technical Specifications in 1987.
5 The original power supply arrangement for the hydrogen recombiners utilized non-safety related loadshed
5 power supplied from the station 4160 Volt Main Feeder Busses. In the event of a LOOP, power could be
5 fed by non-load shed power from Keowee by manually aligning breakers to non-load shed power
5 supplies. For redundancy, each Motor Control Center (MCC) feeding the hydrogen recombiners had a
5 feed from its Unit and also a feed from another Unit. Power could be restored by manual breaker
5 alignment to either Units supply. Subsequent to the approval of the Technical Specification change that
5 added the CHRS, the Auxiliary Power Upgrade Modification (NSM- 1,2,32665) removed the supply for
5 these MCCs from the original sources to their present supplies that are fed from the 230 and 525
5 switchyards. In the event of a LOOP, power may not be immediately available through the switchyard to
5 the Auxiliary Power System and therefore power may not be available to the recombiners. Time is

5 The recombiner is normally not connected to a containment building. When needed post-LOCA, the
5 recombiner and control cabinet will be moved to the affected unit. The control cabinet will be installed
5 on a pad near the recombiner. The recombiner will be anchored to its foundation, and connected by
5 flexible piping to the piping which runs to and from containment penetrations 60 and 61.

5 [Note: During the Technical Specification Change Process that added the CHRS, the flexible hoses
5 connecting the recombiner to the hard piping were described as being metal. These are not metal hoses
5 and the this portion of the system has been declared degraded. The current hose has been evaluated to
5 ensure that the hose is acceptable to perform the intended function. Work is in progress to correct this
5 deviation.]

5 The hydrogen recombiner controls hydrogen by recombining hydrogen with oxygen to form water vapor,
5 which is returned to the reactor building. The air is heated by radiant heaters to a temperature high
5 enough (approximately 1200°F) to begin recombination of the hydrogen and oxygen to form water vapor.
5 Recombination of hydrogen and oxygen is an exothermic reaction and the recombiner is designed to use
5 the heat generated by the reaction to aid in maintaining the process. As temperature increases due to heat
5 generated, controls automatically reduce heater output to maintain proper reaction chamber temperature
5 of approximately 1340°F. The recombiner is capable of processing 90 SCFM with a recombination
5 efficiency of at least 95% for hydrogen concentrations greater than 0.5 volume percent. Although the
5 design flowrate for the recombiner is 90 SCFM, the operating flow rate at Oconee is less since there are
5 several hundred feet of supply and return piping. The minimum required flow rate for post-accident
5 operation is 50 SCFM. The Hydrogen Recombiners meet the environmental standards of NRC Bulletin
5 79-01B. Major component data is listed in Table 15-25.

5 The recombiner is the preferred method of hydrogen control since there is no release of radioactive
5 material to the atmosphere. The air/hydrogen mixture is drawn from the reactor building and the air and
5 water vapor mixture is returned to the Reactor Building.

5 The supply flow path for recombiner operation is from the Reactor Building via Inboard Containment
5 Isolation Valve 1(2)(3)PR-7, Penetration 60, Outboard Containment Isolation Valve 1(2)(3)PR-8, and
5 flexible coupling PR FX0001 to the recombiner unit. The flow path through the recombiner is the
5 blower, the flow element, the radiant heaters, the reaction chamber, and the air blast heat exchanger.

5 The air blast heat exchanger is cooled by the air blast blower, which forces approximately 3000 CFM of
5 air at ambient temperature through the heat exchanger to cool the air/water vapor mixture to near
5 ambient temperature before returning to the Reactor Building. The recombiner will automatically
5 shutdown if outlet temperature reaches 146°F.

5 The return flow path is via flexible coupling PR FX0002, manual valve 1(2)(3)PR-61, Outboard
5 Containment Isolation Valve 1(2)(3)PR-10, Penetration 61, and Inboard Containment Isolation Valve
5 1(2)(3)PR-9.

5 Electric motor operated valves 1(2)(3)PR-7 and 1(2)(3)PR-9 close on an Engineered Safeguards Channel 1
5 signal. Air operated valves 1(2)(3)PR-8 and 1(2)(3)PR-10 close on an Engineered Safeguards Channel 2
5 signal. These are redundant channels which actuate on low RC pressure or high RB pressure to close
5 these containment penetration isolation valves.

5 An alternate supply flow path is provided by Hydrogen Recombiner Inlet Containment Isolation Valve
5 1(2)(3)PR-59 and an alternate return flow path is provided by Hydrogen Recombiner Return
5 Containment Isolation Valve 1(2)(3)PR-60. These valves are normally closed EOVs, capable of being
5 operated electrically from the cable rooms. The valves are installed to ensure a failure of 1(2)(3)PR-7 or

5 radiolysis of water in the sump and water leaking from the RCS. These locations are within the
5 unrestricted main volume of the building and will permit the hydrogen to diffuse rapidly and provide a
5 uniform mixture in this area. This rapid mixing occurs because hydrogen has a high diffusion rate and a
5 low generation rate, and is capable of diffusing in all directions. The hydrogen will diffuse very rapidly
5 giving an even distribution under the conditions existing in the Reactor Building. This situation is not
5 analogous to one where attempts are made to mix streams of gases under dynamic conditions where
5 residence times and mixing distances are critical. In addition, the thermal mixing effects, heating of air
5 above the hot sump water, and possible steam releases from the RCS will move the hydrogen laden air
5 from the points of generation toward the cold external walls and emergency cooling equipment. Although
5 hydrogen is lighter than air, it will not tend to concentrate in high areas because of the high diffusion rate
5 and because of the open design of the Reactor Building.

5 Since the hydrogen is generated primarily from corrosion of aluminum and core radiolysis in the large
5 open areas, the hydrogen must diffuse from the major volumes into those minor volumes which are
5 enclosed. The minor volumes or those not having good communication with the major volumes would
5 be at a lower hydrogen concentration because the hydrogen is diffusing from the higher concentration
5 level to a lower concentration level. Accordingly, pockets, if they exist, will be low concentration pockets
5 rather than high concentration pockets. As the maximum concentration in the major volume will never
5 exceed the 4.0 volume/percent limit, flammable or explosive mixtures will not exist in the minor volumes
5 which might be considered as pocket areas.

5 The ability of hydrogen to diffuse rapidly into all volumes is inferred by a condensing steam
5 environment (CSE) experiment (Reference 8 on page 15-83) which measured the spatial concentration of
5 iodine in the various compartments. The tests showed very good mixing in the main chamber and a rapid
5 interchange by diffusion and mixing with the atmosphere of other chambers which had limited
5 communication. The diffusivity of hydrogen is approximately 10 times that of iodine so a more uniform
5 mixture would be expected for hydrogen than for iodine. Also, the higher concentrations would provide
5 greater concentration gradients for better diffusion than was indicated by the CSE tests.

5 During a DBA LOCA, the operation of Reactor Building sprays and RBCUs will provide mixing in
5 containment. This along with the fact that the hydrogen generation rates are low for the majority of the
5 accident support the conclusion that a nearly uniform hydrogen concentration will exist in containment.
5 Even though the average hydrogen concentration throughout containment may be less than 4 v/0, some
5 small pockets of hydrogen exceeding 4 v/o by a small amount would not be detrimental. Results of
5 experiments summarized in Regulatory Guide 1.7 state that for hydrogen concentrations in the rage of 4
5 to 6 volume percent, partial burning of the hydrogen above 4 percent may occur. However, in this range
5 of 4 to 6 percent, the rate of flame propagation is less than the rate of rise of the flammable mixture.
5 Therefore, whether uniform mixing exists or not, hydrogen concentration at 4 volume percent or slightly
5 higher are not a concern.

## 5 15.16.4 CONTAINMENT HYDROGEN RECOMBINER SYSTEM DESCRIPTION

5 The Containment Hydrogen Recombiner System consists of a portable hydrogen recombiner, control
5 panel for the recombiner, and piping. The Oconee recombiners are Thermal Hydrogen Recombiners
5 developed and constructed by Rockwell International. Two recombiners are normally maintained at the
5 Oconee site. Only one recombiner is required operable per Oconee Technical Specifications. Duke
5 Power Company maintains a lease agreement with Carolina Power and Light (CP&L) and Florida Power
5 and Light (FP&L) for use of the second recombiner when needed at the H.B. Robinson Nuclear Site for
5 CP&L and the Turkey Point Nuclear Site for FP&L. This agreement is based on the sharing of
5 recombiners between sites as described in Regulatory Guide 1.7.

Concentrations of this magnitude are only expected during core damage accidents like those studied in probabilistic risk analyses.

Although a concentration greater than 4 volume/percent may be acceptable, the lower flammability limit of 4 volume/percent specified by Regulatory Guide 1.7 is nevertheless used in this evaluation. Figure 15-86 and Table 15-24 summarize post-LOCA hydrogen generation for Oconee.

### 15.16.3.2 Evaluation of Recombination to Control Hydrogen Concentrations

Prediction of hydrogen generation following the loss-of-coolant accident using the assumptions and method of analysis described in Section 15.16.2, "Post-Accident Hydrogen Generation" on page 15-73 shows that although hydrogen production rate decreases as the post-accident time increases, total hydrogen accumulation can exceed the lower flammability limit of 4 volume percent. The analysis shows that using conservative assumptions, post-LOCA hydrogen concentrations can reach 3 volume percent in approximately 156 hours (6.5 days) and 4 volume percent in approximately 350 hours (14.6 days) (Figure 15-86). A method of control is therefore necessary to prevent hydrogen accumulation from exceeding the Regulatory Guide 1.7 limit of 4 volume percent.

Recombination of hydrogen and oxygen in the reactor building atmosphere is the chosen means of post-accident hydrogen control. The Containment Hydrogen Recombiner System (CHRS) is designed to operate at a flowrate of greater than 50 SCFM with concentrations of 0.5 volume percent and a recombination efficiency of 95%. Additionally, use of the recombiner will not increase offsite releases of radioactive material.

The basic approach evaluated herein is to allow the hydrogen concentration to increase for a minimum of 7 days prior to placing the CHRS into service. This allows time for pressures and temperatures to decrease in the Reactor Building prior to placing the system in service. With hydrogen concentrations conservatively calculated following Regulatory Guide 1.7 methodology not to reach 4 volume percent in containment for 350 hours after the initiation of the event (Figure 15-86), allowing this 7 day time would not cause the 4 limit to be exceeded. Steps are taken to place the recombiner in service when the hydrogen concentration exceeds 0.5 volume percent within the preceeding time limitations and within the pressure/temperature limitations of the recombiner system components. The analysis shows that when recombiner operation is begun as stated above at a flow rate of greater than 50 SCFM, safe hydrogen concentrations will be maintained in containment.

Post accident hydrogen concentrations are indicated by the Containment Hydrogen Monitoring System (CHMS). The CHMS is described in Section 9.3.7, "Containment Hydrogen Monitoring System" on page 9-50 and is shown in Figure 9-15. This instrumentation provides two redundant channels of hydrogen monitoring that can monitor hydrogen concentrations at different levels of the containment including CHRS inlet and return concentrations. Should both trains of hydrogen monitoring be inoperable and no other means of hydrogen measurement be available, the CHRS will be placed in service after 7 days from initiation of the accident to assure hydrogen concentrations are not exceeded.

In order to assure high concentration pockets of hydrogen do not exist and that representative samples of hydrogen can be obtained, adequate mixing of hydrogen throughout containment should exist. Mixing in the Reactor Building atmosphere is expected to be good. The Reactor Building cooling fans or sprays will introduce considerable turbulence to the building atmosphere to provide good mixing of hydrogen in the early stages of the accident. In addition, all the Reactor Building volumes are connected by large vent areas (stair wells, elevator shafts, grating) to promote good air circulation.

Figure 15-89 shows the Reactor Building cross-section. The hydrogen generated will be primarily from the corrosion of aluminum HVAC equipment in the large open area of the containment and from

5 *15.16.2.2.3.1 Corrosion of Plant Materials*

5 Another possible source of hydrogen could occur from metal surfaces exposed to an environment
5 containing high-temperature steam, corrosive sprays, fission products, and radioactivity. Such exposure
5 might result in surface corrosion reactions that produce hydrogen. Corrosive tests have been performed to
5 determine the behavior of various metals that are used in Containment when exposed to a post-LOCA
5 environment. As applied to the quantitative definition of hydrogen production rates, the results of the
5 corrosion tests have shown that only aluminum will corrode at a rate that will significantly add to the
5 hydrogen accumulation in the Containment atmosphere. However, because of the relatively large amount
5 of exposed galvanized and zinc-based painted surfaces in Containment, zinc corrosion must be considered
5 as a contributing hydrogen source.

5 The corrosion of aluminum and zinc may be described by the following reactions:

5 $$2Al + 3H_2O \text{-----}> Al_2O_3 + 3H_2$$
5 $$Zn + H_20 \text{-----}> ZnO + H_2$$

5 The time-temperature cycle considered in the calculation of aluminum and zinc corrosion are based on a
5 conservative representation of the postulated post accident Containment transient. The corrosion data
5 points include the effects of temperature, alloy, and spray solution conditions. NOTE: In Section 5, Part
5 C of Regulatory Guide 1.7 it is stated that values given in Table 1 for evaluating production of
5 combustible gases following a LOCA may be changed on the basis of additional experimental evidence
5 and analyses. As a result the minimum assumed value give for aluminum corrosion rate of 200 mpy is
5 not used in the analysis.

5 **15.16.2.3 Primary Coolant Hydrogen**

5 The maximum equilibrium quantity of hydrogen in the primary coolant is 472 scf. This value includes
5 both hydrogen dissolved in the coolant water at 15-40 cc (STP) per liter of water and corresponding
5 equilibrium hydrogen in the pressurizer gas space. The 472 scf of hydrogen is assumed to be released
5 immediately into Containment at the initiation of the LOCA.

5 **15.16.3 EVALUATION OF RECOMBINATION TO CONTROL HYDROGEN**
5 **CONCENTRATIONS**

5 **15.16.3.1 Hydrogen Flammability Limits**

5 In order to determine the acceptability of any hydrogen removal system, the hydrogen concentration that
5 would constitute a potential hazard if that concentration were exceeded must be established. Regulatory
5 Guide 1.7 defines a concentration limit of 4 volume percent for hydrogen accumulation following a loss of
5 coolant accident.

5 The hydrogen generation which occurs following a design basis LOCA is a slow process driven by sump
5 radiolysis and metal corrosion. Calculations have shown that many days are required to reach the
5 regulatory limit of 4 volume percent. A hydrogen concentration slightly above 4 volume percent is
5 generally accepted as a lower flammability limit. Furthermore, assuming no credit for the Containment
5 Hydrogen Recombiner System, the concentration thirty days following a design basis LOCA is
5 approximately 5.5 volume percent. Studies of containment structural capacity and the effects of hydrogen
5 combustion have shown concentrations much higher than 4 volume percent are required to threaten the
5 integrity of a large dry containment like the Oconee containments. Concentrations in excess of 12 volume
5 percent would be required to present a challenge to the integrity of the Oconee containments.

establish the sump contribution. Both of these hydrogen generation constants are those reported in the literature. The "G" value of 0.45 is used in the core region and is reported as a conservative upper limit for boiling solutions (Reference 2 on page 15-83, Reference 3 on page 15-83). The "G" value of 0.3 is used for the coolant in the sump which represents a slowly moving fluid and it is based upon published ORNL data (Reference 4 on page 15-83).

### 15.16.2.2 Chemical Hydrogen Generation

In addition to the radiolytic hydrogen generation sources (core and sump radiolysis) following a Design Basis Accident, hydrogen may also be evolved from two chemical sources: (1) zirconium-water reaction involving clad material, and (2) from the reaction of zinc and aluminum within the Reactor Building with the borated coolant water.

#### 15.16.2.2.1 Method of Analysis

The quantity of zirconium which reacts with the core cooling solution depends on the performance of the Emergency Core Cooling System. The 10CFR50.46 criteria for evaluation of the Emergency Core Cooling System requires that the zircalloy-water reaction be limited to one percent by weight of the total quantity in the core.

Aluminum is more reactive with the Reactor Building spray solution than other plant materials such as galvanized steel, copper, and copper-nickle alloys. However, because of the relatively large amount of exposed galvanized and zinc-based painted surfaces in the Reactor Building, zinc corrosion must be considered as a contributing hydrogen source.

It should be noted that zirconium-water reaction and the aluminum and zinc corrosion with Reactor Building spray are chemical reactions and thus essentially independent of the radiation field inside the Reactor Building following a LOCA. Radiolytic decomposition of water is dependent on the radiation field intensity. The radiation field inside the Reactor Building is calculated for the maximum credible accident in which the fission product activities given in TID-14844 are used.

#### 15.16.2.2.2 Typical Assumptions

The following discussion outlines the assumptions used in the calculations.

#### 15.16.2.2.3 Zirconium-water Reaction

Hydrogen can be generated during a LOCA by the reaction of hot zirconium cladding with the surrounding steam. The zirconium-water reaction is described by the chemical equation:

$$Zr + 2H_2O \text{ ------> } ZrO_2 + 2H_2 + \text{Heat}$$

The quantity of zirconium which reacts with the core cooling solution depends on the performance of the Emergency Core Cooling System (ECCS). The 10CFR50.46 criteria for evaluation of the ECCS requires that the zirconium-water reaction be limited to one percent by weight of the total quantity of zirconium in the core. For Oconee the maximum of 1% zirconium-water reaction is assumed. Regulatory Guide 1.7 requires that the assumption for hydrogen produced from the zirconium-water reaction equal 5 times the extent of the maximum calculated reaction under 10CFR50.46, i.e., 5.0%. Per Regulatory Guide 1.7, the zirconium-water hydrogen source is assumed to be released over a 2 minute period from the start of the transient, and is assumed to be distributed uniformly throughout Containment.

5 material. Consequently, whenever the volume or mass of the emergency core cooling solution changes,
5 the dose changes. The generation of hydrogen, however, is a function only of the energy absorbed by the
5 solution and does not change simply because the mass of the solution changes.

5 The curve shown in Figure 15-84 includes all the sources previously mentioned, fission product gammas,
5 fission product betas, Np-239 and U-239, and activated structural materials. Also included is the energy
5 absorbed by the water outside the core from sources inside core. This was determined by computing the
5 spatial variation of the gamma fluxes in the water outside the core and integrating over the water volume.
5 This contributed between 3 and 4 percent of the total energy absorbed. The activated cladding and other
5 core hardware sources yield deposited energy which is only a fraction of a percent of that from other
5 sources. The greatest amount of absorbed energy comes from the fission product and Np-239 gamma
5 rays; these contribute approximately 95 percent of the total energy.

5 In addition to the core sources, 100 percent of the noble gases in the Reactor Building atmosphere, 50
5 percent of the halogens and 1 percent of the solids in the cooling water must also be considered.

5 In the Reactor Building atmosphere the activity levels and beta and gamma yield data for the noble gases
5 were used to compute the individual beta and gamma source strengths for each nuclide. These sources
5 were distributed uniformly throughout the free air volume of the Reactor Building. For beta particles the
5 energy absorbed per unit volume of air was assumed to be equivalent to that produced per unit volume.
5 For gamma rays the absorbed energy was computed at a point in the center of the Reactor Building and
5 it was assumed that all other points in the building received this same energy. The gamma flux was
5 calculated with a point kernel integration code. No credit was taken for attenuation by shielding by
5 structures within the Reactor Building.

5 For the 50 percent halogens and 1 percent solid fission products in the cooling water, it was assumed that
5 all of the energy produced by these sources was absorbed in the water. The absorbed energy from the
5 halogens was computed for each nuclide using the individual beta and gamma yields. The contribution
5 from the 1 percent solids was obtained by taking 1 percent of the total fission product decay heat curve
5 and depositing this quantity of energy to the coolant.

5 As can be seen in Figure 15-84, initially the energy is controlled by the fission products in the cooling
5 water. However, by 100 hr the sources in the core have taken over and continue to control as time
5 increases. The contribution from the noble gases in the building atmosphere does not show up on the
5 graph - this was insignificant in comparison to the other sources.

5 Since the core sources, and in particular the fission product gammas, contribute most of the energy to the
5 solution, a comparison has been made between the decay heat gamma sources calculated by B&W with
5 that published by Shure (Reference 1 on page 15-83). Figure 15-85 shows the integrated gamma decay
5 heat (fission products plus U-239 and Np-239) between 10 and 3,000 hr decay time following 620 days
5 irradiation time from the two methods. Over the time span of interest for hydrogen generation (100 -
5 1,000 hr), they are in excellent agreement.

5 **15.16.2.1.3 Radiolytic Hydrogen Generation**

5 The hydrogen generation rates from the radiolytic decomposition of water were calculated utilizing the
5 data presented in Figure 15-84 and a hydrogen generation constant ("G" value) equal to 0.45 molecules of
5 hydrogen per 100 ev of energy absorbed by the fluid in the core region.

5 In the LOCA the integrated energy absorption by the sump solution is small, and this has been lumped
5 with the core energy and a single hydrogen generation constant has been used. Under the Design Basis
5 Accident (DBA) conditions, when the energy absorption in the sump region is significant, as a result of
5 the assumed sources, a "G" value of 0.3 molecules of hydrogen per 100 ev of energy absorbed was used to

5 (BWST). After the BWST is empty, coolant is then circulated from the Reactor Building sump through
5 the LPIS to the reactor vessel.

5 The activity levels of the individual fission product nuclides were determined with B&W's proprietary
5 digital fission product code. This digital code computes the activity of more than 200 fission product
5 nuclides from one or two fissionable materials as a function of reactor operating history. One hundred
5 time steps can be used in the code, and at each time step the program will print the individual nuclide
5 activity along with the total gamma source strengths from all nuclides for each of six gamma energy
5 groups.

5 The activity of the Np-239 and U-239 was obtained using the maximum neutron capture rates that occur
5 in U-238 at any time during a core cycle. Sources from the activated clad and structural materials were
5 calculated assuming saturation activity of these components and using lifetime average neutron fluxes in
5 the core.

#### 15.16.2.1.2 Calculation of Absorbed Energy

5 Table 15-18 summarizes the assumption made in calculating the energy absorbed by the coolant. In a
5 LOCA all of the absorbed energy comes from sources in and near the core with the greatest fraction
5 coming from fission product decay. The energy from fission product decay is about equally divided
5 between gamma rays and beta particles. To determine the energy deposited in the solution by beta
5 particles, the fuel pellets were subdivided into concentric cylindrical source shells, 10 mils in thickness.
5 The amount of beta energy that was transmitted through the fuel cladding was calculated for each of these
5 source shells. These calculations were performed for each fission product nuclide. It was assumed that all
5 of the energy that penetrated the clad was absorbed by the water. The integrated beta energy between 10
5 and 3,000 hr after the LOCA was only two to three percent of that from the gamma rays. Nearly all of
5 the beta energy is absorbed by the oxide and cladding.

5 For gamma rays the fission product source strengths were taken from the output of the fission product
5 code and the gamma sources from U-239 and Np-239 were added to the various energy groups to obtain
5 the total core source strengths. Energy deposition rates to the solution at various times were then
5 calculated for each of the six gamma energy groups. Since these sources contribute most of the energy
5 received by the solution, this calculation was checked using three techniques, all based on the assumption
5 of a homogeneous core with a uniform average source distribution.

5 First, the amount of energy produced per unit volume of core was assumed to equal the amount of energy
5 absorbed per unit volume. The distribution of absorbed energy between the cooling water and the
5 remainder of the core was found by a ratio of the energy absorption coefficient of water to that of the
5 homogenized core for each of the energy groups. In the second technique a receiver point was chosen at
5 the center of the core. The energy absorption rate was calculated at this location with a point kernel
5 integration code. The absorption at all other points in the core was assumed to be in the same as that at
5 this point. For the third calculation the core was represented as an infinite homogeneous medium and the
5 flux equations for an infinite slab were used to calculate the absorbed energy. These latter two
5 calculations require the use of energy absorption buildup parameters for the homogenized core. A
5 comparison between the mass and energy absorption coefficients for the core with those of various
5 materials showed that the coefficients for lead matched those of the core quite well. The energy
5 absorption buildup factors for lead were thus incorporated into the flux equations using the Taylor form
5 of the buildup.

5 The total energy deposited in the solution was obtained by graphical integration of the absorption rate
5 curves. Figure 15-84 shows the energy absorbed by the solution as a function of time following the
5 LOCA. The results have been presented in terms of the energy absorbed by the solution rather than in
5 dose units. The reason is that dose is a measure Absorbed Energy of energy absorbed per unit mass of

# 15.16 POST-ACCIDENT HYDROGEN CONTROL

## 15.16.1 INTRODUCTION

The purpose of this section is to summarize the analyses performed to:

1. Evaluate the hazard caused by hydrogen generation following a LOCA.
2. Evaluate the acceptability of hydrogen recombination as a method for controlling the Reactor Building hydrogen concentration.

In this section the potential for radiolytic hydrogen generation including the dose, or energy deposited in the coolant following the accident, and the basis for the selection of the hydrogen generation constant ("G" value) is analyzed. Since the FSAR analyzes the potential zircaloy-water reaction in other sections, this analysis is not presented herein and a 5 percent zirc-water reaction is assumed in the reference case described in subsequent sections. The potential for hydrogen generation from a zinc-boric acid reaction when borated water spray solution contacts galvanized steel and aluminum in the Reactor Building at the post-accident temperature is also considered. The analysis shows that the radiolytic hydrogen generation rate plus the hydrogen contributed by the zircaloy and other reactions does not result in unacceptable hydrogen concentrations until 350 hr after the initiation of the LOCA.

Post-accident Reactor Building hydrogen concentration is controlled by the use of the Thermal Hydrogen Recombiner. Air is drawn from the reactor building at a flow of greater than 50 scfm to the recombiner. There the air is heated to approximately 1340 deg. F and passed through a reaction chamber causing hydrogen and oxygen molecules to recombine into water vapor. The effluent from this process then flows back to the reactor building. Since all process gas is returned to the reactor building, no release of radioactive material is made to the environment and therefore the hydrogen concentration can be reduced without increasing the radiation dose to the public.

Regulatory Guide 1.7 "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident" has been referenced in several sections of this analysis. Even though the Regulatory Guide has been used for guidance and information, Oconee is not committed to Regulatory Guide 1.7.

## 15.16.2 POST-ACCIDENT HYDROGEN GENERATION

### 15.16.2.1 Radiolytic Hydrogen Generation

#### 15.16.2.1.1 Sources of Radiation

The radiation sources which contribute to the energy absorbed by the coolant following an accident are shown in Table 15-17. For the LOCA the only significant amount of radiation comes from sources in and near the core. In addition to the core sources, the contributions from the gases in the Reactor Building atmosphere, and the fission products in the coolant water itself are also considered. Table 15-18 shows the assumption used in the calculation of the energy deposited in the solution described in the following section.

Figure 15-82 shows the flow path of the post-accident ECCS in the long-term recirculation mode. Following a LOCA, the fluid for the RBSS and the ECCS is supplied from the borated water storage tank

## 15.15.5 REFERENCES

1. Walker, R. J. (B&W), Letter to Gill, R. L., September 11, 1981.
2. Oconee Unit 2, Cycle 6 Reload Report, Babcock & Wilcox, BAW-1691, August 1981.
3. Culkowski, W. M., Deposition and Washout Calculations Based on the Generalized Guassian Plume Model, *ORO-599*.
4. Basic Safety Standards for Radiation Protection, 1967 Edition, Safety Series No. 9, International Atomic Energy Agency, Vienna, 1967.

x = downwind distance (m)

$\sigma_y$ = horizontal dispersion (m)

y = crosswind distance from plume axis (m)

$Q_o$ = release rate ( $\frac{Ci}{sec}$ )

The equation above is conservative since the results do not consider the wind speed or vertical distribution in the cloud. The wind direction is assumed to remain towards Lake Keowee for the 24 hr period with the plume center lines uniformly distributed over this section. Washout is assumed to occur under neutral stability conditions, Pasquill D, which is typical for a rainy day.

The average release rate from the Reactor Building during the 24-hr period following the accident is 0.37 equivalent curies of iodine-131 per sec. Using the above equation, the maximum iodine washout is calculated by assuming that all of the iodine that has washed out remains in the surrounding reservoir and is not affected by runoff. The average number of curies in the reservoir during a one-year period is reduced by a factor of 0.0318 due to the natural decay of iodine. Assuming that this activity mixes in the reservoir and that an adult drinks 0.8 $m^3$ per year (Reference 4 on page 15-71) of the contaminated water, the total dose to the thyroid has been calculated using the methods of TID-14844. The nearest drinking water intake is approximately two miles from the site. At this distance, the total integrated one-year ingestion dose to the thyroid is 1.0 rem. This dose is well below the limits of 10CFR 100.

## 15.15.4 EFFECTS OF ENGINEERED SAFEGUARDS SYSTEMS LEAKAGE

An additional source of fission product leakage during the maximum hypothetical accident can occur from leakage of the engineered safeguards systems external to the Reactor Building during the recirculation phase for long-term core cooling. A detailed analysis of the potential leakage from these systems is
3 presented in Section 6.1.6, "Engineered Safeguards Materials" on page 6-7. This leakage is less than the
3 Technical Specification 4.5.4.1 limit of 2 gallons per hour (gph). Some of this leakage will flash into
3 steam.

It is assumed that the water being recirculated from the Reactor Building sump through the external system piping contains 50 percent of the core saturation iodine inventory. This is the entire amount of iodine released from the RCS. The assumption that all of the iodine escaping from the RCS is absorbed by the water in the Reactor Building is conservative since much of the iodine released from the fuel will
3 be plated out on the building walls. It is assumed that all the iodine contained in the water which is
3 flashed is released to the Auxiliary Building atmosphere. Iodine will also be released from the remaining
3 water, but the total released from both sources is estimated to be less than 5 %.

The Auxiliary Building is ventilated and discharges to the unit vent. The activity is assumed to be
3 continuously released from the unit vent during the recirculation phase (which does not start until 30
3 minutes into the event). Atmospheric dilution is calculated using the 2-hr release dispersion factor of
3 0.00022 seconds per cubic meter. The total integrated dose to the thyroid from this activity is 2.52 rem
3 for 2-hr exposure at the 1 mile exclusion distance. The integrated dose to the thyroid for a 30 day
3 exposure at the 6 mile low population zone boundary is 2.1 rem. Combined with other sources of
3 exposure during a maximum hypothetical accident, this is within the 300 rem guidelines specified in 10
3 CFR Part 100.

# 15.15 MAXIMUM HYPOTHETICAL ACCIDENT

## 15.15.1 IDENTIFICATION OF ACCIDENT

The analyses in the preceding sections have demonstrated that even in the event of a LOCA accident, no significant core melting will occur. However, to demonstrate in a still more conservative manner that the operation of a nuclear power plant at the proposed site does not present any undue hazard to the general public, a maximum hypothetical accident (MHA) involving a gross release of fission products is evaluated. No mechanism whereby such a release occurs is postulated, since this would require a multitude of failures in the engineered safeguards which are provided to prevent such an occurrence. Fission products are assumed to be released from the core as stated in TID-14844, namely, 100 percent of the noble gases, 50 percent of the halogens, and 1 percent of the solids. Further, 50 percent of the iodines released to the
4 Reactor Building are assumed to plate out. The Reactor Building Spray System is credited with removal
4 of a portion of the remaining iodine from the building atmosphere. Other parameters such as meteorological conditions, iodine inventory of the fuel, Reactor Building leak rate, etc., are the same as assumed for the LOCA. The total core fission product inventory of interest is given in Table 15-15 (Reference 1 on page 15-71).

## 15.15.2 ENVIRONMENTAL EVALUATION

The meteorological model employed is the same used in the LOCA analysis. The Reactor Building leak rate is assumed to be 0.25 percent per day by volume for the first 24 hrs, and then 0.125 percent per day for the next 29 days. The other assumptions are consistent with TID-14844.

The direct dose to the whole body following this accident is insignificant as shown in Figure 15-80. The
3 total integrated thyroid doses are 263.6 rem for the 2 hr exposure at the 1-mi exclusion area boundary,
3 and 129 rem for the 30-day exposure at the 6-mi low population zone distance. The corresponding whole
3 body doses are 1.62 rem and 0.383 rem. These dose consequences are within the 10CFR 100 limits. A summary of the dose consequences for all transients and accidents is given in Table 15-16. These doses are revised to account for the effects of high burnup reload core designs (References 1 on page 15-71, 2 on page 15-71).

## 15.15.3 EFFECT OF WASHOUT

To provide a further evaluation of the suitability of the site, the effects of washout on surrounding drinking water reservoirs following the MHA are analyzed. Calculations are made for the case of continuous rain lasting 24 hr covering the general area of the reservoir and the site. The maximum washout rate as a function of distance is calculated from the following equation (Reference 3 on page 15-71):

$$\omega_{max} = \frac{Q_o \, e^{-(y^2/2\sigma_y^2)}}{xe\sigma_y \sqrt{2\pi}}$$

where

$\omega_{max}$ = maximum washout rate $\left(\frac{Ci}{sec - m^2}\right)$

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isotopes of iodine have been equated into dose equivalent curies of iodine-131. The dose equivalency factor is determined by considering the concentration and specific dose of each iodine isotope present over the period of interest. The iodine dose to the thyroid per curie is obtained from the values given in TID-14844. The iodine activity released to the reactor building is $1.43 \times 10^6$ dose equivalent curies of iodine-131.

While the Reactor Building leakage rate will decrease rapidly as the pressure decays, the leakage is assumed to remain constant at the rate of 0.25 percent of Reactor Building volume per day for the first 24 hrs. Thereafter, since the Reactor Building has returned to nearly atmospheric pressure, the rate is assumed to be reduced to 0.125 percent of the Reactor Building volume per day and to remain at this value for the duration of the accident.

It is assumed that 50 percent of the Reactor Building leakage will go into the penetration rooms which will be maintained at a negative pressure. The atmosphere in these rooms is discharged through charcoal filters to the unit vent. The charcoal filters are assumed to be 90 percent efficient for iodine removal. The remaining 50 percent of the Reactor Building leakage is assumed to escape directly to the atmosphere. By this method a maximum of 55 percent of the iodine released from the Reactor Building is ultimately released to the atmosphere. Atmospheric dilution of the leakage discharged from the unit vent is calculated using the elevated release dispersion factor of $3.35 \times 10^{-5}$ sec/m$^3$. Dilution of the other leakage from the Reactor Building is calculated using the ground release dispersion factor of $1.16 \times 10^{-4}$ sec/m$^3$. A breathing rate of $3.47 \times 10^{-4}$ m$^3$/sec is assumed for the 2 hr. exposure. For the 30-day exposure, a breathing rate of $2.32 \times 10^{-4}$ m$^3$/sec is assumed.

The total integrated thyroid doses resulting from this LOCA fission product release are 5.0 rem for the 2 hr. exposure at the 1 mi exclusion distance, and 5.5 rem for the 30-day exposure at the 6 mi low population distance. The corresponding whole body doses are 0.01 rem and 0.01 rem.

## 15.14.8 CONCLUSIONS

A complete spectrum of LOCA's have been conservatively analyzed with the B&W evaluation model which conforms to 10CFR50 Appendix K. The results of these analyses meet the acceptance criteria of 10CFR50.46. The Reactor Building and subcompartment pressure response analyses showed that the structural design limits were not exceeded. The off-site environmental consequences are within the dose limits of 10CFR100. Therefore, the consequences of all design basis LOCA's have been shown to be acceptable.

2 ## 15.14.6 CONFORMANCE WITH ACCEPTANCE CRITERIA

The B&W ECCS Evaluation Model used for the LOCA analysis for Oconee class plants has been shown to be within the guidelines of 10CFR50 Appendix K. This model has been used to perform detailed sensitivity studies to assure that any adverse phenomena are identified and adequately addressed. These analyses have demonstrated that the consequences of hypothetical LOCA's up to and including a double-ended break of the largest pipe in the RCS are within the limits prescribed in 10CFR50.46, as follows:

### 15.14.6.1 Peak Cladding Temperature

The maximum peak cladding temperature was calculated to be 2108°F, which is less than the 2200°F limit.

### 15.14.6.2 Maximum Cladding Oxidation

The maximum local metal-water reaction was calculated to be 3.48 percent, which is less than the 17 percent limit.

### 15.14.6.3 Maximum Hydrogen Generation

The worst case core average hydrogen generation was calculated to be 0.55 percent, which is less than the 1 percent limit.

### 15.14.6.4 Coolable Geometry

Changes in core geometry due to thermal and irradiation effects and mechanical loading have been calculated and show that no gross core blockage or disfiguration will occur. The core will maintain a coolable geometry.

### 15.14.6.5 Long-Term Cooling

Subsequent to the blowdown, refill, and reflood phases of a LOCA, long-term cooling to remove core decay heat for an extended period of time must be established. The ECCS is designed to perform this function. Operator action is assumed to be available fifteen minutes following a LOCA. Several operational modes are available to provide the necessary cooling and also to assure that adequate coolant circulation exists to prevent any concentration of boric acid in a region of the RCS. Redundancy in the design of the ECCS and multiple available flowpaths for removing core heat provide for sufficient long-term cooling.

## 15.14.7 ENVIRONMENTAL EVALUATION

The evaluation of the environmental consequences for the LOCA includes the assumption that one percent of the fuel rods in the core have been defective prior to the initiation of the accident. This results in the coolant fission product inventory given in Table 15-14 (Reference 28 on page 15-67) for the worst time in life (up to 400 EFPD) for each isotope. The fission product release to the Reactor Building includes the coolant activity plus the gap activity from all fuel rods. The total core gap activities are given in Table 15-4.

Of the iodine released, 50 percent is assumed to plate out and the other half is assumed to remain in the Reactor Building atmosphere where it is available for leakage. No credit is taken for removal of airborne iodine by the Reactor Building Spray System (RBSS). To facilitate environmental dose calculations, all

materials could theoretically melt even though the temperature is below the melting point of either material taken singly.

One point of such dissimilar metal contact is between Zircaloy clad fuel rods and Inconel 718 spacer grids. The analysis of the performance of the core flooding tanks during a LOCA indicated that some of the cladding will exceed the zirconium-iron and the zirconium-nickel eutectic points. Since the spacers are located at 21-in. intervals along the assembly and each grid has a very small contact area, only a fraction of the hottest fuel rods would be in contact with Inconel 718 spacer grids.

During the experimental test program referred to above, B&W conducted experimental tests in which specimens of Zircaloy-4 tubing in contact with sections of spacer grids material were subjected to a thermal transient closely approximately that of the clad hot spot following a LOCA. These tests verified that the eutectic reaction is limited to the small region of contact between the clad and the spacer grid tips (dimples), and that it terminates as these materials melt at the point of contact. Both the clad and the grid material maintained their structural integrity because the amount of material involved was small and melting was localized.

Another area of dissimilar metal contact is that of a zirconium guide tube with the stainless steel cladding of the control rod. To determine whether the temperatures in the control rod following a LOCA could become high enough to approach either the temperature required for possible eutectic formation between the clad and the guide tube or the melting temperature of the Ag-In-Cd alloy, the thermal performance of a control rod assembly following a LOCA was examined analytically. A conservative approach was taken in this analysis. In spite of the fact that the core flow is as high or higher than normal core flow during the first two seconds following the rupture, normal steady-state cooling of a control rod is assumed. In two seconds, the core power is essentially down to decay heat levels. However, the following assumptions were made.

1. The average core power after two seconds is 8 percent of ultimate power and remains at this level.
2. All decay heat is absorbed in the core and 50 percent of the decay heat is in the form of gamma rays available for absorption in the control rod. By ratioing the control rod density to the average core density, an average energy deposition rate of 8.50 W/cc in the control assemblies was obtained.
3. The maximum activation product energy in the control rod itself was estimated to be 2.99 W/cc.
4. The highest energy deposition rate at the decay heat level was assumed to be the average times the ratio of peak-to-average power, or 36.18 W/cc.
5. An adiabatic heatup of the control rod with a heat rate of 36.18 W/cc was assumed until the water level reached the point in the core at which the highest peak-to-average power occurs.
6. The temperature of the control alloy is approximately 650°F at the time the rod is assumed to be insulated (2 sec).

Using the assumptions above, the average temperature of the Ag-In-Cd goes up to 1,035°F, at which time the water level in the core reaches the elevation of the hottest spot on the control rod. The temperature of the rod then rapidly decreases.

The lowest temperature for a eutectic formation is that for Zr-Fe, which occurs at 1,710°F. Therefore, the integrity of the control rod assemblies is maintained during and following a LOCA.

2

model predicts more conservative results than the revised SBLOCA evaluation model. Therefore, analyses performed with the previous version of the SBLOCA evaluation model remain bounding.

## 15.14.5 EVALUATION OF NON-FUEL CORE COMPONENT STRUCTURAL RESPONSE

The temperature transient in the core can produce significantly higher than normal temperatures in components other than fuel rods. Therefore a possibility of eutectic formation between dissimilar core materials exists. Considering the general area of eutectic formation in the entire core and reactor vessel internals, the following dissimilar metals are present, with major elements being in the approximate proportions shown:

Type-304 Stainless Steel

19% chromium

10% nickel

remainder iron

Control Rod Poison Material

80% silver

15% indium

5% cadmium

Zircaloy-4

98% zirconium

1-3/4% tin

Inconel

53% nickel

19% chromium

3% molybdenum

5% Nb-Ta

1% titanium

0.5 % aluminum

remainder iron

All these alloys have relatively high melting points (greater than 2,700°F) except those for silver, cadmium, and indium. The melting point of the silver-indium-cadmium alloy is about 1,470°F.

The binary phase diagram indicates that zirconium in the proportion 75 to 80 percent has a eutectic point with either iron, nickel, or chromium at temperatures of approximately 1,710, 1,760, and 2,380°F, respectively. If these dissimilar metals are in contact and if those eutectic points are reached, then the

1 from 16.5 kW/ft to 16.1 kW/ft for all burnups to reflect an error in the evaluation model associated with
1 the BWC critical heat flux correlation (Reference 34 on page 15-67). The LOCA linear heat rate limits as
1 a function of elevation and burnup are shown in Figure 15-51. Plant operation within these limits assures
1 that the 10CFR 50.46 acceptance criteria are not exceeded. Reference 36 on page 15-67 reports a
1 CRAFT2 cross flow error. This error is offset by using the TACO3 code to specify the initial fuel
1 temperatures. Thus, the LOCA linear heat rate limits in Figure 15-51 are not affected by this error.

2 LOCA limit analysis values for MK-B9 fuel are given in Table 15-27, along with a time sequence of
2 events for ECCS equipment. The results indicate a maximum PCT of 1980°F and a local metal water
2 reaction rate of 2.81 percent with 17.0 kW/ft at the 6-ft elevation.

### 15.14.4.3 Small Break LOCA

1 The SBLOCA is considered to be those break sizes less than 0.5 $ft^2$ and greater than the capacity of the
1 normal makeup system. This corresponds to a minimum break size of approximately 0.0008 $ft^2$. In
0 addition to the cold leg break locations, the HPI line break and core flood line with a maximum break
0 size of 0.44 $ft^2$ were considered. The reference analysis considered three cases; the 0.44 $ft^2$ core flood line break, the 0.5 $ft^2$ pump discharge break, and the 0.04 $ft^2$ pump suction break since it had been previously determined to be the limiting small break (Reference 22 on page 15-66). The results of these analyses determined that the core remained covered and assured that the criteria were met.

Subsequent evaluations determined that the worst case small break should be at the pump discharge rather than the pump suction. A break at the pump discharge could result in one half of the HPI going out the break, and an insufficient flow rate would be delivered to the vessel thereby uncovering the core for an extended period of time. As described in Section 15.14.3.3.6, "ECCS Performance and Single Failure Assumption" on page 15-58, the HPIS was modified in order to deliver a higher flowrate. Additional analyses of SBLOCA were performed taking credit for operator action to balance HPIS flow
2 to both loops within ten minutes (Reference 9 on page 15-66). The HPI flow rates assumed in core flood
2 line, RCP discharge, and HPI line small break LOCA analyses are given in Table 15-28, Table 15-29,
2 and Table 15-30, respectively. HPI flow rates are obtained from Reference 37 on page 15-67. Break sizes of 0.04, 0.055, 0.07, 0.085, 0.10, and 0.15 $ft^2$ were performed at the pump discharge to supplement the reference analyses. Two modifications to the evaluation model were also included in the analyses. The results of the analyses showed that minor core uncovery occurred for the 0.055, 0.07, and 0.085 $ft^2$ breaks. The worst case PCT resulting from the 0.07 $ft^2$ break was 1092°F.

The SBLOCA has been analyzed assuming that the reactor coolant pumps trip and coast down on reactor trip. With no forced flow the liquid in the system would collapse to the lower elevations and an inner vessel mixture level could be tracked to determine the occurrence of core uncovery. For cases in which the pumps remained running, the circulation of the two-phase mixture provided adequate core cooling. Subsequent evaluations of the effect of a delayed pump trip revealed that a pump trip at high system void fractions would result in a collapsed mixture level well below the top of the core (Reference 14 on page 15-66). This situation could only occur for a limited range of break sizes and a certain time window when pump trip would be unacceptable. In order to avoid an inadvertent pump trip in the time window, operating procedures were revised to instruct the operator to trip the reactor coolant pumps on loss of primary system subcooled margin (Reference 15 on page 15-66). Analyses have shown that this action will prevent the criteria of 10CFR 50.46 from being exceeded for any SBLOCA.

The SBLOCA evaluation model has been revised due to NUREG-0737, Section II.K.3.30. NUREG-0737, Section II.K.3.31 requires that analyses are performed with the revised evaluation model to show compliance with 10CFR 50.46. Compliance with 10CFR 50.46 is demonstrated by a qualitative assessment of the SBLOCA spectrum and then a quantitative evaluation of the critical break sizes. The analyses documented in Reference 29 on page 15-67 demonstrate that the previous SBLOCA evaluation

A series of large breaks are analyzed from an initial condition where three or two reactor coolant pumps are in operation. Five possible break locations associated with these modes of operation were identified.

Three pump operation:

1. Break in the idle pump discharge.
2. Break in the active pump discharge of the loop with the idle pump.
3. Break in the pump discharge of the loop with two active pumps.

Two pump operation, one idle pump in each loop:

4. Break in the active pump discharge.
5. Break in the idle pump discharge.

An evaluation was made to determine which LOCA initiating from these partial-loop modes of operation would be most limiting. Two pump operating conditions were judged to be less limiting because of the lower initial power level, 51 percent full power (FP) for two pumps compared to 77 percent FP for three pumps. For the three pump cases, the break in the loop with both pumps operating was determined to be less limiting because the two breaks in the idle pump loop result in more adverse core flows during the transient. Therefore, for cases 1 and 2 above, a double-ended break of 8.55 $ft^2$ and $C_D$ = 1.0 was simulated. The maximum PCT of 1766°F occurred for the break at the pump discharge of the active pump in the idle loop. The response to a LOCA from partial-loop modes of operation was shown to be
0 less limiting than the full power case. It should be noted that power operation with only two reactor
0 coolant pumps in operation is no longer allowed by Oconee Technical Specifications.

### 15.14.4.2 Limiting Linear Heat Rate Analysis (LOCA Limits)

The large break spectrum analysis was performed using an axial peaking factor of 1.7 and a peak linear heat rate of 18 kw/ft in order to determine the worst case break size and location. For the limiting power shape or LOCA limits analysis, the location of the 1.7 axial peak was varied along the length of core at the 2,4,6,8, and 10 feet elevations. A LOCA simulation of the worst case break was run for each power shape case. Then, for each case a maximum allowable linear heat rate which results in a core thermal response within the acceptance criteria of 10CFR50.46 was determined. The results of these analyses are presented in Reference 20 on page 15-66.

The LOCA limit at the 6-ft elevation was reanalyzed to assess the effect of the improved system loop pressure distribution (Reference 21 on page 15-66). Due to enhanced core flow during blowdown, lower metal-water reaction rates, and improved reflooding of the core, the PCT decreased 86°F to 2060°F. This assured that the analyses in Reference 20 on page 15-66 were conservative.

The LOCA linear heat rate limits presented in Reference 20 on page 15-66 have been reduced in order to account for the impact of NUREG-0630 and TACO2 (References 12 on page 15-66 and 31 on page 15-67). The TACO2 code, which has replaced TAFY, is used to determine the initial fuel pin pressure, temperature, and gap dimensions. Implementation of TACO2 has resulted in a decrease in the allowable linear heat rate limit at the 2-ft elevation. The NUREG-0630 cladding rupture model has resulted in a decrease in the allowable linear heat rate limit at the 2, 4, and 6 ft elevations. A modified version of the FLECHT-SEASET reflood heat transfer correlation has been incorporated into the Evaluation Model to partially offset the reduction in the LOCA linear heat rate limit at the 2-ft elevation (Reference 12 on page 15-66). The results of these analyses, given in Table 15-7, show a maximum PCT
1 of 2108°F and a local metal water reaction rate of 2.96 percent with 16.5 kw/ft at the 6-ft elevation. It should be noted that the LOCA limits can be restored to the values given in Reference 20 on page 15-66
0 after a burnup of 1000 Mwd/Mtu. The linear heat rate limit at the 6 foot elevation has been reduced

trains injecting at a later time is more limiting than having one ECCS train injecting at an earlier time. The net effect of the changed single failure assumption is less than a 50°F increase in the peak cladding temperature.

A SBLOCA does not progress as rapidly as a large break LOCA. Thus, for a SBLOCA, the timing of ECCS injection is not as significant as with a large break LOCA. For this reason, the worst single failure for a SBLOCA remains the loss of one bus of emergency power. With the selection of an adverse break location, one half of the available HPI train would inject into the broken loop. With these assumptions the ECCS is reduced to the two CFTs, one LPI train, and one half of one HPI train. For conservatism, no credit is taken for the HPIS for large breaks. For a core flood line break, the available equipment are one core flood tank and HPI train.

For the SBLOCA which does not depressurize to below the core flood tank setpoint (600 psig), only one half of one HPI train was available if the break is assumed to be in the cold leg pump discharge. This was identified as an unacceptable scenario (Reference 16 on page 15-66). In order to deliver the required HPIS flow of 350 gal/min at 600 psig (Reference 17 on page 15-66), the HPIS was modified to allow cross connecting of the pump discharges in order to balance the flow from two HPI pumps into the four injection locations (Reference 18 on page 15-66, 19 on page 15-66).

## 15.14.4 BREAK SPECTRUM ANALYSIS

The LOCA analysis has been performed using the B&W Evaluation Model in accordance with 10CFR50 Appendix K for a complete spectrum of break sizes and locations.

This analysis is given on a generic basis for an Oconee type plant in BAW-10103A Rev. 3, "ECCS Analysis of B&W's 177-FA Lowered-Loop NSS" (Reference 20 on page 15-66).

The reference analysis was revised and expanded to account for an improved system loop pressure distribution (Reference 21 on page 15-66), to change the location of the worst small break from the cold leg pump suction to the pump discharge (Reference 9 on page 15-66), and to examine the impact of a delayed reactor coolant pump trip on SBLOCA (Reference 14 on page 15-66). The effects of these reanalyses on the reference analysis will be presented in the following sections.

### 15.14.4.1 Large Break LOCA

A spectrum of large breaks from 0.5 $ft^2$ up to and including the cross sectional area of the largest pipe in the system was analyzed for both double-ended and longitudinal split breaks in all locations. The methodology used to identify the worst break was as follows. A double-ended break with discharge coefficient $C_D$ = 1.0 was analyzed at the hot leg, cold leg pump suction, and pump discharge. The cold leg pump discharge was determined to be the worst break location. The break size was then varied from 0.5 $ft^2$ to the geometric maximum for both double-ended and split breaks. The results of these analyses are shown in Table 15-6 and Figure 15-50. A symmetric power shape with an axial peaking factor of 1.7 and a peak linear heat rate of 18 kw/ft is assumed.

The worst break was identified as the double-ended cold leg break at the pump discharge with $C_D$ = 1.0. This break of 8.55 $ft^2$ area yielded a predicted PCT of 2079°F and a maximum local metal-water reaction of 4.29 percent. The same break size at the pump suction showed a predicted PCT of 1920°F and a metal-water reaction of 3.04 percent. The largest hot leg break of 14.14 $ft^2$ resulted in a PCT of 1953°F and a metal-water reaction of 3.28 percent. The range of break sizes smaller than the full area double-ended break at the pump discharge all showed less severe consequences.