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6.0 Engineered Safeguards

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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, or in the event of secondary system pipe breaks, 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.

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 electrical power or any external actuated component.

Reactor Building integrity is assured by two 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 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. They also serve to reduce Reactor Building pressure and temperature in the event of a main steam line break.

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. This system is no longer required due to adoption of the alternate source term. The system is not credited for event mitigation and serves an ALARA function only.

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, 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. See also Section 6.3.4.3 for additional considerations related to Engineered Safeguards system operability.

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. Other sections of the report contain information which is pertinent to the Engineered Safeguards Systems. Chapter 7 describes the actuation instrumentation of these systems. Chapter 15 describes the analysis of the Engineered Safeguards Systems' capability to provide adequate protection during accident conditions. Chapter 9 discusses functions performed by these systems during normal operation and gives further design details and descriptive information concerning those systems.

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 most of the borated water storage tank inventory is exhausted, the operators initiate steps to transfer the pump suction to the Reactor Building emergency sump for both the reactor core cooling flow and the Reactor Building sprays. System resistance will maintain Reactor Building Spray pumps flow rates between 700 and 1200 gpm to ensure adequate NPSH during injection and recirculation modes, with no throttling required. 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.

Secuity Related Intormation
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1) The Technical Specification minimum initial levels were used for the BWST and the CFT's, with six feet of level remaining in the BWST at completion of switchover to RBES.

2) Some water is maintained in the Reactor Building atmosphere as vapor. The quantity was determined using the results of a FATHOMS Computer Run for a 14.1 ft^2 break.

3) The break is conservatively assumed to occur at the top of the hot leg, thereby keeping the Reactor Coolant System full.

i. Unit-specific and train-specific models were constructed to maximize the friction losses in the LPI and BS pump suction piping.

Results of the NPSH analysis are presented in Table 6-33. Credit is taken for 0.44 psi of reactor building overpressure in the calculation of available NPSH for the RBS and LPI pumps from approximately 3,000 seconds to approximately 30,000 seconds post-accident.

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.

The NRC issued Generic Letter 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps," on October 7, 1997, requesting that licensees submit information necessary to confirm the adequacy of the net positive suction head (NPSH) available for emergency core cooling (including core spray and decay heat removal) and containment heat removal pumps. A review of the current design-basis analyses used to determine the available NPSH for the applicable pumps was performed and report submitted by the letter from M. S. Tuckman to the NRC, dated January 5, 1998 (Reference 2). This report concluded that the available NPSH for these pumps is adequate under all design-basis accident scenarios. As a result of a subsequent review by Duke, the NPSH calculations were determined to be outside the design basis as discussed in a letter from W. R. McCollum to the NRC, dated Septemer 17, 1998 (Reference 3). A License Admendment was submitted to revise the UFSAR to be consistent with the design basis calculations. License Amendments 305/305/305 approving incorporations of the UFSAR changes to the NPSH bases were issued in a letter from D. E. LeBarge (NRC) to W. R. McCollum (Duke), dated July 19, 1999 (Reference 4).

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.
   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 operation. Boundary valves which have been identified to have leakage paths are tested periodically. All 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, a number of 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.

Source		Quantities
	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 for the LPI, HPI, and BS systems is 12 gallons per hour.

Leakage Analysis Conclusions

It was concluded from analysis of the 12 gph limit (in conjunction with the discussion and analysis in Section 15.15.4) that leakage from Engineered Safeguards Systems outside the Reactor Building does not pose a public safety problem.

6.1.4 Quality Control Standards

Quality Control Standards for the Engineered Safeguards Systems are listed in Table 6-3.

6.1.5 Piping Design Conditions

Piping Design Conditions for the Engineered Safeguards Systems are listed on Table 6-4.

6.1.6 Engineered Safeguards Materials

Materials used in Engineered Safeguards components are addressed in applicable sections where appropriate.

6.1.7 References

1. Nuclear Regulatory Commission, Letter to Holders of Operating Licenses for Nuclear Power Plants, Except Those Who Have Permanently Ceased Operations and Have Certified That Fuel Has Been Permanently Removed for the Reactor Vessel, from Jack W. Roe, October 7, 1997, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps (Generic Letter 97-04)."

2. Duke Power Company, Letter from M. S. Tuckman to the NRC, January 5, 1998, re: Response to Generic Letter 97-04, "Assurance of Sufficient Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps."

3. Duke Power Company, Letter from W. R. McCollum to the NRC, dated September 17, 1998. Re: Response to Request for Additional Information Related to Generic Letter 97-04, dated August 11, 1998.

4. D. E. LeBarge (NRC) to W. R. McCollum (Duke), dated July 19, 1999, re: Issuance of Amendments (License Amendments 305/305/305 for NPSH changes related to Generic Letter 97-04).

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6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.1 Containment Structure

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 0.2 percent by volume in 24 hours under the conditions of the maximum hypothetical accident as described below.

The Reactor Building is designed for an external pressure 3.0 psi greater than the internal pressure. The design external pressure of 3.0 psi corresponds to a margin of 0.5 psi above the differential pressure that could be developed if the building is sealed with an internal temperature of 120°F with a barometric pressure of 29.0 inches of Hg and the building is subsequently cooled to an internal temperature of 80°F with a concurrent rise in barometric pressure to 31.0 inches of Hg. The weather conditions assumed here are conservative since an evaluation of National Weather Service records for this area indicates that from 1918 to 1970 the lowest barometric pressure recorded is 29.05 inches of Hg and the highest of 30.85 inches of Hg.

The principal design basis for the structure is that it be capable of withstanding the internal pressure resulting from a loss-of-coolant accident or a secondary line rupture with no loss of integrity. In a LOCA, the total energy contained in the water of the Reactor Coolant System is assumed to be released into the Reactor Building through a break in the reactor coolant piping. In a secondary line break event the energy contained in the water in the secondary coolant system, as well as energy transferred across the steam generator tubes from the Reactor Coolant System is assumed to be released into the Reactor Building through a break in the steam line piping. However, in the case of a secondary line break, the release of energy essentially stops when the faulted steam generator empties and is no longer being 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
Reactor Coolant Stored Energy	Secondary Coolant Stored Energy
Reactor Stored Energy	Secondary System Stored Energy
Reactor Decay Heat	Reactor Coolant Stored Energy
Metal-Water Reactions	Reactor Stored Energy
Secondary Coolant Stored Energy	Reactor Decay Heat
Secondary System Stored Energy	

6.2.1.1.2 Design Features

Since the design of the Engineered Safeguards Systems and their operation is discussed more fully in Section 6.3, 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.

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.

Following a secondary line break at power, main feedwater and turbine-driven emergency feedwater flow to the faulted steam generator are isolated by the Automatic Feedwater Isolation System (AFIS) on low steam line pressure. AFIS will also isolate motor-driven emergency feedwater to the affected steam generator on a high rate of steam line depressurization concurrent with low steam line pressure. For break sizes that do not exceed the AFIS rate of depressurization setpoint, manual operator action is credited at 10 minutes to isolate motor-driven emergency feedwater flow to the affected steam generator. Section 6.2.1.4.4 discusses AFIS actions during secondary line blowdown into containment. Section 7.9 provides a detailed description of AFIS operation.

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.

The low pressure injection coolers remove heat from the containment sump liquid to the Service Water System with heat exchange through tubes.

Section 3.8 provides a detailed description of the Reactor Building design.

6.2.1.1.3 Design Evaluation

6.2.1.1.3.1 LOCA Short Term Containment Pressure Response

This section provides analyses of the short-term (3 minutes) pressure response of the containment to a spectrum of postulated Reactor Coolant System pipe ruptures. The analyses results provide the bounding post-LOCA containment responses with respect to the containment design pressure, with all input assumptions and boundary conditions chosen to provide a conservatively high containment pressure per the methodology presented in Reference 1. The break size and location of each postulated loss of coolant accident is given in Table 6-21. The pressure and temperature response of the four break location sensitivity studies, Cases 1A through 1D, are given in the following figures:

Figure 6-28 Containment pressure for a 14.1 ft^2 break at the reactor vessel outlet (1A)
Figure 6-29 Containment pressure for a 14.1 ft^2 break at the steam generator inlet (1B)
Figure 6-30 Containment pressure for a 8.55 ft^2 break at the RCP discharge (1C)
Figure 6-31 Containment pressure for a 8.55 ft^2 break at the RCP suction (1D)
Figure 6-32 Containment temperature for a 14.1 ft^2 break at the reactor vessel outlet (1A)

Analysis Method and Computer Codes

The analysis method used in this section is described in Reference 1. The computer codes used in this section are RELAP5/MOD2-B&W (Reference 2) for calculating the mass and energy releases and FATHOMS (Reference 3) 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.

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 equal to the upper Technical Specification limit. Instrument uncertainty is taken into account through operation of the Reactor Building purge, before this upper limit is reached.
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% from the Reference 17 value.
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.

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) temperature response of the containment to a spectrum of postulated Reactor Coolant System pipe ruptures. The analyses results provides the bounding long-term containment temperature response for use in Equipment Qualification (EQ) evaluation of equipment within the Reactor Building. The bounding post-LOCA containment pressure responses are discussed in Section 6.2.1.1.3.1, while the bounding short-term containment temperature results are discussed in Section 6.2.1.1.3.3. The long-term large break containment analysis considers only a single break size and location: a double-ended guillotine break located at the A1 cold leg pump discharge. There is no need to analyze a spectrum of large break locations since a suitably bounding site can be chosen by inspection. The qualitative bases for this position is explained in the following paragraphs. Reference 1 also extensively analyzed the mass and energy releases from and containment response to small break LOCAs. The conclusion of these analyses is that small break LOCAs are not more limiting than large break LOCAs with respect to challenging the containment equipment qualification acceptance criteria.

The basis for choosing a cold leg break as opposed to a hot leg break is obvious once the characteristics of each break are considered. Although it is true that an identical quantity of decay heat will be generated regardless of the break location, the manner in which this energy is partitioned between the vapor and liquid break flow streams is the dominant consideration.

Because the long-term containment response is concerned with temperature in containment as a function of time, it is expected that an energy release profile which is dominated by steam relief will generate a more severe containment response. This is because steam relief to the atmosphere will have a greater impact on containment temperature than if the energy is released primarily in the liquid phase, which has only a slight interaction with the containment atmosphere (convection at the pool surface). Indeed, this observation has been validated with the FATHOMS computer code in numerous analyses. It might appear that the Reactor Building Spray System acts to homogenize the containment atmosphere such that the phase in which the energy is released is insignificant. However, when the complicated interactions between the equipment used to cool the containment atmosphere (Reactor Building coolers and sprays) and the equipment used to cool the containment sump (LPI coolers) are examined by analysis, it is apparent that containment will never become completely homogenized. Therefore, the partitioning of energy released to containment between the vapor and liquid phases is the dominant factor in the long-term containment response.

Due to the geometry of a B&W reactor system, steaming from a cold leg break location will never become completely suppressed. This means that steam will always exit the break no matter how much the decay heat power drops. In contrast, it is possible to completely suppress steaming from a hot leg break site as decay power decreases. Decay heat will eventually be absorbed as sensible heat by the injection fluid and thus steaming from the break will cease. Naturally, when this occurs, decay heat will be transferred to containment in the liquid phase, resulting in a less severe containment response.

The cold leg pump discharge break location is selected rather than the pump suction location. For a pump suction break the cold HPI fluid injected into the broken cold leg pump discharge piping will interact with steam exiting the core through the vent valves and condense a large portion of this steam before it reaches the break. Thus, the steam release will be less for a pump suction break. Consequently, the pump discharge break location is limiting.

An accident chronology is presented in Table 6-24 for the most limiting LOCA, an 8.55 ft^2 double-ended guillotine cold leg break at the reactor coolant pump outlet. Table 6-25 presents results for various combinations of initial Reactor Building temperature, LPSW temperature and RBCU heat removal rate. The results of the limiting case are presented in the following figures:

Figure 6-36 Containment pressure
Figure 6-37 Containment atmosphere temperature
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These cases represent the combinations of minimum and maximum LPSW temperature and RBCU heat removal rate. All cases assume LPI cooler heat removal performance which conservatively bounds that for the coolers at all three Oconee units. The equipment qualification criteria for all equipment required for LOCA mitigation are documented in the Oconee Environmental Qualification Criteria Manual, referred to UFSAR Section 3.11. These criteria are met for all cases analyzed.

Analysis Method and Computer Codes

The analysis method used in this section is described in Reference 1. The computer codes used in this section are RELAP5/MOD2-B&W (Reference 2) for calculating the mass and energy releases during the first 30 minutes, BFLOW for calculating the longer term LOCA mass and energy releases, and FATHOMS (Reference 3) 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.

Initial Condition Assumption Conservatisms

Initial condition assumptions in the containment response analyses are adjusted to give a conservative answer:

1. A nominal initial pressure is used, although this parameter has very little effect due to the long duration of this analysis.
2. The initial temperature assumption is conservatively high for full power operation.
3. The nominal containment free volume is reduced by 2% from the Reference 17 value.
4. A high initial relative humidity is used, although this parameter has very little effect due to the long duration of this analysis.

The initial conditions used are tabulated in Table 6-22

Containment Heat Removal Systems

The Reactor Building cooling units (RBCUs) are modeled based on performance relative to a reference value of 80 million Btu/hour. This reference performance is based on the heat removal rate associated with:

1. Low Pressure Service Water (LPSW) temperature of 75°F
2. Containment air temperature of 286°F
3. Containment air mass characteristic of 110°F and 100% relative humidity

The performance of the two operating coolers (refer to single failure discussion below) is parameterized in terms of the percentage of this reference heat removal rate and the LPSW temperature. A reduction below 80 million Btu/hour reflects degradation in RBCU performance. A colder LPSW temperature is used to enhance performance due to a higher ΔT across the RBCU.

Low Pressure Injection (LPI) Cooler test data at various flow rates are 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 to determine an LPI Cooler overall heat transfer coefficient as a function of LPI inlet temperature. Since assumed cooler degradation, fluid flow rates, and LPSW temperature are constant during a simulation, and since LPI inlet temperature changes during the accident, this change determines the LPI cooler heat removal rate for a particular case. No credit is taken for heat removal by the LPI coolers during the injection phase.

Calculations using hydraulic models of the Oconee Reactor Building Spray (RBS) System result in a minimum flow rate of 750 gpm during the injection phase, taking suction from the Borated Water Storage Tank (BWST). Likewise, minimum flow rates of 900 gpm during the sump recirculation phase (taking suction from the Reactor Building Emergency Sump (RBES)) are demonstrated. In the FATHOMS containment response analyses, the RBS flow rate during injection mode is conservatively assumed to be 700 gpm. During sump recirculation mode, the RBS flow rate is assumed to be 900 gpm. 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.

<u>Emergency Core Cooling Systems</u>

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 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.

Liquid injection from the core flood tanks is not explicitly modeled in FATHOMS but is considered in the mass and energy releases discussed in Section 6.2.1.3. The nitrogen cover gas from these tanks is assumed to be injected to increase the containment pressure calculated by FATHOMS. The amount of injected nitrogen is based on the mass which would be present at the pressure and temperature initial conditions of the mass and energy release calculation.

Assumed values for ECCS equipment performance parameters are given in Table 6-26.

<u>Single Failure</u>

While a component single failure generally has little impact on the peak pressure analysis described in the previous section, it has a much greater impact on the long-term containment response. The most limiting single failure is therefore chosen to yield a conservative long-term containment response. The most restrictive single failure is chosen as the one which disables the greatest number of containment heat removal components.

An evaluation was performed to determine the most limiting single failure with respect to containment cooling. This evaluation indicated that the failure of a 4160V switchgear represents the most limiting single failure. Electrical switchgear power a myriad of safety related equipment including injection systems and containment cooling systems. The failure of one of the three available switchgear will result in the loss of the following components:

1. One HPI pump
2. One LPI pump
3. One RBS pump
4. One RBCU

All other ECCS equipment is available following a nominal, transient-specific actuation delay.

The switchgear failure is more limiting than a loss of offsite power (LOOP) and failure of one Keowee hydroelectric unit because the second hydroelectric unit is available to power the standby busses through CT-4 (underground) or through the switchyard (overhead). Therefore, all ECCS equipment would be available after a small time delay.

Structural Heat Sinks

The structural heat sinks within containment are those described in Section 6.2.1.1.3.1 and tabulated in Table 6-23. For the LOCA long-term containment response calculation the surface areas of these heat structures are reduced by 1% for conservatism.

6.2.1.1.3.3 Steam Line Break Containment Pressure and Temperature Response

This section provides analyses of the pressure and temperature response of the containment to postulated secondary system pipe ruptures. A spectrum of break sizes is analyzed to determine the limiting break size for peak containment pressure and temperature. For peak containment pressure, the response depends mainly on the steam mass flow rate. The limiting break size for peak containment pressure and temperature is the double-ended guillotine break (12.6 ft^2). This break size results in the highest initial rate of mass and energy release to the containment and thus maximizes the increase in containment pressure and temperature during the steam line break transient.

The results of the limiting case are given in Table 6-27. The pressure and temperature response of these limiting cases are given in Figure 6-42 (containment pressure) and Figure 6-43 (containment temperature). The period of time during which the calculated temperature exceeds the equipment qualification limit is very short compared to the time that the equipment is exposed to high temperatures during its qualification testing. This short duration of calculated temperatures above the equipment qualification limit is not long enough to cause the equipment internal temperatures to reach values as high as those reached during the qualification testing.

Analysis Method and Computer Codes

The analysis method used in this section is described in Reference 1. The computer codes used in this section are RETRAN-3D (Reference 9) for calculating the steam line break mass and energy releases and FATHOMS (Reference 3) for calculating the containment pressure and temperature response.

Mass and Energy Release Rate Data

The mass and energy release rate data used for the steam line break analyses described in this section are given in Section 6.2.1.4.

Initial Condition Assumption Conservatisms

Initial condition assumptions in the containment response analyses are adjusted to give a conservative answer:

1. The initial pressure assumption is equal to the upper Technical Specification limit for cases in which high initial pressure is conservative.
2. The initial temperature assumption is conservatively high for full power operation. It is known from the LOCA analyses described in the previous section that a lower initial temperature maximizes the containment peak pressures due to a higher initial air mass. However, a higher initial temperature reduces the cooling capacity of the structural heat sinks, outweighing the impact of a higher initial air mass.
3. The nominal containment free volume is reduced by 2% from the Reference 17 value.
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

The Reactor Building cooling units (RBCUs) are modeled as described in Section 6.2.1.1.3.2. The steam line break peak pressure is reached long before the borated water storage tank empties. Therefore the recirculation phase is not simulated, and no credit is taken for heat removal by the LPI coolers during the injection phase. The RBS is modeled as described in Section 6.2.1.1.3.2 for the injection phase.

Single Failure

The assumed single failure is the same as discussed above for LOCA, the failure of a 4160 V switchgear, resulting in the loss of one HPI pump, one LPI pump, one RBS pump and one RBCU.

Structural Heat Sinks

The structural heat sinks within containment are those described in Section 6.2.1.1.3.1 and tabulated in Table 6-23. For the steam line break containment response calculation the surface areas of these heat structures are reduced by 1% for conservatism.

6.2.1.1.3.4 Functional Capability of Normal Containment Ventilation Systems

Normal containment ventilation is provided by four Reactor Building auxiliary cooling units (RBACUs) and two of the three RBCUs. The function of these units during normal operation is described in Section 9.4.6. Upper and lower limits on containment pressure during normal operation are maintained by complying with the Technical Specifications.

6.2.1.1.3.5 Post-Accident Monitoring of Containment Conditions

Post-accident monitoring instrumentation is provided for the following containment parameters:

Reactor Building pressure
Reactor Building air temperature
Reactor Building normal sump level
Reactor Building emergency sump level
Reactor Building wide range sump level
Reactor Building hydrogen concentration

Section 7.5 discusses the range, accuracy, and response of this instrumentation and the tests conducted to qualify the instruments for use in the post-accident containment environment.

6.2.1.2 Containment Subcompartments

The pressure response of the Reactor Building subcompartments following the design basis LOCA has been evaluated using mass and energy release rates calculated by the CRAFT code (Reference 4) using the system model in Reference 5, with the pressure response calculated by the COPRA code (Reference 6). The Reactor Building subcompartments include the reactor compartment and the east and west steam generator compartments. For each compartment the worst case LOCA break size and location is identified, including the effect of piping restraints on the maximum break size. The flow through the subcompartment vents is calculated using a sonic choking model for a homogeneous steam-water-air mixture, with a vent discharge coefficient of 0.6. A discharge coefficient of 1.0 is used for the system blowdown calculation.

The reactor compartment has a volume of 5520 ft^3, one 6 ft^2 vent flowpath, and concrete shield plugs with a total flow area of 69 ft^2. Only the vent flowpath is assumed to be available for pressure relief. Although the maximum break area within the compartment has been determined to be 3.0 ft^2, hot leg 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 CRAFT mass and energy release rates are given in Figure 6-44 and Figure 6-45. The resulting pressure differential across the compartment walls are shown in Figure 6-46. The peak pressure of 160 psi, which occurs for the 8.0 ft^2 hot leg break, is only 78 percent of the design differential pressure of 205 psi.

The west steam generator compartment has a volume of 61,700 ft^3 and a total vent flow area of 1333 ft^2. The east compartment has a volume of 60,400 ft^3 and a flow area of 1222 ft^2. The discharge coefficients for each of the flowpaths and the effective discharge coefficient calculated to result in the correct choked flow are given in Table 6-28 and Figure 6-47. The maximum hot leg break of 14.1 ft^2 was analyzed using the CRAFT mass and energy release rates in Figure 6-44 and Figure 6-45. The resulting pressure differentials across the compartment walls are shown in Figure 6-48. The structural integrity of the compartments is sufficient to withstand 130 percent of the peak differential pressure of 15 psi.

6.2.1.3 Mass and Energy Release Analyses for Postulated Loss-of-Coolant Accidents

6.2.1.3.1 Short-Term Mass and Energy Release Data

From hot leg and cold leg break studies, the limiting break location for peak containment pressure is the hot leg at the steam generator inlet. Studies have shown that reduced reactor coolant system average temperature and differential cold leg temperature cases do not result in the limiting peak containment pressure response. The short-term LOCA peak pressure mass and energy releases are given in Table 6-29. The short-term LOCA peak pressure mass and energy releases using the replacement steam generators are given in Table 6-35.

6.2.1.3.2 Long-Term Mass and Energy Release Date

The long-term LOCA mass and energy releases calculated by RELAP5/MOD2-B&W for the cold leg injection phase of the transient are given in Table 6-30. The mass and energy releases used by the FATHOMS computer code for periods beyond this point in time are given in Table 6-31.

6.2.1.3.3 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. Fission product and actinide decay power is calculated as a function of time based on the methodology from Reference 7. 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 6.2.1.3.6.

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.

6.2.1.3.3.1 Short-Term Energy Sources

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.

The specific conservatisms used in modeling the stored energy sources considered in the short-term LOCA mass and energy release calculations 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 an initial level corresponding to the maximum Technical Specification allowable value of 285 inches plus an indication uncertainty of 25 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 core flood tank (CFT) temperature is assumed to be a conservatively high value of 130°F. This minimizes available sensible heat capacity of the CFT liquid and therefore maximizes the break steaming rate.
6. 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.
7. The CFT liquid volume is the lower Technical Specification limit less an instrument uncertainty of 38 ft3. 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.
8. The main steam safety valve lift setpoints incorporate 3% drift and 4% blowdown to minimize the potential for steam relief. 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.
9. 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.

10. 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.
11. 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 12.5% 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.
12. Emergency feedwater temperature is at a conservatively high value of 130°F. This maximizes the potential energy release to containment.

6.2.1.3.3.2 Long-Term Energy Sources

The long-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.

The specific conservatisms used in modeling the stored energy sources considered in the long-term LOCA mass and energy release calculations 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 an initial level corresponding to the maximum Technical Specification allowable value of 285 inches plus an indication uncertainty of 25 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 core flood tank (CFT) temperature is assumed to be a conservatively high value of 130°F. This minimizes available sensible heat capacity of the CFT liquid and therefore maximizes the break steaming rate.
6. 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.
7. The CFT liquid volume is the lower Technical Specification limit less an instrument uncertainty of 38 ft3. 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.
8. The main steam safety valve lift setpoints incorporate 3% drift and 4% blowdown to minimize the potential for steam relief. 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.

9. 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.
10. 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.
11. 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 10.0% 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.
12. Emergency feedwater temperature is at a conservatively high value of 130°F. This maximizes the potential energy release to containment. This assumption is only relevant for cases that assume a loss of offsite power. Long-term large break analyses with offsite power conservatively use hotter main feedwater.

6.2.1.3.4 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). The methodology for applying this code is given in Reference 1. RELAP5/MOD2-B&W is used to calculate the mass and energy releases during the cold leg injection period of a postulated LOCA. Beyond this point the BFLOW and FATHOMS (Reference 3) codes are used to calculate mass and energy releases for the remainder of the accident as detailed in Reference 1.

6.2.1.3.5 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.

6.2.1.3.6 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 provides the methodology for modeling this energy source. The energy from the metal-water reaction is also considered in the long-term LOCA mass and energy release analysis.

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 to prevent any predicted liquid entrainment from decreasing the break enthalpy below the enthalpy of dry steam.

A spectrum of break sizes is analyzed to determine the limiting break size for peak containment pressure and temperature. For peak containment pressure, the response depends mainly on the steam mass flow rate.

For peak containment temperature, the response depends on both steam mass flow rate and on steam enthalpy The limiting break size for peak containment temperature and pressure is the double-ended guillotine break (6.305 ft^2). This break size results in a higher initial rate of mass and energy release to the containment and thus maximizes the increase in containment temperature during the steam line break transient.

6.2.1.4.1 Mass and Energy Release Data

The mass and energy release data for the limiting break, a 6.3 ft^2 (34" ID pipe) double-ended guillotine break of a main steam line near 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:

1. 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. Note that this single failure continues to be conservatively assumed in the containment response analysis.
2. There are no steam line isolation valves at Oconee.
3. 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.
4. It is assumed that failure of a feedwater control valve to close on a feedwater isolation signal is beyond the licensing basis.
5. 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. Main feedwater is initially assumed to be in manual control. Upon reactor trip, the ICS reverts to automatic and controls steam generator levels.
7. It is conservatively assumed that offsite power is maintained in order to maximize primary-to-secondary heat transfer and feedwater addition to the affected steam generator.

6.2.1.4.3 Initial Conditions

The criteria presented in Reference 8 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 1315°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-3D 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.
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-3D computer code, described in Reference 9 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 with modifications for the containment mass and energy release calculations as described in Reference 1. 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. Reference 1 also discusses and justifies the conservatisms in this calculational method. Reference 9 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 credits the Automatic Feedwater Isolation System (AFIS) to isolate main feedwater and turbine-driven emergency feedwater to the affected steam generator on low steam line pressure. AFIS will also isolate motor-driven emergency feedwater to the affected steam generator on a high rate of steam line depressurization concurrent with low steam line pressure. For break sizes that do not exceed the AFIS rate of depressurization setpoint, manual operator action is credited at 10 minutes to isolate motor-driven emergency feedwater flow to the affected steam generator.

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 temperature. The assumptions used in these analyses result in a conservatively calculated minimum containment pressure response. This method has been shown to result in the maximum peak clad temperature.

The analysis of the minimum containment pressure response for the reflood analysis is performed using the methodology detailed in Reference 11 with Oconee-specific inputs. The mass and energy release to the Reactor Building and the resulting pressure response for the worst case LOCA, 8.55 ft^2 cold leg break at the pump discharge, is shown in Figure 6-50 (mass releases), Figure 6-51 (energy releases), and Figure 6-52 (pressure response). These figures reflect the analysis for the Mark-B11 fuel design. The containment results for the other fuel designs are similar and therefore not presented.

6.2.1.6 Coating Materials

The original coating materials applied to all structures within the containment during plant construction were qualified by withstanding autoclave tests designed to simulate LOCA conditions. The qualification testing of Service Level I substitute coatings now used for new applications or repair/replacement activities inside containment was in accordance with ANSI N 101.2 for LOCA conditions and radiation tolerance. The substitute coatings when used for maintenance over the original coatings were tested, with appropriate documentation, to demonstrate a qualified coating system.

The original, maintenance, and new coating systems defining surface preparation, type of coating, and dry film thickness are tabulated in Table 3-12 (Containment Coatings).

The elements of the Oconee Coatings Program are documented in a Nuclear Generation Department Directive. The Oconee Coatings Program includes periodic condition assessments of Service Level I coatings used inside containment. As localized areas of degraded coatings are identified, those areas are evaluated for repair or replacement, as necessary.

6.2.2 Containment Heat Removal Systems

6.2.2.1 Design Bases

Two engineered safeguards systems, the Reactor Building Spray System and the Reactor Building Cooling System, are provided to remove heat from the containment atmosphere following an accident.

Portions of the Reactor Building Spray system are credited to meet the Extensive Damage Mitigation Strategies (B.5.b) commitments, which have been incorporated into the Oconee Nuclear Station operating license Section H – “Mitigation Strategy License Condition”.

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.

A high Reactor Building pressure signal of less than or equal to 15 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 an emergency low level, the spray pump suction is transferred to the Reactor Building sump manually 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.

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 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 or low speed automatically stop, then restart in low speed after a 3 minute time delay, and any idle unit(s) is also energized at low speed after 3 minute time delay. The LPSW return header will be isolated during a LOOP by the LPSW RB Waterhammer Prevention System (See Section 9.2.2.2.3). Flow is restored once emergency power is available, which is well before the point in time when the RBCU fans restart.

Performance of the cooling system is monitored by flow and temperature instrumentation in the service water supply and return lines for each cooler; by relative humidity and temperature transmitters in the RBCU ductwork; 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 Table 6-3.

The cooling coils for the RBCU's are constructed in accordance with ASME Section III, Class 3 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 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. These curves are included as representative information only and are not intended as performance commitments. For design purposes, actual performance data should be obtained from manufacturer's certified performance test curves.

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.

RB Spray Headers and Nozzles

For each Unit, there are approximately 60 full cone spray nozzles 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.

RBCU Coolers

The cooling surface of the cooling units has been designed for and satisfactorily tested under simulated 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 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. 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.

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 operating in high or low speed automatically stop and then restart in low speed after 3 minutes and any idle unit(s) is also started at low speed (Section 6.2.2.2) after 3 minute time delay. The fans are tested each refueling outage to verify they can pass the required air flow rate across the coils. 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, 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.

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 15 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. Deleted paragraph(s) per 2006 update.

   LPSW system supplies water to RBACs through a separate piping loop that is independent of the RBCUs. During a LOCA, the RBACs and the piping loop are automatically isolated from the LPSW system by air-operated containment isolation valves (LPSW-1054, 1055, 1061, and 1062) on engineered safeguard (ES) signals.

b. The Low Pressure Service Water valves at the discharge of the coolers go to the full open position. Normally, these valves are operating with an intermediate setting (1200-1400 gal/min per loop).The design service water flow rate to the RBCUs is 1,400 gpm. The flow rate under accident conditions may be less than the design flow; however, sufficient containment heat removal is maintained.

c. The idle cooling unit fan(s) is started; after a 3 minute delay and the remaining fans operating in high or low speed automatically stop and then restart in low speed after a 3 minute time delay. The switch to low speed is required due to the changed HP requirements generated 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. Fusible dropout plates have been completely removed from all units on “A” & “C” RBCU ductwork. This prevents the fans from operating in stalled conditions. On all units, the “B” RBCU ductwork has a dropout plate. This plate has fusible links that will melt and drop the plate to ensure a positive path for recirculation of the Reactor Building atmosphere for the “B” RBCU. See Figure 6-3.

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 reached 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. In addition, the blowout plates are not considered functional.

6.2.2.2.8 Environmental Considerations

None of the electrically operated 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 with the Reactor Building Cooling System, is capable of keeping the containment pressure and temperature within environmental qualification (EQ) limits after a loss-of-coolant or steam line break accident. Assuming a single failure, the post-accident Reactor Building cooling load is provided by two cooling units and the Reactor Building Spray System at one-half capacity. The Reactor Building Spray System and Reactor Building Cooling Systems are designed for long term post-accident operation.

Both the Reactor Building Spray System and the Reactor Building Cooling System, with either at full capacity, are individually capable of maintaining the containment pressure below the design limit following a LOCA or MSLB. This capability satisfies the requirements of the design criteria given in Section 3.1.52.

The Reactor Building Spray System can deliver 700-1200 gal/min per train through the spray nozzles within approximately 119 seconds after the Reactor Building pressure reaches the Reactor Building Spray System actuation setpoint (typical value is 10 psig).

The Reactor Building Cooling System provides the design heat removal capacity with two of 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. The design heat removal capacity for these units is shown in Figure 6-6. The safety analyses given in Section 6.2.1 demonstrate system effectiveness.

6.2.2.4 Tests and Inspection

The active components of the Reactor Building Spray System can be tested as follows:

<u>Reactor Building Spray Pumps</u>

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.

<u>Borated Water Storage Tank Outlet Valves</u>

These valves will be tested in performing the pump test above.

<u>Reactor Building Spray Injection Valves</u>

With the pumps shut down and the pump suction valves closed, these valves can each be opened and closed by operator action. These valves are required to be manually operable (both remote and local) for accident mitigation.

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 are performed prior to initial startup after each refueling outage.

The cooling units will be tested periodically as follows:

a. The fans will 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.

6.2.3 Containment Isolation System

6.2.3.1 Design Bases

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.

Reactor Building Essential and Non-essential Isolation occurs on an Engineered Safeguards signal of 3 psig (4 psig Technical Specification value) in the Reactor Building. Reactor Building Non- essential Isolation occurs on an Engineered Safeguards signal of 1600 psig(1590 psig Technical Specification Value). For details on Reactor Building Essential and Non- essential Isolation, refer to Section 7.3, "Engineered Safeguards Protective System" and Table 7-2 and Table 7-3. Valves which isolate the Reactor Building purge flow path will also be closed on a high radiation signal during the movement of recently irradiated fuel. The Reactor Building sump drain flow path will also be isolated by closing a valve on a high radiation signal. Recently irradiated fuel is fuel that has occupied part of a critical reactor core within the previous 72 hours. The radiation monitor signal is not an Engineered Safeguards signal. Although normally open to the Reactor Building, the Reactor Building Gaseous Radiation Monitor penetrations are not closed on a high 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 that are active to close for containment isolation have 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.

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 valves may be either a check valve and an automatic remotely operated valve, two automatic remotely operated valves, or two check valves, depending upon the direction of normal flow.
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 Building. These valves may be either a check valve and an automatic remotely operated valve, or one check valve and one, normally closed manual valve, or two automatic remotely operated valves, or two check valves, depending upon the direction of normal flow. For piping not part of the process flow, double isolation will be used. One or more of the isolations will be a normally closed manual valve located on the vent, drain, or test connection. The other isolation valve may be located on the process piping.
Type C.	Each line not directly connected to the Reactor Coolant System or not open to the Reactor Building atmosphere has at least one valve, either a check valve or an automatic remotely operated valve. This valve is located outside the Reactor Building. A seismic closed loop forms the inside barrier for most Type C penetrations. Since the Component Cooling System has a non-seismic closed loop, penetrations for this system have an additional automatic remotely operated valve or check valve located inside the Reactor Building. A variation to a non-seismic closed loop piping system inside containment is the LPSW piping to and from the Reactor Building Auxiliary Coolers. Penetrations for this piping system have additional automatic remotely operated valve located outside the Reactor Building. Note that the closed loop piping is actually Seismic Category II, but is not treated as such since it is not QA Condition I that is required for containment boundary items (UFSAR Section 3.1.1.1).
Type D.	Each line connected to either the Reactor Building atmosphere or the Reactor Coolant System, but which is not normally open during reactor operation, has two isolation valves. They may be manual valve(s) with provisions for locking in a closed position, check valve(s), and/or remotely operated valve(s), depending

upon the direction of the normal flow.

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 system. For convenience, each different valve arrangement is shown in Table 6-7 and Figure 6-9 of this 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 presented. It may be noted that only electric motor-operated, manual normally closed, or check valves 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. The valves in the reactor building purge flow path are required to be maintained closed in Modes where the engineered safeguards system is required operable. This is a requirement of NUREG 0737, Item II.E.4.2.6. Therefore Engineered Safeguards system testing of these reactor building purge valves is not required.

As the result of Generic Letter 96-06, the issue of thermal overpressurization of certain containment penetrations was addressed by installation of relief valves, check vavles, or other appropriate devices. Additionally, specific penetration(s) required administrative controls to prevent thermal overpressurization. The NRC accepted Oconee's response to Generic Letter 96-06 in correspondence dated December 6, 2007. A check valve provides thermal overpressurization protection for the piping segment between Units 1, 2, 3LP-1 and 1, 2, 3LP-2.

Fluid penetrations which do not require isolation after an accident are also classified as Type A through D, however the redundant containment isolation provisions described above are not applicable. Such 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.

6.2.3.3 Periodic Operability Tests

Each containment isolation valve will be tested periodically during normal operation or during shutdown conditions to assure its operability when needed. A description of periodic testing programs for containment isolation valves and other penetrations is provided in Section 3.8.1.7.4.

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

Local Leak Detection

These tests are discussed in detail in Section 3.8.1.7.4.

6.2.4.2 Continuous Leakage Monitoring

No continuous Reactor Building leakage monitoring system is provided.

The comprehensive program for preoperational testing, inspection, and postoperational surveillance is described in detail in Section 3.8.

6.2.5 References

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4. BAW-10030, "CRAFT - Description of Model for Equilbrium LOCA Analysis Program", Revision 0, October 1971.
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6. Letter from A. C. Thies (Duke) to A. Schwencer (NRC) dated June 28, 1973.
7. ANSI/ANS-5.1-1979, "Decay Heat Power in Light Water Reactors", American Nuclear Society.
8. ANSI/ANS-56.4-1983, "Pressure and Temperature Transient Analysis for Light Water Reactor Containments", American Nuclear Society, December 1983.
9. NP-7450(A), "RETRAN-3D--A Program for Transient Thermal Hydraulic Analysis of Complex Fluid Flow Systems", Electric Power Research Institute, November 2009.
10. DPC-NE-3000-P-A, "Duke Power Company Nuclear Station Thermal-Hydraulic Transient Analysis Methodology", Revision 3, September 2003.
11. BAW-10192PA, "BWNT Loss of Coolant Accident Evaluation Model for Once-Through Steam Generator Plants", June 1998.
12. Deleted Per 1997 Update
13. Deleted Per 1997 Update
14. OM 235-0517-001, "Oconee Nuclear Station, Reactor Building Cooling Coil Performance Analysis"; rev.1.
15. OSC-5683, "Verification of Aerofin's NUCK Program, RBCU Coil Heat Removal Under Post-LOCA Conditions", rev.0.
16. M. S. Tuckman (Duke) letter dated November 11, 1998 to Document Control Desk (NRC), "Response to Generic Letter 98-04: Potential Degradation of the Emergency Core Cooling System and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and Protective Coating Deficiencies and Foreign Material in Containment," Oconee Nuclear Station, Units 1, 2, and 3, Docket Nos. 50-269, -270, and -287.
17. OSC-300, "Containment Volume and Heat Sink Data," Rev. 8.

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6.3 Emergency Core Cooling System

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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 HPI and LPI portions of 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.

The Core Flooding System is passive in nature, receiving no external actuation signal and requiring no electrical motive power. The check valves in the Core Flooding System are technically active components; however, due to the simplicity and inherent safety of their design, no failure of these components is postulated. Both Core Flooding System Tanks and flow paths are required to function for successful mitigation of large break Loss of Coolant Accidents.

The ECCS is designed to operate in the following modes:

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 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 Core Flood 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.

RBES and Strainer Design Bases and Generic Safety Issue (GSI)-191

The Reactor Building Emergency Sump and strainers at Oconee were originally designed by Babcock and Wilcox (B&W) to meet the requirements of B&W Specification 2036 dated February 10, 1972. The specification provided guidance for which included the following elements:

1. Minimum piping lengths to the Low Pressure Injection (LPI) and Reactor Building Spray (BS) pump suctions to minimize friction losses
2. Free communication with the containment basement areas where most of the water has collected
3. Provisions for missile protection
4. A raised lip around the periphery to prevent refuse and dirt from entering the sump
5. Raised sump outlet lines to prevent dirt and debris from entering the recirculation piping
6. Consideration of adequate submergence of the sump outlets to prevent vortex formation and air entrainment
7. Sloping of the sump outlet lines to avoid the entrapment of air

8. Sufficient elevation of the sump above the LPI and BS pump suctions to provide adequate NPSH considering minimum water level, minimum subcooling, maximum piping friction, runout pump flow conditions, and Safety Guide 1 safety margin
9. Adequate provision for draining
10. Protective covering to protect personnel during normal operation, prevent large debris from entering during accident conditions, and allowance for personnel access for maintenance
11. A coarse mesh screen (floor grating) and a fine mesh screen
12. A free flow area that will allow a maximum fluid velocity of one foot per second with a 50% blockage of screen area
13. A vertical screen orientation to promote "self cleaning"
14. Screen cover and supports designed to withstand earthquake loading and prevent collapse from water pressure due to a blockage of 50% of the screen area

In 1979, as a result of industry operating experience and evolving staff concerns about the adequacy of emergency sump designs, the NRC opened Unresolved Safety Issue (USI) A-43, "Containment Emergency Sump Performance". To support the resolution of USI A-43, the NRC undertook an extensive research program, the technical findings of which are summarized in NUREG-0897, "Containment Emergency Sump Performance," dated October 1985. The resolution of USI A-43 was subsequently documented in Generic Letter (GL) 85-22, "Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage", dated December 3, 1985. Through the resolution of USI A-43, the NRC found that the 50 percent blockage assumption (under which most nuclear power plants had been licensed) identified in Regulatory Guide (RG) 1.82, Rev. 0, "Sumps for Emergency Core Cooling and Containment Spray Systems," dated June 1974, should be replaced with more comprehensive guidance. Events at Boiling Water Reactors (BWRs) subsequently challenged the NRC's conclusion that no new requirements were necessary to prevent clogging of ECCS strainers at operating BWRs. A number of generic communications were issued to address this issue, with a focus on BWR plants. Following the successful resolution of the issue with BWRs, the research conducted at the time raised questions concerning the adequacy of PWR sump designs to prevent clogging. These findings prompted the NRC to open GSI-191, "Assessment of Debris Accumulation on PWR Sump Performance" and subsequently led to NRC issuing Bulletin 2003-01, "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors" on June 9, 2003, and GL 2004-02 "Potential Impact of Debris Blockage on Emergency Recirculation During Design Basis Accidents at Pressurized Water Reactors" on September 13, 2004 as a follow-on to the bulletin. The GL requested licensees to perform a new, more realistic analysis of sump performance using an NRC-approved methodology and confirm the functionality of the ECCS and Containment Spray System (CSS) during design basis accidents requiring containment sump recirculation. Through the Nuclear Energy Institute (NEI), industry developed evaluation guidance which was issued as NEI 04-07, "Pressurized Water Reactor Sump Performance Evaluation Methodology" dated November 19, 2004. This guidance document was subsequently incorporated in its entirety, with NRC amendments, and issued in a Safety Evaluation Report dated December 6, 2004.

The approved evaluation methodology requires evaluation of sump performance to include the following elements:

1. Break selection to maximize debris
2. Debris generation evaluation, including conservative Zone of influence and debris characteristics

3. Latent debris evaluation
4. Debris transport evaluation
5. Head loss evaluation
6. Sump structural analysis
7. Vortex evaluation
8. Upstream effects evaluation
9. Downstream effects evaluation in accordance with WCAP 16406-P, Rev. 1, Evaluation of Downstream Sump Debris Effects in Support of GSI-191
10. In-vessel effects evaluation in accordance with WCAP 16793-NP, Rev 0, Evaluation of Long-Term Cooling Considering Particulate, Fibrous, and Chemical Debris in the Recirculating Fluid
11. Chemical effects evaluation in accordance with WCAP 16530-NP, Rev. 0, Evaluation of Post-Accident Chemical Effects in Containment Sump Fluids to Support GSI-191 (Minor deviations from the WCAP were discussed in Duke's GL response dated February 29, 2008.)

Oconee has performed all GL 2004-02 required analyses and evaluations in accordance with the approved methodology. As a result of these evaluations, Oconee made a number of plant modifications and programmatic enhancements, such as:

1. Modification of the Reactor Building Emergency Sump strainers on all three units to increase the strainer surface area. The original surface area was approximately 100 square feet on all units. The current surface area is approximately 4800 square feet on Unit 1 and approximately 5000 square feet on Units 2 and 3.
2. Replacement of HPI pump internals to provide more wear-resistant materials.
3. Replacement of the seal flush orifices and cyclone separators on the HPI, LPI, and BS pumps.
4. Removal of fibrous insulation from areas in containment where it would be potentially affected by a pipe break jet (Zone of Influence or ZOI).
5. Enhancement of plant labeling process to limit potential for tags and stickers to become post-accident debris sources.
6. Enhancement of plant containment coatings program to ensure that degraded coatings identified from maintenance inspections are evaluated for potential effects on RBES evaluations.
7. Enhancement of Foreign Material Exclusion (FME) controls to ensure that any scaffolding remaining in containment during power operation is evaluated for potential chemical effects.
8. Enhancement of plant design change process to ensure that plant modifications are evaluated for impact to RBES evaluations performed in support of GSI-191.
9. Revision to plant Technical Specifications to remove reference to trash racks and screens and add reference to strainers.

A single open item remains to close the GL for Oconee, as the NRC has not yet issued an SER on WCAP-16793-NP. Oconee will address any forthcoming changes to the WCAP when the SER is issued.

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.

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. The High Pressure Injection System is initiated at: (a) a low Reactor Coolant System pressure of 1,600 psig (1590 psig Technical Specification Value) or (b) a Reactor Building pressure of 3 psig (4 psig Technical Specifications 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 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.

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.

In addition to the automatic action described, the pumps and valves may be remote manually operated from the control room.

Operation of the High Pressure Injection System in the emergency mode will continue until the system action is manually terminated. The HPI system is not designed to withstand a single passive failure since the duration of system usage during an accident is not considered to be long term; however, the portion of HPI system piping which is used to return any LPI-to-HPI system leakage to the Reactor Building Emergency sump is evaluated for passive failures since this portion of HPI piping could be utilized during long term cooling following a LOCA.

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.

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. Deleted row(s) per 2002 Update.

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.

Low pressure injection is accomplished through two separate but cross-connected 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. Each pump has minimum flow recirculation loop to protect the pumps from dead-heading. The orifice in the minimun flow recirculation loop was resized in response to NRC Bulletin 88-04 (Potential Safety-Related Pump Loss). The new minimum flow rate, along with procedure guidance for low flow conditions directs operators to take appropriate actions to protect the pump near shut-off head conditions. All ECCS analysis were met with the new orifices.

The initial emergency operation of the Low Pressure Injection System involves pumping water from the borated water storage tank into the reactor vessel. With all ES actuated 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, the operators initiate steps to transfer the pump suction to the Reactor Building emergency sump, permitting recirculation of the spilled reactor coolant and injection water from the Reactor Building emergency sump.

For certain small break LOCA's where the RCS pressure remains above LPI shut-off pressure, the BWST will deplete and the LPI will be aligned in series with the HPI pumps to provide recirculation until the RCS is depressurized (piggy-back mode). In this case, there are short durations where the LPI flowrates are expected to be less than the manufacturer's minimum flow requirements. Appropriate procedures have been revised to alert the operators of the minimum flow conditions and to take appropriate actions. This case would only exist during the transition period before long term cooling is supplied by LPI, at which time, flowrates would satisfy the manufacturer's long term requirements.

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 transfer. The LPI system provides two redundant gravity flow paths from the reactor outlet piping to the Reactor Building Emergency Sump (RBES) to maintain continuous liquid flow through the core and assure post-LOCA boric acid solubility. Additionally, the design of the reactor vessel and vessel internals around the hot leg nozzles provides a third path that can assure post-LOCA boric acid solubility. At least 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 and large Reactor Coolant System pipe failures. It automatically floods the core when the Reactor Coolant System pressure drops below approximately 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 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 Boiler and Pressure Code, Section III and Section VIII, and the TEMA-R 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 accordance with the intent of the ASME Nuclear Power Piping Code B31.7 or FSAR Section 3.2.2.2 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.

6.3.2.3.5 Coolant Storage

The letdown storage tank has a total coolant volume of 600 ft^3 and normally contains approximately 3,200 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.

Provisions are made for sampling the water and adding concentrated boric acid solution or demineralized water.

Each core flooding tank contains approximately 7,000 gallons of borated water with a boron 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 injection 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 HPI and LPI pumps.

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 temperature is illustrated in Figure 6-18. Note that this figure is representative in nature and is provided for information only. It is not intended to constitute design commitments or performance requirements for the coolers.

6.3.2.3.8 Relief Valve Settings

Relief valves are provided to protect the low pressure injection piping and components from overpressure. On Units 1 and 2 these relief valves will be set at 370 psig, the system design pressure at 300°F for the "B" LPI Coolers and at 515 psig, the system design pressure at 250°F for the "A" LPI Coolers. On Unit 3 the 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 manufactured to the Codes and Standards in Table 6-3 or FSAR Section 3.2.2.2 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 of stainless steel, for those portions in contact with borated water. With the exception of the borated water storage tank, the major components in low pressure injection are constructed of stainless steel, for those portions in contact with borated water. 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 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 equipment sizes in the ECCS is demonstrated by the post-accident performance analysis described in Chapter 15.

6.3.2.6.1 High Pressure Injection Operability

A cross connect line with electric operated valves (HP-409 and HP-410, both normally closed) is installed between the "A" and "B" headers to ensure that two paths and two pumps can be aligned to inject to the RCS. During full power operation, two pumps through two trains must be available during an accident to ensure adequate flow reaches the core. In the case of either of valves HP-26 or HP-27 failing to open during an accident situation, the cross connect via valve HP-409 or HP-410 would be utilized to provide flow through the 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

On January 2, 1973, the AEC requested that Duke Power Company determine an acceptable program that would demonstrate operability of active valves. The testing was to simulate conditions associated with normal system operation as well as loading conditions that would appropriately demonstrate seismic and accident vibratory responses. The AEC request was further clarified by stating the "the test program may be based upon selectively testing a representative number of active valves in the piping system according to valve type, seismic and accident load level, size, etc. on a prototype basis". On May 1, 1973, Duke Power Company responded by adding a supplement (Supplement 15) to the FSAR which described various testing (environmental, vibrational, life cycle, etc.) on a subset of active valves. From a historical perspective, the request by the AEC predated the formalization of the established programs and testing requirements which are currently in place that ensure that active valves properly function during normal and post accident conditions. Such programs include, but are not limited to, the following examples: Environmental Qualification (EQ) program (10CFR50.49), MOV Testing and Periodic Verification program (GL 89-10 and GL 96-05), Inservice Testing Program (10CFR50.55a and GL 89-04), Inservice Inspection Program (10FR50.55a), Containment Leakage Program (10CFR50 Appendix J), and Quality Assurance Program (10CFR50 Appendix B). Prior to the formulation and development of such programs, the AEC's request was relevant. However, with the current programs in place, the AEC's 1973 request is deemed historical in nature. The final intent of the request, which was assurance of the operability of active valves, is deemed to be included within current requirements associated with the design, maintenance, and testing of active components.

6.3.2.6.3.1 Deleted per 1999 Update

6.3.2.6.3.2 Deleted per 1999 Update

6.3.2.6.3.3 Deleted per 1999 Update

6.3.2.6.3.4 Deleted per 1999 Update

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 an earthquake discussed in Section 2.5.2.6.

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 there are no possible missile trajectories that could impact the function of the sump suction.

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.

The major 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:

a. Reactor Coolant System pressure transmitters.

b. Reactor Building isolation valves and associated position indications.

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.

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Environmental conditions in the Auxiliary Building are controlled to ensure proper operation of ECCS pumps during accident conditions. Operability of HPI, LPI, and RBS pumps is dependent upon initial pump room temperature and the availability of natural convection flow paths as described in UFSAR Section 9.4.3.

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. A complete listing of the equipment which is evaluated for environmental qualification can be found in the Oconee Nuclear Station Equipment Data Base.

Current material qualification for these components is addressed by the Environmental Qualification (EQ) program discussed in Section 3.11. The Oconee Nuclear Station Environmental Qualification Maintenance Manual, EQMM-1393.01, lists the requirements for maintaining equipment qualifications, and is a major element of the Oconee Nuclear Station EQ Program.

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. For full power operation, the safety analysis in Chapter 15 has shown that two high pressure injection 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.

After receiving an actuation signal, the HPI system valves for injection will open sufficiently to admit the 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

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After receiving an actuation signal, the low pressure injection valves will reach full open within 36 seconds and the low pressure injection pumps will reach full speed within 5 seconds.

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 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.

6.3.3.2.1 Boron Precipitation Evaluation

In response to the RCS depressurization associated with a LOCA, the ECCS actuates and begins injecting borated water into the system to reflood the core, keep the reactor subcritical, and provide for long term cooling. The boiloff of the ECCS delivered water along with flow stagnation in the reactor vessel can result in an increase in the boron concentration. If unrealized, this process could result in localized recrystallization of the boric acid and the potential for deposits to build up on the fuel assemblies and internals and hinder effective heat removal. In order to prevent this occurrence, analytically based operating procedures have been developed to assure sufficient circulation and dilution of the coolant.

In the initial long term phase of post-LOCA heat removal, a natural circulation flowpath from the core through the vent valves to the downcomer occurs which sufficiently circulates the coolant through the core. At some point in time the flowpath through the vent valves will no longer be available as the decay heat becomes insufficient to drive the flow. In addition, natural circulation flow through the gaps between the reactor vessel hot leg nozzles and the reactor internals has also been evaluated to be available. Operator action must be taken to initiate at least one of the two gravity flow paths to provide further assurance that flow is established and post-LOCA boric acid solubility is maintained. The method for performing this function is by means of a drain line from the hot leg to the Reactor Building sump which draws coolant from the top of the core, thereby inducing core circulation. It should be noted that the opening of the primary and alternate boron dilution flow paths must be limited by RCS conditions to prevent damage of the drain flow path, damage of the RBES, and flushing in the LPI pump suction piping (Reference 9 and 10). The system has been designed with redundant drain lines and has been shown to be single failure proof. The boron concentration of the liquid leaving through the drain line is equal to the core boron concentration. Most of the core decay heat is removed by steam flow through the vent valves. ECCS pump flow will continue to be provided to the RCS cold legs and will preclude any boron concentration buildup in the vessel for breaks in the hot leg.

An analysis has been performed to determine the allowable time for the operator to align the post-LOCA boron dilution drain line to prevent unacceptable boron concentrations in the reactor vessel. The analysis determines the rate at which boron concentrates in the reactor vessel following a large cold leg break LOCA with conservative assumptions regarding decay heat, vessel mixing volume, vent valve flow, containment pressure, LPI injection flow and temperature, and initial boron concentrations in the RCS, BWST and core flood tanks. The values of these parameters are given in Table 6-20. The analysis credits a conservative minimum flow through the reactor vessel internals vent valves as predicted by the BFLOW code methodology. The BFLOW code is described in Reference 6.

The results of the analysis show the maximum allowable boric acid concentration establishd by the NRC, which is the boric acid solubility limit minus 4 weight percent, will not be exceeded in the vessel if a boron dilution flow of 40 gpm (Reference 7) from the hot leg to the sump is initiated within 9 hours following a LOCA.

Since there are redundant methods to establish this dilution flow, no diverse means is required to be provided to prevent the buildup of boron concentration. All components of the ECCS are ANS Safety Class 2 and Seismic Category 1.

6.3.3.3 Loss of Normal Power Source

Following a loss-of-coolant accident assuming a simultaneous loss of normal power sources to the LOCA unit, the emergency power source and the Low Pressure Injection Systems will be in full operation within 74 seconds after actuation, even assuming a single failure, and the High Pressure Injection System will be in full operation within 48 seconds after actuation. The electrical power system design is based on the assumption that ESG actuation in one unit occurs simultaneously with a loss of offsite power to all three units. However, accident scenarios in FSAR Section Chapter 15 assume loss of offsite power to the LOCA unit only. Except for large break LOCA (as described in UFSAR Section 15.14.3.3.6), all calculations for the Oconee Units have assumed a 48 second delay from receipt of the actuation signal to start flow for the HPI system and a 74 second delay for the LPI System. 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). 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, to deliver required flow, and 36 seconds respectively at normal bus voltage) and pumps after a 10 second swapover time (required by the single failure), the design injection flow rate (HPI - 450 gal/min, LPI - 3000 gal/min) will be obtained within 48 and 74 seconds, respectively.

6.3.3.4 Single Failure Assumption

UFSAR Section 15.14.3.3.6 discusses ECCS performance and the single failure assumption.

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 inspected for leaks. Items for inspection will be pump seals, valve packing, flange gaskets, heat exchangers, and safety valves for leaks to atmosphere.

6.3.4.3 Gas Management

On January 11, 2008 the NRC issued Generic Letter (GL) 2008-01 to address the issue of gas management in the ECCS, decay heat removal, and containment spray systems. The GL requested licensees to evaluate their licensing basis, design, testing, and corrective action program and to submit information to demonstrate that the subject systems are in compliance with current licensing and design bases and applicable regulatory requirements, and that suitable design, operational, and testing control measures are in place for maintaining this compliance. Oconee Nuclear Station performed the required evaluations for the HPI, LPI, CF, and BS systems, and took the following additional actions:

1. Identified potential gas sources for the systems identified in the GL
2. Identified additional sites where gas accumulation potential exists
3. Performed ultrasonic testing to quantify gas or verify water solid conditions at selected locations where both a potential source and acummulation site were identifed
4. Implemented addional periodic monitoring using techniques to quantify gas
5. Established quantified acceptance criteria for surveillance activities/procedures
6. Added requirements for entry into the corrective action program for gas findings in excess of acceptance criteria
7. Added vent valves in selected locations fo facilitate gas removal
8. Made changes to system fill and vent, startup, operating, maintenance, and test procedures to minimize the potential for introducing gas and to facilitate its effective removal

9. Performed Engineering evaluation of gas which could not be readily removed
10. Strengthened trending of gas accumulation rate to ensure surveillance frequency is adequate

These gas management activities demonstrate that Oconee recognizes the presence of gas in the ECCS, DHR, and BS system as a condition adverse to quality. Oconee responded to the GL by letter describing the results of the required evaluations (Reference 11). The actions described above and in the GL response ensure that Oconee remains in compliance with applicable regulations and the licensing basis as it applies to these systems.

6.3.5 Instrumentation Requirements

The High Pressure Injection System is actuated automatically by a low Reactor Coolant System pressure of 1,600 psig (1590 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. Flow instrumentation is available in each HPI train during an 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. 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.

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. Deleted Per 1997 Update
4. Deleted Per 1997 Update
5. Instruction Manual for Rotork Valve Actuators, OM-245-1023.'
6. DPC-NE-3003-PA, "Duke Energy Corporation Oconee Nuclear Station Mass and Energy Release and Containment Response Methodology", Revision 1b, October 2013.
7. Jones, R. C., Biller, J. R., Dunn, B. M., ECCS Analysis of B&W's 177-FA Lowered-Loop NSSS, Babcock & Wilcox, BAW-10103 Rev. 3, July 1977.
8. Qualification Test Report for Two Valve Operators (11NAZT1 and 90NAZT1) for Rotork Controls, Inc., Report No. 43979-1, Revision A, December 1978.
9. OSC-4678 Boron Dilution Line Discharge Velocity Evaluation.

10. OSC-3862 Uncertainty Estimation for ICCM and OAC Subcooled Margin Indication.
11. Letter from Thomas P. Harrall to U.S. Nuclear Regulatory Commission dated October 13, 2008.

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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 1O CFR 50, however control room habitability was a design consideration at Oconee as discussed in Section 3.1.11.

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.

In 2004 Duke implemented changes to the control room unfiltered air inleakage assumptions as a result of new test data and adopted a revised analysis methodology using the Alternate Source Term. A new licensing basis was established, and re-analysis of the control room radiological consequences of design basis accidents was provided to the NRC (Reference 4).

Duke performed a review of plant shielding to provide for adequate access to vital areas as a result of NUREG-0737, Item II.B.2.2. The review resulted in plant modifications and corrective actions to improve shielding. Based on review of the plant shielding design review, inspection of plant modifications and corrective actions, and performance of an independent assessment of vital area accessibility and personnel doses in a post-accident condition, NRC concluded that the requirements of NUREG-0737, Item II.B.2.2 have been met and are acceptable (Reference 3).

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 Zone and is comprised of the Control Room, 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

The Control Room Ventilation System is described in detail in Section 9.4.1. The ventilation system was designed and installed in accordance with HVAC Industry Standards and practices for commercial and industrial systems. The CRV system is not an "Engineered Safeguards" (ES) System.

6.4.2.3 Leak Tightness

Outside air filter trains are provided as part of the Control Room Ventilation System to provide filtered pressurization air to offset the exfiltration from the control room zone. This minimizes uncontrolled infiltration into the control room zone by creating a positive pressure with respect to adjacent zones.

The Oconee 1 and 2, and Oconee 3 Control Room Ventilation Systems are designed as independent ventilation systems; Two 50% capacity outside air filter trains can maintain their respective control room zones at a positive pressure to prevent 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 Buildings are separate from the Control Room Ventilation System.

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 for disinfection of raw water. Other gases used on site are Ammonia, Hydrazine, Hydrogen, Liquid Nitrogen, and welding gases. Protection of control room operators against potential toxic gas release accidents has been found to be adequate by the NRC (Reference 1).

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.

Greenville Water Works utilizes chlorine at the Adkins Water Treatment plant on Lake Keowee. Potential accidents at this facility have been evaluated and determined not to impact Oconee based on Regulatory Guide 1.78 and 1.95 guidance (Reference 2).

6.4.3 Testing and Inspection

The Control Room Ventilation System is normally operable and is accessible for periodic inspection. The pressurization portion of the system is tested periodically to demonstrate its readiness and operability as required by the Technical Specifications. Temperatures in the Control Rooms, Cable Rooms, and Electrical Equipment Rooms are periodically surveyed, as required by SLC's to ensure the CRVS is functioning properly.

6.4.4 Instrumentation Requirements

Sufficient indications in the form of status lights and performance readouts are provided in the control room to evaluate system operation and indicate system malfunctions.

A radiation monitor is located in the return air side of the Control Room Ventilation System as described in Section 9.4.1.1.

A chlorine detector is located in the Outside Air Intake duct of each Control Room Ventilation System Booster fan as described in Section 9.4.1.2.

6.4.5 References

1. J. F. Stolz (NRC) to H. B. Tucker (Duke) November 24, 1986.
2. OSC-6206, Evaluation of Potential Off-Site Toxic Gas Releases.
3. J.F.Stolz (NRC) to H.B. Tucker (Duke) April 6, 1983.
4. Leonard N. Olshan (NRC) to Mr. Ronald A. Jones (Duke), June 1, 2004, Issuance of Amendments 338, 339, and 339 incorporating changes resulting from use of an alternate source term.

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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. This system is no longer required due to adoption of the Alternate Source Term, Reference 2, and serves an ALARA function only.

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) has shown that Reactor Building leakage is more likely at penetrations than through the liner plates or weld joints.

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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.

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 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.

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:

1. Reactor Building Fuel Transfer Tube and Reactor Coolant SSF Makeup (Penetration No. 11a).
2. Reactor Building Fuel Transfer Tube and Reactor Coolant SSF Letdown (Penetration No. 12a).
3. One Main Steam Line per unit --1B (Penetration No. 28) and 2A & 3A (Penetration No. 26).
4. Normal Personnel Access Lock (Penetration No. 90).

5. Permanent Equipment Hatch which contains a double-gasketed closure (Penetration No. 91).
6. Emergency Personnel Access Lock (Penetration No. 92).
7. Reactor Building Normal Sump Drain (Penetration No. 5a).
8. Reactor Coolant Post-Accident Liquid Sample Lines (Penetration No. 5b).
9. Reactor Coolant Quench Tank Drain (Penetration No. 29).
10. Reactor Building Emergency Sump Recirculation A (Penetration No. 36).
11. Reactor Building Emergency Sump Recirculation B (Penetration No. 37).
12. Reactor Building Emergency Sump Drain (Penetration No. 40).
13. Reactor Coolant Decay Heat Drop Line and Post-Accident Boron Dilution Line (Penetration No. 62 -- Units 2 and 3 only).

Of the listed penetrations, line items 7 through 13 are embedded lines.

The above lines, including 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

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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.

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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).

The equipment in this system is designed and rated in accordance with the following standards:

<u>Pre-Filter</u>- Filter efficiency is determined by the "American Filter Institute Dust Spot Test" utilizing atmospheric dust.

<u>Absolute Filter</u>- 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-3803-1989. 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 may be tested if required to demonstrate 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. External carbon sample canisters are installed on the filters to facilitate sampling if required. Provision is made to check penetration room pressure relative to either the Auxiliary Building or the outside.

Deleted paragraph per 2015 Update.

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.

6.5.1.6 Materials

Carbon steel and suitable coatings are used to obtain desired service life.

6.5.2 Containment Spray Systems

Credit is taken for this system for fission product removal in the MHA off-site dose analyses only. (see 15.15.1).

6.5.3 Deleted per 2001 Update

6.5.4 References

1. Cottrell, W. B. and Savolainen, A. W., Editors, U. S. Reactor Containment Technology, *ORNL-NSIC-5, Volume II*.

2. License Amendment No. 338, 339, and 339 (date of issuance – June 1, 2004); Adoption of Alternate Source Term.

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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.

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 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.

6.6.7 System Pressure Tests

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.

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.

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