Table of Contents

9.0 Auxiliary Systems

9.1 Fuel Storage and Handling
9.1.1 New Fuel Storage
9.1.2 Spent Fuel Storage
9.1.2.1 Spent Fuel Storage - Oconee 1, 2
9.1.2.1.1 Design Bases
9.1.2.1.2 Design Description
9.1.2.2 Spent Fuel Storage - Oconee 3
9.1.2.2.1 Design Bases
9.1.2.2.2 Design Description
9.1.2.3 System Evaluation
9.1.2.3.1 Structural and Seismic Analysis
9.1.2.3.2 Criticality Analysis
9.1.2.3.3 Material, Construction, and Quality Control
9.1.2.3.4 Interface of High Capacity Fuel Storage Rack and Spent Fuel Storage Pool
9.1.2.4 Safety Evaluation
9.1.2.5 Boraflex
9.1.3 Spent Fuel Cooling System
9.1.3.1 Design Bases
9.1.3.1.1 Units 1 and 2 Spent Fuel Pool Cooling System
9.1.3.1.2 Unit 3 Spent Fuel Pool Cooling System
9.1.3.2 System Description
9.1.3.3 System Evaluation
9.1.3.3.1 Normal Operation
9.1.3.3.2 Failure Analysis
9.1.3.4 Safety Evaluation
9.1.4 Fuel Handling System
9.1.4.1 Design Bases
9.1.4.1.1 General System Function
9.1.4.1.2 New Fuel Storage
9.1.4.1.3 Spent Fuel Pool
9.1.4.1.4 Fuel Transfer Tubes
9.1.4.1.5 Fuel Transfer Canal
9.1.4.1.6 Fuel Handling Equipment
9.1.4.2 System Description and Evaluation
9.1.4.2.1 Receiving and Storing Fuel
9.1.4.2.2 Loading and Removing Fuel
9.1.4.2.3 Safety Provisions
9.1.5 Overhead Heavy - Load Handling Systems
9.1.5.1 Introduction and Licensing Background
9.1.5.2 Design Basis
9.1.5.3 Scope of Heavy Load Handling Systems
9.1.5.4 Control of Heavy Lifts Program
9.1.5.4.1 Oconee Commitments in Response to NUREG-0612, Phase 1 Elements
9.1.5.4.2 Oconee Response to NEI Initiative on Heavy Load Lifts
9.1.5.5 Safety Evaluation
9.1.6 References

9.2 Water Systems
9.2.1 Component Cooling System
9.2.1.1 Design Bases

9.2.1.2 System Description and Evaluation
9.2.1.3 Mode of Operation
9.2.1.4 Reliability Considerations
9.2.1.5 Codes and Standards
9.2.1.6 System Isolation
9.2.1.7 Leakage Considerations
9.2.1.8 Failure Considerations
9.2.2 Cooling Water Systems
9.2.2.1 Design Bases
9.2.2.2 System Description and Evaluation
9.2.2.2.1 Condenser Circulating Water System (CCW)
9.2.2.2.2 High Pressure Service Water System (HPSW)
9.2.2.2.3 Low Pressure Service Water System (LPSW)
9.2.2.2.4 Recirculated Cooling Water System (RCW)
9.2.2.2.5 Essential Siphon Vacuum and Siphon Seal Water Systems
9.2.3 Auxiliary Service Water System
9.2.3.1 Design Basis
9.2.3.2 System Description
9.2.4 Ultimate Heat Sink
9.2.5 Control Room Ventilation Chilled Water System (WC)
9.2.5.1 Design Basis
9.2.5.2 System Description and Evaluation
9.2.6 References

9.3 Process Auxiliaries
9.3.1 Chemical Addition and Sampling System
9.3.1.1 Design Bases
9.3.1.2 System Description and Evaluation
9.3.1.2.1 Mode of Operation
9.3.1.2.2 Reliability Considerations
9.3.1.2.3 Codes and Standards
9.3.1.2.4 System Isolation
9.3.1.2.5 Leakage Considerations
9.3.1.2.6 Failure Considerations
9.3.1.2.7 Deleted Per 2001 Update
9.3.2 High Pressure Injection System
9.3.2.1 Design Bases
9.3.2.2 System Description and Evaluation
9.3.2.2.1 Mode of Operation
9.3.2.2.2 Reliability Considerations
9.3.2.2.3 Codes and Standards
9.3.2.2.4 System Isolation
9.3.2.2.5 Leakage Considerations
9.3.2.2.6 Failure Considerations
9.3.2.2.7 Operational Limits
9.3.3 Low Pressure Injection System
9.3.3.1 Design Bases
9.3.3.2 System Description and Evaluation
9.3.3.2.1 Mode of Operation
9.3.3.2.2 Reliability Considerations
9.3.3.2.3 Codes and Standards
9.3.3.2.4 System Isolation
9.3.3.2.5 Leakage Considerations
9.3.3.2.6 Operational Limits
9.3.3.2.7 Failure Considerations
9.3.4 Coolant Storage System

9.3.4.1 Design Bases
9.3.4.2 System Description and Evaluation
9.3.5 Coolant Treatment System
9.3.6 Post-Accident Sampling System
9.3.6.1 Post-Accident Liquid Sampling System
9.3.6.1.1 Design Bases
9.3.6.1.2 System Description and Evaluation
9.3.6.1.3 Deleted Per 2005 Update
9.3.6.2 Post-Accident Containment Air Sampling System
9.3.6.2.1 Design Bases
9.3.6.2.2 Deleted Per 2005 Update
9.3.6.2.3 Deleted Per 2005 Update
9.3.7 Containment Hydrogen Monitoring System
9.3.7.1 Design Bases
9.3.7.2 System Description
9.3.7.3 Safety Evaluation

9.4 Air Conditioning, Heating, Cooling and Ventilation Systems
9.4.1 Control Room Ventilation
9.4.1.1 Design Bases
9.4.1.2 System Description
9.4.1.2.1 Control Room Oconee 1 and 2
9.4.1.2.2 Control Room Oconee 3
9.4.1.3 Safety Evaluation
9.4.1.4 Inspection and Testing Requirements
9.4.2 Spent Fuel Pool Area Ventilation System
9.4.2.1 Design Bases
9.4.2.2 System Description
9.4.2.3 Safety Evaluation
9.4.2.4 Inspection and Test Requirements
9.4.3 Auxiliary Building Ventilation System
9.4.3.1 Design Bases
9.4.3.2 System Description
9.4.3.3 Safety Evaluation
9.4.3.4 Inspection and Testing Requirements
9.4.4 Turbine Building Ventilation System
9.4.4.1 Design Bases
9.4.4.2 System Description
9.4.4.3 Safety Evaluation
9.4.4.4 Inspection and Testing Requirements
9.4.5 Reactor Building Purge System
9.4.5.1 Design Bases
9.4.5.2 System Description
9.4.5.3 Safety Evaluation
9.4.5.4 Inspection and Testing Requirements
9.4.6 Reactor Building Cooling System
9.4.6.1 Design Bases
9.4.6.2 System Description
9.4.6.3 Safety Evaluation
9.4.6.4 Inspection and Testing Requirements
9.4.7 Reactor Building Penetration Room Ventilation System
9.4.7.1 Design Bases
9.4.7.2 System Description
9.4.7.3 Safety Evaluation
9.4.7.4 Inspection and Test Requirements
9.4.8 References

9.5.1 Fire Protection
9.5.1.1 Design Bases
9.5.1.1.1 Defense-in Depth (DID)
9.5.1.1.2 Fundamental Fire Protection Program and Design Elements
9.5.1.1.3 Nuclear Safety Goals, Objectives and Performance Criteria
9.5.1.1.4 Radioactive Release Goal, Objectives and Criteria
9.5.1.2 System Description
9.5.1.2.1 Power Block
9.5.1.2.2 Nuclear Safety Capability Systems and Equipment
9.5.1.2.3 Fire Protection Systems and Features
9.5.1.3 Safety Evaluation
9.5.1.3.1 Fundamental Fire Protection Program and Design Elements
9.5.1.3.2 Nuclear Safety at Power
9.5.1.3.3 Nuclear Safety at Non-Power Operation (NPO)
9.5.1.3.4 Radioactive Release
9.5.1.3.5 Fire PRA
9.5.1.3.6 Summary of NFPA 805 Compliance
9.5.1.4 Inspection and Testing
9.5.1.4.1 Inspection, Testing and Surveillance Methodology
9.5.1.4.2 Monitoring Program Methodology
9.5.1.5 Personnel Qualification and Training
9.5.1.5.1 Fire Brigade
9.5.2 Instrument and Breathing Air Systems
9.5.2.1 Design Basis
9.5.2.2 System Description
9.5.2.3 Instrument Air (IA) System Tests and Inspections
9.6 Standby Shutdown Facility
9.6.1 General Description
9.6.2 Design Bases
9.6.3 System Descriptions
9.6.3.1 Structure
9.6.3.2 Reactor Coolant Makeup (RCM) System
9.6.3.3 Auxiliary Service Water (ASW) System
9.6.3.4 Electrical Power
9.6.3.4.1 General Description
9.6.3.4.2 Diesel Generator
9.6.3.5 Instrumentation
9.6.3.5.1 SSF Reactor Coolant Makeup System Instrumentation
9.6.3.5.2 SSF Auxiliary Service Water Instrumentation
9.6.3.6 Support Systems
9.6.3.6.1 SSF Lighting System Description
9.6.3.6.2 SSF Fire Protection and Detection
9.6.3.6.3 SSF Service Water
9.6.3.6.4 Heating Ventilation and Air Conditioning
9.6.3.6.5 SSF Sump System
9.6.4 System Evaluations
9.6.4.1 General
9.6.4.2 Structure Design
9.6.4.3 Seismic Subsystem Analysis
9.6.4.4 Dynamic Testing and Analysis of Mechanical Components
9.6.4.5 ASME Code Class 1, 2, and 3 Components, Component Supports and Core Support Structures
9.6.4.6 Fire Protection
9.6.4.6.1 Safe Shutdown Systems
9.6.4.6.2 Performance Goals
9.6.4.6.3 Instrumentation Guidelines

9.6.4.6.4 Repairs for Hot Shutdown
9.6.4.6.5 Fire Protection Conclusion
9.6.4.7 Flooding Review
9.6.5 Operation and Testing
9.6.6 References

List of Tables

Table 9-1. Spent Fuel Cooling System Data, Units 1, 2

Table 9-2. Spent Fuel Cooling System Data, Oconee 3

Table 9-3. Component Cooling System Performance Data (For Normal Operation on a Per Oconee Basis)

Table 9-4. Cooling Water Systems Component Data (Component Data on a Per Unit Basis)

Table 9-5. Chemical Addition and Sampling System Component Data

Table 9-6. High Pressure Injection System Performance Data

Table 9-7. High Pressure Injection System Component Data

Table 9-8. Low Pressure Injection System Performance Data

Table 9-9. Low Pressure Injection System Component Data

Table 9-10. Coolant Storage System Component Data (Component Quantities for Three Units)

Table 9-11. Ventilation System Major Component Data

Table 9-12. Deleted Per 2002 Update.

Table 9-13. Component Cooling System Component Data (Component Data on a Per Unit Basis)

Table 9-14. SSF System Main Components

Table 9-15. SSF Primary Valves

Table 9-16. SSF Instrumentation

Table 9-17. Design Basis Tornado Missiles And Their Impact Velocities

Table 9-18. Design Basis Tornado Missiles Minimum Barrier Thicknesses

Table 9-19. Codes and Specifications For Design of Category I Structures

List of Figures

Figure 9-1. Fuel Storage Rack (Module)

Figure 9-2. Fuel Storage Rack (Assembly)

Figure 9-3. Spent Fuel Pool Outline Oconee 1, 2

Figure 9-4. Spent Fuel Pool Outline Oconee 3

Figure 9-5. Spent Fuel Cooling System

Figure 9-6. Deleted per 1990 Update

Figure 9-7. Fuel Handling System (Units 1&2 Page 1 and Unit 3 Page 2)

Figure 9-8. Component Cooling System

Figure 9-9. Condenser Circulating Water System

Figure 9-10. High Pressure Service Water System

Figure 9-11. Low Pressure Service Water System

Figure 9-12. Low Pressure Service Water System

Figure 9-13. Recirculated Cooling Water System

Figure 9-14. Deleted Per 1997 Update

Figure 9-15. Chemical Addition and Sampling System

Figure 9-16. Chemical Addition and Sampling System

Figure 9-17. High Pressure Injection System

Figure 9-18. High Pressure Injection System

Figure 9-19. Low Pressure Injection System

Figure 9-20. Coolant Storage System

Figure 9-21. Coolant Treatment System

Figure 9-22. Post-Accident Liquid Sample System

Figure 9-23. Post-Accident Containment Air Sample System

Figure 9-24. Control Room Area Ventilation and Air Conditioning System

Figure 9-25. Spent Fuel Pool Ventilation System Unit 1 and 2

Figure 9-26. Spent Fuel Pool Ventilation System Unit 3

Figure 9-27. Auxiliary Building Ventilation System Unit 1 and 2

Figure 9-28. Auxiliary Building Ventilation System Unit 3

Figure 9-29. Deleted Per 1998 Update

Figure 9-30. SSF General Arrangements Longitudinal Section

Figure 9-31. SSF General Arrangements Plan Elevation 777' and 754'

Figure 9-32. SSF General Arrangements Plan Elevation 797+0

Figure 9-33. SSF General Arrangements Plan Elevation 817+0

Figure 9-34. SSF General Arrangements Transverse Section

Figure 9-35. SSF RC Makeup System

Figure 9-36. SSF Auxiliary Service Water System

Figure 9-37. SSF HVAC Service Water System & SSF Diesel Cooling Water System

Figure 9-38. SSF Diesel Air Starting System

Figure 9-39. SSF Sump System

Figure 9-40. SSF 4160V/600V/208V Electrical Distribution

Figure 9-41. SSF 125 VDC Auxiliary Power Systems

Figure 9-42. Essential Siphon Vacuum System

Figure 9-43. Siphon Seal Water System

9.0 Auxiliary Systems

The Auxiliary Systems required to support the reactor during normal operations and servicing of the Oconee Nuclear Station are described in this section. Some of these systems have also been described and discussed in Chapter 6, since they serve as engineered safeguards. The information in this section deals primarily with the functions served by these systems during normal operation.

The design of the Auxiliary Systems has included consideration of system sharing, where feasible, between the three Oconee Nuclear Station units. This section describes the equipment for each unit and states where equipment is shared.

The majority of the components in these systems are located within the Auxiliary Building. Those systems connected by piping between the Reactor Building and the Auxiliary Building are equipped with Reactor Building isolation valves as described in Chapter 6.

THIS IS THE LAST PAGE OF THE TEXT SECTION 9.0.

THIS PAGE LEFT BLANK INTENTIONALLY.

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

New fuel will normally be stored in the spent fuel pool serving the respective unit. New or irradiated fuel assemblies with initial nominal enrichments up to 5.00 weight percent U-235 which do not meet the requirements for unrestricted storage must be placed in a restricted loading pattern.

Deleted paragraph(s) per 2002 update.

Reactivity analyses for these assemblies, stored in a checkerboard type configuration in the spent fuel pool, were performed using the methods discussed in Section 9.1.2.3.2.

New fuel may also be stored in the fuel transfer canal. The fuel assemblies are stored in five racks in a row having a nominal center-to-center distance of 2 ft 1-3/4 inches. One rack is oversized to receive a failed fuel assembly container. The other four racks are normal size and are capable of receiving new fuel assemblies.

New fuel may also be stored in shipping containers.

9.1.2 Spent Fuel Storage

9.1.2.1 Spent Fuel Storage - Oconee 1, 2

The Spent Fuel Pool common to Oconee Units l and 2 has been re-racked to increase the spent fuel storage capacity to 1312 fuel assemblies. This modification is pursuant to License Amendment Nos. 90, 90 and 87 for License Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station.

9.1.2.1.1 Design Bases

The Spent Fuel Pool designed for Oconee 1 and 2 is an integral part of the Oconee 1 and 2 Auxiliary Buildings and conforms to Safety Guide 13, “Fuel Storage Design Basis.” The fuel pools were designed for tornado wind and missiles, turbine generator missile, and seismic conditions as listed in Table 3-23. The Spent Fuel Pools were analyzed for the postulated cask drop accident as described in Section 3.8.4.4.

The spent fuel pool is constructed of reinforced concrete lined with stainless steel plate. The fuel pool concrete, reinforcing steel, liner plate and welds connecting the liner plate to the fuel pool floor concrete embedments are analyzed based on consideration of the new racks and additional fuel. Design criteria including loading combinations and allowable stresses are in compliance with Oconee FSAR Section 3.8.4 for Class I structures. The determination of Ta (abnormal thermal load condition to be used in combination with E') is based on the failure of one pump or cooler during normal operating conditions.

The function of the spent fuel storage racks is to provide for storage of spent fuel assemblies in a flooded pool, while maintaining a coolable geometry, preventing criticality, and protecting the fuel assemblies from excess mechanical or thermal loadings.

A list of design criteria is given below:

1. The racks are designed in accordance with the “NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications,” dated April 14, 1978 and revised January 18, 1979.
2. The racks are designed to meet the nuclear requirements of ANSI/ANS-57.2-1983. Oconee complies with the criticality accident requirements of 10CFR50.68(b) (Reference 27). The effective multiplication factor, K_{eff}, in the spent fuel pool is less than or equal to 0.95, including all uncertainties and under all credible conditions with partial credit for soluble boron.

3. The racks are designed to allow coolant flow such that boiling in the water channels between fuel assemblies does not occur.
4. The racks are designed to Seismic Category 1 requirements, and are classified as ANS Safety Class 3 and ASME Code Class 3 Component Support structures.
5. The racks are designed to withstand loads which may result from fuel handling accidents and from the maximum uplift force of the fuel handling crane.
6. Each storage position in the racks is designed to support and guide the fuel assembly in a manner that will minimize the possibility of application of excessive lateral, axial and bending loads to fuel assemblies during fuel assembly handling and storage.
7. The racks are designed to preclude the insertion of a fuel assembly in other than design locations.
8. The materials used in construction of the racks are compatible with the storage pool environment and do not contaminate the fuel assemblies

9.1.2.1.2 Design Description

The Oconee fuel storage racks are composed of individual storage cells made of stainless steel interconnected by grid assemblies to form integral module structures as shown in Figure 9-1. Each cell has a lead-in opening which is symmetrical and is blended smooth to facilitate fuel insertion. The cells are open at the top and bottom to provide a flow path for convective cooling of spent fuel assemblies through natural circulation. The fuel assembly storage cells are structurally connected to form modules through the use of plates and box beams which limit structural deformations and maintain a nominal center-to-center spacing between adjacent storage cavities during design conditions including the Safe Shutdown Earthquake. The racks utilize a neutron absorber, Boraflex, which is attached to each cell. However, due to degradation of the absorber material, no reactivity holddown credit is taken for any remaining Boraflex in the storage cells. The modules are neither anchored to the floor nor braced by the pool walls. The following information applies to the Oconee 1 and 2 fuel storage pool.

Number of Cells	1312
Number of Modules	4 – 8 x 11
	10 – 8 x 12
Deleted row(s) per 2002 Update.	
Center-to-Center Spacing	10.65 in.
Deleted row(s) per 2004 Update.	
Approximate Rack Assembly Dimensions and Maximum Weights	8 x 10 – 85.5 x 107 x 172 – 18, 060 lbs. 8 x 12 – 85.5 x 128 x 172 – 21,800 lbs.

The pool outline and rack arrangements are shown in Figure 9-3 and Figure 9-4.

9.1.2.2 Spent Fuel Storage - Oconee 3

The Spent Fuel Pool serving Oconee Unit 3 has been re-racked to increase the spent fuel storage capacity to 822 fuel assemblies, plus 3 additional storage spaces for failed fuel containers. This modification is pursuant to License Amendment Nos. 123, 123, and 120 for License Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station.

9.1.2.2.1 Design Bases

The Oconee 3 Spent Fuel Pool has the same Design Bases as the Oconee 1 and 2 pool described in Section 9.1.2.1.1.

9.1.2.2.2 Design Description

The Oconee 3 Spent Fuel Pool storage racks are similar to the Oconee 1 and 2 racks described in Section 9.1.2.1.2. The following information applies to Oconee Unit 3 spent fuel storage racks.

Number of Cells	822 plus storage locations for 3 failed fuel containers
Number Rack Arrays	7 – 8 x 11 2 – 8 x 12 1 – 8 x 10 x/3 container locations
Deleted row(s) per 2002 Update.	
Center-to-Center Spacing	10.60 in.
Deleted row(s) per 2004 Update.	
Approximate Rack Assembly Dimensions and Maximum Weights	8 x 10 – 85.5 x 107 x 172 – 18, 060 lbs. 8 x 12 – 85.5 x 128 x 172 – 21,800 lbs.

The pool outline and rack arrangements are shown in Figure 9-3 and Figure 9-4.

9.1.2.3 System Evaluation

9.1.2.3.1 Structural and Seismic Analysis

Fuel assembly storage rack and associated structures are designed to withstand the maximum forces generated during normal operation combined with the Safe Shutdown Earthquake according to the requirements of a Seismic Class l structure. For these conditions, the storage rack design is such that all stresses fall within the allowable stress limits specified in the AISC Specifications for Design, Fabrication and Erection of Structural Steel.

Normal operating loads include dead weight (in air) and thermal expansion loads. Lateral and vertical seismic loads along with the fluid forces generated by seismically generated pool water sloshing are considered to be acting simultaneously.

The seismic input spectra conform to the requirements of Regulatory Guide 1.60, “Design Response Spectra for Seismic Design of Nuclear Power Plants.”

Reference is made to Project 81 PSAR, Docket Nos. STN50-488 through -493, Section 3.7. The smoothed response spectra shown on Figure 2E-2A were normalized to 10 percent g for Safe Shutdown Earthquake (SSE). An earthquake acceleration-time history compatible with these spectra, as shown in Figures 2E-2B through 2E-2E, was used as a base motion on the model of the Auxiliary Building.

The seismic response of the Auxiliary Building to the base excitation is determined by a dynamic analysis. The dynamic analysis is made by idealizing the structure as a series of lumped masses with weightless elastic columns acting as spring restraints. The base of the structure is considered fixed. The choice of the location of the mass-joints depends on the distribution of masses in the real structure.

The seismic analysis of the racks was performed in two phases:

First a seismic time history analysis of a simplified non-linear 2-dimensional model was conducted. The model consisted of spring, mass, damping, friction, and gap elements to simulate a fuel bundle in a simplified model of a rack. The fuel assembly-to-cell impact loads, support pad lift-off values, rack sliding, and overall rack response were determined from the non-linear analysis. Coefficients of friction were varied between minimum and maximum possible values in order to determine worst case conditions of sliding and tipping respectively. Rack-to-rack impacts were precluded by spacing the racks beyond maximum possible excursion. The gap spaces are large enough to accomodate lateral module motion due to earthquake forces. In order to account for 3-dimensional effects, the results of independent orthogonal loadings were combined by the SRSS method.

Next, a seismic response spectrum analysis of a 3-dimensional finite element model of the racks, using inputs from the results of the non-linear analysis, and superimposed with other applicable loads, was conducted. Design stresses and safety margins for appropriate components in the racks were tabulated and found to be acceptable.

The structural damping values used are 4 percent for an SSE and 2 percent for an OBE.

The maximum uplift load available from the fuel handling crane on the storage rack is limited to 3000 lbs or less by the hoist interlock. A separate fuel assembly drop analysis was performed. A 3000 pound object was postulated to impact the top of the rack from a height of 6 feet. Calculations show that the resulting stresses are within acceptable stress limits.

Structural design precludes placing a fuel assembly between cells, and the rack will withstand the loadings imposed by a postulated dropped fuel assembly.

9.1.2.3.2 Criticality Analysis

The design methodology which ensures the criticality safety of the fuel assemblies in the spent fuel storage rack is discussed in Section 9.1.2.3.2.3 and in Reference 8.

9.1.2.3.2.1 Neutron Multiplication Factor

Criticality of fuel assemblies in the spent fuel storage rack is prevented by the design of the rack which limits fuel assembly interaction. This is done by fixing the minimum separation between assemblies and inserting neutron poisons between assemblies.

The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor (k_{eff}) of the fuel assembly array will be less than 1.0 in unborated spent fuel pool water, and less than 0.95 with partial credit for soluble boron, in accordance with Reference 26. Oconee complies with the criticality accident requirements of 10CFR50.68(b) (Reference 27). The acceptance criteria for criticality is further discussed in Section 9.1.2.3.2.5.

9.1.2.3.2.2 Normal Storage

Under normal storage conditions, the following assumptions were used in the criticality analysis.

1. Credit is taken for the decrease in reactivity associated with the fuel assembly burnup.
2. The fuel assembly is the most reactive fuel assembly to be stored based on a minimum burnup. The fuel designs analyzed include the fuel assembly designs described in Chapter 4, and earlier designs. Additionally, a small number of alternate fuel configurations are analyzed (e.g. lead test asssemblies, failed rod canisters, and rod consolidation canisters).
3. The moderator is at the temperature within the design limits of the pool which yields the largest reactivity, and contains at least 430 ppm boron (pursvant to License Amendment Nos. 323, 323, 324

for License Nos. DPR-38, DPR-47, and DPR-55), to maintain $K_{eff} \leq 0.95$ for normal storage conditions. Full credit for soluble boron is taken for postulated accident conditions and during fuel movement. For accident conditions the double contingency principle of ANSI N16.1-1975 is applied. This principle states that it shall require at least two unlikely, independent, and concurrent events to produce a criticality accident. During fuel movement the presence of dissolved boron in the spent fuel pool water is assumed since this is only a temporary condition and only a single assembly is handled at a time.

4. The array is either infinite in the lateral extent or is surrounded by a conservatively chosen reflector, whichever is appropriate for the design. The nominal case calculation is infinite in the lateral extent. However, poison plates are not necessary on the periphery of the modular array and between widely spaced modules because calculations show that this finite array is less reactive than the nominal case infinite array. The assemblies are also infinite in the axial extent. A reactivity bias is included in all burned-fuel criticality calculations to conservatively account for reactivity differences between a detailed 3-D axial burnup model and the 2-D average burnup model employed for nominal calculations.

5. Mechanical uncertainties and biases due to mechanical tolerances during construction are treated by either using "worst case" conditions or by performing sensitivity studies and obtaining appropriate values. The items included in the analysis are:

   a. Deleted row per 2002 Update.

   b. Deleted row per 2002 Update.

   c. Can ID

   d. Stainless steel thickness

   e. Center-to-center spacing

   f. Fuel enrichment

   g. Fuel pellet density

   h. Fuel pellet OD

   Other applicable uncertainties and biases are discussed in Section 9.1.2.3.2.3.

6. No credit is taken for the assembly spacer grids.

7. No credit is taken for fuel assembly control components which can be removed (e.g. burnable poisons and control rods).

8. Credit is taken for the inherent neutron absorbing effect of some of the rack structure materials in accordance with ANSI/ANS-57.2-1983 and Reference 26.

9.1.2.3.2.3 Criticality Analysis Methodology

Criticality of fuel assemblies outside the reactor is precluded by adequate design of fuel transfer, shipping and storage facilities and by administrative control procedures. The two principal methods of preventing criticality are limiting the fuel assembly array size and limiting assembly interaction by fixing the minimum separation between assemblies and/or inserting neutron poisons between assemblies.

The design basis for preventing criticality outside the reactor is that, considering possible variations, there is a 95 percent probability at a 95 percent confidence level that the effective multiplication factor (k_{eff}) of the fuel assembly array will be less than or equal to 0.95, with partial credit for soluble boron. Oconee complies with the criticality accident requirements of 10CFR50.68(b) (Reference 27). The conditions that are assumed in meeting this design basis are outlined in Section 9.1.2.3.2.2.

In order to justify storage of fuel up to 5.0 w/o, the burnup credit approach was utilized in the spent fuel pools. The burnup credit approach to fuel rack criticality analysis requires calculation and comparison of reactivity values over a range of burnup and initial enrichment conditions. In order to accurately model characteristics of irradiated fuel which impact reactivity, a criticality analysis method capable of evaluating arrays of these irradiated assemblies is needed. The advanced nodal methodology combining CASMO-3/TABLES-3/SIMULATE-3 is used for this purpose. CASMO-3 (Reference 4) is an integral transport theory code, SIMULATE-3 (Reference 6) is a nodal diffusion theory code, and TABLES-3 (Reference 5) is a linking code which reformats CASMO-3 data for use in SIMULATE-3. This methodology permits direct coupling of incore depletion calculations and resulting fuel isotopics with out-of-core storage array criticality analyses. The variable effects of fission product poisoning, fissile material production and utilization and other related effects are accurately modeled with the CASMO-3/TABLES-3/SIMULATE-3 methodology. Applicable biases and uncertainties are developed and become inputs to the methodology.

The results for the criticality methodology are validated by comparison to measured results of fuel storage critical experiments. The criticality experiments used to benchmark the methodology were the Babcock and Wilcox close proximity storage critical experiments performed at the CX-10 facility (Reference 7). The B&W critical experiments used are specifically designed for benchmarking reactivity calculation techniques. The experiments are analyzed, and the statistical accuracy of the calculated reactivity results are assessed.

The bias associated with the benchmarks is -0.00142 Δk with a standard deviation of 0.00412 Δk. The 95/95 one-sided tolerance limit factor for 10 values is 2.911. Therefore, there is a 95 percent probability at a 95 percent confidence level that the uncertainty in reactivity due to the method is not greater than 0.01199 Δk.

For burned fuel, the maximum reactivity occurs approximately 100 hours after shutdown due to the decay of Xe^{135}. Therefore, all fuel assemblies in the spent fuel pool are modeled at no xenon conditions.

An additional bias and uncertainty are required to quantify the reactivity of burned nuclear fuel assemblies. Two burnup uncertainties associated with this methodology are accounted for in the criticality analysis. The first penalty accounts for uncertainties in the reactivity due to uncertainties in the burnup of the assembly, while the second penalty accounts for the reactivity holddown effect of lumped burnable absorbers.

The exposure reactivity uncertainty accounts for the uncertainty on the assembly burnup. Since the final burnup qualification curves are based on a code calculated burnup, the uncertainty in that calculated burnup must be considered. Rather than determining the uncertainty on the actual burnup, the uncertainty on reactivity due to burnup was applied to account for the burnup uncertainty. A 95/95 one-sided tolerance was determined to account for the maximum reactivity error associated with the burnup of the fuel.

As required by the standards, no removable poisons are accounted for in the criticality analyses. Thus, all assemblies are modeled with no burnable poisons (BPs). However, this can be slightly non-conservative due to the increase in reactivity associated with the removal of the BP. Thus a burnable poison removal (BP-Pull) penalty is developed to account for this effect. BPs are used in the core design to hold down reactivity, and hence peaking of fresh assemblies. Thus, the reactivity of the BPd assembly is less than the non-BPd assembly. However, once the BP is removed (from the previously BPd assembly), a reactivity increase is seen due to the shadowing effect the BPs had on the assembly. This difference in reactivity is applied as an additional bias on reactivity.

The basic approach in the burnup credit methodology is to use reactivity equivalencing techniques to construct burnup versus enrichment curves which represent equivalent and acceptable reactivity conditions over an applicable range of burnups and initial enrichments. These burnup versus enrichment curves are established for each type of storage, e.g. unrestricted and restricted storage.

Generation of the applicable burnup credit curves requires a two part calculation process. The first part is to create two types of reactivity versus burnup curves. The first type of curve defines the maximum reactivity for the spent fuel pool such that the appropriate design criteria are met including allowances for both calculational uncertainties and manufacturing tolerances. The second type of curve represents the reactivity versus burnup for a particular enrichment, and is generated for the range of enrichments. The intersection of the maximum design reactivity curve with the multiple enrichment curves provides data points for the second part of the process.

The second part of the process generates the burnup versus initial enrichment curves by plotting the burnup where the maximum design reactivity equals the reactivity of a particular enrichment for each enrichment. Two curves are generated which represent the qualification criteria for a particular storage configuration. Each burnup versus enrichment curve shows the minimum amount of burnup required to qualify fuel for storage in the applicable loading pattern as a function of the fuel's initial enrichment. Additional details of the methods used can be found in Reference 8.

The SCALE-4 system of computer codes (Reference 10) was used to analyze the boundary restrictions between Checkerboard, Restricted, and Unrestricted storage configurations to assure that the storage configurations at the boundary do not cause an increase in the nominal k_{eff} for the individual regions. This analysis is performed to determine if there is a need for new administrative restrictions at the boundaries.

This methodology utilizes two dimensional Monte Carlo theory. Specifically, this analysis method used the CSAS25 sequence contained in Criticality Analysis Sequence No. 4 (CSAS4). CSAS4 is a control module contained in the SCALE-4.2 system of codes. The CSAS25 sequence utilizes two cross section processing codes (NITAWL and BONAMI) and a 3-D Monte Carlo code (KENO Va) for calculating the effective multiplication factor for the system. The 27 Group NDF4 cross section library was used exclusively for this analysis.

Acceptable interface boundary conditions between storage configurations were determined by varying the boundaries between various storage regions to determine the worst case configurations for coupling between assemblies in different regions. The boundaries were then reflected to simulate an infinite array. The k_{eff} of these infinite boundary arrays were compared to the base k_{eff} of infinite arrays of either fuel storage region creating the boundary. If the infinite boundary array k_{eff} did not represent an increase in the k_{eff} of the regions making the boundary, then no storage restrictions were imposed at the interface. When the worst case did represent an increase, conservative storage restrictions were applied.

These methods conform with ANSI N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," Section 5.7, Fuel Handling System; ANSI/ANS-57.2-1983, “Design Requirements for LWR Spent Fuel Storage Facilities at Nuclear Power Stations,” Section 6.4.2.2; ANSI N16.9-1975, "NRC Standard Review Plan," Section 9.1.2 and the NRC guidance contained in Reference 26.

9.1.2.3.2.4 Postulated Accidents

As part of the criticality analysis for the Oconee spent fuel pools, abnormal and accident conditions are considered to verify that acceptable criticality margin is maintained for all conditions. Most accident conditions will not result in an increase in k_{eff} of the rack. For an assembly dropped on top of the storage rack, the section of the rack structure which is essential for preventing criticality is not excessively deformed. Furthermore, the dropped assembly has more than eight inches of water separating it from the active fuel height of stored assemblies which precludes any interaction between the dropped assembly and the stored assemblies. Although the dropped assembly is more reactive outside of the storage cell rather than inside, the assembly is no more reactive dropped on top of the storage rack than located anywhere else in the pool outside the storage rack.

However, accidents can be postulated which would increase reactivity. Misloading of an assembly would increase reactivity; in particular, misloading the highest reactive assembly in place of the lowest reactive assembly. This is either the misplacement of a fresh assembly in an empty cell in the checkerboard pattern or in a filler cell in the restricted pattern.

For loss of spent fuel pool cooling scenarios, the reactivity increases with decreasing water density for the Oconee fuel storage racks and the current analyzed fuel designs. Two accident scenarios are postulated: heat load due to the loss of one cooling train and cold water emergency makeup. The emergency makeup event encompasses a dilution event, since one source of makeup is Lake Keowee.

For accident conditions, the double contingency principle is employed. The double contingency principle of ANSI/ANS-57.2-1983 states that it is not required to assume two unlikely, independent concurrent events to ensure protection against a criticality accident. Thus, for accident conditions, the presence of soluble boron in the storage pool water can be assumed as a realistic initial condition, since not assuming its presence would be a second unlikely event.

The acceptance criteria for criticality are further discussed in 9.1.2.3.2.5.

9.1.2.3.2.5 Acceptance Criteria for Criticality

The acceptance criteria for the spent fuel pools will be $k_{eff} \leq 0.95$. This assumes full credit may be taken for soluble boron under accident conditions as allowed by the double contingency principle in ANSI/ANS-57.2-1983, and that only partial credit is taken for soluble boron under normal conditions, per Reference 26. Oconee complies with the criticality accident requirements of 10CFR50.68(b) (Reference 27).

Deleted paragraph(s) per 2002 update.

9.1.2.3.2.6 Cask Drop Accident

Cask drop accidents are analyzed for criticality consequences in Section 15.11.2.5.1.

9.1.2.3.2.7 Criticality Analyses for Loading NUHOMS Dry Storage Canisters (DSC)

The criticality analysis of the NUHOMS®-24P/24PHB DSC, for loading and unloading operations in the Oconee spent fuel pools, has been performed in accordance with the requirements of 10CFR50.68(b). The evaluation takes partial credit for soluble boron in the spent fuel pools. Minimum burnup requirements were developed for fuel to be placed without location restrictions in the NUHOMS®-24P/24PHB DSC. These burnup requirements, applicable for eligible fuel assemblies with a minimum 5 years post-irradiation cooling time, are a function of initial U-235 enrichment.

The criticality analysis demonstrated that the current minimum boron concentration required in the Oconee spent fuel pools is adequate to maintain the maximum 95/95 K_{eff} below 0.95 for all normal conditions and credible accident scenarios associated with loading fuel assemblies into the NUHOMS®-24P/24PHB DSCs.

Consistency was maintained between the spent fuel pool rack and DSC normal and accident analyses. Accidents analyzed included assembly dropped on top of the storage rack, misloading of an assembly, and loss of spent fuel pool cooling scenarios.

9.1.2.3.3 Material, Construction, and Quality Control

The entire fuel assembly storage rack is constructed of type 304 stainless steel, with Boraflex panels attached to each cell. All welded construction is used in the fabrication of the fuel assembly storage rack. The all-welded construction ensures the structural integrity of the storage modules and provides assurance of smooth, snag-free passage in the storage cavities so that it is highly improbable that a fuel assembly could become stuck in the rack.

The material, construction and quality control procedures are in accordance with the quality assurance requirements of Duke Power Company, as described in Duke Power Company Topical Report, DUKE-1.

9.1.2.3.4 Interface of High Capacity Fuel Storage Rack and Spent Fuel Storage Pool

The pool floor will support the high capacity storage rack as a free-standing structure during all design conditions. During installation, no racks are moved over spent fuel assemblies in the pool. All spent fuel assemblies in Unit 3 are removed prior to removing existing racks.

For the free-standing rack structure, conservative analysis shows that under simultaneous forces from vertical and lateral seismic excitation, the residual displacement of the rack relative to the pool floor is less than 1 inch for full-loaded condition (i.e., much less than minimum clearance of 2.75 inches to pool walls and installed equipment.)

The maximum sliding distance of the Westinghouse free-standing fuel rack is obtained by equating the kinetic energy developed in the fuel rack, in response to the SSE seismic event, to the energy dissipated by friction between the fuel rack supports and the pool floor, during sliding. The maximum kinetic energy in the fuel rack, produced by the SSE seismic event, is calculated from the spectral response to the SSE response spectrum. The horizontal displacement of the rack is 1.414 times the sum of the deflecton of the top of the rack (0.245 in) and the maximum sliding distance (0.432). The coefficient of friction is assumed to be 0.20.

The rack/pool floor normal force on which the lateral friction forces used in the analysis are based includes the effect of vertical seismic acceleration.

The maximum lateral seismic force exerted by any rack module on pool floor is 189000 pounds and results in a stress of 2440 psi in the floor liner and 3296 psi in the weld connecting the floor liner to embedments in the concrete. The maximum combined seismic and thermal stress in the floor liner is 21640 psi and 30610 psi in the weld between liner and embedments. The maximum stresses are below the design allowable stress of 27,000 psi in the liner and 32000 psi in the welds.

9.1.2.4 Safety Evaluation

The storage rack is designed and constructed to retain the integrity of the structure under all anticipated loads, including the Safe Shutdown Earthquake, with the maximum number of fuel assemblies occupying the storage locations.

The rack design provides protection against damage to the fuel and precludes the possibility of a fuel assembly being placed between cells. Although not required for safe storage of spent fuel assemblies, the spent fuel pool water is normally borated to a concentration of at least 2220 ppm, or higher as specified by the Core Operating Limits Report (COLR). The rack design also assures a K_{eff} of less than 1.0 even when the entire array of fuel assemblies, assumed to be in their most reactive condition and within the limits specified in the Technical Specifications, are immersed in unborated water at room temperature. Furthermore, if the pools were filled with the most reactive fuel allowed, which is clearly in violation of the Technical Specifications, K_{eff} would be $\simeq$ 0.85 with full credit for soluble boron. Under these conditions a criticality accident during refueling or storage is not considered credible.

9.1.2.5 Boraflex

The spent fuel storage racks contain Boraflex, which is the trade name for a silicon polymer that contains a specified amount of Boron 10 that was originally used as the neutron absorber to assure that the design basis for criticality control was met through the service life of the racks. The Boraflex is affixed to each of the four exterior sides of the fuel storage cell by means of stainless steel wrappers. Boraflex was originally used in spent fuel storage racks for the nonproductive absorption of neutrons such that the NRC established acceptance criterion of k_{eff} no greater than 0.95 was maintained. However, due to degradation of the absorber material, no reactivity holddown credit is taken any longer for the remaining Boraflex in the storage cells.

Since reactivity hold-down credit is no longer being taken for Boraflex in the Spent Fuel Pool storage cells, the License Renewal commitment to inspect the Boraflex panels is no longer required, and the inspection program has been discontinued.

Deleted paragraph(s) per 2002 update.

9.1.3 Spent Fuel Cooling System

9.1.3.1 Design Bases

9.1.3.1.1 Units 1 and 2 Spent Fuel Pool Cooling System

The primary function of Spent Fuel Pool Cooling System for Units 1 and 2 is to provide decay heat removal for the spent fuel stored in the Units 1 and 2 spent fuel pool. The cooling system design requirements are the criteria imposed by the 1980 re-racking (References 11, 12). Other system functions are to maintain the pool inventory, clarity and chemistry at acceptable levels.

Revised criteria have been imposed during the 1980 re-racking modification, pursuant to Amendments 90, 90, and 87 for License Nos. DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station. The thermal-hydraulic analyses associated with the spent fuel pool racks assumes that the bulk spent fuel pool temperature remain at or below 150°F, for normal heat loads (Reference 11). The Units 1 and 2 Spent Fuel Cooling System is designed to keep the pool bulk water temperature:

1. Below 150°F for normal heat loads and two or three pump-cooler configurations in operation (Reference 11)
2. Below 150°F for abnormal heat loads and three pump-cooler configurations in operation (Reference 11)
3. Below 205°F for abnormal heat loads and any two pump-cooler configurations in operation (Reference 11).

For the Units 1 and 2 spent fuel cooling system, the design basis normal heat load assumes that Units 1 and 2 are refueled consecutively, and the rack positions are filled with previous discharges, except for 118 spaces reserved for a full core discharge (Reference 11). The design basis abnormal heat load assumes that Units 1 and 2 are refueled consecutively, followed by a full core discharge after a short period of operation. In this case, all rack positions contain spent fuel (References 11 and 12). The licensing basis decay heat predictions were performed with the methodology outlined in Reference 13. Various operational evolutions may utilize decay heat predictions based on the ORIGEN methodology (e.g., ORIGEN-ARP or SAS2H/ORIGEN-S) presented in References 24 and 25.

It should be noted that, while all temperature conditions above represent design criteria associated with specific analytical assumptions, only the higher temperature of 205°F represents an actual operating limit. Analyses have been performed to ensure that seismic and structural integrity of the pool liner, supporting concrete, and fuel racks are not compromised at this temperature limit. Thermal - hydraulic analysis of the racks has also shown that boiling within the fuel cells does not occur with pool temperatures maintained at or below this limit, provided normal operating pool level is maintained.

In addition to the primary function of decay heat removal, the system provides for purification of the spent fuel pool water, the fuel transfer canal water, and the contents of the borated water storage tank, in order to remove fission and corrosion products and to maintain water clarity for fuel handling operations. The system also provides inventory makeup for the fuel transfer canal and the incore instrument handling tank.

The system is designed to withstand the effects of a seismic event and meet the requirements of Quality Group C classification.

Portions of the Spent Fuel Cooling 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.

9.1.3.1.2 Unit 3 Spent Fuel Pool Cooling System

The Unit 3 Spent Fuel Pool Cooling System duplicates the equipment used for the Units 1 and 2 system. The Unit 3 system is designed to remove the decay heat from the stored fuel in the Unit 3 spent fuel pool. The cooling system heat removal requirements are as set forth in NRC Standard Review Plan Section SRP-9.1.3 (References 14, 15). Other system functions are to maintain the pool inventory, clarity and chemistry at acceptable levels.

The Unit 3 system heat removal design requirements, as stipulated by Standard Review Plan 9.1.3, are:

1. For the maximum normal heat load with the normal cooling systems in operation, and assuming a single active failure, the temperature of the pool water shall be maintained at or below 140°F and the liquid level in the pool should be maintained.
2. For the abnormal maximum heat load with the normal cooling systems in operation, the pool water temperature should be kept below boiling and the liquid level in the pool should be maintained. A single active failure need not be considered.

The design basis maximum normal and abnormal decay heat loads are as defined in SRP 9.1.3 (Reference 15), for fuel racks with greater than 1 1/3 core storage capacity. The licensing basis decay heat predictions were performed with the methodology outlined in Reference 13. Various operational evolutions may utilize decay heat predictions based on the ORIGEN methodology (e.g., ORIGEN-ARP or SAS2H/ORIGEN-S) presented in References 24 and 25.

It should be noted that, while both temperature conditions above represent design criteria associated with specific analytical assumptions, only the boiling criterion represents an actual design limit. An operating limit of 205°F is imposed for conservatism. Analyses have been performed to ensure that seismic and structural integrity of the pool liner, supporting concrete, and fuel racks are not compromised at this temperature limit. Thermal - hydraulic analysis of the racks also has shown that boiling within the fuel cells does not occur with pool temperatures maintained at or below this limit, provided normal operating pool level is maintained.

In addition to the primary function of decay heat removal, the system provides for purification of the spent fuel pool water, the fuel transfer canal water, and the contents of the borated water storage tank, in order to remove fission and corrosion products and to maintain water clarity for fuel handling operations. The system also provides inventory makeup for the fuel transfer canal and the incore instrument handling tank.

The system is designed to withstand the effects of a seismic event and meet the requirements of Quality Group C classification.

Portions of the Spent Fuel Cooling 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".

9.1.3.2 System Description

The Spent Fuel Cooling System (Figure 9-5, Units 1, 2 and 3) provides cooling for the spent fuel pool to remove fission product decay heat energy. System performance data are shown in Table 9-1 (Units 1 and 2) and Table 9-2 (Unit 3). Major components of the system are briefly described below.

Spent Fuel Coolers

The spent fuel coolers are designed to maintain the temperature of the spent fuel pool as noted in Section 9.1.3.1. There are three coolers for Oconee 1 and 2, and three coolers for Unit 3, arranged in parallel.

Spent Fuel Coolant Pumps

The spent fuel coolant pumps take suction from the spent fuel pool and recirculate the fluid back to the pool after passing through the coolers. A portion of the flow is demineralized and filtered depending on conditions. There are three pumps for Oconee Units l and 2, and three pumps for Oconee 3. The spent fuel coolant pumps are also used for filling the fuel transfer canal or incore instrumentation handling tank with borated water from the borated water storage tank.

Spent Fuel Coolant Demineralizers

One spent fuel coolant demineralizer will process approximately one-half of the spent fuel pool volume in 24 hours. There is one demineralizer for Units 1 and 2, and one for Unit 3.

Spent Fuel Coolant Filters

The spent fuel coolant filters are designed to remove particulate matter from the spent fuel pool water. They are sized for the same flow rate as the demineralizers (180 gpm). There are two filters for Units l and 2, and two for Unit 3.

Borated Water Recirculation Pump

This pump removes water from the borated water storage tank for demineralization and filtering. The pump may also be used while demineralizing and filtering the water in the fuel transfer canal during a transfer of fuel. It may also be used for emptying the fuel transfer canal if spent fuel coolant pumps are unavailable for use. There is one pump for Units 1 and 2, and one for Unit 3.

9.1.3.3 System Evaluation

9.1.3.3.1 Normal Operation

The normal operation of the Spent Fuel Cooling System provides several functions. The most safety significant of these functions is to maintain pool inventory so that stored fuel is always covered with water. In order to protect against loss of inventory by boil-off, the system maintains the pool temperature below the design bases limits specified in Section 9.1.3.1. The system also maintains the pool clarity and chemistry at acceptable levels.

Spent fuel pool heat removal is accomplished by recirculating spent fuel coolant water through heat exchangers and then back to the pool. The spent fuel pumps take suction from the spent fuel pool and transport the flow through the coolers, which are arranged in parallel. The waste heat is removed from the shell side of the coolers by the Recirculated Cooling Water System. The cooled spent fuel pool water is then directed back to the spent fuel pool.

The spent fuel pool water temperature is a direct function of the decay heat load produced by the fuel in the racks, in conjunction with the heat removal capability of the spent fuel cooling system. The total heat removal capacities are the same for the Units 1 and 2 and the Unit 3 spent fuel pool coolant systems. Both systems use the same numbers of pumps and coolers, with the same design specifications and overall equipment configurations. The expected decay heat loads vary with the number of fuel assemblies present in the pool, the burnups of the various fuel assemblies, and the post-irradiation decay times.

At the time that the Units 1 and 2 spent fuel pool was re-racked, its spent fuel cooling system was upgraded to handle the higher total heat load expected from the increased number of stored fuel assemblies. The heat removal capability of the upgraded spent fuel cooling system has been sized to meet the design limits specified in Section 9.1.3.1. A specific analysis of expected maximum normal and abnormal heat loads was performed, as described in Reference 58. The Spent Fuel Cooling System was analyzed to predict the pool temperatures which would result from these heat loads. Temperatures meet the design requirements as specified in Section 9.1.3.1. Core offloads are controlled such that the ultimate heat load from these analyses are not exceeded.

At the time that the Unit 3 spent fuel pool was re-racked, its spent fuel cooling system was upgraded to handle the higher total heat load expected from the increased number of stored fuel assemblies. The heat removal capability of the upgraded spent fuel cooling system has been sized to meet the design limits specified in Section 9.1.3.1. A specific analysis of expected maximum normal and abnormal heat loads was performed, as described in Reference 59. Again, the Spent Fuel Cooling System was analyzed to predict the pool temperatures resulting from these heat loads. These temperatures meet the design requirements as specified in Section 9.1.3.1. Core offloads are controlled such that the ultimate heat load from these analyses are not exceeded.

During an actual refueling outage for any unit at ONS, it is now common practice to offload a full core (177 fuel assemblies) into the pool. The resulting heat load under this condition will be less than the abnormal heat load cases evaluated in Sections 9.1.3.1.1 and 9.1.3.1.2 for the Units 1 and 2 fuel pool and Unit 3 fuel pool respectively. In addition, the resulting temperature will be less than 205°F in the fuel pools in the abnormal heat load case, assuming a single active failure. Normal practice at Oconee during the abnormal heat load case is to limit the maximum pool temperature to 150°F. This is accomplished via plant procedures. The seismic structural integrity of the storage racks, pools, and supporting structures has been evaluated at or above this temperature, and found to be adequate. Also, the thermal-hydraulic analysis of the storage racks indicates that localized boiling will not occur if water entering the storage cells reaches this temperature, as long as normal pool level is maintained.

A bypass purification loop is provided to maintain the purity of the water in the spent fuel pool. This loop is also utilized to purify the water in the borated water storage tank following refueling, and to maintain clarity in the fuel transfer canal during refueling. Water from the borated water storage tank or fuel transfer canal can be purified by using the borated water recirculation pump.

9.1.3.3.2 Failure Analysis

An analysis of the maximum fuel cladding temperature has been performed for the postulated case of complete loss of coolant circulation to the pool. The analysis assumes maximum anticipated heat load in the pool, with the hottest assembly located in the least cooled storage area. The maximum cladding temperature will occur at the location of maximum heat flux. For a fuel assembly having the maximum value for decay heat power of 80 kw, and for an axial peak to average power density ratio of 1.2, the maximum local fuel rod heat flux is 1200 BTU/hr-ft^2. Natural circulation flow rates within the storage tubes have been calculated which give confidence that convection film coefficients in excess of 50 BTU/hr-ft^2 °F can be expected. Assuming this low value for conservatism, the clad surface temperature is 24°F above the coolant temperature. Because the heat flux is small, very large uncertainties in the film coefficient are acceptable without causing prohibitively high clad temperatures. For example, a reduction by a factor of five in the film coefficient would result in a clad surface temperature of 120°F above the coolant temperature. A reduction by a factor of ten, from 50 BTU/hr-ft^2 °F to 5 BTU/hr-ft^2 °F would result in a clad surface temperature of 240°F above the coolant temperature. These temperatures are below 650°F, which is the normal operating temperature of the fuel clad in the core.

9.1.3.4 Safety Evaluation

The Spent Fuel Cooling System provides adequate capacity and component redundancy to assure the cooling of stored spent fuel, even when large quantities of fuel are in storage. Multiple component failures or complete cooling failures permit ample time to assure that protective actions are taken. The system is arranged so that loss of fuel pool water by piping or component failure is highly improbable. The system performs no emergency functions. Alarms are provided to alert operator of abnormal pool level and temperature.

The Spent Fuel Cooling System has one process line connecting to the Reactor Coolant System through the SSF RC Makeup line. Its major penetration to the Reactor Building is through the fuel transfer tube. The fuel transfer tube is isolated inside the Reactor Building by a blind flange connection in the fuel transfer canal.

9.1.4 Fuel Handling System

9.1.4.1 Design Bases

9.1.4.1.1 General System Function

The fuel handling system shown on Figure 9-7 (sheets 1 & 2) is designed to provide a safe, effective means of transporting and handling fuel from the time it reaches the station in an unirradiated condition until it leaves the station after postirradiation cooling. The system is designed to minimize the possibility of mishandling or maloperations that could cause fuel assembly damage and/or potential fission product release.

Separate fuel handling equipment is provided for each reactor. A common fuel storage area serves Oconee 1 and 2, while a separate fuel storage area is provided for Oconee 3.

The reactors are refueled with equipment designed to handle the spent fuel assemblies underwater from the time they leave the reactor vessels until they are placed in a cask for shipment from the spent fuel pools. Underwater transfer of spent fuel assemblies provides an effective, economic, and transparent radiation shield, as well as a reliable cooling medium for removal of decay heat. Use of borated water assures reactor subcriticality during refueling.

9.1.4.1.2 New Fuel Storage

New Fuel Storage is described in Section 9.1.1.

9.1.4.1.3 Spent Fuel Pool

Each spent fuel pool is a reinforced concrete pool located in its respective Auxiliary Building. The Oconee 1, 2 pool is lined with stainless clad plate. The Oconee 3 pool is lined with stainless steel plate. The unit 1 and 2 spent fuel pool will hold 1312 fuel assemblies. The unit 3 spent fuel pool will hold 822 assemblies plus 3 spaces for failed fuel canisters. Fuel components (such as control rods, BP's, or APSR's) requiring removal from the reactors are stored in the spent fuel assemblies or in brackets suspended from the top of the fuel racks.

9.1.4.1.4 Fuel Transfer Tubes

Two horizontal tubes are provided to convey fuel between each Reactor Building and the respective Auxiliary Building. These tubes contain tracks for the fuel transfer carriages, gate valves on the spent fuel pool side, and a means for flanged closure on the Reactor Building side. The fuel transfer tubes penetrate the spent fuel pool and the fuel transfer canal at their lower depth, where space is provided for the rotation of the fuel transfer carriage baskets.

9.1.4.1.5 Fuel Transfer Canal

The fuel transfer canal is a passageway in the Reactor Building extending from the reactor vessel to the Reactor Building wall. It is formed by an upward extension of the primary shield walls. The enclosure is a reinforced concrete structure lined with stainless clad plate to form a canal above the reactor vessel which is filled with borated water for refueling.

Space is available in the deeper portion of the fuel transfer canal for underwater storage of the reactor vessel internals upper plenum assembly. This portion of the fuel transfer canal can also be used for storage of the reactor vessel internals core barrel and thermal shield assembly by storing the upper plenum assembly in the upper end of the fuel transfer canal.

9.1.4.1.6 Fuel Handling Equipment

This equipment consists of fuel handling bridges, fuel handling mechanisms, fuel storage racks, fuel transfer mechanisms, and shipping casks. In addition to the equipment directly associated with the handling of fuel, equipment is provided for handling the reactor vessel closure head and the upper plenum assembly to expose the core for refueling.

9.1.4.2 System Description and Evaluation

9.1.4.2.1 Receiving and Storing Fuel

New fuel assemblies are received in shipping containers, unloaded and stored in the appropriate spent fuel pool. After reactor shutdown, new fuel assemblies can be transferred from the spent fuel pool to the Reactor Building with the use of the fuel transfer mechanisms and the fuel transfer tubes.

9.1.4.2.2 Loading and Removing Fuel

Following the reactor shutdown and Reactor Building entry, the refueling procedure is begun by removal of the reactor closure head. Prior to this it is necessary to uncouple the control rods from the drive mechanisms. An auxiliary hoist (the CRDM crane, located over the fuel transfer canal) is used for this and any other special purposes that may be required during refueling. The electrical and water connections to the head assembly are disconnected.

To close the annular space between the reactor vessel flange and fuel transfer canal floor, a seal plate is lowered into position and bolted to the canal shield flange with appropriate gaskets. The isolation valves on the spent fuel pool end of the fuel transfer tubes are closed and the tubes drained. The blind flanges on the reactor building end of the transfer tubes are then removed.

Head removal and replacement time is minimized by the use of multiple tensioners. The stud tensioners are hydraulically operated to permit preloading and unloading of the reactor vessel closure studs at cold shutdown conditions. The studs are tensioned to their operational load in discrete steps in a predetermined sequence. Required stud elongation after tensioning is verified by an elongation gauge.

Following removal of the studs from the reactor vessel tapped holes, the studs and nuts are supported in the closure head bolt holes with specially designed spacers. The studs and nuts are then removed from the reactor closure head for inspection and cleaning using special stud and nut handling fixtures. Two special alignment studs are installed in stud location Nos. 15 and 45. The lift of the head and replacement after refueling is guided by these studs. These studs are also used to locate the index fixture used for aligning the plenum assembly during removal and replacement. Storage racks are provided for the closure head studs and the alignment studs.

The reactor closure head is lifted out of the canal onto a head storage stand on the operating floor by a head and internals handling fixture attached to the polar crane. The stand is designed to protect the gasket surface of the closure head.

The upper plenum assembly is removed from the reactor, using the head and internals handling fixture and adaptors attached to the polar crane with an internals handling extension, and stored in the deeper portion of the fuel transfer canal on a stand on the canal floor. The reactor vessel stud holes except for locations Nos. 15 and 45 are closed with special plugs that prevent water and/or other foreign substances from entering the holes. The fuel transfer canal is then filled with borated water.

The original plant design provided provisions for optimizing refueling operations by using two fuel handling bridges in each Reactor Building, a Main Bridge and an Auxiliary Bridge, which spanned the fuel transfer canal. The Main Bridge was used to shuttle spent fuel assemblies from the core to the transfer station and new fuel assemblies from the transfer station to the core, while the Auxiliary Bridge was used to relocate partially spent fuel assemblies within the core as specified by the fuel management program. The full core off-load refueling practice is now normally used. Fuel shuffling is performed by completely unloading the core using the Main Bridge, shuffling the control components in the spent fuel pool using manual tools suspended from an overhead hoist mounted on the Spent Fuel Bridge, and then reloading the core. Since the Auxiliary Bridges were no longer needed for their original design purpose (Main Bridge could be used if 'in-core' shuffling *of fuel assemblies* became necessary) and they were an interference for fuel handling activities, the Auxiliary Bridges were physically removed from the Reactor Buildings (ref. NSM X2914).

In the original plant design, each unit's Main Bridge was equipped with two trolley-mounted hoists. One hoist (fuel handling mechanism) was equipped with a fuel grapple and the second hoist (control rod handling mechanism) housed the control rod grapple. (The Unit 3 Main Bridge was later upgraded to one trolley mounted multiple purpose hoist equipped with both fuel and component grapples). The Main Bridges now have one trolley mounted hoist equipped with a fuel grapple only. (ref. NSM-X2914) The Auxiliary Bridges (which consisted of one trolley-mounted hoist with fuel grapple only) for each unit has been removed. Each fuel handling bridge uses a pneumatic system for grapple operation. (ref. NSM X2914)

The Main Bridge moves a spent fuel assembly from the core underwater to the transfer station where the fuel assembly is lowered into the fuel transfer carriage fuel basket. The Main Bridges have a fuel mast only and are not capable of handling components (ref. NSM-X2914). Components are shuffled in the spent fuel pool (after complete core off load) using manual tools suspended from an overhead hoist mounted on the Spent Fuel Bridge, and then reloaded the core

Spent fuel assemblies removed from the reactors are transported to the spent fuel pool from the Reactor Building via fuel transfer tubes by means of the fuel transfer mechanism. The fuel transfer mechanisms are carriages that run on tracks extending from each spent fuel pool through the transfer tubes and into the respective Reactor Building. Each of the two independently operated fuel transfer mechanisms which serve Oconee 1 and 2 is designed to operate in two directions so that either of the two Reactor Buildings can be serviced by one or two mechanisms as required. A rotating fuel basket is provided on each end of each fuel transfer carriage to receive fuel assemblies in a vertical position. The hydraulically operated fuel basket is rotated to a horizontal position for passage through the transfer tube, and then rotated back to a vertical position in the spent fuel pool or Reactor Building for vertical removal or insertion of the fuel assembly.

The spent fuel assemblies are removed from the fuel transfer carriage fuel basket using a fuel handling bridge equipped with a fuel handling mechanism and fuel grapple. This bridge spans the spent fuel pool and permits the refueling crew to store or remove new and spent fuel assemblies in any one of the storage rack positions. Spent fuel assemblies may be moved within the spent fuel pools by use of the fuel handling bridge auxiliary hoist and appropriate remote handling tools. In addition, a Post Irradiation Examination jib crane, with associated grapple that may be used to move fuel, is installed in the Unit 1 and 2 spent fuel pool.

Once refueling is completed, the fuel transfer canal is drained through a pipe located in the deep transfer station area. The canal water is pumped to the borated water storage tank to be available for the next refueling.

During operation of the reactors, the fuel transfer carriages are stored in the respective spent fuel pools, thus permitting a blind flange to be installed on the Reactor Building side of each tube.

Space is provided in each spent fuel pool to receive a spent fuel shipping cask as well as provide for required fuel storage. The layout of the fuel pool is shown on Figure 1-4 through Figure 1-8. The cask area is located at the north end of the fuel pools and adjacent to the fuel racks. Following a decay period, the spent fuel assemblies are removed from storage and loaded into the spent fuel shipping cask under water for removal from the site. The spent fuel shipping cask does not pass over fuel storage racks, or any systems or equipment important to safety when being moved to or from the spent fuel pool.

The spent fuel cask handling facility consists of a 100-ton capacity overhead bridge crane with a 13 foot 6 inch span. The hoist controls are five step magnetic, contactor reversing, secondary resistor type with time delay acceleration and a maximum speed of 9 feet per minute. The hoist is equipped with AC solenoid-operated brake system and an eddy-current brake. The bridge controls are the same as the hoist controls and are equipped with AC solenoid operated brake system and has a maximum speed of 50 feet per minute. The trolley is a single speed, four feet/minute, magnetic contactor reversing type controller with AC solenoid-operated brake system. The cranes were designed in accordance with Electric Overhead Crane Institute's Specification No. 61, Class A.

The cranes were tested in the shop by performing a running test, and load tested at the Oconee site to 98 percent of capacity. The running and load test results were satisfactory. Maintenance of the cranes is in accordance with ANSI B30.2. The structural and mechanical components of the crane are designed to have a minimum factor of safety of 2.5 based on yield strength and rated capacity. The hoist brake system consists of the dynamic AB 707 eddy-current control brake and a 13-inch solenoid-operated shoe brake (Whiting SESA). The bridge is equipped with a solenoid-operated shoe brake for operating the crane by pendant control from the floor. The trolley is equipped with a solenoid-operated shoe brake. The hoist system is equipped with a 75 horsepower motor that produces 328 foot-pounds of torque at full load, 1200 rpm. The starting and instantaneous stalling torque is 902 foot-pounds. The hoist is equipped with a geared lower limit switch for block travel and a paddle-type upper limit switch to prevent a two-blocking situation from occurring.

The cranes are equipped with a sister type hook with safety latch. The hook was load tested and non-destructive tested in the shop. Bethanized wire rope with a safety factor of 6 was used. A lifting adapter to be used between the yoke and the crane hook is also designed to support three times the load. The lifting adapter is a stainless steel member approximately 24 feet long, used to lift the cask from the platform to the bottom of the spent fuel pool.

A decontamination area is located in the building adjacent to each spent fuel pool where the outside surfaces of the casks can be decontaminated prior to shipment by using water, detergent solutions and manual scrubbing to the extent required.

9.1.4.2.3 Safety Provisions

Safety provisions are designed into the fuel handling system to prevent the development of hazardous conditions in the event of component malfunctions, accidental damage or operational and administrative failures during refueling or transfer operations.

All fuel assembly storage facilities employ neutron poison material and/or maintain an eversafe geometric spacing between assemblies to assure fuel storage arrays remain subcritical under all credible storage conditions. The fuel storage racks are designed so that it is impossible to insert fuel assemblies in other than the prescribed locations, thereby assuring the necessary spacing between assemblies. Fuel handling and transfer containers are also designed to maintain an eversafe geometric array. Under these conditions, a criticality accident during refueling or storage is not considered credible.

Fuel handling equipment is designed to minimize the possibility of mechanical damage to the fuel assemblies during transfer operations. If fuel damage should occur, the amount of radioactivity reaching the environment will present no hazard. The fuel handling accident is analyzed in Chapter 15.

All spent fuel assembly transfer operations are conducted underwater. The water level in the fuel transfer canal provides a nominal water level of 9 feet over the active fuel line of the spent fuel assemblies during movement from the core into storage to limit radiation at the surface of the water. The fuel storage racks provide a nominal 23.5 feet of water shielding over the stored assemblies. The minimum water depth over the stored fuel assemblies is equal to, or greater than 21.34 feet. The minimum depth of water over the fuel assemblies and the thickness of the concrete walls of the storage pool are sufficient to limit radiation levels in the working area.

Water in the reactor vessel is cooled during shutdown and refueling as described in Section 9.3.3. Adequate redundant electrical power supply assures continuity of heat removal. The spent fuel pool water is cooled as described in Section 9.1.3. A power failure during the refueling cycle will create no immediate hazardous condition due to the large water volume in both the transfer canal and spent fuel pool. With a normal quantity of spent fuel assemblies in the storage pool and no cooling available, the water temperature in the spent fuel pool would increase very slowly (Section 9.1.3).

During reactor operations, bolted and gasketed closure plates, located on the reactor building flanges of the fuel transfer tubes, isolate the fuel transfer canal from the spent fuel pool. Both the spent fuel pool and the fuel transfer canals are completely lined with stainless clad steel plate for leak tightness and for ease of decontamination. The fuel transfer tubes will be appropriately attached to these liners to maintain leak integrity. The spent fuel pool cannot be accidentally drained by gravity since water must be pumped out.

During the refueling period the water level in both the fuel transfer canal and the spent fuel pool is the same, and the fuel transfer tube valves are open. This eliminates the necessity for interlocks between the fuel transfer carriages and transfer tube valve operations except to verify full-open valve position.

The fuel transfer canal and spent fuel pool water will have a boron concentration as specified by the Core Operating Limits Report. Although this concentration is sufficient to maintain core shutdown if all of the control rod assemblies were removed from the core, only a few control rods will be removed at any one time during the fuel shuffling and replacement. Although not required for safe storage of spent fuel assemblies, the spent fuel pool water will also be borated so that the transfer canal water will not be diluted during fuel transfer operations.

The fuel transfer mechanisms permit initiation of the fuel basket rotation from the building in which the fuel basket is being loaded or unloaded. Carriage travel and fuel basket rotation are interlocked to prevent inadvertent carriage movement when the fuel basket is in the vertical position. Rotation of the fuel baskets is possible only when the carriages are in the rotating frame at the end of travel.

Interlocks are provided to prevent operation of the bridges or trolleys with a fuel assembly until the assemblies have been hoisted to the upper limit in the mast tube. Mandatory slow zones are provided for the hoisting mechanisms as the grapples approach the core and fuel baskets during insertion of fuel assemblies. The slow zones will be in effect during entry into the reactor core or fuel storage rack and just before and during bottoming out of the fuel assemblies. The controls are appropriately interlocked to prevent simultaneous movement of the bridge, trolley or hoists. The grapple mechanisms are interlocked with the hoists to prevent vertical movement unless the grapples are either fully opened or fully closed. The fuel grapple is so designed that when loaded with the fuel assembly, the fuel grapple cannot be opened as a result of operator error, electrical, or pneumatic failure.

All operating mechanisms of the system are located in the fuel handling and storage area for ease of maintenance and accessibility for inspection prior to start of refueling operations. All electrical equipment, with the exception of some limit switches, is located above water for greater integrity and ease of maintenance. The hydraulic systems which actuate the fuel basket rotating frame use demineralized water for operation.

Deleted paragraph(s) per 2005 update

The Main fuel handling bridges have a fuel mast only and are not capable of handling components. The original design of the Main fuel bridges included separate hoists, which allowed control components to be exchanged between fuel assemblies within the Reactor building. This capability has been removed. (ref NSM-X2914) All lifts for handling of reactor closure heads and reactor internal assemblies will be made using the Reactor Building Polar crane.

Travel speeds for the fuel handling bridges, hoists and fuel transfer carriages will be controlled to assure safe handling conditions.

Since 1990, Oconee has been involved in transferring spent fuel from the Unit 1 and 2 and the Unit 3 Spent Fuel Pools to an on-site Independent Spent Fuel Storage Installation. A specially designed transfer cask and associated handling equipment is used for this operation. Cask handling accidents are addressed in Chapter 15. More detailed information on cask loading and handling activities can be found in the ONS Site Specific and General License System ISFSI UFSARs.

9.1.5 Overhead Heavy - Load Handling Systems

9.1.5.1 Introduction and Licensing Background

As a result of Generic Task A-36, “Control of Heavy Loads Near Spent Fuel," NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” (Reference 28) was developed. Following the issuance of NUREG-0612, Generic Letter 80-113, dated December 22, 1980 (Reference 29), as supplemented on February 3, 1981, by Generic Letter 81-07 (Reference 30), was sent to all operating plants, applicants for operating licenses and holders of construction permits requesting that responses be prepared to indicate the degree of compliance with the guidelines of NUREG-0612. Phase 1 responded to Section 5.1.1 of NUREG-0612 and addressed applicable codes and standards for the subject cranes and special lifting devices, crane operator training and qualification and procedures for heavy load handling. Phase II responded to Sections 5.1.2, 5.1.3, 5.1.5, and 5.1.6 (5.1.4 was specific to BWRs) of NUREG-0612 and addressed the need for mechanical stops or electrical interlocks, the need for single- failure-proof handling systems and load drop consequence analyses. By correspondence dated June 26, July 30, August 31, October 1, 1981, and February 1, October 8, November 5, and December 22, 1982, (References 31, 32, 33, 34, 35, 36, 37, and 38) Duke provided the Oconee responses.

On April 20, 1983, the NRC issued its Safety Evaluation Report (SER) (Reference 39) for Oconee Nuclear Station (ONS), concluding that “... the guidelines of NUREG-0612, Section 5.1.1, have been satisfied.” The SER further states that “... Phase 1 actions taken for the Oconee units are acceptable."

On June 28, 1985, the NRC issued Generic Letter 85-11 (Reference 40). This generic letter concluded that Phase 1 had provided improvements in heavy load handling and that Phase II was no longer required. By correspondence dated October 2, 1987 (Reference 41), Duke concluded that implementation of any actions identified in Phase II are not a requirement. NRC responded with a letter dated November 2, 1987 (Reference 42), stating NRC has no objections with the statement that Duke will implement only those Phase II commitments which Duke considers appropriate. On October 31, 2005, the NRC issued Regulatory Issue Summary (RIS) 2005-25, "Clarification of NRC Guidelines for Control of Heavy Loads" (Reference 45), as a result of recommendations developed through Generic Issue (GI) 186, "Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants." (Reference 44) The RIS reemphasized the guidelines of NUREG-0612 and identified relevant operating experience and inspection information related to the movement of heavy loads. On May 29, 2007, the NRC issued Supplement 1 of RIS 2005-25 (Reference 46) addressing remaining recommendations associated with GI 186 and communicated regulatory expectations related to safe load handling.

On September 14, 2007, an "Industry Initiative on Heavy Load Lifts" (Reference 55) was initiated by the Nuclear Energy Institute (NEI) to specify those actions to be taken by each plant to ensure that heavy lifts continue to be conducted safely and that each plant's licensing bases accurately reflected those plant practices.

9.1.5.2 Design Basis

The design bases of the overhead heavy load systems are to:

1. Assure that the potential for a load drop is extremely small,
2. Assure that in the event of a postulated reactor vessel head drop, the core remains covered and cooled, and,
3. Assure the consequences of a load drop in the spent fuel pool meet the acceptance criteria of NUREG-0612.

9.1.5.3 Scope of Heavy Load Handling Systems

All cranes and hoists lifting heavy loads over spent fuel or safe shutdown equipment comply with the guidelines of NUREG-0612 and are consistent with Duke's responses and commitments related to the handling of heavy loads.

9.1.5.4 Control of Heavy Lifts Program

The control of heavy lifts consists of the following:

1. Duke's commitments in response to NUREG-0612, Phase 1 elements
2. Duke's response to the NEI Initiative on Heavy Load Lifts
3. Reactor pressure vessel head lift load drop analysis assumptions (lift height and medium present) are incorporated into plant procedures
4. Load drop analyses have been performed for loads over the spent fuel pool.

Duke maintains a Lifting Program to minimize the potential for adverse interaction between overhead load handling operations and: 1) nuclear fuel assemblies to ensure a sub-critical configuration and preclude radiological consequences and; 2) structures, systems and components (SSCs) selected to ensure safe shutdown of the plant following a postulated heavy load drop event. A "heavy load" has been defined as one weighing 1500 lbs. or more. No suspended loads of more than 3000 lbs shall be transported over fuel stored in the spent fuel pool. The bases of the NRC acceptance of Duke's program is summarized in the April 20, 1983 SER. The objective of the program is to ensure that all load handling systems are designed, operated, and maintained such that their probability of failure is uniformly small and their use appropriate for the critical tasks in which they are employed.

9.1.5.4.1 Oconee Commitments in Response to NUREG-0612, Phase 1 Elements

The Duke Lifting Program is based on the NEI "Industry Initiative on Heavy Load Lifts" and the following general guideline areas of NUREG-0612, Section 5.1.1:

Guideline 1 - Safe Load Paths

Guideline 2 - Load Handling Procedures

Guideline 3 - Crane Operator Training

Guideline 4 - Special Lifting Devices

Guideline 5 - Lifting Devices (not specifically designed)

Guideline 6 - Cranes (inspection, testing and maintenance)

Guideline 7 - Crane Design

The following sections summarize the commitments made by Duke in compliance with Section 5.1.1 of NUREG-0612:

Safe Load Paths

NUREG-0612, Section 5.1.1 defines a "Safe Load Path" as one which minimizes the potential for heavy loads, if dropped, to impact irradiated fuel in the reactor vessel and in the spent fuel pool, or to impact safe shutdown equipment.

Oconee has established safe load paths for all load handling systems identified as "handling heavy loads in the vicinity of vital equipment." Heavy loads are considered to be those weighing 1500 lb or more. Vital equipment includes those systems necessary for safe shutdown and decay heat removal and those involved with spent fuel handling.

The safe load paths for cranes follow beams and avoid vital systems where possible. The safe load paths for monorails are the vertical projections of the beams onto the floor.

Safe load paths are indicated on general arrangement drawings. Guidance for following safe load paths is contained in site procedures and directives.

Load Handling Procedures

Load handling requirements are specified in site procedures and directives which describe; safe load paths, instructions for special lifts, appropriate procedures, and any restrictions placed on the crane or hoist.

Crane Operator Training

Crane operators are qualified, trained and conduct themselves in accordance with ANSI B30.2-1976.

Special Lifting Devices

Special lifting devices at Oconee comply with applicable ANSI standards and NUREG-0612 guidelines. Many special lift devices at Oconee were designed and procured prior to the publication of ANSI N14.6-1978 and therefore were not designed in specific accordance with that standard. As a result, the NRC identified exceptions and "approaches consistent with this guideline" in "Synopsis of Issues Associated with NUREG 0612 dated May 4, 1983" (Reference 57). This provided information to better determine which ONS special lift devices require specific inspections.

Lifting Devices (not specifically designed)

All lifts are made by qualified personnel who, by experience and/or training, are cognizant in the movement of loads.

The use of lifting devices at Oconee Nuclear Station complies with the applicable ANSI standards and NUREG-0612 guidelines. Lifting devices used consist of the appropriate size and number of rigging hardware, such as chain-falls, chokers, and slings as determined by the rigger. In making a selection, the rigger draws on experience and training. Choker and sling sizing is determined by a conservative estimated weight of the load.

Slings are required to be inspected before each use.

Dynamic loads on slings are properly accounted. Typically, these dynamic loads can be neglected due to being a reasonably small percentage of the overall static load. (Based on hoisting speed less than 30 fpm) (Reference 39)

Inspection, Testing and Maintenance

The Oconee Crane Inspection Program is discussed in Section 18.3.5 of the Oconee UFSAR (Reference 50).

Oconee Nuclear Station crane inspection, testing, and maintenance programs comply with the requirements of ANSI B30.2-1976, Section 2-1.

Crane Design

Oconee Nuclear Station evaluated its overhead heavy load handling systems for design compliance with CMAA-70 and ANSI B30.2-1976. The generator room crane in the standby shutdown facility (SSF) is exempt from CMAA-70 and ANSI B30.2-1976 design requirements because it is a manually operated, single girder overhead traveling crane. With the exception of the SSF generator room crane, the cranes listed in Section 2.1.1(a) of the Oconee SER, dated April 20, 1983, were designed in accordance with Duke Power Company specifications, Electric Overhead Crane Institute (EOCI) Specification 61, and USAS B30.2.0-1976. A comparative study of CMAA-70 versus EOCI-61 identified 13 items of difference between the two specifications. The 13 items of difference are enumerated within the April 20, 1983, SER.

Oconee Nuclear Station cranes substantially meet the intent of this guideline on the basis that the cranes were originally built to EOCI-61. In addition, for those criteria in CMAA-70 noted to be more restrictive than the requirements of EOCI-61, Oconee demonstrated compliance with CMAA-70 or provided reasonable assurance that the existing design meets the intent of the CMAA criteria.

9.1.5.4.2 Oconee Response to NEI Initiative on Heavy Load Lifts

9.1.5.4.2.1 Reactor Vessel Head Lifting Procedures

In response to the September 14, 2007 NEI "Industry Initiative on Heavy Load Lifts," Oconee procedures used to control the lift and replacement of the reactor vessel head were verified to contain limits of load height above the reactor vessel flange. These load height limits are based on the existing Oconee load drop analysis performed in March, 1983. These load height limits provide additional assurance that the core will remain covered and cooled in the event of a postulated reactor vessel head drop.

The Oconee reactor vessel head load drop analysis meets the guidance and acceptance criteria developed by NEI as part of its initiative.

9.1.5.4.2.2 Load Drops in the Spent Fuel Pool Building

The Spent Fuel Pool (SFP) slab was designed for the postulated cask drop accident. Fill concrete was placed from sound rock to the bottom of the fuel pool slab in the area covered by the cask crane to prevent the shearing of a large plug from the pool slab in the event the cask was accidentally dropped.

The SFP concrete floor slab is designed to withstand a 100 ton cask drop. However, localized concrete could be crushed and the steel liner plate punctured in the area of the dry storage cask impact. For the purpose of analyzing the event, a gap of 1/64 inch for a perimeter of 308 inches in the liner plate was assumed. The calculated leakage of pool water through the gap is 21.3 gallons per day. This amount of water loss is within the capability of the SFP makeup sources.

The evaluation and consequence of fuel shipping cask drops is discussed in Section 15.11.2.4 of the UFSAR (Reference 48). The evaluation and consequence of dry storage transfer cask drops is discussed in Section 15.11.2.5 of the UFSAR (Reference 49).

The radiological consequence of either a fuel shipping cask drop or a dry storage transfer cask drop is within Regulatory Guide 1.183 (Reference 56) limits.

9.1.5.5 Safety Evaluation

The Duke Lifting Program provides a defense-in-depth approach which ensures that all load handling systems are designed, operated, and maintained such that the probability of their failure is very small and the use of said handling systems appropriate for the tasks in which they are employed. In addition, procedures to lift and replace the reactor vessel head ensure the core remains covered and cooled when a reactor vessel head drop is postulated.

9.1.6 References

1. Calculation OSC-1870, "Oconee Nuclear Station Unit 3 Poison Spent Fuel Storage Racks"
2. Calculation OSC-6574, "Oconee Nuclear Station Unit 1 and 2 Poison Spent Fuel Storage Racks"
3. Calculation OSC-1875, "Slashing Effect of Water in Spent Fuel Pools"
4. Studsvik, "CASMO-3 A Fuel Assembly Burnup Program," STUDSVIK/NFA-89/3, Revision 4.4, January, 1991.
5. Studsvik, "TABLES-3 Library Preparation Code for SIMULATE-3", STUDSVIK/SOA-92/03, Revision 0, April, 1992.
6. Studsvik, "SIMULATE-3 Advanced Three-Dimensional Two-Group Reactor Analysis Code", STUDSVIK/SOA-92/01, Revision 0, April, 1992.
7. M.N. Baldwin, et. al., "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel", The Babcock and Wilcox Company, BAW-1484-7, July, 1979.
8. Duke Power Company, Letter to U.S. Nuclear Regulatory Commission Document Control Desk, from J.W. Hampton, November 22,1994, "Oconee Nuclear Station Docket Nos. 50-269,-270,-287 Unit 3 Cycle 16 Reload Technical Specifications".
9. U.S. Nuclear Regulatory Commission letter to All Power Reactor Licensees, from B.K. Grimes, April 14, 1978, "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications".

10. Oak Ridge National Laboratory, "SCALE 4.2, A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation", Volumes I-IV, NUREG/CR-200, Revision 4, April 1995.
11. Letter from W. O. Parker, Jr. (DPC) to H. R. Denton (USNRC), dated July 1, 1980.
12. Letter from W. O. Parker, Jr. (DPC) to H. R. Denton (USNRC), dated July 25, 1980.
13. USNRC Branch Technical Position (BTP) APCSB 9-2, "Residual Decay Energy for Light Water Reactors for Long Term Cooling".
14. Letter from H. B. Tucker (DPC) to H. R. Denton (USNRC), dated March 10, 1983.
15. NUREG-0800, Standard Review Plan Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System".
16. Safety Evaluation by the Office of Nuclear Reactor Regulation related to Amendment No. 209 to Facility Operating License DPR-38, Amendment No. 209 to Facility Operating License DPR-47, Amendment No. 206 to Facility Operating License DPR-55, Duke Power Company, Oconee Nuclear Station, Units 1, 2, and 3, Docket Nos. 50-269, 50-270, and 50-287.
17. Letter from J. W. Hampton to USNRC, dated November 22, 1994, Oconee Nuclear Station Docket Nos. 50-269, -270, -287 Unit 3 Cycle 16 Reload Technical Specifications, May 3, 1995.
18. License amendment 123, 123, and 120 for Units 1, 2 and 3 respectively.
19. License amendment 90, 90, and 87 for Units 1, 2 and 3 respectively.
20. Calculations OSC-1870, Rev D3, "Oconee Nuclear Station Unit 3 Poison Spent Fuel Storage Racks".
21. Calculations OSC-6574, Rev 0, "Oconee Nuclear Station Unit 1 and 2 Poison Spent Fuel Storage Racks".
22. Application for Renewed Operating Licenses for Oconee Nuclear Station, Units 1, 2, and 3, submitted by M. S. Tuckman (Duke) letter dated July 6, 1998 to Document Control Desk (NRC), Docket Nos. 50-269, -270, and -287.
23. NUREG-1723, Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3, Docket Nos. 50-269, 50-270, and 50-287.
24. NUREG/CR-0200, Section S2, "SAS2H: A Coupled One-Dimensional Depletion and Shielding Analysis Module."
25. NUREG/CR-0200, Section D1, "ORIGEN-ARP: Automatic Rapid Process for Spent Fuel Depletion, Decay, and Source Term Analysis."
26. "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light-Water Reactor Power Plants," L. Kopp, U.S. NRC, August 19, 1998.
27. Letter from Bruce H. Hamilton to USNRC dated March 1, 2006, "License Amendment to Reconcile 10CFR50 and 10CFR72 Criticality Requirements for Loading and Unloading Dry Spent Fuel Storage Canisters in the Spent Fuel Pool," LAR No. 2005-009.
28. NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" Resolution of Generic Technical Activity A-36, July 1, 1980.
29. Letter from D. G. Eisenhut (NRC) to all licensees dated December 22, 1980. Subject: Control of Heavy Loads (Generic Letter 80-113).
30. Letter from D. G. Eisenhut (NRC) to all licensees dated February 3, 1981. Subject: Control of Heavy Loads (Generic Letter 81-07).

31. Letter from W. O. Parker, Jr. (Duke) to H. R. Denton (NRC) dated June 26, 1981. Subject: Response to the Dec 22, 1980 Generic Letter 80-113.
32. Letter from W. O. Parker, Jr. (Duke) to H. R. Denton (NRC) dated July 30, 1981. Subject: Control of Heavy Loads, NUREG-0612.
33. Letter from W. O. Parker, Jr. (Duke) to H. R. Denton NRC) dated August 31, 1981. Subject: Control of Heavy Loads, NUREG-0612.
34. Letter from W. O. Parker, Jr. (Duke) to H. R. Denton (NRC) dated October 1, 1981. Subject: Control of Heavy Loads, NUREG-0612.
35. Letter from W. O. Parker, Jr. (Duke) to H. R. Denton (NRC) dated February 1, 1982. Subject: Control of Heavy Loads, NUREG-0612.
36. Letter from H. B. Tucker (Duke) to H. R. Denton (NRC) dated October 8, 1982. Subject: Control of Heavy Loads, NUREG-0612.
37. Letter from H. B. Tucker (Duke) to H. R. Denton (NRC) dated November 5, 1982. Subject: Control of Heavy Loads, NUREG-0612.
38. Telephone Conversation of S. Roberts (FRC) and P. Wagner (NRC) dated December 22, 1982. Subject: Control of Heavy Loads, NUREG-0612. (No transcripts available; this is listed as Ref. 11 in Oconee SER dated April 20, 1983).
39. Letter from J. F. Stolz (NRC) to H. B. Tucker (Duke) dated April 20, 1983. Subject: Control of Heavy Loads [Safety Evaluation Report].
40. Letter from H. L. Thompson, Jr. (NRC) to All Licensees for Operating Reactors dated June 28, 1985. Subject: Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants", NUREG-0612 (Generic Letter 85-11).
41. Letter from H. B. Tucker (Duke) to Document Control Desk (NRC) dated October 2, 1987. Subject: Control of Heavy Loads.
42. Letter from Lawrence P. Crocker (NRC) to H. B. Tucker (Duke) dated November 2, 1987. Subject: Control of Heavy Loads, Phase II.
43. Letter from M. S. Tuckman (Duke) to Document Control Desk (NRC) dated May 13, 1996. Subject: Response to NRC Bulletin 96-02: Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety-Related Equipment.
44. Generic Issue 0186, “Potential Risk and Consequences of Heavy Load Drops in Nuclear Power, dated April 1, 1999.
45. NRC Regulatory Issue Summary 2005-25, “Clarification of NRC Guidelines for Control of Heavy Loads” dated October 31, 2005.
46. NRC Regulatory Issue Summary 2005-25, Supplement 1, “Clarification of NRC Guidelines for Control of Heavy Loads” dated May 29, 2007.
47. Oconee UFSAR Section 3.8.4.4, “Design and Analysis Procedures”.
48. Oconee UFSAR Section 15.11.2.4, “Fuel Shipping Cask Drop Accidents”.
49. Oconee UFSAR Section 15.11.2.5, “Dry Storage Transfer Cask Drop Accident in Spent Fuel Pool Building”.
50. Oconee UFSAR, Section 18.3.5, “Crane Inspection Program”.

51. Selected Licensee Commitment (SLC) 16.9.16, Reactor Building Polar Crane and Auxiliary Hoist (RCS System Open).
52. Selected Licensee Commitment (SLC) 16.12.5, Loads Suspended Over Spent Fuel in Spent Fuel Pool.
53. Duke Energy Nuclear Lifting Program.
54. NRC Enforcement Guidance Memorandum 07-006, "Enforcement Discretion for Heavy Load Handling Activities," dated September 28, 2007.
55. NEI "Industry Initiatives on Heavy Load Lifts," dated September 14, 2007.
56. Regulatory Guide 1.183, "Alternative Radiological Source Term for Evaluating Design Basis Accidents at Nuclear Power Reactors".
57. NRC "Synopsis of Issues Associated with NUREG 0612" dated May 4, 1983.
58. Safety Evaluation Report dated June 19, 1979 relating to the modification of the Oconee Units 1/2 common Spent Fuel Pool.
59. Safety Evaluation Report dated September 29, 1983 relating to the modification of the Oconee Unit 3 Spent Fuel Pool.

THIS IS THE LAST PAGE OF THE TEXT SECTION 9.1.

9.2 Water Systems

9.2.1 Component Cooling System

9.2.1.1 Design Bases

The Component Cooling System is designed to provide cooling water for various components in the Reactor Building as follows: letdown coolers, reactor coolant pump cooling jacket and seal coolers, quench tank cooler, and control rod drive cooling coils. The design cooling requirement for the system is based on the maximum heat loads from these sources. The system also provides an additional barrier between high pressure reactor coolant and service water to prevent an inadvertent release of activity.

9.2.1.2 System Description and Evaluation

The Component Cooling System is shown schematically on Figure 9-8, and the performance requirements of the system are tabulated in Table 9-3. The following is a brief functional description of the major components of the system and their sharing between nuclear units of the station:

Component Cooler

Each component cooler is designed for the total Component Cooling System heat load for a reactor unit. Oconee 1 and 2 each have a single component cooler with a shared common spare. Oconee 3 has two coolers. The coolers reject the heat load to the Low Pressure Service Water System.

Component Cooling Pumps

Each component cooling pump is designed to deliver the necessary flows to the letdown coolers, reactor coolant pump cooling jackets and seal coolers, quench tank cooler, and control rod drive cooling coils. Each unit has one operating pump and one spare.

Component Cooling Surge Tank

This tank allows for thermal expansion and contraction of the water in this closed-loop system. It also provides the required NPSH for the component cooling pumps.

Control Rod Drive Filters

Two filters are provided in the cooling water circuit to the control rod drives to prevent particulates from entering the drive cooling coils. Only one filter is used at a time, with the second as a spare. A bypass is also provided.

9.2.1.3 Mode of Operation

During operation, one component cooling pump and one component cooler recirculate and cool water to accommodate the system heat loads for each reactor unit. The component cooling surge tank accommodates expansion, contraction, and leakage of coolant into or out of the system. The surge tank provides a reservoir of component cooling water until a leaking cooling line can be isolated. Makeup water is added to the system in the surge tank. Corrosion inhibiting chemicals are added to the system in the surge tank or the chemical addition feeder (pot).

9.2.1.4 Reliability Considerations

The Component Cooling System performs no emergency functions. Redundancy in active components is provided to improve system reliability. The pumps, coolers, surge tank, and most of the instrumentation are located in the Auxiliary Building and are accessible for inspection and maintenance.

9.2.1.5 Codes and Standards

The components of the system are designed to the codes and standards given in Table 9-13.

9.2.1.6 System Isolation

Since the Component Cooling System is not an engineered safeguards system, Reactor Building isolation valves are automatically closed on a high Reactor Building pressure signal to provide building isolation. The Reactor Building inlet lines are isolated by two check valves, one on the outside and one on the inside of the Reactor Building. The Reactor Building outlet line is isolated by an electric motor-operated valve on the inside and by a pneumatic valve on the outside of the Reactor Building.

9.2.1.7 Leakage Considerations

Water leakage from piping, valves, and other equipment in the system is not considered to be detrimental since the cooling water is normally nonradioactive. Welded construction is used throughout the system to minimize the possibility of leakage except where flanged connections are required for servicing.

In-leakage of reactor coolant to the system is detected by a radiation monitor (RIA-50) located in the pump recirculation line and is also indicated by an increase in surge tank level. A defective coil or thermal barrier tube of a reactor coolant pump can be remotely isolated by an electric motor-operated valve on the outlet cooling line and a stop-check valve on the inlet line. On Unit 1 the RCS leak can be isolated. On Units 2 and 3 the RCS leak will be vented to containment through CC System relief valves. A letdown cooler leak can be remotely isolated with motor-operated valves on the reactor coolant side of the cooler. The cooling water side can be completely isolated by closing a remotely operated, motor-actuated valve on the inlet of the cooler and the manual valves on the outlet cooling lines. Leakage from the quench tank cooler can be isolated by manual valves on the reactor coolant side. The cooling water side can be completely isolated by two manual valves. Access to the manual valves is not available at power operations.

9.2.1.8 Failure Considerations

Since the system serves no engineered safeguards function, the only consideration following a loss-of-coolant accident is the operation of the containment isolation valves. Redundant isolation valves are provided as described in Chapter 6. Failures and malfunction of components during normal operation were evaluated. Operation of the Component Cooling System is essential to normal reactor operation. In the event of loss of a component cooling pump, the standby pump will automatically start and maintain cooling water flow. The complete loss of cooling water flow does not require immediate reactor shutdown. However, procedures will require the operator to shutdown the reactor to protect the control rod drive coils. The reactor coolant pumps can be operated without component cooling water if seal injection flow is available.

9.2.2 Cooling Water Systems

9.2.2.1 Design Bases

The cooling water systems for the station are designed to provide redundant cooling water supplies to insure continuous heat removal capability both during normal and accident conditions.

The Low Pressure Service Water (LPSW) and portions of the Condenser Circulating Water (CCW) systems are designed so no single component failure will impair emergency safeguards operation. Redundant pumping capability is provided, heat exchangers and pumps can be isolated and pressure reducing valves are provided with bypasses.

All cooling systems are designed to be operated and monitored from the control room. Component design parameters are given in Table 9-4.

The design purpose of each of the cooling water systems is outlined below:

Condenser Circulating Water (CCW) System - This system provides for cooling of the condensers during normal operation of the plant. The system generally uses lake water as the ultimate heat sink for decay heat removal during cooldown of the plant. In some events, such as the loss of Lake Keowee, the water trapped in the CCW piping is used as the ultimate heat sink. The CCW System is the suction source for other service water systems, including HPSW, LPSW, ASW, and SSF ASW. In addition, CCW provides a heat sink for the RCW system. Following a design basis event involving loss of the CCW pumps, the Emergency Condenser Circulating Water (ECCW) System supplies suction to the LPSW pumps.

High Pressure Service Water (HPSW) System - This system provides a source of water for fire protection throughout the station. In the event of a loss of the normal LPSW supply, HPSW automatically supplies cooling water to the HPI pump motor coolers. For loss of A.C. power, HPSW via the Elevated Water Storage Tank automatically supplies cooling water to the Turbine Driven Emergency Feedwater Pump Oil Cooler and the LPSW Leakage Accumulator for all Units.

Low Pressure Service Water (LPSW) System - This system provide cooling water for normal and emergency services throughout the station. Safety related functions served by this system are:

1. Reactor Building cooling units.
2. Decay heat removal coolers.
3. High pressure injection pump motor bearing coolers.
4. Motor-Driven Emergency Feedwater Pump motor air coolers.
5. Deleted Per 2006 Update.
6. Siphon Seal Water.

Recirculated Cooling Water (RCW) System - This is a closed loop system to supply corrosion inhibited cooling water to various components. This system has no direct safety related functions.

Essential Siphon Vacuum (ESV) System - This system supports the Condenser Circulating Water (CCW) system by removing air from the CCW Intake header during normal and siphon modes of operation. The nuclear safety-related functions are:

1. Remove air from the CCW Intake Headers during normal operation to ensure that the operable Intake Headers are primed at the start of an event requiring the siphon mode of operation.
2. Remove air from the CCW Intake Headers during the siphon mode of operation to ensure that the siphon does not fail due to air accumulation during a Design Basis Accident involving loss of power to the CCW pumps.

Paragraph(s) Deleted Per 2000 Update.

Siphon Seal Water (SSW) System - This system's nuclear safety-related function is to support the ESV system by providing operating liquid to the ESV pumps. The ESV pumps are liquid ring vacuum pumps which require a continuous supply of water in order to create a vacuum. Additionally, it has a non-nuclear safety-related function of providing sealing and cooling water to the CCW pumps and motors.

On July 18, 1989 the NRC Issued Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment," requesting holders of operating licenses to supply information about their respective service water systems to assure the NRC of compliance with the recommended actions of

Generic Letter 89-13, and to confirm that the safety functions of their respective service water systems are being met. Oconee's responses to Generic Letter 89-13 are contained in references 7, 8, 9, 10, 12. In order to assure the adequacy of the Oconee service water systems and safety related heat exchangers to perform their functions as designed, a Service Water System Program has been established in accordance with NSD-312. The Service Water System Program consists of all those activities related to the service water systems and components, including periodic inspections, repairs, replacements, monitoring and testing.

9.2.2.2 System Description and Evaluation

9.2.2.2.1 Condenser Circulating Water System (CCW)

The Little River arm of Lake Keowee is the source of water for the CCW systems. Figure 2-4 shows the arrangement of the systems with respect to the two branches of Lake Keowee. Each unit has four condenser circulating water pumps supplying water via two 11 ft. conduits into a common condenser intake header under the turbine building floor. The discharge from the condenser is returned to the Keowee River arm of Lake Keowee.

The suction of the condenser circulating water pumps extends below the maximum drawdown of the lake. The intake structure is provided with screens which can be manually removed for periodic cleaning.

The CCW system is designed to take advantage of the siphon effect so the pumps are required only to overcome pipe and condenser friction loss.

The CCW system has an emergency discharge line to the Keowee hydro tailrace. This discharge line is connected to each of the three condensers of each unit. Under a loss-of-power situation, the emergency discharge line will automatically open and the CCW system will continue to operate as an unassisted siphon system supplying sufficient water to the condenser for decay heat removal and emergency cooling requirements. This siphon system is the Emergency Condenser Circulating Water (ECCW) System and can be divided into two distinct parts. The "first siphon" takes suction from the CCW intake canal and supplies flow to the CCW crossover header in the Turbine Building basement, where the LPSW System takes its suction. The "second siphon" takes suction from the condenser inlet piping, supplies flow through the condenser, and discharges to the Keowee Hydro tailrace. A loss of function of the second siphon would not affect the capability of the first siphon to perform its function.

In a loss of off-site power (LOOP) situation, the CCW pumps will be tripped by a load shed command. The ECCW System first siphon is required to supply suction to the LPSW System until a CCW pump can be manually restarted by the control room operator. Restart of a CCW pump is not required since the ESV system can maintain the first siphon for the duration of the event. Gravity flow (without relying on the siphon) to the suction of the LPSW pumps is possible if the lake level is sufficiently above the bottom of the CCW intake piping to maintain the required NPSH and flow demand. Refer to Section 16.9.7, Selected Licensee Commitments Manual, for additional requirements regarding the CCW Supply to the LPSW System.

During a loss of all AC power situation (Station Blackout), the CCW System is not required to supply suction to the LPSW System since power to the LPSW pumps would not be available. The second siphon is not required. Decay heat removal can be accomplished by venting steam to the atmosphere using the main steam safety valves or the manual atmospheric dump valves. The CCW piping has sufficient inventory to cope with a four-hour Station Blackout by supplying suction to the SSF Auxiliary Service Water System. (Reference 8.3.2.2.4.)

During normal operation, the continuous vacuum priming system removes noncondensible gases from portions of the CCW System. An emergency steam air ejector (ESAE) is available to enhance operation of the second siphon if the vacuum priming pumps are lost due to a loss of power. The essential siphon vacuum (ESV) system is connected to the CCW inlet header to remove non-condensible gases during normal and siphon operations.

Pursuant to the recommendations of the Oconee Probabilistic Risk Assessment study a pushbutton has been installed in the control room for sending a close signal to the CCW pump discharge valves. The capability to close the CCW valves is needed to protect against the possibility of CCW siphoning into the turbine building basement, causing flooding.

The intake canal that supplies water from Lake Keowee to the suction of the CCW pumps contains a submerged weir. The purpose of this weir is to provide an emergency pond of cooling water if the water supply from Lake Keowee were lost. This emergency pond could be recirculated through the condensers and back to the intake canal for decay heat removal as long as the intake canal level remains sufficient. However, during the operating license review of Oconee Units 2 and 3, the Atomic Energy Commission (AEC) staff requested a reanalysis of the capability of the weir to withstand hydraulic forces using more conservative assumptions than those used in the original design. Based on results obtained, the AEC staff concluded that a rapid drawdown of Lake Keowee could cause considerable displacement of the riprap used to face the weir. However, the AEC staff did not require Duke Power to redesign the weir, since the water trapped in the condenser intake and discharge piping below elevation 791 ft. MSL is adequate to supply the three Oconee units with steam generator boil off for safe shutdown for a period of 37 days (Reference 1). The Auxiliary Service Water (ASW) System is capable of using the inventory trapped in the CCW piping for decay heat removal (Reference 9.2.3). Therefore, the licensing basis does not rely on the weir nor recirculation of the intake canal for decay heat removal after a loss of Lake Keowee event (Reference 2).

9.2.2.2.2 High Pressure Service Water System (HPSW)

The schematic arrangement of the HPSW system is shown on Figure 9-10. This system is used primarily for fire protection throughout the Oconee station. In the event of a loss of the normal LPSW supply, HPSW automatically supplies cooling water to the HPI pump motor coolers. For loss of AC power, HPSW via the elevated water storage tank automatically supplies cooling water to the turbine driven emergency feedwater pump oil cooler for all units. HPSW is also used as a backup supply to the SSW system. Refer to Sections 16.9.7 and 16.9.8 for specific requirements to support the LPSW System.

Two full size (6000 gal/min at 117 psig) and one reduced size (500 gal/min at 117 psig) high pressure service water pumps supply the high pressure system. A 100,000 gallon elevated water storage tank provides inventory for a backup supply of water.

The 500 gal/min pump will normally operate to keep pressure on the fire headers. In the event of a fire, one full size pump provides adequate capacity for automatically maintaining the elevated water storage tank inventory. The second full size pump is an installed spare. The HPSW pumps take suction from the CCW system. The HPSW and LPSW pump suctions are connected to the 42 inch cross-connection between the Condenser Circulating Water inlet headers for the three units. Manual isolation valves are provided so that service water may be supplied from any or all of the inlet headers.

Portions of the High Pressure Service Water 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.

9.2.2.2.3 Low Pressure Service Water System (LPSW)

The schematic arrangement of the LPSW system is shown on Figure 9-11 and Figure 9-12. Oconee 1 and 2 share three 15,000 gal/min LPSW pumps. The LPSW pumps and the HPSW pumps take suction from the 42 inch crossover line between the condenser inlet headers; two LPSW pumps are supplied by one suction line and the other pump is supplied by the other suction line. The HPSW system is connected to

LPSW at the LPSW pump discharge, but the interconnections are not used. The alignment of HPSW to LPSW is not credited to mitigate any design basis accident or design event.

Suction is provided to the LPSW pumps via gravity flow or siphon flow from the CCW System (ECCW mode) following a design basis accident where the CCW pumps are not running. Lake level is administratively controlled to maintain sufficient NPSH for the LPSW pumps under these conditions.

The LPSW pumps have a minimum continuous flow rate of 4250 gpm based on manufacturer's recommendation. On Oconee Units 1 & 2, two LPSW pumps are normally operating with the third pump in standby. Therefore, on Oconee Units 1 & 2, the potential for interaction between running LPSW pumps is possible whenever the total demand from system loads is minimized. The potential exists where the stronger pump may close the weaker pump's discharge valve and keep it closed. The weaker pump would then be exposed to extended dead-head conditions. To minimize the potential for deadheading, procedural guidance has been provided to ensure LPSW flow will be maintained greater than 4,000 gpm on the shutdown unit whenever either Unit 1 or Unit 2 is shutdown in refueling. If this flow rate cannot be maintained on the shutdown unit, the LPSW system must be reduced to one pump operation. (References 3, 4)

On an engineered safeguards signal, the standby LPSW pump(s) starts resulting in three Unit 1 & 2 LPSW pumps operating or two Unit 3 LPSW pumps operating. Under this condition the potential exists for the LPSW pumps to be operated below the recommended minimum continuous flow rate of 4250 gpm per pump, or for a stronger pump to deadhead a weaker pump during low flow conditions. To avoid pump damage due to low flow conditions, a minimum flow line is provided for each LPSW pump. (Reference 5)

The Standby LPSW pump auto-start circuit actuates the Standby LPSW pump automatically for the Units 1&2 and Unit 3 LPSW system. The circuit actuates following a Loss of Offsite Power (LOOP) event when a running LPSW pump fails to restart and LPSW header pressure fails to return to normal operating values. The auto-start circuit will also start the Standby LPSW pump during normal operation when LPSW header pressure falls below an acceptable value.

The LPSW system provides cooling for components in the Turbine Building, the Auxiliary Building, and in the Reactor Building. Two separate 24 inch lines provide LPSW to the components in the Auxiliary and Reactor Buildings. These two supply lines are further divided into four separate supply headers, two supplying the components in Oconee l and two supplying the components in Oconee 2. The decay heat removal coolers and the Reactor Building cooling units are supplied by separate LPSW supply lines. The return lines from the decay heat removal coolers and the Reactor Building coolers maintain separation to a point beyond a remote-operated isolation valve.

For Oconee 3, each of the two 15,000 gal/min LPSW pumps take their suction from the CCW crossover. These pumps provide cooling water via separate supply lines to engineered safeguards equipment in the Reactor Building and the Auxiliary Building similar to Oconee 1 and 2. The return lines from the Oconee 3 engineered safeguards maintain separation to a point beyond a remote-operated isolation valve.

The Turbine Building requirements for LPSW are supplied from other separate headers. The three pumps associated with Oconee 1 and 2 have a Turbine Building header serving the Turbine Building requirements for Oconee l and 2. The two pumps associated with Oconee 3 also have a Turbine Building header to supply the Oconee 3 requirements.

The separate flow paths serving the emergency safeguards equipment can be isolated by remote-operated isolation valves.

The LPSW system is monitored and operated from the control room. Isolation valves are incorporated in all LPSW lines penetrating the Reactor Building.

The three (per unit) Reactor Building coolers (RBCUs) (“A”, “B”, and “C”) are supplied by individual lines from the separate LPSW supply headers. Each inlet line is provided with a motor operated shutoff valve located outside the Reactor Building. Similarly, each discharge line from the coolers is provided with a motor operated valve located outside the Reactor Building. This allows each cooler to be isolated individually.

LPSW flow is provided to the Reactor Building Auxiliary Cooling Units (RBACs) through a separate piping loop that is independent of the RBCUs. RBAC flow can be throttled to supplement RBCU cooling. During normal plant operation, the three RBCUs “A”, “B”, and “C” can be throttled to provide cooling of the Reactor Building. During times when LPSW temperature is high and greater cooling is desired inside the Reactor Building, chilled water can be provided to the Auxiliary Coolers in lieu of LPSW by a temporary chilled water system during modes 1-4 and to the Auxiliary Coolers and/or the “B” RBCU during modes 5, 6, and no mode. LPSW flow path to and from RB auxiliary cooling units is automatically isolated by air-operated containment isolation valves on engineered safeguard signals. The Containment Isolation Valves (CIVs) are also automatically closed upon low LPSW supply header pressure to prevent column closure waterhammers upon LPSW pressure restoration.

On an engineered safeguards signal the outlet valves on the three RBCUs fully open to assure emergency flow through coolers.

The LPSW System provides sufficient flow to the Low Pressure Injection (LPI) coolers and Reactor Building Cooling Units (RBCUs) to ensure sufficient heat transfer capability following a design basis accident and a single active failure. The worst case design basis accident involves a LOCA/loss of offsite power with a loss of instrument air. The worst case single failures for achieving desired flows to the RBCUs and LPI coolers are 1) failure of a single LPSW pump, and 2) failure of a 4160 volt bus which fails an LPSW pump, an RBCU fan, and an LPI cooler isolation valve. Analysis and testing have been performed to demonstrate system performance under worst case conditions.

The LPSW System can provide sufficient flow to the required loads following a seismic event. Valves 1LPSW-139, 2LPSW-139, and 3LPSW-139 are remotely-operated, seismically-qualified valves which can isolate the non-seismic, non-essential header from the safety-related portions of the system. Non-seismic connections to the system exist which cannot be remotely isolated. Analysis has demonstrated that a seismically-induced single pipe break of a non-seismic connection that cannot be remotely isolated will not cause loss of system safety function.

LPSW flow to the LPI coolers is normally throttled using air-operated valves LPSW-251 and LPSW-252. During a design basis accident involving a loss of instrument air, these valves fail open to their travel stops. Motor-operated valves LPSW-4 and LPSW-5 will be used to throttle LPSW flow to the LPI coolers under these conditions. Travel stops are in place on LPSW-251 and LPSW-252 to ensure LPSW flow through an LPI cooler does not exceed the design limit of 7500 gpm under worst case conditions.

The LPSW flow to and from each Reactor Building cooler is measured. Provisions are available to indicate cooler leakage.

LPSW is a non-radioactive cooling water system that is monitored for radioactivity. Monitoring is required per Section 11.5.1 since LPSW provides cooling to normally radioactive systems. Components from these normally radioactive systems could potentially leak radioactivity into LPSW. Upon any indication of radioactivity, the component suspected of leaking may be individually isolated.

The LPSW pumps are connected to the 4160 volt buses which supply power to engineered safeguards equipment. The emergency power supply is adequate to operate all LPSW pumps upon a loss of off-site power.

During normal operation, the cooling requirements are supplied by operating one LPSW pump per unit. The LPSW requirement following a loss of coolant accident can also be supplied by one pump per unit.

The spare pump is started by the engineered safeguards actuation signal to provide redundancy for single failure criteria.

LPSW supplies water to the SSW system.

Generic Letter 96-06 required consideration of effects inside containment due to the change in environment during a Loss of Coolant Accident (LOCA). This consideration identified the potential for waterhammers in cooling water systems serving containment following a Loss of Offsite Power (LOOP) concurrent with a LOCA or Main Steam Line Break (MSLB). Analysis and system testing in response to GL 96-06 concluded that waterhammers could occur in the Low Pressure Service Water (LPSW) system during all LOOP events (e.g., LOCA/LOOP, MSLB/LOOP). The LPSW piping supplies the Reactor Building Cooling Units (RBCU), the Reactor Building Auxiliary Coolers (RBAC), and the Reactor Coolant Pump Motor Coolers (RCPMC). During Loss of Offsite Power (LOOP) events or Loss of Coolant Accident (LOCA) events coupled with a LOOP it was possible to create a column Closure Waterhammer (CCWH) or Condensation Induced Waterhammer (CIWH) in the LPSW piping and components inside containment. CCWH could have occurred when the LPSW pumps restart following a LOOP and rapidly close vapor voids within the system. CIWH could have occurred when heated steam voids interact with sub-cooled water in long horizontal piping sections.

The LPSW RB Waterhammer Prevention system (WPS) was designed to maintain the LPSW piping inside containment water solid during events which cause a loss of LPSW such a LOOP, LOCA/LOOP, or MSLB/LOOP. The system's major components consist of check valves in the supply headers (LPSW-1111, 1116), pneumatic discharge isolation valves (LPSW-1121, 1122, 1123, and 1124), pneumatic vent valves (a.k.a., controllable vacuum breakers) (LPSW-1150, 1151), and associated actuation circuitry. The discharge header from containment is a common header. The header splits into two parallel headers each of which contain two of the pneumatic discharge isolation valves. The controllable vacuum breakers are located on the common header downstream of the pneumatic discharge isolation valves. See Figure 9-12. The actuation circuitry consists of four pressure measurement loops along with necessary components to cause the pneumatic discharge isolation valves to close and the controllable vacuum breakers to open on low LPSW supply header pressure. The circuitry resets and causes a) the pneumatic discharge isolation valves to reopen and b) the controllable vacuum breakers to reclose on increasing LPSW supply header pressure. The circuitry is designed to be single failure proof to open and close the valves. Failure of the pneumatic discharge isolation valves to reopen following system actuation will prevent flow through the Reactor Building Cooling Units as well as other containment loads such as the Reactor Building Auxiliary Coolers and the Reactor Coolant Pump coolers. Provisions to manually fail open the valves are provided. The failure of the controllable vacuum breakers to reclose is inconsequential (i.e., containment heat removal can be accomplished with the valves in the open position). Each pneumatic valve is provided with an air accumulator to provide a source of air to move the valve and maintain the desired end state for a short period of time. Only for the case of a Station Blackout (SBO) could the air in the accumulator be insufficient to maintain closure of the pneumatic discharge isolation valves for the duration of the SBO. In this case, reliance on the Supplemental Diesel Air Compressors is needed to provide air to make-up any leakage to maintain closure.

The system includes a "leakage accumulator" to allow a reasonable amount of boundary valve leakage while the piping inside containment is being maintained water solid. The leakage accumulator consists of a quantity of water with an air overpressure. The air overpressure will force water into the isolated portion of LPSW should the pressure decrease due to leakage in order to prevent voiding. The leakage accumulator is a passive device and is normally kept charged by LPSW. During an SBO, a HPSW connection to the accumulator provides extended make-up for leakage. During times when the WPS is out of service, piping code allowable stresses may be exceeded, but pipe rupture is not expected, if an event occurs that produces a waterhammer.

9.2.2.2.4 Recirculated Cooling Water System (RCW)

The RCW system for the Oconee station is shown schematically in Figure 9-13. This system provides inhibited closed cycle cooling water to various components outside the Reactor Building including:

1. RC pump seal return coolers
2. Spent fuel cooling
3. Sample coolers
4. Evaporator systems
5. Various pumps and coolers in the Turbine Building

The RCW system consists of two parallel loops which are normally isolated from each other. One loop supplies cooling for shared station loads, Unit 1 and 2 loads and secondary loads on Unit 3. It consists of four motor-driven pumps and four RCW heat exchangers. A 25,000 gallon surge tank provides a surge volume to accommodate temperature changes and leakage. Condenser circulating water is used to cool the RCW heat exchangers. The other loop supplies cooling for Unit 3 primary loads. It consists of two motor-driven pumps and two RCW heat exchangers. It contains a 7,700 gallon surge tank and also utilizes condenser circulating water to cool the RCW heat exchangers. RCW effluent from the Auxiliary Building is monitored for radioactivity. Leakage of radioactive fluids from any of the coolers in the Auxiliary Building will be indicated by these monitors. Separate monitors are provided on the return lines from the Oconee l and 2 Auxiliary Building and the Oconee 3 Auxiliary Building.

The number of RCW pumps and RCW heat exchangers in operation varies depending on the spent fuel heat load and lake water temperature. The isolation valves, which normally separate the two parallel loops, can be opened, however; it is not a necessary configuration.

The RCW provides no engineered safeguards functions and does not penetrate the Reactor Building.

9.2.2.2.5 Essential Siphon Vacuum and Siphon Seal Water Systems

The Essential Siphon Vacuum (ESV) and the Siphon Seal Water (SSW) systems are discussed together due to their inherit relatedness. Simplified schematic diagrams of the systems are shown in Figure 9-42 and Figure 9-43.

The ESV system consists of three (3) liquid ring vacuum pumps per unit. These pumps, one of which is an installed spare, are connected to two (2) tanks. These tanks are connected to the CCW Intake headers (one tank per header). A float valve is used to minimize CCW water passage into the ESV system. A minimum flow line for the ESV pumps is provided on the tanks to ensure that a minimum amount of air is passing through the ESV pumps. Without this minimum amount of air, the vacuum created in the ESV pumps will cause cavitation, which, over a long period of time, can cause pump degradation. Short periods of time (e.g., over a month) without minimum flow operation will not degrade the pumps.

During normal operations, an ESV pump and tank are aligned to a given CCW Intake header. Air accumulation in the CCW Intake Header is removed by the ESV system in order to maintain the CCW Intake Header primed during normal operations. During emergency operations, the ESV pump minimum flow line is isolated and the ESV pumps remove any air accumulation that occurs in the CCW Intake Header. This allows full ESV pump capacity to be directed toward the siphon until the event is mitigated.

The ESV pumps are controlled from the Control Room. Vacuum Tank pressure indication and pump operating status are located in the control room. Float valve heat trace current and valve temperature indications are also available in order to allow monitoring of float valve condition during sub-freezing weather. During emergency operations, the ESV pump restart is delayed for a short period of time in order to allow for other, more time-critical loads to load onto the emergency power system. A variety of non-nuclear safety-related data points associated with the ESV/SSW/ECCW systems are sent to the plant computer.

The SSW System consists of two headers that are supplied water from the Low Pressure Service Water (LPSW) system. Only one header is needed to supply all loads. However, both SSW headers are normally in service so that a single failure in the LPSW system cannot cause a loss of safety function. The SSW supply water routes from the Turbine Building to the ESV Building, where it is strained. Once strained, SSW routes to the ESV pumps and to the CCW pumps. SSW provides an operating liquid for the ESV pumps and provides sealing and cooling water the CCW pump shaft seal and motor bearing cooler. The nuclear safety-related function of the SSW System is to provide the operating liquid to the ESV pumps. The ESV pumps are liquid ring vacuum pumps which require a continuous supply of water in order to create a vacuum. As the header branches to the ESV pumps and then branches to each ESV pump individually, a solenoid valve is contained at each pump. This solenoid valve is interlocked with the ESV pump control circuitry. The valve opens when the pump starts and closes when it stops. A failure of one of these solenoids would cause a single ESV pump to be inoperable. The SSW system function would not be affected, since it could successfully deliver water to the remaining ESV pumps.

The SSW system contains provisions for connection of a submersible pump to supply sealing/cooling water to the CCW pumps. Both the ESV and SSW systems are designated as QA Condition I systems. They are seismically designed and designed to continue functioning with a single, active failure. However, they are not designed for tornado loads. Interfacing structures existing prior to the installation of these systems are designated QA Condition 4. The ESV Building shell is also a QA Condition 4 structure.

9.2.3 Auxiliary Service Water System

9.2.3.1 Design Basis

The Auxiliary Service Water System is designed for decay heat removal following a concurrent loss of the main feedwater system, Emergency Feedwater System, and Decay Heat Removal System. The system will maintain decay heat removal for a minimum of 37 days.

Portions of the Station Auxiliary Service Water 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.

9.2.3.2 System Description

The Auxiliary Service Water System utilizes the plant CCW intake and discharge conduits as a source of raw cooling water for decay heat removal (Figure 10-8). These conduits are interconnected by crossovers and unwatering lines. An Auxiliary Service Water Pump located in the Auxiliary Building at Elev. 771 takes its suction from the Oconee 2 intake conduit and discharges into the steam generators of each unit via separate lines into the emergency feedwater headers. The raw water is vaporized in the steam generator removing residual heat and dumped to the atmosphere.

The auxiliary service water pump is an end suction centrifugal pump with a rated capacity of 3000 gal/min at a total head of 180 feet.

It has been submitted to the following tests:

1. A non-witness ASME hydro test
2. Witnessed performance test
3. Sonic testing of shaft

4. Mill test certificates for casing, impeller, and shaft
5. Certified caliper measurements

The pump power supply is taken from the 4160 volt Standby Bus No. 1.

All valves required for operation of the Auxiliary Service Water System are either check valves or manually operated. The pump suction is equipped with a manually operated butterfly valve and the discharge with a check valve and manually operated gate valve. The pump is equipped with a minimum flow path to the CCW discharge crossover line, which is isolated by a globe valve. The individual lines to each steam generator auxiliary feedwater header are equipped with a check valve and one normally closed gate valve which is used to control flow. The majority of non-embedded piping is Duke Class F.

Atmospheric steam dumps on each main steam line are equipped with one normally closed gate valve and one normally closed control valve which must be opened to reduce steam generator shell side pressure before placing the Auxiliary Service Water System into operation.

9.2.4 Ultimate Heat Sink

Lake Keowee supplies the Condenser Circulating Water (CCW) System and the lake water generally serves as the ultimate heat sink for Oconee Nuclear Station. In some events, such as loss of Lake Keowee, the water trapped in the CCW piping serves as the ultimate heat sink The CCW system is described in Section 9.2.2.2.1.

9.2.5 Control Room Ventilation Chilled Water System (WC)

The WC System is shown schematically on Figure 9-24.

9.2.5.1 Design Basis

The WC System provides chilled water for the Control Room Ventilation System for all three units. The major equipment of the chilled water system is arranged in two parallel redundant trains with one supply and return line and each train capable of supplying the required cooling capacity. A temporary cooling train and piping may be installed in parallel with the permanent chilled water system equipment. The temporary cooling train and piping will connect to the system supply and return piping and be capable of supplying the required cooling capacity. The bases to one of the Technical Specification 3.7.16 addresses the use a temporarily installed full capacity control area cooling train as one of the Tchnical Specification 3.7.16 required WC trains.

9.2.5.2 System Description and Evaluation

The WC permanently installed chillers are each made up of a compressor, an evaporator and a refrigerant condenser. The single stage compressor is driven by an open drip proof motor. Both the evaporator and condenser are horizontal shell and finned tube design with individually replaceable tubes. Two chillers are provided for this system, each with 100% capacity.

For the permanently installed WC cooling trains condenser water temperature is measured by a thermistor located upstream of the condenser. This sensor is used to generate a signal that modulates a three-way bypass valve located downstream of the condenser, to maintain proper condenser water temperature. (i.e. as the temperature increases, the bypass port on the three-way valve is modulated towards closed, to maintain proper entering condenser water temperature for operating the chiller.)

A temporary cooling train with piping connected to the WC system chilled water return and supply piping may be used, providing 100% cooling capacity.

Cooling of a specific area is controlled by a chilled water control valve located downstream of the corresponding AHU. Temperatures throughout the Control Room Area are monitored by individual room thermostats. On a rise in room temperature, the AHU controls will modulate open the corresponding chilled water valve. On a decrease in room temperature, the chilled water valve will be gradually closed.

9.2.6 References

1. Safety Evaluation Report for Oconee Units 2 and 3, dated July 6, 1973.
2. Letter from J. W. Hampton (Duke) to USNRC Document Control Desk, dated May 31, 1995, Service Water Issues.
3. Letter from H. B. Tucker (Duke) to USNRC Document Control Desk, dated December 5, 1989, NRC Bulletin No. 88-04 Potential Safety-Related Pump Loss Action 4 Report Status Update.
4. Letter from J. W. Hampton (Duke) to USNRC Document Control Desk, dated January 7, 1993, "NRC Bulletin No. 88-04 Potential Safety-Related Pump Loss Revised Response".
5. Letter from L. A. Wiens (NRC) to J. W. Hampton (Duke), dated June 10, 1993, "Revised Response to NRC Bulletin 88-04, "Safety Related Pump Loss".
6. Safety Evaluation Report for License Amendment 217/217/214, dated August 19, 1996.
7. Letter from H. B. Tucker (Duke) to USNRC Document Control Desk, dated January 26, 1990, "Response to NRC Generic Letter 89-13, Service Water System Problems Affecting Safety-Related Equipment".
8. Letter from H. B. Tucker (Duke) to USNRC Document Control Desk, dated May 31, 1990, "Supplemental Response to NRC Generic Letter 89-13, Service Water System Problems Affecting Safety-Related Equipment".
9. Letter from J. W. Hampton (Duke) to USNRC Document Control Desk, dated September 1, 1994, "Generic Letter 89-13".
10. Letter from J. W. Hampton (Duke) to USNRC Document Control Desk, dated April 4, 1995, "Supplemental Response #3 Generic Letter 89-13".
11. Letter from L. A. Wiens (NRC) to M. S. Tuckman (Duke), dated February 8, 1991, "NRC Generic Letter 89-13, Service Water System Problems Affecting Safety-Related Equipment".
12. Letter from J. W. Hampton (Duke) to USNRC Document Control Desk, dated July 12, 1995, "Supplemental Response #4 to G. L. 89-13".
13. PIP 0-098-3629 Operability Evaluation

THIS IS THE LAST PAGE OF THE TEXT SECTION 9.2.

9.3 Process Auxiliaries

9.3.1 Chemical Addition and Sampling System

9.3.1.1 Design Bases

Chemical addition and sampling operations are required to change and monitor the concentration of various chemicals in the Reactor Coolant System and Auxiliary Systems. The Chemical Addition and Sampling System is designed to add boric acid to the Reactor Coolant System for reactivity control, lithium hydroxide for pH control, and hydrazine and/or carbohydrazide for oxygen control. The Chemical Addition and Sampling System can also be used for hydrogen peroxide additions to induce 'crud' bursts during unit shutdowns to enhance corrosion product removal and, therefore, reduce equipment/system/component dose rates. Following a LOCA, a passive design system is used to modify the pH of the reactor coolant system.

9.3.1.2 System Description and Evaluation

The Mechanical Chemical Addition and Sampling System is shown schematically on Figure 9-15 and Figure 9-16. The Passive TSP Baskets are described below. The Sampling System has separate sampling stations for reactor coolant and steam generator sampling for each of the three units. Two auxiliary systems sampling stations are provided, one for Oconee l and 2 and one for Oconee 3.

Two chemical addition systems are also provided, one for Oconee l and 2 and one for Oconee 3. These systems permit chemical addition to and sampling of the Reactor Coolant System and other Reactor Auxiliary Systems during normal reactor operation.

The Chemical Addition and Sampling System performs no emergency functions (Refer to Section 9.3.6 for information on Post-Accident Sampling System). Guidelines for maintaining feedwater and reactor coolant quality are derived from vendor recommendations and the current revisions of the EPRI PWR Secondary and Primary Water Chemistry Guidelines, respectively. Detailed operating specifications for the chemistry of these systems are addressed in the Chemistry Section Manual. A brief functional description of the major system components follows.

Boric Acid Mix Tank

Two Boric Acid Mix Tanks, one shared between units 1 and 2, and one for unit 3, are provided as a source of concentrated boric acid solution. Tank heaters and electrically heat traced transfer lines maintain the fluid temperature above that required to assure solubility of the boric acid.

Boric Acid Pumps

Six boric acid pumps, three shared between units l and 2, and three for Unit 3, are provided to transfer the concentrated boric acid solution from the boric acid tank to the borated water storage tank, letdown storage tanks, spent fuel storage pool, or the core flood tanks. Two pumps, each with a l gal/min capacity, supply boric acid to the core flood tanks. The other four pumps, which each have 10 gal/min capacities, supply boric acid to other tanks, systems, and locations (Figure 9-15 and Figure 9-16).

Reactor Building TSP Baskets

Granulated Tri-Sodium Phosphate (TSP) is stored in screen-sided baskets at the lowest elevation of the Reactor Building. During events that flood the Reactor Building, the TSP dissolves and maintains the pH of the water in the Reactor Building Emergency Sump at a level that minimizes gaseous iodine production and H_2 production from zinc-boric acid reactions during Reactor Building Spray operation. This Post LOCA pH Adjustment System was installed to replace the Caustic Addition System.

Caustic Mix Tank

The caustic addition portion of the system is no longer used during emergency conditions. Previously the Caustic Addition System was used to add sodium hydroxide to the LPI system following a LOCA. The addition of the TSP baskets described above replaced this function. Previously, this system was used to control the pH in the RC bleed and miscellaneous waste evaporators and to regenerate the resins in the deborating demineralizers, but it is no longer used in this capacity. A single caustic mix tank is provided for Units 1 and 2, and one tank is provided for Unit 3. This system can be used to add chemicals (as needed) to the RCS and Auxiliary Systems.

Caustic Pump

The caustic pump provides the capability to transfer sodium hydroxide from caustic bulk storage containers or the caustic mix tank to the LPI system. It is no longer used for this purpose since installation of the Reactor building TSP Baskets. A single pump is provided for Units 1 and 2 and one is provided for Unit 3. These pumps can be used to add other chemicals (as needed) to the RCS and Auxiliary Systems.

Lithium Hydroxide Tank

Lithium hydroxide is mixed and added to the Reactor Coolant System for pH control from the lithium hydroxide tank. A single tank is provided for Units l and 2, and one tank is provided for Unit 3.

Lithium Hydroxide Pump

The lithium hydroxide pump transfers lithium hydroxide from the LiOH tank to the letdown line upstream of the letdown filters. A single pump is provided for Units l and 2, and one pump is provided for Unit 3.

Hydrazine Pump

The hydrazine pump transfers hydrazine to the letdown line upstream of the letdown filters. The hydrazine pump, after sufficient demineralized water flushes, is also used to transfer hydrogen peroxide. A single pump is provided for units 1&2, and one pump is provided for unit 3. These pumps can also be used as a backup to the Lithium Hydroxide pump or to add other chemicals (as needed) to the RCS.

Pressurizer Chemical Addition Pump

A Pressurizer Chemical Addition pump transfers an oxygen scavenger from a small container backwards through the pressurizer water space sample line to the pressurizer. Each unit has its own separate pump.

Pressurizer Sample Cooler

This cooler cools the effluent sample taken from the pressurizer steam or water space. One cooler is provided per unit.

Steam Generator Sample Cooler

This cooler cools the effluent sample taken from the secondary side of the steam generator. Two coolers are provided per unit.

9.3.1.2.1 Mode of Operation

The chemical addition portion of this system delivers the necessary chemicals to other systems as required. Boric acid is provided to the spent fuel pool, borated water storage tank, letdown storage tank, and core flooding tanks as makeup for leakage or to change the concentration of boric acid in the associated systems. Following a LOCA, tri-sodium phosphate (TSP) is mixed with the LPI system to maintain the pH of the water via use of screen-sided baskets that contain granulated TSP. The TSP dissolves during Reactor Building flooding. The sampling portion of this system is used to take samples to assure that water qualities and boric acid concentrations are maintained. Sampling locations and the samples taken at each location are as follows:

Liquids

Primary Sample Basin

Steam Generator Sample Sink

Secondary Side of Steam Generator

Reactor Coolant Sample Sink

Pressurizer Water Space

Pressurizer Steam Space

Low Pressurizer Injection Cooler Outlet

Core Flooding Tanks

Total Gas Sample

Reactor Coolant

Waste Disposal Sample Basin

Auxiliary Systems Sample Sink

Purification Demineralizer Inlet and Outlet

Deborating Demineralizer Outlet

Letdown Storage Tank Water Space

RC Bleed Evaporator Feed Pump Discharge (out of service)

Deborating Demineralizer Outlet (Regeneration)

Waste Evaporator Feed Pump Discharge (out of service)

RC Bleed Evaporator (Concentrate) (out of service)

Concentrated Boric Acid Storage Tank Pump Discharge

RC Bleed Evaporator (Distillate) (out of service)

Waste Evaporator (Concentrate) (out of service)

RC Bleed Transfer Pump Discharge

Waste Transfer Pump Discharge

High Activity Waste Transfer Pump Discharge

Low Activity Waste Transfer Pump Discharge

Condensate Test Tank Pump Discharge (out of service)

RC Bleed Evaporator Demineralizer Outlet (out of service)

Reactor Building Normal Sump

Waste Evaporator (distillate) (out of service)

Gaseous

Hydrogen Analyzer

Containment

Gas Analyzer (Unit 1 and 3) (out of serice)

Waste Holdup Tank

High Activity Waste Tank

RC Bleed Holdup Tanks

Waste Gas Vent Header

RC Bleed Evaporator (Unit 1 only)

Waste Evaporator (Unit 1 only)

Waste Gas Decay Tanks

H_2 Purge Station

Sample Containers (to be analyzed for a variety of substances)

Letdown Storage Tank Gas Space

Pressurizer Steam and Water Space

Gas Analyzer Sample (out of service)

9.3.1.2.2 Reliability Considerations

The Chemical Addition and Sampling System is not required to function during an emergency condition. Redundant boric acid pumps and flow paths are provided to guard against a single component failure rendering the system inadequate for boron addition. In addition to the boric acid mix tank, boric acid is also available for boration in 5-percent by weight solution from the concentrated boric acid storage tank. To prevent precipitation, heating/heat tracing is installed on components and lines used to transfer concentrated boric acid. The pumps, tanks, coolers, and instrumentation are located in the Auxiliary Building and are accessible for inspection and maintenance.

9.3.1.2.3 Codes and Standards

The components of the Chemical Addition and Sampling System are designed to the codes and standards noted in Table 9-5.

9.3.1.2.4 System Isolation

The pressurizer sample line, core flood sample line, and both steam generator sample lines are the only system lines that penetrate the Reactor Building. All these lines contain electric motor-operated isolation valves inside the Reactor Building and pneumatic valves outside, which are automatically closed by an engineered safeguards signal (except for the core flood sample line which has a manual isolation valve).

9.3.1.2.5 Leakage Considerations

Leakage of radioactive reactor coolant from this system within the Reactor Building will be collected in the Reactor Building normal sump. Leakage of radioactive gases from this system outside the Reactor Building is collected by placing the sampling stations under hoods exhausting to the unit vent.

9.3.1.2.6 Failure Considerations

Since the system serves no engineered safeguards function, the only consideration immediately following a loss-of-coolant accident is the operation of the isolation valves. Redundant isolation valves are provided to assure isolation of the Reactor Building as described in Section 9.3.1.2.4.

9.3.1.2.7 Deleted Per 2001 Update

9.3.2 High Pressure Injection System

9.3.2.1 Design Bases

The High Pressure Injection System is designed to accommodate the following function during normal reactor operation:

Supply the Reactor Coolant System with fill and operational makeup water.

Provide seal injection water for the reactor coolant pumps.

Provide for purification of the reactor coolant to remove corrosion and fission products.

Control the boric acid concentration in the reactor coolant.

In conjunction with the pressurizer, the system will accommodate temporary changes in reactor coolant volume due to small temperature changes.

Maintain the proper concentration of hydrogen and corrosion inhibiting chemicals in the Reactor Coolant System.

Provides continuous flow for cooling the normal HPI nozzles (see FSAR Section 5.4.7.2) to minimize thermal shock.

Provides auxiliary pressurizer spray control for cooldown when normal pressurizer spray is unavailable.

The specific design bases for various parts of the system are as follows:

Letdown Capability

The system will accommodate letdown required as a result of coolant volume expansion when heating the reactor coolant to operating temperature at a rate of 100°F/h while maintaining constant pressurizer level. The letdown is cooled before leaving the Reactor Building.

Purification

Filters and demineralizers are provided to remove reactor coolant impurities. The letdown filters and purification demineralizers are sized for full flow through the letdown orifice.

Makeup

The system will accommodate makeup requirements during design reactor coolant system transients and for Reactor Coolant System cooldown at the design rate.

9.3.2.2 System Description and Evaluation

The High Pressure Injection System is shown schematically on Figure 9-17 and Figure 9-18. Table 9-6 and Table 9-7 list the system Performance requirements and data for individual components. The following is a brief functional description of system components:

Letdown Cooler

The letdown cooler reduces the temperature of the letdown flow from the Reactor Coolant System to a temperature suitable for demineralization and injection to the reactor coolant pump seals. Heat in the letdown coolers is rejected to the Component Cooling System.

Letdown Flow Control

The letdown flow rate at reactor operating pressure is limited by a fixed block orifice. A parallel, normally closed, remotely operated valve can be opened to maintain the desired flow rate. In addition there is a second parallel, normally closed valve which may be manually positioned for flow control.

Purification Demineralizer

The letdown flow is passed through the purification demineralizer to remove reactor coolant impurities other than boron. The design purification letdown flow is equal to one Reactor Coolant System volume in 24 hours. One demineralizer is provided for each unit. In addition, a spare demineralizer is shared between Oconee 1 and 2, and another spare is installed for Oconee 3. The spare demineralizer may be used to remove lithium from the reactor coolant system to maintain system chemistry and/or used to remove cesium from the reactor coolant system in the event of fuel defects. Chapter 11 describes coolant activities, coolant handling and storage, and expected limits on activity discharge.

Letdown Filters

Two letdown filters in parallel are provided to prevent particulates from entering the Reactor Coolant System and subsequently the pump seal filters. One filter is normally in use.

High Pressure Injection Pumps

The high pressure injection pumps are designed to return coolant which is letdown for purification to the Reactor Coolant System, and to supply the seal water to the reactor coolant pumps. The pumps are sized to permit one pump to provide normal operating makeup and seal water flow.

Reactor Coolant Pump Seal Injection Filters

Two reactor coolant pump seal filters are provided to prevent particulates from entering the pump seals. One is normally in use.

Seal Return filter

A single filter is installed in the seal return line upstream of the seal return coolers to remove particulate matter. A bypass is installed to permit servicing during operation.

Reactor Coolant Pump Seal Return Coolers

The seal return coolers are sized to remove the heat added by the high pressure injection pumps and the heat picked up in passage through the reactor coolant pump seals. Heat from these coolers is rejected to the Recirculated Cooling Water System. Two coolers are provided and one is normally in operation.

Letdown Storage Tank

The letdown storage tank serves as a receiver for letdown, seal return, chemical addition, and system makeup. The tank also accommodates temporary changes in system coolant volume.

9.3.2.2.1 Mode of Operation

During normal operation of the Reactor Coolant System, one high pressure injection pump continuously supplies high pressure water from the letdown storage tank to the seals of each of the reactor coolant pumps and to makeup line connections on two of the reactor inlet lines. Makeup flow to the Reactor Coolant System is regulated by a flow control valve, which operates on signals from the pressurizer level controller.

A control valve in the common injection line to the pump seals automatically maintains the desired total injection flow to the seals. Manual Throttle valves in each pump seal injection line provide a capability to balance the seal injection flow rates. A portion of the water supplied to the seals enters the Reactor Coolant System. The remainder returns to the letdown storage tank after passing through one of the two reactor coolant pump seal return coolers. A small amount which leaks through the final seal is also collected and routed to the quench tank.

Paragraph(s) Deleted Per 2000 Update.

Seal water inleakage to the Reactor Coolant system requires a continuous letdown of reactor coolant to maintain the desired pressurizer level. Letdown is also required for removal of impurities and boric acid from the reactor coolant. The letdown is cooled by one of the letdown coolers, reduced in pressure by the letdown orifice, and then passed through the purification demineralizer to a three-way valve which directs the coolant to the letdown storage tank or to the Coolant Storage System.

Normally, the three-way valve is positioned to direct the letdown flow to the letdown storage tank. If the boric acid concentration in the reactor coolant is to be reduced, the three-way valve is positioned to divert the letdown flow to the Coolant Storage System. Boric acid is removed by directing the letdown flow through a deborating demineralizer with the effluent returned directly to the letdown storage tank, or by the feed and bleed method. Feed and bleed is the process of directing the letdown flow to a coolant bleed holdup tank and maintaining the level in the letdown storage tank with demineralized water pumped from a supply of unborated water. The flow of demineralized water is measured and totaled by inline flow instrumentation. The flow of demineralized or borated water returning to the letdown storage tank is controlled remotely by the makeup control valve. During normal operation the inline instrumentation or the control rod drive interlock will terminate makeup flow.

The letdown storage tank also receives chemicals for addition to the reactor coolant. A hydrogen overpressure is maintained in the tank to assure a slight amount of excess hydrogen in the circulating reactor coolant. Other chemicals are injected in solution into the tank.

System control is accomplished remotely from the control room with the exception of the reactor coolant pump seal return cooling. The letdown flow rate is set by remotely positioning the letdown flow control valve to pass the desired flow rate. The spare purification demineralizer can be placed in service by remote positioning of the demineralizer isolation valves. The letdown flow to the Coolant Storage System is diverted by remote positioning of the three-way valve and the valves in the Coolant Storage System. The reactor coolant volume control valve is automatically controlled by the pressurizer level controller.

A continuous cooling flow is maintained through the HPI nozzle warming lines. Flow is monitored via the Operator Aid Computer with signals from a flow transmitter on each warming line.

Auxiliary pressurizer spray is remote manually controlled from the control room. No means exists for directly monitoring auxiliary pressurizer spray flow. Instead, pressurizer level is utilized for process monitoring of auxiliary pressurizer spray.

For emergency operation as a High Pressure Injection System, the normal letdown coolant flow line and the normal pump seal return line are closed, and additional makeup flow is supplied through the high pressure injection emergency lines. The pumps and pump motors are designed to be able to operate at the higher flow rates and lower discharge pressures associated with emergency high pressure injection requirements. Emergency operation of this system is described in Chapter 6.

9.3.2.2.2 Reliability Considerations

This system provides essential functions for the normal operation of the unit. Redundant components and alternate flow paths have been provided to improve system reliability.

Each unit has three high-pressure injection pumps, each capable of supplying the required reactor coolant pump seal and makeup flow. One is normally in operation while another is in standby status to be used as needed. The third pump is used only for emergency injection. There are two letdown coolers and two seal return coolers. One cooler in each group will perform the required duty while the other may be used as a spare.

One of the two letdown filters or reactor coolant pump seal filters is normally in use while the other is a spare.

9.3.2.2.3 Codes and Standards

Each component of this system will be designed to the code or standard, as applicable, noted in Table 9-7.

9.3.2.2.4 System Isolation

The letdown line and reactor coolant pump seal return line are outflow lines which penetrate the Reactor Building. Both lines contain electric motor-operated isolation valves inside the Reactor Building and pneumatic valves outside which are automatically closed by an engineered safeguards signal. The injection lines to the reactor coolant pump seals are inflow lines penetrating the Reactor Building. These lines contain a check valve on the inside and on the outside of the Reactor Building. Check valves in the discharge of each high pressure injection pump provide further backup for Reactor Building isolation. The two emergency coolant injection lines are used for injecting coolant to the reactor vessel after a loss-of-coolant accident. After use of the lines for emergency injection is discontinued the electric motor-operated isolation valves in each line outside the Reactor Building may be closed for isolation. The HPI nozzle warming line and auxiliary pressurizer spray line are inflow lines penetrating the Reactor Building. These lines each contain a check valve on the inside and on the outside for Reactor Building isolation.

9.3.2.2.5 Leakage Considerations

Design and installation of the components and piping in the High Pressure Injection System considers the radioactive service of this system. Except where flanged connections have been installed for ease of maintenance, the system is an all-welded system.

9.3.2.2.6 Failure Considerations

The effects of failure and malfunctions in the High Pressure Injection System concurrent with a loss-of-coolant accident are presented in Chapter 6. These analyses show that redundant safety features are provided where required.

For pipe failures in the High Pressure Injection System, the consequences depend upon the location of the rupture. If the rupture were to occur between the reactor coolant loop and the first isolation valve or check valve, it would lead to an uncontrolled loss-of-coolant from the Reactor Coolant System. The analysis of this loss-of-coolant Accident is included in Chapter 15. If the rupture were to occur beyond the first isolation valve or outside the Reactor Building, the release of radioactivity would be limited by the small line sizes and by closing of the isolation or check valve.

A single failure will not prevent boration when desired for reactivity control, since several alternate paths are available for adding boron to the Reactor Coolant System. These are: (a) through the normal makeup lines, (b) through the reactor coolant pump seals, and (c) through the emergency injection lines. If pump suction is unavailable from the letdown storage tank, a source of borated water is available from the borated water storage tank during normal operation.

9.3.2.2.7 Operational Limits

Alarms or interlocks are provided to limit variables or conditions of operation that could cause system upsets. The variables or conditions of operation that are limited are as follows:

Letdown Storage Tank Level

Low water level in the letdown storage tank is alarmed and interlocked to the three-way bleed valve. Low water level will switch the three way valve from the bleed position to the normal position.

Letdown Line Temperature

A high letdown temperature in the letdown line downstream of the letdown coolers is alarmed and interlocked to close the pneumatic letdown isolation valve, thus protecting the purification demineralizer resins.

Dilution Control

The dilution cycle is initiated by the operator. Several safeguards are incorporated into the design to prevent inadvertent excessive dilution of the reactor coolant.

The dilution valves have an automatic feature such that the operator may preset the desired quantity of dilution volume before initiating the dilution cycle. The dilution cycle will terminate when flow has integrated to the desired batch size. This interlock may be manually bypassed. Operation in the automatic mode is the preferred method of dilution.

Interlocks on the regulating control rod bank automatically terminate the dilution cycle regardless of the mode of operation the controller is in, automatic or manual, if the regulating rod group (Group 6) is inserted into the core beyond 25 percent.

The operator may manually terminate the dilution cycle at any time.

9.3.3 Low Pressure Injection System

9.3.3.1 Design Bases

The Low Pressure Injection System removes decay heat from the core and sensible heat from the Reactor Coolant System during the latter stages of cooldown. It provides the means for filling and draining the fuel transfer canal. The system maintains the reactor coolant temperature during refueling and reduced inventory operation. The LPI and support system(s), selected components of the RCS and HPI are dedicated to prevention and mitigation of loss of Decay Heat Removal events. (See Section 16.5.3 in the Selected Licensee Commitments Manual.)

In the event of a loss-of-coolant accident, the system injects borated water into the reactor vessel for longterm emergency cooling. The emergency functions of this system are described in Chapter 6. Performance data is listed in Table 9-8.

9.3.3.2 System Description and Evaluation

The Low Pressure Injection System is shown schematically in Figure 9-19. An independent system is provided for each unit. The Low Pressure Injection System normally takes suction from the reactor coolant outlet line and delivers the water back to the reactor through the core flooding nozzles after passing through the low pressure injection pumps and coolers. The Low Pressure Injection System may be lined up when the reactor pressure is below the system suction piping design pressure for cooldown of the system to refueling temperatures. The decay heat is transferred to the Low Pressure Service Water System by the decay heat removal coolers. Component data are shown in Table 9-9.

The major system components are described as follows:

Decay Heat Removal Pumps

Three decay heat removal pumps are arranged in parallel with electric motor operated valves in the suction line to each pump. Each pump has a separate minimum flow recirculation line with an orifice between pump discharge and pump suction. The bore of each orifice was increased to address considerations detailed in IEB 88-04, Safety Related Pump Loss. The two outboard pumps are normally available for emergency operation, and the center pump is valved off on both the suction and discharge sides of the pump. During decay heat removal, any two of the three pumps are lined up to the decay heat removal coolers.

The design flow is that required to cool the Reactor Coolant System from 250°F to 140°F in 14 hours. The steam generators are used to reduce the Reactor Coolant System from operating temperature to the 250°F temperature.

Decay Heat Removal Coolers

The decay heat removal coolers, during a routine shutdown, remove the decay heat from the circulated reactor coolant. Both coolers are designed to cool the circulated reactor coolant from 250°F to 140°F in 14 hours.

Borated Water Storage Tank

The borated water storage tank is located outside the Reactor Building and the Auxiliary Building. It contains borated water with boron concentration maintained in accordance with the Core Operating Limits Report. It is used for filling the fuel transfer canal during refueling and for filling the incore instrumentation handling tank. The borated water storage tank also provides borated water for emergency core cooling and the Reactor Building Spray System. Liquid level in the borated water storage tank is monitored by redundant level instrumentation.

9.3.3.2.1 Mode of Operation

Two pumps and two coolers normally perform the decay heat removal function for each unit. The steam generators reduce the reactor coolant temperature to approximately 250°F and pressure to approximately 300 psig. These conditions represent upper limits for starting an LPI pump so as to avoid exceeding system design limits. For Oconee Units 1 and 2, when these temperatures and pressures are reached, decay heat removal will be initiated by aligning the system in one of three possible configurations. The first path aligns A and C pumps to RCS through high pressure piping. With either the A or C pump operating, fluid is returned to the RCS through the "A" train of LPI. The second path aligns the B cooler to the RCS and the outlet of the cooler is routed to the suction of the A and C pumps. In this alignment, the pump in service will return fluid to the RCS through the "B" train of LPI. The third path aligns the B cooler to the RCS and the outlet of the cooler is routed to the suction of the A and C pumps. In this alignment, the pump in service will return fluid to the RCS through the “A” train of LPI (including the A cooler). After the RCS pressure has been reduced to approximately 125 psig, the system is aligned so that two pumps take suction from the reactor outlet line and discharge through two coolers.

For Oconee 3 decay heat cooling is initiated at 290 psig/250°F by aligning pumps to take suction from the reactor outlet line and discharge through the coolers into the reactor vessel. The equipment utilized for decay heat cooling is also used for low pressure injection during accident conditions.

During refueling, the decay heat from the reactor core is rejected to the low pressure injection coolers in the same manner as it is during cooldown to 140°F. At the beginning of the refueling period, both coolers and both pumps are required to maintain 140°F in the core and fuel transfer canal. Later, as core decay heat decreases, one cooler and pump can maintain the required 140°F.

The fuel transfer canal may be filled by switching the suction of the decay heat removal pumps from the reactor outlet to the borated water storage tank. When the transfer canal is filled, suction to the pumps is switched back to the reactor outlet pipe. (Normally filled with the spent fuel cooling pumps as described in Section 9.1.3.)

After refueling, the transfer canal is drained by switching the discharge of one of the pumps from the reactor injection nozzle to the borated water storage tank. The other pump will continue the recirculation mode of decay heat removal.

9.3.3.2.2 Reliability Considerations

Since the equipment is designed to perform both normal and emergency functions, separate and redundant flow paths and equipment are provided to prevent a single component failure from reducing the system performance below a safe level. All rotating equipment and most valves are located in the Auxiliary Building to facilitate maintenance and periodic operational testing and inspection. See also Section 6.3.4.3 for additional considerations related to DHR system operability.

9.3.3.2.3 Codes and Standards

Each component of this system will be designed to the code or standard, as applicable, as noted in Table 9-9.

9.3.3.2.4 System Isolation

The Low Pressure Injection System is connected to the reactor outlet line on the suction side and to the reactor vessel on the discharge side. The system is isolated from the Reactor Building on the suction side by two electric motor-operated valves located inside the Reactor Building and one electric motor-operated valve located outside the Reactor Building. The discharge side is isolated from the Reactor Building by a check valve inside and an electric motor-operated valve outside the Reactor Building. All of these valves are normally closed whenever the reactor is in the operating condition. In the event of a loss-of-coolant accident, the valve on the discharge side opens, but the valves between the reactor vessel and the suction side of the pumps remain closed throughout the accident.

9.3.3.2.5 Leakage Considerations

During reactor power operation, all equipment of the Low Pressure Injection System is idle, and all isolation valves are closed. Under loss-of-coolant accident conditions, fission products may be recirculated in the coolant through the exterior piping system. Potential leaks have been evaluated to obtain the total radiation dose to the public due to leakage from this system. The evaluation is discussed in Chapter 12.

9.3.3.2.6 Operational Limits

Alarms or interlocks are provided to limit variables or conditions of operation that might affect system or station safety. These variables or conditions of operation are as follows:

Decay Heat Removal Flow Rate

Low flow from the pumps during the decay heat removal mode of operation is alarmed to signify a reduction or stoppage of flow and cooling to the core.

Reactor Coolant Pressure Interlock

The first valve from the Reactor Coolant System in the suction line to the low pressure injection pumps is interlocked with the Reactor Coolant System pressure instrumentation to prevent inadvertent overpressurization of the Low Pressure Injection System piping while the Reactor Coolant System is still above Low Pressure Injection System design pressure.

Reactor Coolant Leaving Decay Heat Removal Coolers

High temperature of the reactor coolant discharging from the decay heat removal coolers is alarmed to signal a loss of cooling capability in the respective cooler.

9.3.3.2.7 Failure Considerations

The effects of failure and malfunctions in the Low Pressure Injection System concurrent with a loss-of-coolant accident are presented in Section 6.3.3.4. Redundant safety features are provided where required.

For pipe failures in the Low Pressure Injection System, the consequences depend upon the location of the rupture. If the rupture were to occur between the first check valve upstream of the core flood nozzle and the vessel, this would lead to a loss-of-coolant accident. The analysis of this loss-of-coolant accident is included in Chapter 15. Section 15.14.4.3 addressed this failure as one of the limiting small break. Reference ECCS Analysis of B&W 177 FA LOWERED-LOOP NSS Rev. 3 (BAW-10103A, Rev. 3 Topical Report July 1977).

9.3.4 Coolant Storage System

9.3.4.1 Design Bases

The Coolant Storage System for each unit is designed to accommodate the accumulated coolant bleed over a core cycle, including startup expansion and coolant letdown to storage for boric acid reduction.

Two coolant bleed holdup tanks, each with a capacity of 11,000 ft^3, are provided for each unit. One tank provides storage for the reactor coolant bleed prior to treatment by the Radwaste Facility or makeup to the Reactor Coolant System. The other tank provides additional storage and is used to store clean water for use as feed to the Reactor Coolant System. An additional tank is provided for storage of the concentrated boric acid from the boric acid mix tank. The RC Bleed Evaporator and associated equipment is not used for coolant processing. Coolant processing is performed by the Radwaste Facility.

The storage of reactor coolant bleed requires approximately 55 percent of the volume of the bleed holdup tanks for each unit. The tanks for all three units are arranged so that they can be utilized to store liquid from the other units if so desired.

The design volume of coolant removed from one unit during heatup and dilution from MODE 5 is approximately 9600 ft^3. This occurs near the end of the core cycle when boric acid concentrations are reduced. Earlier in core life, coolant is removed in smaller quantities to reduce boric acid concentrations.

An additional requirement for coolant storage is the partial drain which occurs during refueling. The coolant is removed in a batch of approximately 6100 ft^3. per unit and returned to the Reactor Coolant System upon completion of refueling. Thus, it occupies storage capacity only during the period of refueling. The required storage volume for refueling operations of 6100 ft^3. is less than 10 percent of the total available capacity.

A quench tank, located inside the Reactor Building, condenses and contains any effluent from the pressurizer safety valves. The quench tank is sized to condense one normal pressurizer steam volume without relieving to the Reactor Building atmosphere. A quench tank drain pump is provided for pumping the quench tank contents into the letdown storage tank. The reactor coolant which has leaked into the quench tank can be pumped directly back into the coolant system to avoid routing this leakage through the waste disposal system.

9.3.4.2 System Description and Evaluation

The Coolant Storage System is used for the collection and storage of reactor coolant liquid. The liquid is received from the High Pressure Injection System both as a result of reactor coolant expansion during startup and for boric acid concentration reduction during startup and normal operation. It is either conveyed to coolant bleed holdup tanks for storage or passed through deborating demineralizers for boric acid removal and returned as unborated makeup to the High Pressure Injection System. A spray nozzle in the coolant bleed tanks on the inlet line allow some of the gases to be released. Recirculating the tank allows further stripping action to occur. Liquid from the coolant bleed holdup tanks can be pumped to the Radwaste Facility for processing. This is schematically shown in Figure 9-21 and Figure 9-18. Component data is shown in Table 9-10.

The quench tank, located inside the Reactor Building, condenses and contains effluent from the pressurizer safety valves and various vents. Liquid in the quench tank can be circulated through a cooler for temperature control, sampled and the excess liquid pumped to the Letdown Storage Tank, coolant bleed holdup tanks or the Liquid Waste Disposal System. This portion of the Coolant Storage System is shown schematically on Figure 9-20.

The deborating demineralizers may also be loaded with mixed bed resin and used as purification demineralizers to support normal purification and boron/lithium coordination programs.

The coolant bleed holdup tanks and the concentrated boric acid storage tanks are vented to the gaseous waste vent header to provide for filling and emptying without overpressurization or causing a vacuum to exist. In addition, each tank is equipped with a relief valve and a vacuum breaker. Pressurized nitrogen can be supplied to each tank to allow purging.

Instruments and controls for operation of this system are located in the control rooms. Instruments and controls for the coolant bleed holdup tanks and pumps and for the concentrated boric acid storage tanks and pumps are duplicated on the auxiliary control boards.

9.3.5 Coolant Treatment System

The Coolant Treatment System was originally designed and installed to both store reactor coolant bleed and to treat RC bleed for recycling. Since the boron recycling portion of the original Coolant Treatment System never functioned properly, the coolant storage portion is the only part of the system still in use at Oconee. The Coolant Storage System is described in Section 9.3.4. Radwaste processing is described in Section 11.6.3.

9.3.6 Post-Accident Sampling System

9.3.6.1 Post-Accident Liquid Sampling System

9.3.6.1.1 Design Bases

This system provides the capability to obtain and analyze a liquid Reactor Coolant System sample under accident (Reg. Guide 1.3 or 1.4 release of Fission products) conditions without incurring a radiation exposure to any individual in excess of five (5) rems whole body dose or 75 rems to extremities. The system is currently only used to take a reactor coolant boron sample during a Steam Generator Tube Rupture event or an event with radiation exposure potential equal to or less than a Steam Generator Tube Rupture.

9.3.6.1.2 System Description and Evaluation

The Post-Accident Liquid Sample System consists of a sample panel that houses the tubing, valving, instrumentation and system components. The system is controlled and monitored remotely from the sample control panel. The system is schematically illustrated in Figure 9-22. Note that the only portions of this system currently used are those needed to obtain an RCS boron sample. There is one separate system for each Oconee unit, located at elevation 771+0 in the auxiliary building.

The control panel actuates valves that are outside containment and before/after the sample enters/leaves the sampling panel. The remaining valves are controlled either by Operations from the control room or by Operations manually. All valves involving penetrations to the reactor building are normally closed except when sampling.

Deleted paragraph(s) per 2005 update.

9.3.6.1.3 Deleted Per 2005 Update

9.3.6.2 Post-Accident Containment Air Sampling System

9.3.6.2.1 Design Bases

The system is no longer used to obtain and analyze a containment air sample under accident conditions. Even though the system is no longer used, Figure 9-23 schematically illustrates the system.

9.3.6.2.2 Deleted Per 2005 Update

9.3.6.2.3 Deleted Per 2005 Update

9.3.7 Containment Hydrogen Monitoring System

9.3.7.1 Design Bases

The containment Hydrogen Monitoring System provides continuous indication of hydrogen concentration in the containment atmosphere. The measurement capability is provided over the range of 0% to 10% hydrogen concentration under both positive and negative ambient pressures. A continuous indication of the hydrogen concentration is not required in the control room at all times during normal operation. If continuous indication of the hydrogen concentration is not available at all times, continuous indication and recording shall be functioning within 90 minutes of the initiation of the safety injection.

9.3.7.2 System Description

The Containment Hydrogen Monitor System withdraws a sample from the containment under normal, LOCA or Post LOCA conditions. The sample is analysed and returned to the containment. The monitoring system is designed to monitor containment gas for percentage volume of hydrogen.

A system of sample taking tubing is installed in the containment to draw air samples from 5 different levels or areas. Each of the sample intake lines has a solenoid valve which is remotely operated from a control panel in the ventilation room. At the control panel a selector solenoid valve is used to provide air flow to the Hydrogen Analyser from the selected intake port. The Hydrogen Analyser panels and associated remote control panels are located in the ventilation room. Remote alarm and indication is provided in the control room. There are two trains of equipment for each unit.

Ten Hydrogen Analyzer intake ports are installed, (two each) in the following locations:

The top of the Containment Building Dome, Elevation 983' ± 5"

The operational level as close to the vessel as practical, Elevation 844 + 0' ± 10'

The basement area, Elevation 788' + 0" ± 10'

The radiation monitor/hydrogen recombiner inlet header, Elevation 827' + 4"

The radiation monitor/hydrogen recombiner outlet header, Elevation 824' + 0"

Hydrogen Measurement

Analysis is accomplished by using the well established principle of thermal conductivity measurements of gases. This technique utilizes a self-heating filament fixed in the center of a temperature-controlled metal cavity. The filament temperature is determined by the amount of heat conducted by the presence of gas from the filament of the cavity walls. Thermal conductivity varies with gas species, thereby causing the filament temperature to change as the gas in the cavity changes. Filament resistance changes with temperature therefore, by using two filaments in separate cavities and connecting them in an electrical bridge, the difference in thermal conductivity of gases in the separate cavities may be determined electrically.

Electrical zero is set by first introducing the same gas to both cavities, then adjusting the electrical bridge to balance, resulting in a zero output. As different gases are introduced to the two individual cavities, the bridge will become unbalanced, and the electrical output will amplify with increasing differences in thermal conductivity of the gases used.

The measurement of hydrogen in the presence of nitrogen, oxygen and water vapor is possible because the thermal conductivity of hydrogen is approximately seven times higher than nitrogen, oxygen or water vapor, which have nearly the same thermal conductivities (at the filament operational temperature of approximately 550°K). The measurement is accomplished by using a thermal conductivity measurement cell and a catalytic reactor. The sample first flows through the reference section of the cell, then passes through the sample section of the measuring cell that includes the catalyst. The catalyst is chosen so that post-LOCA iodine will not poison the catalyst bed. The change in sample composition, due to the catalytic reaction is therefore indicated by the difference in thermal conductivity of the sample hydrogen content, as measured in the sample and reference sides of the cell.

If an excess amount of oxygen does not exist in the sample for recombining all the hydrogen, oxygen can be provided ahead of the hydrogen analyzer. The amount of oxygen added is determined by the highest range of the analyzer.

Alarms

Alarms are provided for high hydrogen concentration, cell failure and loss of power. These alarms are available on the analyzer itself and as signals to the control room annunciator. Additional alarms on the analyzer itself include low instrument temperature, low sample flow, low gas pressure and common failure.

9.3.7.3 Safety Evaluation

The Containment Hydrogen Monitor System (CHMS) meets the requirements of NUREG-0737, Item II.F.1.6. The CHMS has both indicator and recorder readouts in the control room on one of the two redundant channels and a indicator readout on the second channel. The CHMS has a range of 0% to 10% of Hydrogen. The CHMS indicator loop has a system accuracy of 3.0% of the full scale. The CHMS hardwired recorder loop and all the CHMS plant process computer loops have a system accuracy of 2.6% of the full scale. These values will provide information over the intended range of the CHMS that is sufficiently accurate and useful to allow the plant operator to adequately assess the hydrogen concentration within containment. There are five ports to draw samples for each of the redundant hydrogen monitors. The system provides capability to rapidly detect Hydrogen from the reactor and determine its concentration throughout the containment.

THIS IS THE LAST PAGE OF THE TEXT SECTION 9.3.

9.4 Air Conditioning, Heating, Cooling and Ventilation Systems

9.4.1 Control Room Ventilation

9.4.1.1 Design Bases

The Control Room Ventilation and Air Conditioning Systems are designed to maintain the environment in the control area which is comprised of the Control Room, Cable Room and Electrical Equipment Rooms as indicated on Figure 9-24 within acceptable limits for the operation of unit controls as necessary for equipment and operating personnel. In addition, redundant air conditioning and ventilation equipment is provided, as summarized below, to assure that no single failure of an active component within these systems will prevent proper control area environmental control.

1. Two 100 percent capacity supply fans with filter banks and chilled water coils for cooling the control rooms and the Unit 3 cable & electrical equipment rooms.
2. Two 100 percent capacity chillers.
3. Two 100 percent capactiy chilled water pumps.
4. Two 100 percent capacity chiller condenser service water pumps.
5. Two 50 percent capacity outside air booster fans.
6. Four supply fans with filter banks and chilled water coils serving the Units 1&2 cable & electrical equipment rooms.
7. Four motorized control dampers in the cable shafts between the Units 1 and 2 Cable and Electrical Equipment Rooms.

Acceptable limits for equipment in the cable rooms and for the electrical equipment rooms is 120°F and 100°F for the Control Room.

Design conditions for the Control Room are 74°F and 50 percent maximum relative humidity. The Equipment Room is designed for 86°F and all other areas, i.e., the Control Room Zone and Cable Room are designed for 74°F. Outdoor design conditions are 95°F dry bulb and 76°F wet bulb. The ventilation and air conditioning systems are designed for continuous operation.

The radiation monitor, RIA-39, has a continuous sample of control room air pumped through the detector. High radiation level and loss of sample flow are annunciated separately. If high radiation level is detected, the operator starts the outside air filter trains if not already started by Emergency Procedures or Abnormal Procedures. Emergency Procedures and Abnormal Procedures direct operators to start the outside air filter trains regardless of radiation levels inside the Control Rooms. If loss of sample flow occurs, backup sampling or other alternate operator actions are performed, as required, until RIA-39 is restored. The outside air filter trains act to filter particulate matter from the outside air to minimize uncontrolled infiltration into the Control Room.

Control area temperatures related to Station Blackout are addressed by Selected Licensee Commitment 16.8.1. The pressurization and filtration of the control room envelope is discussed further in Section 6.4.

9.4.1.2 System Description

9.4.1.2.1 Control Room Oconee 1 and 2

The Control Room for Oconee 1 and 2 is shared for the operation of both units. The Control Room is primarily served by two large air handling units. The units are 100 percent capacity and only one unit is required to operate at a time. Cooling is provided to the Unit l Cable Room, Unit 2 Cable Room, Unit l Equipment Room, and Unit 2 Equipment Room by a total of four air handling units. An automated damper control system will operate to maintain acceptable temperatures in the cable and electrical equipment rooms if one of the cable rooms AHUs is out of service.

All of the air handling units described above consist of roughing filters, chilled water cooling coils, and centrifugal fans. Chilled water is supplied to the units from the plant WC chilled water system. Electric duct heaters are installed in the ductwork to provide heat to the different areas when necessary.

Outside air is supplied to the Control Room for pressurization purposes, from dual intakes on the Turbine Building roof. Air passes through filter trains which consist of pre-filters, 99.5 percent efficient HEPA filters, 97.5 percent efficient charcoal filter beds, and a centrifugal fan. There are two 50 percent filter trains and the system is capable of operating with one train or both trains. During normal plant operations, the filter trains are not energized and require operator action to start. The outside air is supplied to the return air intake of the large air handling units which serve the Control Room. A radiation monitor is provided in the return air intake of the air handling units to alert the operators in the Control Room on a high radiation reading at which time the operators start the outside air filter trains if not already started by Emergency Procedures or Abnormal Procedures. The filter trains are designed for a flow of 1350 cfm each. The pressurization system was not designed or licensed to maintain a positive pressure in the Control Room assuming a single failure.

A chlorine monitor is provided in the Outside Air Intake Duct to each Control Room Outside Air Booster fan. Detection of high chlorine by either monitor will actuate an alarm in the Control Room, de-energize the Booster Fans, and close the Control Room Ventilation Dampers.

Cooling is provided to the Cable Rooms and Electrical Equipment Rooms by four air handling units located in the vicinity of the rooms.

Table 9-11 is a list of the air handling units and operation requirement for the Control Room and Control Room Zone air conditioning system. Figure 9-24 is a schematic description of the ventilation and air conditioning systems for the Control Room and Control Room Zone.

9.4.1.2.2 Control Room Oconee 3

The Oconee 3 Control area is comprised of the Control Room, the Cable Room, and the Electrical Equipment Room. These areas are served by six air handling units. Two 100 percent air handling units serve the Control Room, two 100 percent air handling units serve the Cable Room, and two 100 percent air handling units serve the Electrical Equipment Room. The air handling units consist of roughing filters, chilled water cooling coils, and centrifugal fans. Chilled water is supplied to the air handling units by the Plant WC Chilled Water System.

Outside air is supplied to the Control Room for pressurization purposes by two 50% trains taking suction from dual intakes on the Turbine Building roof. The outside air passes through a filter system composed of a prefilter, 99.5 percent efficient HEPA filter, 97.5 percent efficient charcoal filter beds and centrifugal fan. Outside air is supplied to the return air intake of the air handling units which serve the Control Room. The outside air system is started by the plant operators. The pressurization system was not designed or licensed to maintain a positive pressure in the Control Room assuming a single failure.

A chlorine monitor is provided in the Outside Air Intake Duct to each Control Room Outside Air Booster Fan. Detection of high chlorine by either monitor will actuate an alarm in the Control Room, de-energize the Booster Fans, and close the Control Room Ventilation Dampers.

A radiation monitor is provided to sample the return air entering the Control Room and Control Room Zone air handling units. The monitor alarms on a high radiation signal and alerts the operators to energize the outside air filter system if not already started by Emergency Procedures or Abnormal Procedures to minimize the infiltration of unfiltered air into the Control Room.

Table 9-11 lists the air handling unit and operation requirements. Figure 9-24 is a schematic representation of the air conditioning system.

9.4.1.3 Safety Evaluation

The Control Room is served by redundant air handling units. The chilled water for the air handling units is supplied from the Plant WC Chilled Water System which is capable of supplying sufficient chilled water for all necessary systems with one of two chillers in service or a temporarily cooling train.

Return air from the Control Room is continuously monitored by a radiation monitor before recirculating back to the Control Room. A high radiation level will alert the operators to energize the outside air filter trains if not already started by Emergency Procedures or Abnormal Procedures. The filter trains are 50 percent, each train consisting of a prefilter, HEPA filter, 97.5 percent efficient charcoal filter bed and centrifugal fan. The filters act to filter particulate matter from the outside air supplied to minimize uncontrolled infiltration into the Control Room.

9.4.1.4 Inspection and Testing Requirements

The Control Room Ventilation System is in continuous operation and is accessible for periodic inspection. The Control Room 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 the Electrical Equipment Rooms are periodically monitored, as required by SLC's to ensure proper system operation.

9.4.2 Spent Fuel Pool Area Ventilation System

9.4.2.1 Design Bases

The Spent Fuel Pool Area Ventilation System is designed to maintain a suitable environment for the operation, maintenance and testing of equipment and also for personnel access. The ventilation system is designed to maintain the Spent Fuel Pool Area at a maximum inside temperature of 113°F and a minimum temperature of 60°F.

The path of ventilating air in the Spent Fuel Pool Area is from areas of low activity toward areas of progressively higher activity for discharge to the unit vent.

An air handling unit consisting of roughing filters, steam heating coil, cooling coil supplied by low pressure service water, and a centrifugal fan supply 100 percent outside air to the Spent Fuel Pool Area. During periods of increased work in Spent Fuel Pool areas, Air Handling Units normal LPSW supply may be replaced temporarily by chilled water from the CW System by station procedures to increase capacity and protect workers from heat stress. Two methods of exhausting air from the Fuel Pool Area are provided, a filtered exhaust system and an unfiltered exhaust system. Normal operation is with the unfiltered system in operation. In the filter mode, the Fuel Pool Area ventilation air passes through a filter train consisting of prefilters, high efficiency particulate (HEPA) filters, charcoal filter and two 100 percent vane axial fans. The filtered exhaust system is operable whenever fuel handling operations involving recently irradiated fuel assemblies in the fuel pool are in progress.

The Spent Fuel Pool Area air is continuously monitored by radiation monitor, RIA-41.

9.4.2.2 System Description

Ventilation air for the Spent Fuel Pool Area is supplied by an air handling unit which consists of roughing filters, steam heating coil, cooling coil supplied by low pressure service water, and a centrifugal fan. Temperature is maintained in the Spent Fuel Pool Area by throttling steam to the heating coil or low pressure service water to the cooling coil. During periods of increased work in Spent Fuel Pool areas, Air Handling Units normal LPSW supply may be replaced temporarily by chilled water from the CW System by station procedures to increase capacity and protect workers from heat stress.

In the normal mode of operation, the air from the Spent Fuel Pool Area is exhausted directly to the unit vents by the general Auxiliary Building exhaust fans. When fuel handling operations involving recently irradiated fuel assemblies are in progress, the filtered exhaust system must be operable so in the event of an emergency the air leaving the Fuel Pool Area can be filtered.

The filtered exhaust system consists of a single filter train and two 100 percent capacity vane axial fans. The filter train utilized is the Reactor Building Purge Filter Train. The filter train is comprised of prefilters, HEPA filters, and charcoal filters. An attempt to start the main Reactor Building purge fan will stop the Spent Fuel Pool filtered ventilation.

To control the direction of air flow, i.e., to direct the air from the Fuel Pool Area to the Reactor Building Purge Filter Train, a series of pneumatic motor operated dampers are provided along with a crossover duct from the Fuel Pool to the filter train.

Figure 9-25 and Figure 9-26 are detailed diagramatics of the Spent Fuel Pool Area Ventilation System. The flow paths as well as air quantities are given in the diagram. With the adoption of the alternate source term and installation of various modifications, the Spent Fuel Pool Ventilation System is not credited in dose analysis calculations.

9.4.2.3 Safety Evaluation

Prior to handling recently irradiated fuel assemblies in the Spent Fuel Pool Area, the Spent Fuel Pool Ventilation System must be made operable as required by the Technical Specifications.

There are two 100 percent capacity vane axial fans which direct the Spent Fuel Pool air through the Reactor Building Purge Filter Train prior to being released to the unit vent. Only one fan is required for operation. The fans are manually energized by the operators should it become necessary to filter the exhaust air from the Fuel Pool Area. The automatic control sequence is such that the damper alignment, to redirect air flow through the Reactor Building Purge Filters, is automatically done when one of the fans is energized.

An alarm is provided when the fuel pool filtered flow drops below 70 percent of design flow.

A radiation monitor is provided to continuously monitor the fuel pool air and will alarm on a high radiation level.

The analysis of the limiting fuel handling accident, the cask drop accident, assumes that a certain number of fuel assemblies are damaged. The DBA analysis for the cask drop accident does not assume operation of the SFPVS in order to meet requirements of 10CFR50.67. The assumptions and analysis are consistent with guidance provided in Regulatory Guide 1.183.

9.4.2.4 Inspection and Test Requirements

The normal mode of the Spent Fuel Pool Area Ventilation System is in continuous operation and is accessible for periodic inspection. The filtering mode of the Spent Fuel Pool Area Ventilation system is tested prior to movement of recently irradiated fuel assemblies to demonstrate its readiness and operability as required by the Technical Specifications.

9.4.3 Auxiliary Building Ventilation System

9.4.3.1 Design Bases

The Auxiliary Building Ventilation System is designed to provide a suitable environment for the operation, maintenance and testing of equipment and also for personnel access.

The Auxiliary Building Ventilation System serves all areas of the Auxiliary Building with the exception of the Control Room Area and the Penetration Rooms. The ventilation system indoor design conditions are 104°F and 60°F during summer and winter respectively. During normal operation, the system maintains temperatures within limits for equipment operation.

Ventilation air is supplied to both clean and potentially contaminated areas within the Auxiliary Building. The flow path of the ventilation air in the Auxiliary Building is from clean or low activity areas towards areas of progressively higher activity.

All potentially contaminated air from the Auxiliary Building is directed to the unit vent stacks at which point it is exhausted and continuously monitored by a radiation monitor which alarms on high radiation levels. In addition, a radiation monitor samples air throughout the Auxiliary Building Ventilation System. The detector output is logged on a recorder in the Control Room. All air from the Hot Machine Shop is exhausted to the atmosphere after being measured by an air flow monitor. Periodically, radiation levels are checked in the air flow using an air flow totalizer and particulate sampler.

The exhaust fans and supply fans are manually balanced such that the exhaust flow exceeds the supply air flow to minimize outleakage.

9.4.3.2 System Description

The Auxiliary Building Ventilation System is comprised of the Auxiliary Building Ventilation System proper and the Hot Machine Shop as shown in Figure 9-27 and Figure 9-28. Air is supplied to the Auxiliary Building by a low pressure fan duct system. Air is taken in through outside air intake louvers by supply units consisting of roughing filters, steam coil, and cooling coil supplied by low pressure service water. There are six main supply fans, each required for normal plant operation. Auxiliary Building air is exhausted from the building, via exhaust duct and exhaust fans, through three unit vent stacks.

The Hot Machine Shop air is supplied by two recirculating local cooling units. Each unit consists of roughing filters, a compressor, evaporator and condenser coils, and centrifugal fan. These units supply recirculated air with a small amount of make-up air throughout the Hot Machine Shop via a low pressure duct system. Air is exhausted from the Hot Machine Shop via exhaust duct and filter train and is discharged to the atmosphere through an independent vent stack.

Table 9-11 is a list of the primary equipment which comprises the Auxiliary Building Ventilation System and the Hot Machine Shop Ventilation System. The list includes number of installed components and normal operation requirements.

Temperatures are maintained in the Auxiliary Building by throttling steam to the steam coils or low pressure service water to the cooling coils as required. Temperatures are maintained in the Hot Machine Shop by electric unit heaters in the supply ductwork. The Hot Machine Shop uses direct expansion (DX) cooling.

Remote recirculating fan-coil type units provide standby spot cooling in the pump rooms and other high heat load areas. The fan coil units are also served by the Low Pressure Service Water System.

9.4.3.3 Safety Evaluation

Under normal operating conditions, the Auxiliary Building Ventilation System supply fans and exhaust fans are balanced such that the exhaust air flow exceeds the supply air flow in order to minimize outleakage.

All exhaust air from potentially contaminated areas of the Auxiliary Building is directed to the unit vents where it is monitored prior to being released to the atmosphere. All exhaust air from the Hot Machine Shop is monitored prior to being released to the atmosphere through an independent vent stack.

HPI and LPI/RBS pump room temperatures are maintained within pump temperature limits by natural convection if the Auxiliary Building Ventilation System is unavailable. The natural convection flow path is for air to enter the pump rooms through duct openings and escape through stairwell and piping openings.

9.4.3.4 Inspection and Testing Requirements

The Auxiliary Building Ventilation System and the Hot Machine Shop Ventilation System are in continuous operation and are readily accessible for periodic inspection and maintenance.

9.4.4 Turbine Building Ventilation System

9.4.4.1 Design Bases

The Turbine Building Ventilation System is designed to provide a suitable environment for the operation of equipment and personnel access as required for inspection, testing and maintenance.

9.4.4.2 System Description

The Turbine Building is ventilated using 100 percent outside air. Air is supplied through wall openings along the east wall and is exhausted by fans mounted in the roof and along the west wall.

There are twelve roof mounted exhaust fans. Eighteen additional exhaust fans are located along the west wall. Each of the thirty fans are independently operated so that all or a portion of the fans can run as needed to maintain conditions within the Turbine Building.

Table 9-11 is a list of the primary equipment which includes the Turbine Building Ventilation System Exhaust Fans. The list includes number installed and normal operation requirements.

9.4.4.3 Safety Evaluation

The Turbine Building Ventilation System operates to maintain suitable environmental conditions in the Turbine Building during normal plant operation.

9.4.4.4 Inspection and Testing Requirements

The Turbine Building Ventilation System is in continuous operation during normal plant operation and is readily accessible for periodic inspection and maintenance.

9.4.5 Reactor Building Purge System

9.4.5.1 Design Bases

The Reactor Building Purge System purges the Reactor Building with fresh air during unit outages.

During operation, outside air is introduced into the Reactor Building through a supply system which has dual isolation valves at the containment wall. Outside air is circulated throughout the Reactor Building by the normal Reactor Building Ventilation System. Air is then exhausted from the Reactor Building by the Reactor Building purge exhaust filter train.

The filter train consists of prefilters, HEPA filters, and charcoal filters. A centrifugal fan is positioned downstream of the filter train. There are double isolation valves in the piping running from the Reactor Building to the filter train.

The isolation valves are automatic, are normally closed, and are opened only for the purging operation. The valves are arranged so the purge supply piping and the purge exhaust piping each have a electrically actuated valve inside the Reactor Building and a pneumatically actuated valve outside the Reactor Building.

There are two modes of operation possible for the Reactor Building Purge System; normal purge, and mini-purge. The system also has a recirculation mode, however it is not used because of duct leakage concerns. The purge filter train can also be used to provide filtered exhaust as discussed in Section 9.4.2.

9.4.5.2 System Description

The “Reactor Building Purge System” (Figure 6-4) purges the Reactor Building with fresh air to reduce airborne contaminant levels inside the Reactor Building.

The supply portion of this system consists of an outside air intake louver, roughing filters, a steam heating coil, associated ductwork and dual isolation valves at the reactor building wall. The exhaust portion of this system consists of a filter train, fans, associated ductwork, and dual isolation valves at the Reactor Building wall. The filter train consists of prefilter, HEPA filter, and charcoal filter. The isolation valves are automatic, normally closed in accordance with the requirements of NUREG 0737, Item II.E.4.2.6, and are opened only for the purging operation. The valves are so arranged that the supply portion and exhaust portion of the system each have an electrically actuated isolation valve inside the Reactor Building and two (2) pneumatically operated valves outside the Reactor Building (one is an isolation valve). A bleed valve between the two (2) outer valves vents any leakage from the Reactor Building into the penetration room.

There are two modes of operation possible for the “Reactor Building Purge System”: 1) the normal purge, and 2) the mini-purge.

The normal purge mode purges the Reactor Building with 35,000 cfm of fresh air which enters by way of the supply portion and leaves by way of the exhaust portion described above. The filtered exhaust air is all released to the atmosphere via the unit vent.

The mini-purge mode of operation provides a means to purge the Reactor Building at a reduced flow rate when activity levels are higher than desired for full purging. A 10,000 cfm vane-axial fan is provided to by-pass the normal purge exhaust fan. A series of pneumatically operated dampers provide isolation and control. During mini-purge, flow from the Reactor Building is through the purge filter train and can be modulated up to a maximum of 10,000 cfm. The vane-axial mini-purge fan is constant volume and to maintain 10,000 cfm flow, Reactor Building air is mixed with outside air, i.e., the more air being purged from the Reactor Building, the less air drawn from the outside air make-up intake. The mini-purge fan and normal purge fan cannot operate simultaneously.

9.4.5.3 Safety Evaluation

Each Reactor Building Purge System supply and exhaust penetration of the Reactor Building wall is equipped with dual isolation valves. The valves inside the Reactor Building are electrically operated and the valves outside the Reactor Building have pneumatic actuators. The valves operate independently of one another and are in the closed position unless the purge is in operation.

The Purge System discharge to the unit vent is monitored and alarmed to prevent the release from exceeding acceptable limits.

9.4.5.4 Inspection and Testing Requirements

The Reactor Building Purge System is normally not in operation. The equipment and component are accessible for periodic maintenance. Parts of the system are maintained and tested in accordance with the Technical Specifications.

9.4.6 Reactor Building Cooling System

9.4.6.1 Design Bases

The Reactor Building Cooling Systems are designed to remove the heat in the containment atmosphere during normal plant operation and post accident operation.

A portion of the Reactor Building Cooling System is described in Section 6.2.2 as an Engineered Safety Feature.

The Reactor Building Cooling System is composed of two subsystems: Reactor Building Coolers and Reactor Building Auxiliary Coolers.

All components of the Reactor Building Cooling System are inside the Reactor Building. The only penetrations into and out of the Reactor Building that are related to the cooling system are the low pressure service water supply and return lines and isolation valves are provided on these lines at the penetrations.

9.4.6.2 System Description

The Reactor Building Cooling System shown in Figure 6-3 consists of the following subsystems and components:

1. Three Reactor Building Cooling Units (RBCUs), each consisting of a 2-speed vane axial fan, four cooling coils and distribution ductwork. These three cooling units are Engineered Safety Systems.
2. Four Reactor Building Auxiliary Cooling Units, each consisting of a 2-speed vane axial fan, four cooling coils, and distribution ductwork.

Deleted paragraph(s) per 2006 update.

The LPSW flow is provided to four Reactor Building Auxiliary Cooling Units (RBACs) through a separate piping loop that is independent of the RBCUs. During normal plant operation, any combination of the three RB cooling units (“A”, “B”, and “C”) may operate in the high or low speed mode to provide normal cooling of the Reactor Building.

The RB cooling units circulate Reactor Building air over low pressure service water supplied cooling coils and distribute the cool air throughout the lower portion of the Reactor Building. The Auxiliary Cooling Units distribute the cool air via a duct system to the upper portion of the Reactor Building. The temperature in the Reactor Building can be controlled by varying the number of Auxiliary Cooling Units or RBCUs running, changing their speed, or by supplying chilled water from a temporary chiller to the Auxiliary Coolers in lieu of LPSW in modes 1-4 and to the Auxiliary Coolers and/or the “B” RBCU in modes 5, 6, and no mode.

During an emergency, the Reactor Building Cooling System mode of operation changes automatically. Upon receipt of the signal from the Engineered Safeguards Actuation System, the operating Reactor Building Cooling Units change to low speed operation and any idle unit(s) is energized at low speed. This change occurs after a three (3) minute delay. Upon an ES signal, the Reactor Building Cooling Units operating in high or low speed automatically stop, and then restart in low speed operation after a three (3) minute time delay, and any idle unit(s) is also energized at low speed after a three (3) minute time delay. The fans are run at the slower speed because of the changed horsepower requirements generated by the denser building atmosphere. The LPSW flow path to RBACs is automatically isolated by the closure of air-operated containment isolation valves (LPSW-1054, 1055, 1061, and 1062) on ES signals. Additionally, all Low Pressure Service Water valves at the discharge of the three RBCUs go to the full open position.

The accident may impose severe stresses on the lower portion of the duct work, causing possible collapse or deformation. Therefore, the fusible links holding the dropout plates provided in the duct work below the coils melt and drop off, assuring that a positive path for recirculation of the Reactor Building atmosphere is available. 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 fusible dropout plate. See Figure 6-3.

9.4.6.3 Safety Evaluation

The three Reactor Building Cooling Units (RBCUs) are an engineered safety feature. These units alone can provide the design heat removal capacity to keep containment pressure below the design limit following a loss-of-coolant accident with all three coolers operating by continuously circulating the steam-air mixture past the cooling tubes to transfer heat from the containment atmosphere to the low pressure service water.

Inside the Reactor Building, the cooling units are located outside the secondary shield at an elevation above the water level in the bottom of the Reactor Building during post-accident conditions. In this location, the units are protected from being flooded.

The major equipment of the Reactor Building Cooling Units is arranged in three independent strings with three duplicate service water supply lines. In the unlikely event of a failure in one of the three cooling units, half of the Reactor Building Spray System capacity combined with the remaining two cooling units, is capable of keeping the containment temperature and pressure within environmental qualification (EQ) limits and is capable of keeping containment pressure below the design limit after a loss-of-coolant or steam line break accident. Acceptable fan-motor operation is verified by testing each refueling outage or every 24 months.

A failure analysis of the cooling units is presented in Table 6-6.

The NRC issued Generic Letter 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Conditions," on September 30, 1996, requesting that licensees determine if containment air cooler cooling water systems are susceptible to either water hammer or two-phase flow conditions during postulated accident conditions and to determine if piping systems that penetrate containment are susceptible to thermal expansion of fluid so that overpressurization of piping could occur. Evaluations of affected Oconee Nuclear Station systems and components were completed with response to the NRC submitted in the letter from M. S. Tuckman to the NRC, dated January 28, 1997. The evaluations determined that the Oconee Nuclear Station containment air cooler cooling water systems were not susceptible to significant two-phase flow conditions, but that some types of water hammer could occur during accident conditions. Commitments were provided in a letter from W. R. McCollum to the NRC, dated 9/30/02 to implement two modifications to mininmize water hammer potential: (1) changes to the LPSW piping in containment to prevent drainage from the system, and (2) modifications to the containment ventilation system to separate the RBACU from the RBCU trains. Regarding the thermal overpressure concern, further evaluations performed concluded that certain piping systems that penetrate the containment were susceptible to thermal expansion of fluid so that overpressurization of piping could occur. Commitments provided in a letter from W. R. McCollum to the NRC, dated 12/17/98, identified a list of containment penetrations and associated piping that warranted modifications to provide overpressure protection. The NRC approved the Oconee responses and closed the Generic Letter 96-06 in a letter to B. H. Hamilton, dated 12/06/07. The referenced modifications for containment ventilation system train separation and containment penetration overpressure protection were completed. The NRC-approved LPSW piping drainage modifications are pending future Oconee refueling outages.

9.4.6.4 Inspection and Testing Requirements

See "Tests and Inspections" under Section 6.2.2.

9.4.7 Reactor Building Penetration Room Ventilation System

9.4.7.1 Design Bases

Prior to the adoption of the alternate source term, Reference 2, the Penetration Room Ventilation system was required to collect and process post-accident Reactor Building leakage by establishing a vacuum in the Penetration Rooms and processing the leakage through a prefilter, an absolute filter, and a carbon filter prior to release by way of the unit vent. This system is still available but no longer required to serve an accident mitigation function. Reference Figure 6-4 for a schematic of the system.

This system is designed to collect and process potential Reactor Building penetration leakage to minimize environmental activity levels resulting from post-accident Reactor Building leaks. Experience has shown that Reactor Building leakage is more likely at penetrations than through the liner plates or weld joints.

The main function of the system is to control and minimize the release of radioactive materials from the Reactor Building to the environment in post-accident conditions.

Leakage into each of the penetration rooms is discharged to the unit vent through a pair of filter assemblies each consisting of a prefilter, an absolute filter, and a charcoal filter in series. The entire system is designed to operate under negative pressure up to the fan discharge.

The Penetration Room Ventilation System is not vulnerable to control malfunctions since it is controlled manually. Instrumentation is used only to monitor system performance and has no control function other than to guide the operator in adjusting the final control elements.

More detailed information concerning radiation levels and leakage requirements are discussed in Section 6.5.1.

9.4.7.2 System Description

The Penetration Room Ventilation System is provided with two fans and two filter assemblies. Both fans discharge through a single line to the unit vent. A schematic of the system is shown in Figure 6-4.

During normal operation, this system is held on standby with each fan aligned with a filter assembly. The engineered safeguards signal from the Reactor Building pressure will actuate the fans. The Control room, as well as remote instrumentation, monitors operation.

The design flow rate from the penetration room far exceeds the maximum anticipated Reactor Building leakage. The design leak rate of 0.1 volume percent per day from the Reactor Building to the penetration room (this is one-half of the total design leak rate out of the Reactor Building referenced in Section 6.2.1) amounts to approximately 6.2 scfm compared to a design evacuation rate of 1000 scfm for each half of the system. The three valves in each purge line penetration will be closed by Reactor Building isolation signal. The Reactor Building Purge Equipment, if running, will be shut down from an interlock on the Reactor Building Purge isolation valves. After closing of the external valves, a small normally open valve vents the leakage, if any, from the two outermost valves into the penetration room. The Reactor Building Purge Equipment is not activated when the reactor is above cold shutdown conditions.

Following a loss-of-coolant accident, a Reactor Building isolation signal will place the system in operation by starting both full-size fans. Two power-operated butterfly valves which open when the fans start are provided at the discharge of each fan. This valve will be closed to prevent recirculation if one fan fails. A check damper is also provided at the discharge of each fan to prevent recirculation on failure of a fan. In the event of a fan failure, the normally closed tie valve (PR-20) can be opened from its remote manual station to maintain cooling air through the idle filter train. Even if air flow is lost through a filter train, Reference 2 has shown that the charcoal ignition temperature will not be reached and operation of PR-20 is not required.

The system utilizes remote manual control valves PR-13 and PR-17 in conjunction with constant speed fans. Locations of penetrations and openings in the penetration room are shown on Figure 6-23 and Figure 6-24.

The remote manual control valve is also used to compensate for filter loading. Initially, it will be partially closed; and as the filter loads up causing a decrease in flow and negative penetration room pressure, the valve will gradually be opened so that the pressure drop across the filter-valve combination remains constant. By periodically adjusting the remote manual control valve to offset the effect of increased leakage and filter loading, the system characteristic remains constant.

The communicative paths between various parts of the penetration room are very large in comparison with the minute leakage that might exist due to imperfect seals. It therefore can be assumed that no pressure differentials exist in the room so that an instrument string sensing pressure at a single point can be used. Penetration room pressure is displayed in the control room and excessive and insufficient vacuum are annunciated.

Fan status and radiation level of filter effluent are displayed in the control room and excessive radiation is annunciated. Filter ΔP is displayed locally. Filter flow is displayed remotely adjacent to the remote manual control valves PR-13 and PR-17 remote control stations.

The system may be actuated by an operator during normal operation for testing. It may also operate intermittently during normal conditions as required to maintain satisfactory temperature in the penetrations rooms.

Particulate filtration is achieved by a medium efficiency pre-filter and a high efficiency (HEPA) filter.

The pre-filter consists of multiple horizontal tubular bags attached to a vertical metal plate header. The bags are made of ultra fine glass fibers and are supported so that adjacent bags do not touch and reduce the flow area. At the filter train design flow of 1000 cfm, the pre-filter is operating at one-half its rated flow.

The HEPA filter will intercept any particulates that pass through the pre-filter. The filter consists of a single cell of fiber glass media mounted in a metal frame. The cell has face dimensions of 24 inches x 24 inches and a depth of 11½ inches and is rated at 1150 scfm.

Adsorption filtration is accomplished by an activated charcoal filter. The filter consists of three horizontal removable type double tray carbon cells. Flow through the trays is essentially vertical. Each tray has a face area of 4.2 sq ft and a bed depth of 2 inches. At rated flow (167 cfm), the average face velocity is 40 ft/min and the residence time is 0.25 seconds. Each tray contains 40 lbs of carbon. The carbon is impregnated so that it will adsorb methyl iodide as well as elemental iodine.

9.4.7.3 Safety Evaluation

The Penetration Room Ventilation system is no longer required due to the adoption of the alternate source term, Reference 2.

9.4.7.4 Inspection and Test Requirements

The Penetration Room Ventilation System is not normally in operation, but the equipment is accessible for periodic inspection. The entire system can be tested during normal operation.

9.4.8 References

1. Deleted per 2005 update.
2. License Amendment No. 338, 339, and 339 (date of issuance – June 1, 2004); Adoption of Alternate Source Term.

THIS IS THE LAST PAGE OF THE TEXT SECTION 9.4.

Other Auxiliary Systems

Deleted Paragraph(s) per 2011 update.

9.5.1 Fire Protection

Fire protection and fire safety play a critical role in overall safety for nuclear power. The history of the industry shows that while fires are rare, consequences can be quite high depending on the location and severity of a fire within the plant. Therefore, the risk from fire is not insignificant in nuclear power facilities.

Fire protection is comprised of numerous interworking elements. The most visible elements include those fire protection design features installed in the plant such as fire detection systems, fire pumps, sprinklers, hose stations and fire extinguishers. Less visible, but of equivalent importance are passive design features such as fire-rated structures and barriers and equipment separation to prevent a fire in one area from damaging an entire system function within the same area or fire zone. All of these designed protection features are supplemented through programmatic elements designed to further reduce the risk of fire that can be introduced by the human element present in all operating environments.

The combination of these design and programmatic elements are referred to as the Fire Protection Program which is captured in the Fire Protection Design Basis Document (DBD).

(Clarifying Note: Throughout this UFSAR section on Fire Protection, general reference is made to the Fire Protection Design Basis Document (DBD) in accordance with FAQ 12-0062 Revision 1, UFSAR Content (ADAMS ML121430035) and NEI 98-03, Revision 1, Guidelines for Updating Final Safety Analysis Reports. General reference of this DBD is only intended to reduce unnecessary detail in the UFSAR and direct the reader for additional information and is in no way to be construed as "incorporation by reference" as defined in NEI-98-03, Revision 1.)

On July 16, 2004, the National Fire Protection Association's (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants", 2001 Edition (hereafter referred to as NFPA 805 or the standard) was endorsed by the NRC with exceptions through the issuance of a new subsection (c) to 10CFR50.48, "Fire Protection".

Fire protection requirements predating this amendment were prescriptive in nature as they were established well before the emergence of risk-informed (RI) and performance-based (PB) techniques. As probabilistic risk (safety) assessment (PRA/PSA) technology developed and significant operating insights were gained, more effective and flexible design approaches for achieving fire safety could be justified.

NFPA 805 establishes an overall performance-based framework for fire protection design. The standard established plant-level performance criteria in the areas of nuclear safety (including radioactive release) and provides evaluation methodologies to reasonably ensure the Fire Protection Program (FPP) will achieve those criteria through all modes of plant operation.

The introduction of NFPA 805 for industry adoption was accomplished through a pilot plant process (of which ONS was one of two participants), so that the standard's new methodologies could be tested and the industry could benefit from lessons learned. To that end, the Nuclear Energy Institute (NEI) played a role in facilitating the pilot process through the issuance of NEI 04-02, Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10CFR50.48(c). Additionally, an NFPA 805 Frequently Asked Question (FAQ) process was jointly developed by NEI and NRC to facilitate timely clarifications of NRC positions.

10CFR50.48(c) provides for the voluntary use of the NFPA 805 performance-based approach as an alternative to 10CFR50.48 Section (b) and Appendix R. Oconee Nuclear Station has adopted, through License Amendment Request 2008-01 (April 14, 2010) and NRC Issuance of Amendments and Safety

Evaluation (December 29, 2010), this new fire protection licensing basis which complies with the requirements in 10CFR50.48(c), 10CFR50.48(a) and the guidance provided in Regulatory Guide (RG) 1.205, Revision 1.

9.5.1.1 Design Bases

The design basis of NFPA 805 is comprised of several related elements. As in most engineering applications, the standard maintains the traditional fire safety reliance on layers of defense to increase the margin of safety – the concept of defense-in-depth. Additionally, and particularly for fire safety, both programmatic elements as well as design features are necessary to achieve defense-in-depth. The standard identifies these elements and provides minimum requirements for effectiveness.

Different than past fire protection requirements in the nuclear industry, the design basis provided by this standard provides minimum nuclear safety goals and criteria and for all plant modes and configurations of operation. The standard requires at least one success path for each safety criteria, given a fire in any fire area of the plant. To accomplish this objective, specific methodology is provided by the standard for evaluating each fire area. That methodology includes the allowance of past deterministic, as well as use of risk-informed, performance-based means of compliance.

9.5.1.1.1 Defense-in Depth (DID)

Building on past fire protection design philosophy, the NFPA 805 fire protection standard acknowledges the importance of a defense-in-depth approach for fire safety which is accomplished through a balance of each of three lines of defense:

1. Preventing fires from starting through effective plant controls that closely manage combustible materials and sources of ignition.
2. Rapidly detecting, controlling and extinguishing a fire that might occur, thereby limiting damage.
3. Providing an adequate level of fire protection for structures, systems, and components important to nuclear safety, so that a fire that is not promptly extinguished will not result in a loss of essential plant safety functions necessary to maintain the nuclear fuel in a safe and stable condition.

Keys to this third level of defense are:

- Identification of necessary safety functions for each mode of plant operation.
- Identification of systems and equipment needed to support each safety function.
- Analyzing fire damage effects (including from fire suppression) within each fire area containing the necessary equipment, to determine if a safety function is impeded.
- Determining appropriate improvements in fire safety design when adverse impacts are unacceptable.

9.5.1.1.2 Fundamental Fire Protection Program and Design Elements

NFPA 805 Chapter 3, identifies those required fundamental elements of the Fire Protection Program (FPP) and specifies the minimum requirements for fire protection systems and features that are necessary to provide a fire safety program with defense-in-depth. Section 4.3.1 of NEI 04-02 provides a systematic process for determining the extent to which the plant meets these criteria and for identifying any changes that would be necessary for compliance.

Significant program elements include:

- A documented fire protection design basis document (DBD) capturing elements of the Fire Protection Program.
- Administrative and design controls that limit combustibles and ignition sources.
- General fire safety training for all plant workers
- Establishment of a qualified industrial fire brigade.
- Documented inspections of program elements and corrective action follow through.

Design requirements focus on fire protection equipment and features including:

- Fire detection and alarm systems.
- Fire suppression systems.
- Passive fire protection features such as rated walls, ceilings, floor assemblies, fire doors, fire dampers and through fire barrier penetration seals. Passive fire protection features also include electrical raceway fire barriers.
- Special material applications such as insulation, paints and roofing.
- Nuclear plant equipment with unique risk elements (e.g., RCP motors, bulk hydrogen storage and distribution).

10 CFR 50.48(c) takes exception to three specific requirements of NFPA 805, Chapter 3, and provides alternative requirements as follows:

- Section 3.1 – performance-based compliance methods not allowed: Permits the use of performance-based methods outlined elsewhere in the standard
- Section 3.3.5.3 - electric cable construction: In lieu of installing cables meeting flame propagation tests, a flame-retardant coating may be applied to the electric cables, or an automatic fixed fire suppression system may be installed to provide an equivalent level of protection.
- Section 3.6.4 – manual fire suppression for safe shutdown earthquake areas: The italicized exception to this section is not endorsed. Licensees who wish to use the exception must submit a request for a license amendment.

9.5.1.1.3 Nuclear Safety Goals, Objectives and Performance Criteria

Prior to NFPA 805, fire safety regulation was deterministic, containing prescriptive design criteria. Instead, NFPA 805 provides for a performance-based approach to fire protection:

- by establishing high level performance goals, objectives and criteria that fire protections systems and features must accomplish,
- for a postulated fire in any area of the plant, and
- for all modes of plant operation.

To satisfy these safety goals, NFPA 805 requires the use of a systematic evaluation methodology; the framework for which provides flexibility for achieving performance.

The standard's Nuclear Safety Goal requires reasonable assurance that a fire during any operational mode or plant configuration will not prevent the nuclear fuel from being maintained in a safe and stable condition. Safe and stable conditions are defined for two plant states as:

- With fuel in the reactor vessel and reactor vessel head tensioned, the ability to maintain K_{eff}<0.99 and RCS temperature at or below the requirements for hot standby.
- For all other plant configurations, the ability to maintain K_{eff}<0.99 and fuel coolant temperature below boiling.

Associated Nuclear Safety Performance Objectives ensure the plant will:

- Retain reactivity control to rapidly achieve and/or maintain subcritical conditions.
- Retain fuel cooling capability through decay heat removal and inventory control functions.
- Retain fission product boundary by preventing fuel clad damage.

These performance goals and objectives are translated into Nuclear Safety Performance Criteria (NSPC) that shall be met to ensure the plant is not placed in an unrecoverable condition, in the event of a fire. From Section 1.5.1 of the standard, the NSPC are:

1. **Reactivity control** shall be capable of inserting negative reactivity to achieve and maintain subcritical conditions within a timeframe that prevents exceeding fuel design limits.
2. **Inventory and pressure control** shall be capable of controlling RCS level so that sub-cooling is maintained when fuel is in the reactor vessel and the reactor vessel head is fully tensioned.
3. **Decay heat removal** capability shall be sufficient to remove heat from the reactor core or spent fuel and maintain a safe and stable fuel condition.
4. **Vital auxiliaries** shall be capable of supporting those systems required under (a), (b), (c) and (e) to perform their nuclear safety functions.
5. **Process monitoring** shall be capable of providing the necessary data to ensure criteria (a) through (d) have been achieved and are maintained.

The NRC promulgated the adoption of NFPA 805 in 10CFR50.48(c) and took exception to NFPA 805 Section 1.5.1 (b) and (c) by disallowing the use of feed-and-bleed as the sole method for achieving either of these NSPC.

9.5.1.1.3.1 At Power Analysis

NFPA 805 directs the use of a Nuclear Safety Capability Assessment (NSCA) to demonstrate compliance with the standard's Nuclear Safety Performance Criteria (NSPC) during at power conditions. The NSCA provides two analytical approaches for evaluating compliance, deterministic and performance-based.

In both approaches, the system(s) and equipment required to achieve a given NSPC are identified by success paths. This includes components required to directly fulfill the nuclear safety functions as well as components (equipment and cabling) whose fire-induced failure could jeopardize the operation those components needed to meet the nuclear safety criteria.

Physical plant locations for all identified components are determined on a fire area basis to determine if a success path for all performance criteria exist in a single fire area.

NFPA 805 requires that one success path necessary to achieve and maintain the nuclear safety performance criteria shall be maintained free of fire damage by a single fire. The effects of fire suppression activities on the ability to achieve the nuclear safety performance criteria shall also be evaluated.

Deterministic Analysis

To satisfy performance criteria, one of a set of conservative design methods for achieving fire safety (e.g., encapsulation or separation by distance with/without suppression and detection) must be ensured so that one success path is protected in the fire area where all success paths are potentially challenged. The deterministic approach does not consider likely actual fire scenarios, but conservatively assumes a fire destroys all equipment in that location, except for that protected by a pre-defined conservative design method. Engineering equivalency evaluations or approved exemptions/deviations from earlier license bases that provide an equivalent level of compliance can be utilized in accordance with NEI 04-02.

Recovery actions, as defined by NFPA 805 (activities to achieve the NSPC that take place outside the main control room or primary control station(s)) are not allowed to meet deterministic compliance of the NSPC. These typically were referred to in the industry as operator manual actions.

The deterministic analysis is bounded by the following high-level assumptions on plant conditions as stated in the standard:

1. Independent failures (i.e., failures that are not a direct consequence of fire damage) of systems, equipment, instrumentation, controls, or power supplies relied upon to achieve the nuclear safety performance criteria do not occur before, during, or following the fire. Therefore, contrary to other nuclear power plant design basis events, a concurrent single active failure is not required to be postulated.
2. No abnormal system transients, behavior, or design basis accidents precede the onset of the fire, nor do any of these events, which are not a direct consequence of fire damage, occur during or following the fire.

Additional detailed assumptions regarding deterministic analysis methods are contained in NEI 00-01, Guidance for Post-Fire Safe Shutdown Analysis.

Performance-based (PB) Approach

A performance-based approach provides for the use of engineering analyses to demonstrate one success path is free of fire damage in the fire area where all success paths are potentially challenged. One PB approach involves the use of qualitative judgement that is supported by quantitative analyses such as PRA/PSA. This approach is termed risk-informed, performance-based (RI/PB). Another PB approach utilizes detailed fire modeling to precisely determine the nature and effects of the fire and involves complex analytical methods. Although available as a PB option, fire modeling was not employed at ONS for initial adoption of NFPA 805.

The RI/PB approach begins with a variance-from-deterministic-requirement (VFDR) that specifies what deterministic design features are missing or incomplete. This includes the use of recovery actions that are not allowed for deterministic compliance. A Fire Risk Evaluation (FRE) is conducted that utilizes a Fire PRA model to evaluate overall plant risk changes due to the variation identified. The Fire PRA more closely models the projected fire's frequency and effects, when utilizing additional details on the fire area and variation. If overall risk changes are not acceptable, proposed changes to the variation are reevaluated.

The overall acceptance of the Fire Risk Evaluation is conducted in three parts:

- Quantitatively, using the ΔCDF (Core Damage Frequency) and ΔLERF (Large Early Release Frequency) criteria from RG 1.174, as clarified in RG 1.205 Regulatory Position C.2.2.4. That is, to ensure the variation does not contribute to higher plant risk as compared to full deterministic compliance.
- Qualitatively, by reviewing the impact of the change on defense-in-depth as outlined in NEI 04-02.
- Qualitatively, by reviewing of the impact of the change on safety margin.

9.5.1.1.3.2 Non-Power Operations (NPO) Analysis

The nuclear safety goal of NFPA 805 requires evaluation of fire effects during any mode and plant configuration, including non-power operations (NPO) (Modes 3 and below). Because of the significant body of risk-informed guidance on non-power operation, the evaluation uses a risk-informed approach that addresses the effects of fire during previously defined, high risk NPO configurations (in accordance with FAQ 07-0040).

NPO involves a number of plant configurations within each Technical Specification mode. Numerous studies have been conducted to characterize the risk associated with these configurations, including using Core Damage Frequency (CDF) as a risk metric. While many NPO configurations, termed Plant Operating States (POSs) are of relatively low risk, the following POSs represent the highest risk and are evaluated for NFPA 805 regarding NPO:

- POS 1B (when SG heat removal unavailable) - The RCS is closed and RHR is in service. This POS ends when the RCS is vented to preclude SG heat removal. Includes Mode 4 (hot shutdown) and portions of Mode 5 (cold shutdown).
- POS2 - This POS starts when the RCS is vented such that a sufficient vent path exists for feed-and-bleed. This POS includes portions of Mode 5 (cold shutdown) and Mode 6 (refueling). Reduced inventory operations and mid-loop operations with a vented RCS are subsets of this POS.
- POS3 - This POS represents the shutdown condition when the refueling cavity water level is at or above the minimum level required for movement of irradiated fuel assemblies within containment (Mode 6).

The systems and equipment necessary to achieve performance criteria for NPO have previously been established through the development of the Key Safety Function (KSF) concept outlined in NUMARC 91-06 and adopted in plant shutdown risk programs. For the purpose of evaluating the effectiveness of the fire protection program during NPO, the following KSFs apply:

- Decay Heat Removal
- RCS Inventory Control,
- Reactivity Control, and
- Power Availability, including support functions.

Containment Control is not a selected KSF (with regards to fire safety) since it does not directly support the nuclear safety goal of NFPA 805. However, maintaining a high confidence capability to close the equipment hatch prior to a release that would exceed the NFPA 805 radiological release criteria is deemed important.

All systems and equipment necessary to support each KSF are identified. Since each KSF may have several methods for success, equipment is categorized into success paths. KSF success paths for each evaluated POS are determined and their respective physical plant locations identified. Fire Area Assessments (by fire zone) are conducted to determine locations where some or all success paths of a credited KSF could be damaged by a fire. When all success paths for one or more KSFs are postulated lost in a given fire zone, the location is considered a “pinch point” (IAW FAQ 07-0040).

Fire zones with KSF success path impacts are categorized based on fire risk vulnerability and changes are implemented to fire risk and outage management procedures and processes. In addition, during High Risk Evolutions (HREs) as defined in NUMARC 91-06, additional fire protection and prevention measures are evaluated. At ONS the specific HREs of concern are referred to as Higher Risk Plant Operating States (HRPOS).

9.5.1.1.4 Radioactive Release Goal, Objectives and Criteria

NFPA 805 establishes a high level Radioactive Release Goal that provides reasonable assurance a fire will not result in a radiological release that adversely effects the public, plant personnel or the environment. The standard's objective for this goal is to ensure that during all operational modes and plant configurations either:

- Containment integrity is capable of being maintained, or
- The source term is capable of being limited.

The standard's Nuclear Safety Goal, Objectives and Performance Criteria require the prevention of fuel cladding damage. This is accomplished through the Nuclear Safety Capability Assessment and may be met either deterministically or by a risk-informed, performance-based approach via measure of CDF and LERF (along with defense-in-depth and safety margin considerations). Therefore, a separate examination of radioactive release from fuel cladding damage is not required. This effectively limits the source of radiation (source term). As a result, specific examination of containment integrity, including containment isolation valves is not required to be included in the scope of the NSCA evaluation. (Reference NEI 04-02 and FAQ 09-0056).

The Radioactive Release Performance Criteria states:

Radiation release to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) shall be as low as reasonably achievable (ALARA) and shall not exceed applicable 10 CFR Part 20 limits.

Fuel clad damage is addressed through the Nuclear Safety Performance Criteria; therefore, radiological release evaluation only involves releases to unrestricted areas that are due to the direct effects of fire suppression activities and must address the loss of boundary and/or source term control in contaminated spaces.

NEI 04-02 and FAQ 09-0056 direct the use of fire pre-plans for this evaluation. Fire pre-plans are developed and maintained for Fire Brigade use on a fire zone basis. These NFPA 805 required plans, detail fire area boundaries, fire hazards, major nuclear safety components, radiation hazards and fire protection systems that are present. Where fire areas potentially contain radioactive or contaminated materials, associated pre-plans are evaluated. Where prudent and necessary, these plans are modified for effective control in monitoring and containing potentially contaminated smoke and fire suppression material.

9.5.1.2 System Description

9.5.1.2.1 Power Block

NFPA 805 utilizes both terms "Power Block" and "Plant" to delineate those structures or plant areas that contain equipment required for safe and reliable nuclear power operation. Examples include Reactor, Auxiliary and Turbine Buildings and SSF. A detailed list is contained in the Design Basis Specification for Fire Protection (OSS-0254.00-00-4008).

The Power Block defines those areas of the plant that are required to be included in the Fire Protection Program to ensure that NFPA 805 nuclear safety and radioactive release goals are met.

9.5.1.2.2 Nuclear Safety Capability Systems and Equipment

A comprehensive list of systems and equipment are identified that are required to achieve the Nuclear Safety Performance Criteria, generally in accordance with NEI 00-01. The list shall include components needed to achieve and maintain the nuclear safety functions as well as cables and related equipment, whose fire-induced failure (e.g., hot shorts, open circuits, shorts to ground) may defeat (i.e., flow diversion, degraded cooling) NSPC-required equipment from achieving their nuclear safety function.

The Design Basis Specification for Fire Protection (OSS-0254.00-00-4008) identifies these systems and equipment.

9.5.1.2.3 Fire Protection Systems and Features

Fire protection systems and features are those systems, components and passive features that are credited with mitigating the effects of a fire on equipment necessary to achieve the nuclear safety and radioactive release performance criteria.

A majority of these systems and features are identified and detailed in the Design Basis Specification for Fire Protection (OSS-0254.00-00-4008).

9.5.1.2.3.1 Codes of Record

The primary code of record is NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition. Chapter 6, Referenced Publications, of this standard contains a comprehensive list of NFPA publications, industry codes and guides and was utilized in code compliance reviews that were performed for those elements requiring transition to NFPA 805. The applicable codes and standards are identified in the Design Basis Specification for Fire Protection (OSS-0254.00-00-4008).

9.5.1.3 Safety Evaluation

On December 29, 2010, the NRC granted License Amendments (Numbers 371, 373 and 372) to Renewed Facility Operating Licenses DPR-38, DPR-47 and DPR-55 for the Oconee Nuclear Station, Units 1, 2, and 3, respectively and the approved Safety Evaluation. The license amendments consist of changes to the licenses and technical specifications in response to License Amendment Request (LAR) dated April 14, 2010 to allow for the transition of the fire protection program in accordance with 10 CFR 50.48(c). The following sections describe salient aspects for the NRC Safety Evaluation relative to NFPA 805 implementation.

9.5.1.3.1 Fundamental Fire Protection Program and Design Elements

The Safety Evaluation concluded that a systematic approach described in NEI 04-02 Revision 2 (and endorsed in Regulatory Guide 1.205, Revision 1) was utilized to assess Chapter 3 requirements. Documentation was provided that adequately substantiated deterministic compliance for each requirement (element).

For eleven (11) elements, a performance-based approach was used to demonstrate compliance. In accordance with the regulations, specific approval was requested in the LAR and received. The following summarizes each approved variance:

- Wood utilized in temporary concrete forming applications is not required to be fire-retardant.
- Flammable gas cylinder storage is acceptable for four in-plant locations: The Chemistry Labs located on the 796’ elevation of the Unit 2 and 3 Auxiliary Building and at the Post-Accident Monitoring instrumentation located on the 838’ elevation of the Auxiliary Building in the Units 1/2 air handling unit (AHU) and Unit 3 AHU rooms.
- Existing wiring above suspended ceilings in the Control Room, Control Room lobbies, Turbine Building office areas, Auxiliary Building stair and elevator lobbies, Auxiliary Building office

areas, Auxiliary Building change areas and 838’ elevation Auxiliary Building corridor is acceptable.

- Existing low voltage video/communication/data cables are acceptable for continued use.
- The lack of an oil mist collection system for the RCP motors is acceptable.
- Circulation relief valves are not required on the HPSW and Keowee fire pumps.
- The use of LPSW to supply the RB hose stations/standpipes is acceptable.
- The automatic stop function on ONS HPSW pumps (fire water) and the Keowee fire pump, as well as the remote operation of the HPSW pumps from the control room are acceptable.
- A fire main loop and fire hydrants are not required for Keowee Hydro Station.
- The HPSW and Keowee Service Water systems, while each having dual purposes, are acceptable as the fire protection water supply systems.
- The single, fixed motor-driven fire pump for Keowee is acceptable.

9.5.1.3.2 Nuclear Safety at Power

Nuclear Safety Capability Assessment (NSCA) Methods

The LAR submittal identified differences from the methodology required to conduct the NSCA as provided in NFPA 805, NEI 00-01 and associated documents. The Safety Evaluation reviewed those differences as well as compliance for all aspects of the NSCA methods utilized to select NSCA equipment and to ensure capability to achieve performance criteria given a fire in each area. The following are the salient issues from this part of the Safety Evaluation.

Timing of Fire-induced Failures

As part of the LAR, the NRC was requested to formally document as a prior approval (as allowed in RG 1.205, Section 2.2.1) recognition that during the 10 minutes required to activate the SSF, fire growth will not have reached a point where fire damage will preclude operator actions from the Control Room, nor will any spurious operations or loss of offsite power conditions occur within the first 10 minutes following identification of a confirmed active fire.

In supplementary information, ONS agreed to eliminate the 10-minute free of fire damage assumption and to perform a risk-informed evaluation of scenarios that previously relied upon that. This evaluation utilized a sensitivity study to determine that cumulative delta risk from these scenarios was within the available Protected Service Water (PSW) risk offset margin for all fire areas where the SSF is credited (i.e., Auxiliary Building Fire Area). Based on this evaluation, the elimination of the 10-minute free of damage assumption is acceptable. The PSW modification is carried as an implementation item with respect to the NFPA 805 transition.

72-hour Coping Duration

The NFPA 805 standard does not explicitly define a time period for which a safe and stable condition should be evaluated in the deterministic approach. Regardless of the chosen safe end state (i.e., hot standby, hot shutdown, or cold shutdown) it is recognized that mitigation systems are typically designed for defined mission times due to ultimate system limitations. Therefore, even a deterministic approach cannot be defined by a final plant state but must incorporate the concept of a reasonable duration that provides for the completion of additional recovery actions.

The LAR determined that the nuclear safety goal of NFPA 805 was accomplished by identifying systems and equipment required to achieve and maintain hot standby for 72 hours, following a fire from at-power conditions. The need to maintain a unit in hot standby for greater than 72 hours is not anticipated, as assessment and repair activities would commence soon after stabilization at hot standby to provide for RCS cool-down in a controlled manner.

For the most limiting fire scenarios, an extended cool-down past hot standby would approach 250°F in the RCS (hot shutdown) and provide an extended coping period due to ample system capabilities in this plant configuration.

The 72 hour coping period was determined to provide reasonable assurance for completion of necessary response actions and was accepted by the NRC. A predetermined strategy with supporting procedures and repair equipment to aid in the transition to long-term decay heat removal are in place.

Current Transformer (CT) Circuit Analysis

The LAR stated that ONS does not align with guidance contained in Section 3.5.2.1 of NEI 00-01 because it disagreed that an open circuit in the secondary winding circuit of a CT could cause a secondary fire. ONS later revised the position to align with NFPA 805 and updated the NSCA to include all CT cables and ensure such failures will not impact the ability to achieve the NSPC.

Offsite Power

ONS credits the power supplies from Keowee Hydro Station (KHS) and the SSF Diesel Generator, and neither requires offsite power to function. Also, the adverse consequences of offsite power being available are considered in the NSCA. Appropriate analyses were revised to reword the alignment basis and clearly state that offsite power is not credited for the deterministic analysis and therefore not analyzed for its availability in the deterministic analysis. Offsite power, however is assumed to be available for the analysis of spurious component actuations.

Applicability of Feed-and-Bleed

10CFR 50.48(c)(2)(iii) limits the use of feed and bleed to achieve the NSPC. Specifically:

In demonstrating compliance with the performance criteria of Sections 1.5.1 (b) and (c), a high-pressure charging/injection pump coupled with the pressurizer power-operated relief valves (PORVs) as the sole fire-protected SSD path for maintaining reactor coolant inventory, pressure control, and decay heat removal capability (i.e., feed-and-bleed) for pressurized-water reactors (PWRs) is not permitted.

Although loss of pressurizer heaters was considered possible, the SE confirmed that all fire area analyses include the SSD equipment necessary to provide decay heat removal without relying on feed-and-bleed and the PORV is not the only means of pressure control for potential solid water operation.

Multiple Spurious Operations (MSOs)

NFPA 805, Section 2.4.2.2.1, Circuits Required in Nuclear Safety Functions requires that circuits be identified where fire-induced failures could prevent the proper operation of other systems and equipment that are necessary to achieve nuclear safety performance criteria.

ONS utilized an expert panel to identify potential MSO combinations, which needed to be considered in the NSCA, as well as to assess the plant-specific vulnerabilities associated with these MSO combinations. The expert panel utilized NRC and NEI guidance on expert panels.

Additionally, the list of fire-induced MSO scenarios provided by the PWR Owners Group were compared and considered. The results of this effort were incorporated into the NSCA and Fire PRA.

Transition of Operator Manual Actions to Recovery Actions

During resolution of VFDRs, ONS identified recovery actions to satisfy the defense-in-depth requirements of NFPA 805. These recovery actions were subjected to a feasibility review in accordance with NEI 04-02 guidance. Based on that review, the following actions will be taken:

- Hand-held radios relied upon for communication will be verified to operate in the locations of the recovery actions.
- Procedures do not exist for non-SSF shutdown fire areas. New SSD procedures will be developed for Fire Areas RB1, RB2 and RB3.
- Training will be provided to operators on these new SSD procedures.
- Drills will be conducted on the recovery actions.

Completion of these items is an implementation item.

In response to RAIs, ONS defines primary control stations (PCS) as:

- Actions inside the Main Control Rooms (MCR),
- Actions inside the SSF control room,
- Actions inside the SSF facility to transfer control from the MCR to the SSF,
- Actions inside the SSF facility to operate manual valves.

Fire Modeling

NFPA 805 allows for the use of fire modeling as a performance-based alternative. ONS only used fire modeling to support development of the Fire PRA for use in performing FRE's. The fire modeling approach of NFPA 805, Section 4.2.4.1 was not used.

Nuclear Safety Capability Analysis (NSCA) Results

This section of the Safety Evaluation addresses the results of the NSCA. The following salient issues are discussed.

ONS is a three-unit plant and is divided into 15 fire areas. For its transition to NFPA 805, the previous Balance-of-Plant Fire Area (BOP) was split into two new fire areas, AB (Auxiliary Building) and TB (Turbine Building). The NSCA was conducted on a fire area basis for each of those fire areas. For each fire area, the following were documented:

- The approach used – deterministic or performance-based.
- The SSCs required for achieving the nuclear safety performance criteria (NSPC).
- An evaluation of the effects of fire suppression activities on achieving the NSPC.
- The disposition of each VFDR using either modifications (completed or committed) or the performance of a Fire Risk Evaluation (In accordance with NFPA 805, Section 4.2.4.2).

For a fire in the following Fire Areas:

- Unit 1 and 2, Unit 3 Block Houses (BH1&2, BH3),
- Reactor Buildings (RB1, RB2, RB3),
- Standby Shutdown Facility (SSF),
- Turbine Building (TB),
- West Penetration Rooms (WP1, WP2, WP3), and
- Yard Areas (YARD),

Safe and stable plant conditions are achieved utilizing the PSW pump and credited systems powered from the PSW power supply with control from the Main Control Rooms.

For a fire in Fire Areas CT-4 Block House (CT-4), Keowee Hydro Station (KEO), or PSW Building (PSW), normal shutdown systems are not affected and the plant will be shutdown, if desired using normal systems and operating procedures.

For a fire in the Fire Area AB the SSF will perform as a dedicated shutdown facility used to establish safe and stable plant conditions.

FREs identified the need to improve general area and/or hazard detection in several fire zones, either to support assumptions made in the Fire PRA or to provide defense in depth. ONS will upgrade and/or install new automatic fire detection systems in many fire zones throughout the plant in accordance with NFPA 72, National Fire Alarm Code*,* as required by NFPA 805.

Each fire area was summarized for VFDRs, their resolution, and applicable Licensing Actions including required modifications for performance-based compliance. Refer to attachments to the ONS Design Basis Specification for Fire Protection (OSS-0254.00-00-4008).

9.5.1.3.3 Nuclear Safety at Non-Power Operation (NPO)

The same basic deterministic methodology utilized for the nuclear safety capability analysis was utilized when assessing fire risk during NPO modes. One difference involved the use of fire zones instead of the larger fire areas as the assessed space.

The NPO analysis was conducted and documented to identify "pinch points", defined as where all of the credited success paths for a given KSF are lost in a fire, in accordance with FAQ 07-0040. Where degraded KSF functions were identified, additional fire protection elements were recommended.

Due to a lack of rated fire barriers between all fire zones, PRA insights were used to determine if a fire was expected to spread from one zone to another. Additional fire prevention recommendations were made for the identified zones where compartment-to-compartment interactions could potentially take place. Fire modeling was not used to eliminate any fire zone from being a 'pinch point.' The fire zone analysis resulted in the following grouping of results and recommendations:

- About 25% of fire zones were found to have an adequate number of KSF success paths so that all KSFs are preserved. This included accounting for multi-compartment analysis involving adjacent compartments. Limited recommendations were made for these zones.
- About 40% of fire zones were found to involve loss of all success paths for one or more KSFs for one unit due to a fire in that fire zone. These KSF success paths were preserved through recommended use of fire protection program controls. Several of these zones have barrier separation from adjacent compartments; however, where applicable, multi-compartment analysis was used to ensure conservative treatment of adjacent compartments.
- About 25% of fire zones were found to involve the loss of a given KSF in 2 or more of the 3 units, due to equipment credited for multiple units (e.g., power supply, support systems). These zones include locations such as the main control rooms and cable/equipment rooms. With the exception of the control rooms which are constantly manned, these zones have been recommended for limitation of hot work and transient combustible storage during HREs/HRPOS, as well as other Fire Protection Program controls. Many of these zones have sufficient barrier separation from adjacent compartments to preclude the need for multi-compartment analysis considerations; however, where applicable, multi-compartment analysis was used to ensure conservative treatment of adjacent compartments.
- The remaining fire zones were found to involve loss of all KSFs for all three units, primarily due to impacts on the main power feeder and/or standby buses located in these zones. In addition to the recommendations for fire zones as described above, recommendations for more conservative fire protection program controls were provided. These fire zones have sufficient barrier

separation from adjacent compartments to preclude the need for multi-compartment analysis considerations.

In accordance with the method endorsed in NEI 04-02 and FAQ 07-0040, the primary mechanism being used to meet the nuclear safety performance criteria during NPO conditions is through the expanded utilization of fire protection defense in depth elements that reduce the risk of fire. Examples include use of additional fire watch patrols and establishing more specific controls to govern ignition sources and combustibles. During HREs/HRPOS, risk reduction is achieved by further enhancing defense in depth actions to reduce the frequency, severity or impact of fires so that key pinch points are protected. Examples of enhanced defense in depth actions include:

- Prohibition or limitation of hot work in vulnerable fire zones.
- Provision for continuous fire patrols or other surveillance means.
- Required rescheduling of work to a period of lower risk.

9.5.1.3.4 Radioactive Release

Compliance with radioactive release performance criteria was verified through review of all ONS pre-fire plans and fire brigade training materials. Guidance for performing this review is contained in NEI 04-02, Section 4.3 and Appendix B. The docketed NEI response to FAQ 09-0056 was also utilized.

For pre-fire plans, plant locations that had potential for radioactive release were updated with specific guidance for containing and monitoring potentially contaminated smoke and fire suppression water. Specified engineering and procedural controls were reviewed and verified effective. Additionally, a new fire brigade Standard Operating Guideline (SOG) was developed to address potentially contaminated areas that are established on a short term basis, such as for outages or maintenance evolutions.

Fire brigade training materials were reviewed and verified consistent with pre-fire plans in terms of containment and monitoring of smoke and fire suppression water run-off. Also, the new Standard Operating Guideline was integrated into the fire brigade training program.

Because ONS fire suppression activities, as defined in pre-fire plans and fire brigade operating guidelines, are developed for any plant operating mode, the NFPA 805 radioactive release performance objective of limiting the source term during all operational modes is met.

NFPA 805 specified performance criteria for radioactive release due to fire suppression references applicable 10CFR Part 20, Limits. Oconee's prior licensed performance criteria for liquid effluent release was approved as 10 times that of 10 CFR Part 20, Appendix B, Table 2, Column 2 (by NRC Safety Evaluation dated January 6, 1993). This criteria was accepted as equivalent to the NFPA 805 performance criteria.

9.5.1.3.5 Fire PRA

RI-PB evaluations are an integral element of an NFPA 805 fire protection program. Key components of RI-PB evaluations include the use of a Fire PRA, fire modeling and the Fire Risk Evaluation (FRE) Process.

The ONS Fire PRA model was developed using the guidance provided by NUREG/CR-6850/EPRI TR-1011989 (including FAQ 08-00148) and reviewed against the requirements of Part 3 "Internal Fires at Power Probabilistic Risk Assessment Requirements," of the ASME and ANS combined PRA Standard, ASME/ANS RA-S-008(draft), hereafter referred to as Fire PRA Standard.

The ONS base internal events PRA was the starting point for the Fire PRA. This PRA has undergone a self-assessment by a team of Duke Energy PRA personnel, an independent peer review of model, data and documentation and an assessment of impacts on the NFPA 805 program. The internal events PRA was modified to capture the effects of fire as an initiator of an event and as the potential failure mode for affected circuits or individual targets.

The Fire PRA is a unit-specific analysis and determines the risk impact from various fire events from the perspective of a single unit, even though another unit may be credited with providing a support function (with consideration that the support function may not be available as in a unit outage). Additionally, as a spatial analysis, the Fire PRA considers differences in equipment locations and cable routing. For example, a Unit 3 fire compartment or zone may be risk significant in the Unit 3 Fire PRA, but the same compartment may have an insignificant impact on the fire risk for Unit 1 or 2.

The Unit 3 Fire PRA was developed based on the fault tree and basic event database associated with the Full Power Internal Events (FPIE) PRA model, which was actually a Unit 3 PRA model. Accordingly, the associated documentation developed for the Fire PRA was written primarily from a Unit 3 perspective.

A fault tree specific to Unit 2 was developed but was limited to assessing fire risk. The basic event mapping (to components) process was performed consistent with the process followed for Unit 3. The component footprint (i.e., cable location) necessary for performing a compartment analysis from a Unit 2 perspective was already included in the utilized database. Similar component footprint information was later added to the database for Unit 1; however, a separate fault tree for that unit was not developed. Following completion of the cable selection process for Unit 1 and the attendant database update, a comparative analysis of the failures and ignition frequencies for comparable fire compartments between Units 1 and 2 was performed. The comparative screening analysis indicated that a separate Unit 1 fault tree and Fire PRA quantification file were not necessary. For the limited number of cases where the Unit 2 results were not considered to be bounding, a method for adjustment of the Unit 2 results for application to Unit 1 was provided.

Fire modeling was utilized as part of the Fire PRA development to support Fire Risk Evaluations (NFPA 805 Section 4.2.4.2). The approach taken by ONS to simplify the fire modeling incorporated several fire model tools covered by NUREG 1824, as well as additional features. The approach was collectively referred to as the Fire Modeling Generic Treatments. Acceptable levels of verification and validation for these approaches were conducted.

ONS was a pilot plant for 10 CFR 50.48(c); therefore, as described in RG 1.205, an assessment was performed by NRC personnel between the ONS draft PRA and the requirements of the draft Fire PRA Standard. This was in lieu of a review by independent industry analysts normally required by RG 1.200, prior to a risk-informed submittal. ONS was later requested to have an independent Industry Peer review of the Fire PRA conducted against the ASME/ANS RA-SA-2009 PRA Standard.

Fire PRA Overall Risk Insights

Risk insights were documented as part of the development of the Fire PRA. The total plant fire CDF/LERF was derived using the NUREG/CR-6850 methodology for Fire PRA development and was useful in identifying the areas of the plant where fire risk is greatest and where specific contributors might be mitigated via modification.

Risk Change Due to NFPA 805 Transition

In accordance with the guidance in RG 1.205, risk increases or decreases for each fire area using Fire Risk Evaluations and the overall plant should be provided. RG 1.205 states in part:

Any increase or decrease in risk (both in terms of core damage frequency (CDF) and large early release frequency (LERF)) should be evaluated and provided for each fire area that uses a fire risk evaluation.

Additionally, RG 1.205 states in part:

The total increase or decrease in risk associated with the implementation of NFPA 805 for the overall plant should be calculated by summing the risk increases and decreases for each fire area. The total risk increase should be consistent with the acceptance guidelines in Regulatory Guide 1.174.

The risk increase (CDF/LERF) associated with all VFDRs resulted in a collective risk increase (CDF/LERF) above the threshold for self-approval.

As allowed by RG 1.205, risk credits (decreases) associated with non-fire related modifications that affect the Fire PRA results have been calculated and more than offset the NFPA 805 transition risk increase.

9.5.1.3.6 Summary of NFPA 805 Compliance

Compliance with the requirements of NFPA 805 was achieved through the appropriate application of the standard's methodologies, completion of required reviews and analyses and assessment of results against performance criteria, including determined risk levels.

Major areas of compliance include verification of Chapter 3 fundamental fire protection elements and minimum design requirements, verification of Chapters 2 and 4 assessment of nuclear safety capability for power and NPO plant states, development and use of fire PRA and associated risk evaluation methods for determining acceptable risk impacts changes, and development of an effective performance-based monitoring program that continues to verify underlying assumptions regarding risk.

9.5.1.4 Inspection and Testing

9.5.1.4.1 Inspection, Testing and Surveillance Methodology

Inspection and testing procedures are utilized to monitor the current effectiveness of fire protection systems, features and programs. The corrective action process will be utilized when unanalyzed fire hazards are identified.

9.5.1.4.2 Monitoring Program Methodology

NFPA 805, Section 2.6 establishes a monitoring programmatic requirement that ensures the availability and reliability of fire protection systems and features are maintained and the Fire Protection Program retains effective performance.

All FPP systems, structures and components (SSCs) and programmatic elements credited in the NSCA, Fire PRAs or other NFPA 805 supporting analyses to reduce fire risk are monitored for performance. In cases where a monitored function is performed by a number of individual components for a given fire area, Performance Monitoring Groups (PMGs) are utilized that monitor performance at the functional (versus component) level. Systems and equipment, other than fire protection components, and credited with achieving the NSPC of NFPA 805 are monitored as part of the Maintenance Rule (10CFR 50.65).

ONS has utilized the screening thresholds of FAQ 10-059 and based on the results of the Fire PRA identified high safety significant fire protection elements. As defined by these thresholds, high safety significant fire zones (and all PMGs within) and high safety significant fire protection components that are amenable to risk measurement will be monitored as described in FAQ 10-059.

ONS has established criteria for acceptable levels of availability, reliability, and performance, or appropriate action levels, for each PMG, with the intent to establish conservative values of availability and reliability, such that assumptions made in the applicable supporting analyses are bounded. Target and action levels for availability will be primarily based on site-specific data reflecting expected out-of-service times to support maintenance and inspection activities. Planned impairments, with appropriate functional compensatory measures, will not be assessed against availability criteria. Target and action values for reliability will be based primarily on industry guidance in EPRI Technical Report (TR)

1006756, Fire Protection Surveillance Optimization and Maintenance Guide for Fire Protection Systems and Features, with adjustments to reflect site-specific operating experience, Fire PRA assumptions, and equipment types. Availability and reliability targets for NFPA 805 monitored SSCs will be selected, reviewed, and maintained to ensure that the assumptions of the applicable supporting analyses (e.g., Fire PRA, NSCA, etc.) remain valid. Performance of programmatic elements such as fire brigade, fire-watches, and combustible controls will be evaluated using the existing ONS plant health process.

When monitoring programs reveal established levels of performance are not met, the corrective action program shall be utilized appropriately.

9.5.1.5 Personnel Qualification and Training

All plant employees and contractors shall be trained on fire safety principles including familiarization with plant fire prevention procedures, fire reporting responsibilities and plant emergency alarms.

Plant personnel that are required to respond with the fire brigade but that are not part of the fire brigade are trained on their responsibilities, potential hazards and interface requirements with the fire brigade.

9.5.1.5.1 Fire Brigade

An industrial fire brigade (in accordance with NFPA 805) is required, including dedicated members for each shift, effective training and drills, and all necessary equipment.

Fire brigade members are required to perform the strenuous activities associated with manual fire-fighting, including the use of portable respiratory protection equipment. Annual physical examination is required to ensure fire brigade members can satisfy these requirements.

The fire brigade leader and at least two brigade members shall have sufficient training and knowledge of nuclear safety systems to understand the effects of fire and fire suppressants on nuclear safety performance criteria. This requirement may be alternately met through the dedicated assignment of a Shift Technical Advisor to fire brigade support.

Fire brigade members receive training consistent with applicable NFPA standards and receive quarterly training and practice in fire-fighting, including radiation protection considerations. A written program details and documents the fire brigade training program, including records of member training.

9.5.2 Instrument and Breathing Air Systems

9.5.2.1 Design Basis

The Instrument and Breathing Air Systems are designed to provide clean, dry, oil free instrument air to all air operated instrumentation and valves. Instrument air is supplied to ANSI/ISA-7.0.01-1996 standards, and breathing air is supplied at ANSI Z86.1 Grade D standards to minimize personnel exposure in areas of airborne contamination.

9.5.2.2 System Description

The Instrument Air (IA) System consists of a) one primary IA compressor with two filter/dryer trains, b) three backup IA compressors with two filter/dryer trains, c) distribution headers, d) receiver tanks and e) components supply lines. The IA System is shared by all three Oconee Units and the Radwaste Facility; therefore, the IA System is required to operate continuously.

Normal operation for the IA System is for the primary IA compressor to supply all IA demands. Should the primary IA compressor trip, be required to be removed from service for maintenance, or the IA

System demand exceed the primary IA compressor capacity, the backup IA compressors and any available Service Air System compressor capacity reserves are used in supplying IA System demands.

An Auxiliary Instrument Air (AIA) System provides a backup auxiliary source of instrument air to key plant components in order to minimize operator burden during a normal loss of IA event while reaching and maintaining a safe shutdown. This system is composed of three (one per unit) compressors, combination filters, and desiccant dryers. Separate distribution headers and supply lines are provided to these key components to ensure AIA availability. The AIA System is designed such that a failure will not fail Instrument Air or affect operating equipment.

Although the AIA System may be available, it is not required for performing or supporting any operation. Each of the key plant components supplied backup AIA fails in a safe condition and has an alternate procedurally controlled method to control the process.

The Unit 1 and 2 Breathing Air System and the Unit 3 Breathing Air System each consist of one primary and one backup compressor package. These packages consist of one a) two stage inlet air filter, b) compressor, c) air/oil separator, d) and oil cooler/aftercooler. After the compressor the air is passed through a) an air/water separator, b) a filter package, c) two purification packages in parallel (Unit 3 'A' train has only one purifier package), d) into two parallel receiver tanks, and e) finally into the breathing air manifolds. Breathing air is supplied to all areas and elevations by headers and individual supply stations where the pressure is regulated for personnel use. Units 1 & 2 have one primary and one backup compressor total for both Units, and Unit 3 has one primary and one backup for its use. The breathing air systems are cross connected in such a way that any of the compressors can supply either of the Units' breathing air needs.

9.5.2.3 Instrument Air (IA) System Tests and Inspections

The Instrument Air System is always in service, supporting the continued operation of three nuclear units, and therefore continually demonstrates it is capable of performing its intended function. The Primary IA components are normally in service, the backup IA components are placed in service when PM's are performed on the Primary IA Components. The air supplied by the Instrument Air System is periodically tested to verify that the quality of the air is acceptable (e.g., acceptable dew point, oil content, and particulate contamination). Safety related air-operated valves are periodically tested to ensure that they fail in their safe position during a loss of Instrument Air event. The Instrument Air Preventive Maintenance Program and ongoing monitoring of the system operation also help ensure adequate system performance. All of these tests, inspections and programs ensure that the Instrument Air System is capable of performing its intended function and will not adversely affect safety-related demand equipment in accordance with Generic Letter 88-14, "Instrument Air Supply System Problems affecting Safety-Related Equipment."

THIS IS THE LAST PAGE OF THE TEXT SECTION 9.5.

THIS PAGE LEFT BLANK INTENTIONALLY.

9.6 Standby Shutdown Facility

9.6.1 General Description

The Standby Shutdown Facility (SSF) houses stand-alone systems that are designed to maintain the plant in a safe and stable condition following postulated emergency events that are distinct from the design basis accidents for which the plant systems were originally designed. The system provides additional "defense in-depth" protection for the health and safety of the public by serving as a backup to existing safety systems. As such, the SSF provides an alternate means to achieve and maintain mode 3 with an average Reactor Coolant temperature ≥ 525°F (RCS cold leg temperature ≤ 555°F and RCS pressure $\simeq$ 2155 psig) following postulated fire, sabotage, or flooding events, and is designed in accordance with criteria associated with these events.

Loss of all other station power does not impact the SSF's capability to mitigate each event. The SSF is also credited as the alternate AC (AAC) power source and the source of decay heat removal required to demonstrate safe shutdown during the required station blackout coping duration.

SSF-designated events are not postulated to simultaneously occur with standard design basis events such as an earthquake or LOCA; therefore, the single failure criterion is not applicable or required. However, SSF systems are required to be designed such that failures or inadvertent operation of existing plant systems will not be caused by these SSF systems. The SSF requires manual activation that would occur under adverse fire, flooding or sabotage events when normal plant systems may have been damaged or have become unavailable.

The SSF is designed to:

1. Maintain a minimum water level above the reactor core, with an intact Reactor Coolant System, and maintain Reactor Coolant Pump Seal cooling.
2. Assure natural circulation and core cooling by maintaining the primary coolant system filled to a sufficient level in the pressurizer while maintaining sufficient secondary side cooling water.
3. Transfer decay heat from the fuel to an ultimate heat sink.
4. Maintain the reactor 1% shutdown with the most reactive rod stuck fully withdrawn, after all normal sources of RCS makeup have become unavailable, by providing makeup via the Reactor Coolant Makeup Pump System which always supplies makeup of a sufficient boron concentration.

The SSF consists of the following:

1. SSF Structure
2. SSF Reactor Coolant Makeup (RCM) System
3. SSF Auxiliary Service Water (ASW) System
4. SSF Electrical Power
5. SSF Support Systems

System Main Components are listed in Table 9-14. SSF Primary Valves are listed in Table 9-15. SSF Instrumentation is listed in Table 9-16.

9.6.2 Design Bases

FIRE PROTECTION CRITERIA

Oconee transitioned to NFPA 805, Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants, 2001 Edition, in accordance with 10CFR50.48(c). NFPA 805 establishes a nuclear safety goal that requires reasonable assurance that a fire during any operational mode or plant configuration will not prevent the plant from being maintained in a safe and stable condition. Safe and stable is defined as maintaining K_{eff}<0.99 with the RCS at or below the requirements for hot standby.

To accomplish this goal, fire protection systems and features must be capable of ensuring at least one success path of equipment remains free of fire damage following a fire in a single fire area. For one fire area of Oconee, the SSF provides the single success path necessary to achieve the NFPA 805 nuclear safety goal.

The nuclear safety goal of NFPA 805 does not prescribe a transition to cold shutdown within 72 hours following a fire; rather, only that the plant be maintained safe and stable in hot standby for up to a 72 hour coping duration while repairs are made to achieve a licensed end state of hot shutdown.

The hypothesized fire is to be considered an "event", and thus need not be postulated concurrent with non-fire-related failures in safety systems, other plant accidents, or the most severe natural phenomena. This includes not postulating the most reactive control rod being stuck in the fully withdrawn position during a fire event.

Deleted Paragraph(s) per 2012 update.

TURBINE BUILDING FLOOD CRITERIA

Components of the SSF systems and the associated structures are designed to achieve and maintain mode 3 with an average Reactor Coolant temperature ≥ 525°F (RCS cold leg temperature ≤ 555°F and RCS pressure $\simeq$ 2155 psig) in the event the Turbine Building is subjected to internal flooding.

The turbine building flood began as an event which was not considered to occur with any other plant events (i.e. seismic). The NRC reviewed and approved the design of the SSF to mitigate fire, turbine building flood, and sabotage.

During the seismic qualification review of the Oconee EFW system in the 1980s, the turbine building flood event was redefined as being the result of a seismic event. It was concluded that the EFW system would not be available following a maximum hypothetical earthquake since the turbine building would be flooded and the EFW pumps would be inundated with water and considered unavailable. As a result, the NRC reviewed the Oconee design to determine if adequate decay heat removal would be available following the maximum hypothetical earthquake and a concurrent single active failure. The requirement for a concurrent single active failure was viewed as a potential backfit issue. The NRC reviewed the need to require the SSF to be single failure proof and determined that additional modifications to the SSF were not warranted. The NRC determined that additional modifications were not warranted based on the availability of HPI feed and bleed as an alternate means of decay heat removal. With the combination of HPI feed and bleed and the SSF auxiliary service water system, the NRC closed the issue concerning the concurrent single active failure.

Since the SSF was not backfitted to be single failure proof, the initial licensing requirements of crediting the SSF for mitigation of a turbine building flood were not changed. The availability of HPI feed and bleed was credited in the backfit analysis but is not credited in the Oconee licensing basis for mitigation of the turbine building flood. However, HPI feed and bleed does play an important role in the reduction of the core melt frequency for a turbine building flood event and is described in the safety analysis report as an alternate means of decay heat removal. Thus, changes to HPI feed and bleed should be evaluated for their impact on the consequences of a turbine building flood event.

In addition to HPI feed and bleed, Duke indicated on a number of occasions that the Station auxiliary service water system was an alternate means of decay heat removal. Duke did not credit the Station auxiliary service water system in the mitigation of the turbine building flood. In addition, the NRC did not credit the Station auxiliary service water system for the mitigation of the turbine building flood event in the licensing basis or backfit analysis.

The reactor building spray pumps are described with respect to the waterproofing of the walls between the auxiliary building and the turbine building. However, Duke did not credit the reactor building spray pumps in the mitigation of the turbine building flood. In addition, the NRC did not credit the reactor building spray pumps for the mitigation of the turbine building flood event in the licensing basis or backfit analysis.

ELECTRICAL SEPARATION CRITERIA

Selected motor operated valves and selected pressurizer heaters are capable of being powered and controlled from either the normal station electrical systems or the SSF electrical system. Suitable electrical separation is provided in the following manner. Electrical distribution of the SSF is identified in Figure 9-40 and Figure 9-41 is provided by the SSF motor control centers (MCC's). Loads fed from MCC's 1XSF, 2XSF, 3XSF, and XSF are capable of being powered from either an existing plant load center or the SSF load center through key interlocked breakers at the MCC's. These breakers provide separation of the power supplies to the SSF loads.

Loads fed from MCC PXSF are capable of being powered from either Unit 2 B2T or the SSF Diesel via switchgear OTS1. Breakers feeding OTS1 are electrically interlocked and provide separation of the power supplies to the SSF loads.

During normal operation, these loads are powered from a normal (non-SSF) load center via the SSF MCC's 1XSF, 2XSF, 3XSF (Group B) or switchgear OTS1 via SSF MCC PXSF (Group C).

During operation of the SSF, these loads are powered from the SSF diesel generator via the SSF load center/switchgear and SSF MCC's.

9.6.3 System Descriptions

9.6.3.1 Structure

The Standby Shutdown Facility (SSF) is a reinforced concrete structure consisting of a diesel generator room, electrical equipment room, mechanical pump room, control room, central alarm station (CAS), and ventilation equipment room. The general arrangement of major equipment and structures is shown in Figure 9-30, Figure 9-31, Figure 9-32, Figure 9-33 and Figure 9-34.

The SSF has a seismic classification of Category 1. The following load conditions are considered in the analysis and design:

1. Structure Dead Loads
2. Equipment Loads
3. Live Loads
4. Normal Wind Loads
5. Seismic Loads
6. Tornado Wind Loads
7. Tornado Missile Loads
8. High Pressure Pipe Break Loads

9. Turbine Building Flooding Potential

WIND AND TORNADO LOADS

The design wind velocity for the SSF is 95 mph, at 30 ft. above the nominal ground elevation. This velocity is the fastest wind with a recurrence interval of 100 years. A gust factor of unity is used for determining wind forces. The design tornado used in calculating tornado loadings is in conformance with Regulatory Guide 1.76, Revision 0, with the following exceptions:

1. Rotational wind speed is 300 mph.
2. Translational speed of tornado is 60 mph.
3. Radius of maximum rotational speed is 240 ft.
4. Tornado induced negative pressure differential is 3 psi, occurring in three seconds.

The spectrum and characteristics of tornado-generated missiles are covered later in this section.

Revision 1 to Regulatory Guide 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," was released in March 2007. Revision 1 to Regulatory Guide 1.76 was incorporated into the SSF licensing basis in the 4th quarter of 2007. The design of all future changes to and/or analysis of SSF-related systems, structures, and components subject to tornado loadings will conform to the tornado wind, differential pressure, and missile criteria specified in Regulatory Guide 1.76, Revision 1.

FLOOD DESIGN

Flood studies show that Lake Keowee and Jocassee are designed with adequate margins to contain and control floods. The first is a general flooding of the rivers and reservoirs in the area due to a rainfall in excess of the Probable Maximum Precipitation (PMP). The FSAR addresses Oconee's location as on a ridgeline 100' above maximum known floods. Therefore, external flooding due to rainfall affecting rivers and reservoirs is not a problem. The SSF is within the site boundary and, therefore, is not subject to flooding from lake waters.

The grade level entrance of the SSF is 797.0 feet above mean sea level (msl). In the event of flooding due to a break in the non-seismic condenser circulating water (CCW) system piping located in the Turbine Building, the maximum expected water level within the site boundary is 796.5 ft. Since the maximum expected water level is below the elevation of the grade level entrance to the SSF, the structure will not be flooded by such an incident.

The SSF will stabilize the plant at mode 3 with an average Reactor Coolant temperature ≥ 525°F. As a PRA enhancement the SSF is provided with a five foot external flood wall which is equipped with a water tight door near the south entrance of the SSF. A stairway over the wall provides access to the north entrance. The yard elevation at both the north and the south entrance to the SSF is 796.0 feet above mean sea level (msl). Based on the as-built configuration of the 5' flood wall provided at the north entrance and a flood wall at the south entrance to the SSF, SSF external flood protection is provided for flooding that does not exceed 801 feet above mean sea level.

MISSILE PROTECTION

The only postulated missiles generated by natural phenomena are tornado generated missiles. The SSF is designed to resist the effects of tornado generated missiles in combination with other loadings. Table 9-17 lists the postulated tornado generated missiles.

Penetration depths are calculated using the modified NDRC formula and the modified Petry formula.

Modified N.D.R.C Formula:

$$\text{Penetration depth, }(x) = \sqrt{4KNWd\left(\frac{\nu_o}{1{,}000d}\right)^{1.80}} \text{ for } x/d \le 2.0$$

$$= \sqrt{KNWd\left(\frac{\nu_o}{1{,}000d}\right)^{1.80}} + d \text{ for } x/d > 2.0$$

Where:

N = missile shape factor = 0.72 for flat nosed bodies, 1.14 for sharp nosed bodies

K = concrete penetrability factor = $\frac{180}{\sqrt{f_c}}$

W = Weight in pounds

v_o = striking velocity

D = effective projectile diameter = $\sqrt{4Ac/\pi}$

A_c = projectile contact Area in in^2

Modified Petry Formula:

Penetration depth,(x) $= 12K_pA_p\log_{10}(1 + V^2/215{,}000)$

Where:

K_p = a coefficient depending on the nature of the concrete

= 0.00426 for normal reinforced concrete

A_p = weight of missile per unit of impact area

= W/A_c

A_c = Impact Area

V = striking velocity of projectile

Table 9-18 lists the calculated penetration depths and the minimum barrier thicknesses to preclude perforation and scabbing, hence eliminating secondary missiles.

Revision 1 to Regulatory Guide 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," was released in March 2007. Revision 1 to Regulatory Guide 1.76 was incorporated into the SSF licensing basis in the 4th quarter of 2007. The design of all future changes to and/or analysis of SSF-related systems, structures, and components subject to tornado loadings will conform to the tornado wind, differential pressure, and missile criteria specified in Regulatory Guide 1.76, Revision 1.

SEISMIC DESIGN

The design response spectra correspond to the expected maximum bedrock acceleration of 0.1 g. The design response spectra were developed in accordance with the procedures of Reg. Guide 1.60. The seismic loads as a result of a base excitation are determined by a dynamic analysis. The dynamic analysis is made utilizing the STRUDL-DYNAL computer program. The base of the structure is considered fixed.

With the geometry and properties of the model defined, the model's influence coefficients (the flexibility matrix) are determined. The contributions of flexure as well as shearing deformations are considered. The resulting matrix is inverted to obtain the stiffness matrix, which is used together with the mass matrix to obtain the eigenvalues and associated eigenvectors.

Having obtained the frequencies and mode shapes and employing the appropriate damping factors, the spectral acceleration for each mode can be obtained from Design Ground Motion response spectra curves. The standard response spectrum technique is used to determine inertial forces, shears, moments, and displacements for each mode. The structural response is obtained by combining the modal contributions of all the modes considered. The combined effect is represented by the square root of the sum of the squares.

The analytical technique used to generate the response spectra at specified elevations is the time history method. The acceleration time history of each elevation is retained for the generation of response spectra reflecting the maximum acceleration of a single degree of freedom system for a range of frequencies at the respective elevation. The structure will withstand the specified design conditions without impairment of structural integrity or safety function.

9.6.3.2 Reactor Coolant Makeup (RCM) System

The SSF RCM System is designed to supply borated makeup to the Reactor Coolant System (RCS) to provide Reactor Coolant Pump Seal cooling and RCS inventory. An SSF RCM Pump located in the Reactor Building of each unit will supply makeup to the RCS should the normal makeup system and the reactor coolant pumps become inoperative because of a station blackout condition caused by the loss of all other on-site and off-site power. The system is designed to ensure that sufficient borated water is available from the spent fuel pools to allow the SSF to maintain mode 3 with an average Reactor Coolant temperature ≥ 525°F (the initiating event may cause average RCS temperature to drop below 525°F) for all three units for approximately 72 hours. This time period is based on drawing the water level in the spent fuel pool down to a minimum of one foot above the top of the spent fuel racks. The SSF RCM System is operated and/or tested from the Standby Shutdown Facility. The SSF RCM System is shown on Figure 9-35. The SSF RCM Pump is capable of delivering borated water from the Spent Fuel Pool to the RC pump seal injection lines. A portion of this seal injection flow is used to makeup for RC pump seal leakage while the remainder flows into the RCS to makeup for other RCS leakage.

The SSF RCM Pump is a positive displacement pump driven by an induction motor, powered from the SSF Power System. The pump is located in the Reactor Building basement sufficiently below the spent fuel pool water level to assure that adequate net positive suction head is available.

A SSF RCM Filter is supplied downstream of the SSF RCM Pump to collect particulate matter larger than five microns that could be harmful to the seal faces. The filter is sized to accept three times the flow output of the SSF RCM Pump. Fouling of this filter is not considered to be a problem since the filter has been conservatively sized.

SSF controlled pressurizer heaters support achieving and maintaining RCS natural circulation flow by offsetting pressurizer heat loss due to ambient heat loss from the pressurizer and pressurizer steam space leakage. Pressurizer heater Group B, Bank 2 that is normally controlled from the main unit's control room may be controlled from the SSF Control Panel during SSF events. Pressurizer heater Group C, Bank 2 can only be controlled from the SSF Control Panel. Pressurizer level control can be accomplished from proper control of ASW flow to the steam generators, and proper control of the SSF RC letdown line flow. Additional RCS inventory control can be accomplished using the RV head vent. SSF D/G power can be connected to the RV head vent valves. Control of the RV head vent valves will be accomplished using a portable control panel.

During an accident that requires operation of the SSF, the following RCS isolation valves are closed to preserve RCS inventory once control of these valves is transferred to the SSF (Reference Table 9-15):

1,2,3HP-3
1,2,3HP-4
1,2,3HP-20
1,2,3RC-4
1,2,3RC-5
1,2,3RC-6

9.6.3.3 Auxiliary Service Water (ASW) System

The SSF ASW System is designed to cool the RCS during a station blackout and in conjunction with the loss of the normal and Emergency Feedwater System by providing steam generator cooling.

The SSF ASW pump is the major component of the system. One motor driven SSF ASW pump, powered from OTS1 Switchgear, serves all three units and is located in the SSF. The suction supply for the SSF ASW pump, the SSF HVAC service water pumps, and the SSF DSW pump is lake water from the embedded Unit 2 condenser circulating water piping. A portable submersible pump that can be installed in the intake canal and powered from the SSF is available to replenish the water supply in the embedded CCW pipe if both forced CCW and siphon flow through the CCW pipe are lost.

The SSF ASW flow rate provided to each unit's steam generators is controlled using the motor operated valves on each unit's SSF ASW supply header. Manually operated bypass valves, installed in parallel with the motor-operated valves, are also available to:

1. Provide SSF ASW Flow control at low SSF ASW Flow rates.
2. Provide more precise SSF ASW Flow control when used in parallel with the motor-operated valves.

The SSF ASW pump is sized to provide enough flow to all 3 Oconee units to adequately remove decay heat from the RCS and maintain natural circulation in the RCS. An SSF ASW pump minimum flow line is provided to ensure that the pump minimum flow requirements are met. The SSF ASW system, pump and valves are operated and tested from the SSF only. The SSF ASW system is shown on Figure 9-36.

Auxiliary service water enters the steam generators via the normal emergency feedwater ring headers.

The SSF ASW System provides the motive force for the SSF ASW suction pipe air ejector. The air ejector is needed to maintain siphon flow to the SSF HVAC service water pump, the SSF DSW pump, and the SSF ASW pump when the water level in the U2 CCW supply pipe becomes too low.

The SSF ASW System provides adequate SG cooling to reduce and maintain RCS pressure below the pressure where the SSF RC makeup pump discharge relief valve, HP-404, begins to pass flow. Therefore, full SSF RC Makeup System seal injection flow will be provided to the RC pump seals in time to prevent seal degradation or failure.

Though not a requirement for operability, the SSF diesel generator should be aligned to carry SSF loads and the SSF ASW pump should be operated to provide a large enough load so that diesel souping concerns are not a problem when the Emergency Start pushbutton is used to start the SSF diesel engines and continued operation of the SSF diesel engines is desired. While continued operation of the SSF diesel engines when they are lightly loaded is possible (i.e. one, two or three SSF RC makeup pumps operating without operating the SSF ASW pump), lightly loading the engines in this manner is not preferred due to the potential for a fire in the diesel exhaust if a large load is added after souping of the engine occurs.

9.6.3.4 Electrical Power

9.6.3.4.1 General Description

The Standby Shutdown Facility (SSF) Electrical Power System includes 4160VAC, 600VAC, 208VAC, 120VAC, and 125VDC power. This system supplies power necessary to maintain mode 3 with an average Reactor Coolant temperature ≥ 525°F for the reactors of each unit, in the event of loss of power from all other power systems. It consists of switchgear, load center, motor control centers, panelboards, batteries, battery chargers, inverters, a diesel-electric generator unit, relays, control devices, and interconnecting cable supplying the appropriate loads.

The 120VAC power system in conjunction with the 125VDC instrumentation and control power system supplies continuous control power to all loads that are required for achieving mode 3 with an average Reactor Coolant temperature ≥ 525°F of each reactor.

Following the loss of all normal and emergency power, on-site and off-site, the diesel-electric generating unit will be manually started by initiating its start signal from the SSF Control Panel in the SSF. SSF Systems cannot operate without receiving power from the diesel for SSF scenarios when power from the Unit 2 Main Feeder Bus is not available. The diesel generator and its associated auxiliaries are housed in a Class 1 structure and are protected against seismic events.

The 4160VAC SSF Power System bus will then be connected to its diesel-electric, backup source of power by manually closing the appropriate 4160VAC generator breaker.

Schematics of the SSF electrical system are shown on Figure 9-40 and Figure 9-41.

9.6.3.4.2 Diesel Generator

The SSF Power System is provided with standby power from a dedicated diesel generator. This SSF diesel generator is rated for continuous operation at 3500 kW, 0.8 pf, and 4160 VAC. The SSF electrical design load does not exceed the continuous rating of the diesel generator. The auxiliaries required to assure proper operation of the SSF diesel generator are supplied entirely from the SSF Power System. The SSF diesel generator is provided with manual start capability from the SSF only. It uses a compressed air starting system with four air storage tanks. Each set of two tanks will provide sufficient air to start the diesel unit three successive times. An independent fuel system, complete with a separate underground storage tank, duplex filter arrangement, a fuel oil transfer pump, and one-hour day tank, is supplied for the diesel-electric generating unit.

The diesel generator protection system initiates automatic and immediate protective action to prevent or limit damage to the SSF diesel generator. The following protective trips are provided to protect the diesel generator at all times and are not bypassed when the diesel generator is in the emergency mode:

1. Engine Overspeed
2. Generator Differential Protection
3. Low-low Lube Oil Pressure
4. Generator Overcurrent

9.6.3.5 Instrumentation

9.6.3.5.1 SSF Reactor Coolant Makeup System Instrumentation

Each unit is provided with instrumentation to monitor RCM System flow, pressure and temperature; RC Loop A and B pressure and temperature; pressurizer level and pressure; and reactor incore temperature.

Five (5) Incore Thermocouples per unit may be used to monitor the incore temperature. Six (6) RTD's per unit will be used to monitor Loop A and B RC System Hot & Cold Leg temperature. Readout is displayed on the SSF control panel. Table 9-16 provides a listing of instrumentation.

9.6.3.5.2 SSF Auxiliary Service Water Instrumentation

Each unit is provided with Steam Generator A & B level instrumentation labeled as listed in Table 9-16. Readout is displayed on the SSF control panel. Each unit's SSF ASW piping is also provided with instruments to monitor SSF ASW System flow and pressure. Each unit's flow is displayed on the SSF control panel. The SSF ASW pump recirculation piping is provided with instrumentation to monitor SSF ASW System recirculation flow and pressure. The recirculation flow is displayed on the SSF control panel.

9.6.3.6 Support Systems

The Standby Shutdown Facility (SSF) Support Systems are designed to provide for the SSF:

1. Lighting
2. Fire Protection
3. Fire Detection
4. Service Water
5. Heating Ventilation and Air Conditioning (HVAC)
6. Sump Drainage
7. Potable Water

The diesel engine service water and the HVAC service water piping are designed in accordance with ASME Section III, Class 3, which includes seismic design. The fire protection water, carbon dioxide, potable water, and sewage piping systems are seismically restrained in areas above seismically designed equipment. Portions of the SSF Sump System are seismically restrained to prevent flooding of the SSF Pump Room. The lighting system and the fire detection system are not seismically designed. The water and carbon dioxide fire protection systems and the fire detection system are designed and constructed to meet or exceed National Fire Codes.

9.6.3.6.1 SSF Lighting System Description

Normal lighting for the SSF is provided by fluorescent and HID lighting units. These lighting units are located to provide adequate levels of light with good distribution throughout the structure.

Emergency AC lighting for the SSF is provided. These units are located to provide adequate levels of lighting in all areas of the structure.

Emergency DC lighting for the SSF is provided by self-contained 12VDC battery pack lighting units. These units are located to provide adequate levels of lighting for control panel operation and for entering and leaving the structure. These battery pack lights are energized automatically upon an undervoltage in the normal lighting system power supply.

9.6.3.6.2 SSF Fire Protection and Detection

The SSF contains two fire protection systems, a water system and a carbon dioxide system.

The water system is provided with manually valved hose reels in the stairwell at each floor elevation and inside the entrance to the diesel room. From these locations the hose lengths are such that the entire SSF can be served by the primary fire protection system.

The low pressure carbon dioxide system provided is actuated by thermal detectors to automatically flood the diesel area. Carbon dioxide is stored in a refrigerated storage tank in sufficient quantity to provide twice the required coverage for the area.

Portable carbon dioxide extinguishers are also provided.

Detection devices are located throughout the SSF and will annunciate with a single alarm to the Unit Control Rooms, SSF Control Room and Security. Specific alarms annunciate on the Fire Alarm Control Unit located in the SSF vestibule.

9.6.3.6.3 SSF Service Water

The SSF Service Water System consists of two subsystems: The HVAC Service Water System and the Diesel Engine Service Water System.

The HVAC Service Water System, which operates continuously, contains two pumps and supplies cooling water to the HVAC condensers. Only one pump will operate at any given time with the other idle pump acting as a backup.

The Diesel Engine Service Water System, which normally operates only when the diesel is operating or when system components are being tested, contains one pump and provides service water to the diesel engine jacket water heat exchangers.

This flow is monitored during periodic operational test or emergency operation. All three pumps take their suction from the embedded CCW piping and return the flow to the CCW piping after passing through their respective system. SSF Diesel Engine Service Water is diverted to the yard drain during an SSF event to avoid overheating the water contained in the SSF ASW supply piping.

The SSF Diesel Engine Service Water System is shown on Figure 9-37.

9.6.3.6.4 Heating Ventilation and Air Conditioning

The SSF HVAC system consists of two subsystems, a ventilation system and an air conditioning system. Both systems are powered by the SSF Power System. Sections of each system are shut down in event of fire in the area served. The SSF HVAC System supports operation of systems and equipment located in the SSF by maintaining temperature in the SSF within design limits.

VENTILATION SYSTEM

The diesel generator room, switchgear room, pump room, and HVAC room do not require close control of temperature, and the relatively high heat loads are dissipated with a variable volume ventilation system. The purpose of the ventilation system is to provide filtered outside air which is tempered if necessary to maintain a minimum temperature of 60°F and a maximum temperature as follows:

1. HVAC Room 120°F
2. Switchgear Room 120°F
3. Pump Room 120°F
4. Diesel Generator Room 125°F

AIR CONDITIONING SYSTEM

Certain rooms in the SSF require close control of temperature and have year-round heat loads of such magnitude to necessitate mechanical refrigeration. Normal operating conditions for these rooms are 72°F and 50 percent RH with a minimum of outside air for ventilation. During an SSF event, air conditioned rooms are maintained within the following design temperature limits:

1. SSF Control Room 100°F
2. SSF Battery Rooms 113°F
3. Computer Room (no limit for SSF power system operability)

The air conditioning system supplies each area with a constant volume of air. A heating coil located in each area with a local control tempers the air as required to maintain the desired temperature.

9.6.3.6.5 SSF Sump System

The SSF Sump System provides a collection and discharge function for normal equipment drainage within the SSF. The main components of the system are the sump and two sump pumps which handle the flow routed to the sump via the floor drain system located throughout the SSF.

9.6.4 System Evaluations

9.6.4.1 General

The design of the SSF was reviewed to meet the requirements of Appendix R of 10CFR 50, Sections III.G.3 and III.L, and those requirements applicable for flooding and seismic events. Since the transition to NFPA 805, some original SSF design criteria for fire events only no longer align with Appendix R.

The SSF, the associated mechanical and electrical systems and power supplies meet or exceed the applicable criteria contained in the Oconee FSAR Chapter 3. Additionally, ASME and IEEE codes are utilized as appropriate, in the design of various subsystems and components. The SSF and systems/components needed for safe shutdown are designed to withstand the Safe Shutdown Earthquake (SSE). The SSF systems required for safe shutdown are designed with adequate capacity to achieve and maintain mode 3 conditions with an average Reactor Coolant temperature ≥ 525°F (the initiating event may cause average RCS temperature to drop below 525°F) of all three Oconee units.

The SSF power system is designed with adequate capacity and capability to supply the necessary loads, and is physically and electrically independent from the station electrical distribution system power supply. Additionally, the AC and DC power systems and equipment required for the SSF essential functions have been designed and installed consistent with the Oconee QA program for Class 1E equipment.

These systems are not designed to meet the single failure criterion, but are designed such that failures in the systems do not cause failures or inadvertent operations of existing plant systems. The electrical systems in the SSF are manually initiated, that is, multiple actions must be performed to provide flow to existing plant safety systems.

9.6.4.2 Structure Design

The SSF is statically and dynamically analyzed and designed as a three-dimensional space frame subjected to the applicable loads summarized in Section 9.6.3.1. The Structural Design Language (STRUDL) computer program is used to perform the analyses. The design is in accordance with the codes and criteria listed in Table 9-19. Design loads and loading combinations are in accordance with the NRC Standard Review Plan, Section 3.8.4.

The SSF is designed to withstand the effects of wind and tornado loadings, without loss of capability of the systems to perform their safety functions. The basis for the selected wind velocity is reference 1 of Section 3.3. Buildings and structures with a height to minimum horizontal dimension ratio exceeding five should be dynamically analyzed to determine the effect of gust factors (ref. American National Standard, "Building Code Requirements for Minimum Design Loads in Buildings and Other Structures," ANSI A58.1-1972, New York, New York). The SSF has a height/width ratio of less than five, and therefore, the gust factor of unity is used for determining wind forces. The design tornado used in calculating tornado loadings is in conformance with Regulatory Guide 1.76 except as noted in Section 9.6.3.1.

The relatively small surface area of the structure and its location result in an extremely low probability that a turbine missile would strike the facility. Turbine missile impact is not considered a viable load condition due to the location of the SSF with respect to the turbine. All postulated missiles are per the NRC Standard Review Plan Section 3.5.1.4 Rev. 1 and Regulatory Guide 1.76, Revision 0. The barrier thicknesses for the structure are such that they preclude any perforation and/or scabbing from the postulated tornado generated missiles. Minimum barrier thickness is three times the postulated missiles calculated depths of penetrations (see Table 9-18).

See Section 9.6.3.1 for information regarding Regulatory Guide 1.76, Revision 1, and future changes to and/or analysis of SSF-related systems, structures, and components subject to tornado loadings.

The dynamic analysis is made utilizing the STRUDL-DYNAL computer program. The design response spectra were developed in accordance with the procedures of Regulatory Guide 1.60. It corresponds to the expected maximum bedrock acceleration of 0.1g. Damping values are per Regulatory Guide 1.61.

The structure will withstand the specified design conditions without impairment of structural integrity or safety function.

9.6.4.3 Seismic Subsystem Analysis

The seismic analysis of Category I pipe is performed using dynamic modal analysis techniques. No static seismic analysis is used for SSF ASME Code piping. Modal response spectrum methods are used. Response of individual modes is combined by the Grouping Method of Regulatory Guide 1.92. An adequate number of masses or degrees of freedom are included in the model to determine the response of significant modes. The response due to each of three components of earthquake motion is combined by the square-root-of-the-sum-of-the-square rule as described in Regulatory Guide 1.92. Pipe supported from multiple levels or structure is designed for an envelope of the response spectra for all supporting structures.

Constant vertical static factors are not used. Vertical response is obtained from a dynamic modal analysis. Modal damping ratios are consistent with Regulatory Guide 1.61.

The location of the SSF non-Category I piping has been reviewed to determine those areas of proximity to Category I piping or safety related equipment. Where Category I piping or safety related equipment is in the proximity area, the non-Category I piping has been seismically qualified and supported or rerouted out of the problem area.

The SSF auxiliary service water buried piping is seismically designed for stresses resulting from SSE and OBE events. The design and analysis were based on the current state-of-the-art for initial effects and the effects of static resistance of the surrounding soil.

9.6.4.4 Dynamic Testing and Analysis of Mechanical Components

Procedures were established for the startup testing of the Class B and C piping in the SSF to verify the following information under different operating modes:

1. Physical Compliance with Piping Design: An "as built" verification procedure is utilized to verify that piping, components and support-/restraints have been erected with design tolerance.
2. Vibration Monitoring for Equipment: The purpose of this monitoring program is to verify that vibration levels for system components are within acceptance criteria. Pump vibration is monitored during testing in accordance with IWP-3210 to verify vibrations are less than or equal to the maximum allowable per the specific vendor's requirements.

9.6.4.5 ASME Code Class 1, 2, and 3 Components, Component Supports and Core Support Structures

Piping systems for the SSF are designed in accordance with the appropriate ASME Code based on the Quality Group classifications outlined in Regulatory Guide 1.26. Where part of an existing QA 1 piping system was used by an SSF subsystem to perform its function, the existing piping system was not "upgraded" to the pipe class and code used for piping when the SSF was constructed. The load combinations and stress limits contained in the requirements of SRP 3.9.3.II and referenced in Regulatory Guide 1.48 are met, except Code Case 1606 is used for the faulted load combination.

The SSF RC Makeup System is designed per the requirements stated in ASME Section III Class 2 (1974 Edition, Summer 1975 Addendum) to Oconee Class B. Portions of the HPI seal injection piping used by the SSF RC Makeup System to deliver flow to the RC pump seals are designed to Duke Class C.

The SSF ASW System has a portion (crossover between emergency feedwater lines) in each Reactor Building that was designed per the requirements stated in ASME Section III, Class 2 (1974 Edition, Summer 1975 Addendum) to Oconee Class B. The remainder of the SSF ASW System was designed per the requirements stated in ASME Section III Class 3 (1974 Edition, Summer 1975 Addendum) to Oconee Class C. Portions of the EFW System piping used by the SSF ASW System to deliver flow to the steam generators are designed to Duke Class F.

The loads from pressure relief valves with an open discharge are evaluated in accordance with Code 1569, "Design of Piping for Pressure Relief Valve Station", assuming multiple valves on the same pipe open in the most conservative sequence. A dynamic load factor of two is used to determine the transient loads unless a lower value is justified by analysis.

Relief valves discharging into a closed system or a system with long discharge piping are reviewed to identify any significant transient loadings. Any significant loading is analyzed using dynamic analyses to include the effects of changes in momentum due to fluid flow changes of direction and any potential water slugs. The piping will be adequately supported such that piping stresses associated with the defined transient loads satisfy applicable Code requirements.

The loading combinations and stress limits contained in the requirements of SRP 3.9.3.II.4 and referenced in Regulatory Guide 1.48 are met. However, ASME Code Section III Subsection NF did not provide faulted condition allowable stress limits for Class 2 and 3 component supports until the 1977 edition. The allowables for Class 1 components in the 1974 edition of Subsection NF and subsequent applicable addenda for its Class 2 and 3 component supports faulted stress allowables were utilized.

9.6.4.6 Fire Protection

Resulting from the Nuclear Safety Capability Assessment conducted as required by NFPA 805, the SSF is credited for achieving and maintaining safe and stable plant conditions following a fire in specific locations within the Auxiliary Building, including the main control rooms.

9.6.4.6.1 Safe Shutdown Systems

Safe shutdown of the reactor is initially performed by the insertion of control rods from the control room. Insertion can also be accomplished by removing power to the control rod drive mechanisms. When normal and emergency systems are not available, reactor coolant inventory and reactor shutdown margin are maintained, from the SSF Control Panel by the SSF RC makeup pump taking suction from the spent fuel pool. Primary system pressure can be maintained by the pressurizer heaters or by use of charging combined with letdown.

Deleted Paragraph(s) per 2012 update.

9.6.4.6.2 Performance Goals

The performance goals for post-fire safe and stable conditions (as defined in NFPA 805) can be met using the SSF for those specific fire events that require SSF control.

The process monitoring instruments to be used for a post fire shutdown include reactor coolant hot leg and cold leg temperatures, reactor coolant pressure, pressurizer level and pressure, steam generator level, SSF RC makeup pump flow, and SSF ASW system flow to each unit.

STEAM GENERATOR PRESSURE

Reactor coolant system (RCS) heat removal for achieving mode 3 with an average Reactor Coolant temperature ≥ 525°F can be directly monitored by RCS parameters and controlled by SG level without SG pressure indication, provided that SG pressure is regulated.

SG pressure should be regulated by the main steam code safety valves, which will relieve at their setpoints. Secondary side depressurization is limited by isolating selected main steam branch line boundary valves. RCS conditions can be monitored by primary coolant temperature and pressure, pressurizer level and SG level. Should RCS overcooling occur, corrective actions can be taken from the SSF to reinstate proper cooling by controlling the SSF ASW flow rate provided to a unit's SGs, and by restoring steam generator level for applicable events, in order to restore T-cold.

The SSF is designed to achieve and maintain mode 3 with an average Reactor Coolant temperature ≥ 525°F (RCS cold leg temperature ≤ 555°F and RCS pressure $\simeq$ 2155 psig) for one or more of the three Oconee units. The SSF is not designed to independently bring the reactor from mode 3 with an average Reactor Coolant temperature ≥ 525°F (RCS cold leg temperature ≤ 555°F and RCS pressure $\simeq$ 2155 psig) to shutdown. Shutdown will be achieved and maintained through the use of normal plant systems and equipment.

SOURCE RANGE FLUX MONITOR

The SSF is designed to achieve and maintain mode 3 with an average Reactor Coolant temperature ≥ 525°F (RCS cold leg temperature ≤ 555°F and RCS pressure $\simeq$ 2155 psig) for any or all of the Oconee units. Prior to leaving the Unit 1/2 or Unit 3 control room, all control rods for the unit under consideration are required to be inserted. No non-borated sources tie into the SSF makeup/boration flow path. RCS makeup and boration following transfer of control to the SSF RCM is from the spent fuel pool. Thus, boron dilution events are highly unlikely.

Oconee Units 1, 2, and 3 can achieve and maintain controlled cooling to mode 3 with an average Reactor Coolant temperature ≥ 525°F (RCS cold leg temperature ≤ 555°F and RCS pressure $\simeq$ 2155 psig) safely from the SSF without the need for remote SG pressure instrumentation or a remote source range monitor.

The need for source range instrumentation is not necessary since boron sampling can be utilized to ensure shutdown margin.

9.6.4.6.3 Instrumentation Guidelines

NFPA 805 states that shutdown systems installed for ensuring post-fire shutdown capability need not be designed to meet seismic Category I criteria, single failure criteria, or other design basis accident criteria, except where required for other reasons, e.g., because of interface with or impact on existing safety systems, or because of adverse valve actions due to fire damage. Since the monitors for the above listed parameters, in Section 9.6.4.6.2, will not interface with or impact existing safety systems, the monitors need not be "safety grade".

9.6.4.6.4 Repairs for Hot Shutdown

NFPA 805 requires that the plant achieve safe and stable conditions after any single fire. For scenarios requiring the use of the SSF, safe and stable conditions can be maintained in mode 3 (hot standby) with an average Reactor Coolant temperature ≥ 525°F for up to a 72 hour coping duration to allow for the repair of any damaged equipment necessary to reach hot shutdown. Repairs might include replacement of power cabling, pump motors and switchgear associated with the HPI system required for hot shutdown. Stored on-site are all components necessary to achieve all repairs. Guidelines are available to implement the required repairs and replacements.

9.6.4.6.5 Fire Protection Conclusion

While many fire areas have credited success paths for achieving safe and stable plant conditions from the Control Room, a select number of fire scenarios only credit the SSF for providing the requisite one train of systems necessary to achieve and maintain safe and stable conditions.

9.6.4.7 Flooding Review

The SSF will not be affected by the following postulated flood events:

1. Turbine Building Flood caused by a break in the non-seismic condenser circulating water (CCW) piping system.
2. Infiltration of normal groundwater.

The structure meets the requirements of GDC 2, and the guidelines of Regulatory Guide 1.102 with respect to protection against flooding.

9.6.5 Operation and Testing

The SSF will be placed into operation to mitigate the consequences of the following events:

1. Flooding
2. Fire
3. Sabotage
4. Station Blackout

For fire events that require activation of the SSF for the unit affected, following local confirmation of the fire, the operator will staff the SSF and perform the electrical isolation/control transfer of the 600VAC Motor Control Center in the SSF as promptly as possible after confirmation of the fire. Following the control transfer, the operator will establish continuous communications with the Control Room of the unit affected awaiting instructions regarding the need to start and utilize the available SSF Diesel Generator, RCMU system and establish SSF Auxiliary Service Water flow to the steam generators as needed and close all of the Reactor Coolant System isolation valves that are controlled from the SSF.

Additionally, for fire events where SSF activation is required, main steam boundary valves must also be promptly closed to maintain proper control of RCS parameters while the SSF is made operational.

For flooding, sabotage, station blackout and those fire events where the SSF is credited for safe shutdown, operators will be sent to the SSF. When directed by the shift supervisor or procedure, the operator will start the RCM system and establish SSF Auxiliary Service Water flow to the steam generators as needed, as well as close SSF controlled Reactor Coolant System pressure boundary valves.

Deleted Paragraph(s) per 2012 update.

In-service testing of pumps and valves will be done in accordance with the provision of ASME Section XI except for the Submersible Pump which is used to supply makeup water to the Unit 2 embedded condenser circulating piping. This pump should be tested every other year to verify flow capability. A recirculation flow path with flow and pressure instrumentation is available for SSF ASW pump testing.

The electrical power system components will be tested consistent with Duke Power's Testing Philosophy as described in the nuclear station directives.

9.6.6 References

1. Safety Evaluation by the Office of Nuclear Reactor Regulation Oconee Nuclear Station Standby Shutdown Facility, Docket Nos. 50-269, 50-270, and 50-287, April 28, 1983
2. Safety Evaluation for Station Blackout (10 CFR 50.63) - Oconee Nuclear Station, Units 1, 2, and 3 (TACS M68574/M68575/M68576), Docket Nos. 50-269, 50-270, 50-287, March 10, 1992
3. Safety Evaluation for Station Blackout (10 CFR 50.63) - Oconee Nuclear Staton, Units 1, 2, and 3 (TACS M68574/M68575/M68576), Docket Nos. 50-269, 50-270, 50-287, December 3, 1992
4. Safety Evaluation Report on Effect of Tornado Missiles on Oconee Emergency Feedwater System (TACS 48225, 48226, and 48227), July 28, 1989
5. Safety Evaluation Report for Implementation of Recommendation for Auxiliary Feedwater Systems, August 25, 1981
6. Evaluation of the Oconee, Units 1,2,&3 Generic Safety Issues (GSI-23 & GSI-105) Resolution, March 24, 1995
7. Letter from WO Parker (Duke) to EG Case (NRC), dated 1/25/78, Response to NRC Questions
8. Letter from WO Parker (Duke) to EG Case (NRC), dated 2/1/78, SSF System Description
9. Letter from WO Parker (Duke) to EG Case (NRC), dated 6/19/78, Response to Staff Questions Concerning Oconee Nuclear Station Safe Shutdown System
10. Letter from WO Parker (Duke) to HR Denton (NRC), dated 3/28/80
11. Letter from WO Parker (Duke) to HR Denton (NRC), dated 2/16/81, Response to NRC Request for Information
12. Letter from WO Parker (Duke) to HR Denton (NRC), dated 3/18/81, Modifications Needed to Meet Appendix R Requirements
13. Letter from WO Parker (Duke) to HR Denton (NRC), dated 3/31/81, Response to NRC Request for Information
14. Letter from WO Parker (Duke) to HR Denton (NRC), dated 4/30/81, Cable Routing and Separation
15. Letter from WO Parker (Duke) to HR Denton (NRC), dated 1/25/82, Response to NRC Concerns for Source Range Instrumentation and Steam Generator Pressure

16. Letter from HB Tucker (Duke) to HR Denton (NRC), dated 9/20/82, Response to NRC Request for Information

17. Letter from HB Tucker (Duke) to HR Denton (NRC), dated 12/23/82, Requested Supplemental Information

18. Letter from HB Tucker (Duke) to HR Denton (NRC), dated 7/15/83, Request for Exemption from 10CFR50 Appendix R, Section III.L.2

19. Letter from JF Stolz (NRC) to HB Tucker (Duke), dated 8/31/83, Exemption from Source Range Flux and Steam Generator Pressure Instrumentation for the SSF

20. Deleted per 2012 update.

21. Deleted per 2012 update.

22. Deleted per 2012 update.

23. Regulatory Guide 1.76, “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants,” Revision 1

24. OSC-9086, Calculation for the USQ Review of Regulatory Guide 1.76, Revision 1

25. Letter from CA Julian (NRC) to HB Tucker (Duke), dated 10/4/89, NRC Inspection Report

26. O-320Z-3 SSF External Barrier Walls Concrete

27. 10CFR Part 50 Appendix R Section III.L Alternative and Dedicated Shutdown Capability

28. NFPA 805, “Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants”, 2001 Edition

29. 10CFR50.48 (c), “Fire Protection”

30. ONS NFPA 805 License Amendment Request 2008-01 (April 14, 2010)

31. NRC Issuance of ONS NFPA 805 Amendments and Safety Evaluation (December 29, 2010)

THIS IS THE LAST PAGE OF THE TEXT SECTION 9.6.

THIS PAGE LEFT BLANK INTENTIONALLY.