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3.0 Design of Structures, Components, Equipment, and Systems

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3.1 Conformance with NRC General Design Criteria

Deleted Paragraph(s) per 2009 Update

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

3.1.1 Criterion 1 - Quality Standards (Category A)

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

Discussion

1. Essential Systems and Components

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

   a. Reactor Coolant System

   b. Reactor vessel internals

   c. Reactor Building

   d. Engineered Safeguards System

   e. Electric emergency power sources

2. Codes and Standards

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

Item	Codes	Quality Control	Testing
Reactor Coolant System	Section 5.2.2	Section 5.2.3.11	Sections 5.2.3.11; 4.4.4
Reactor Vessel Internals	Section 4.5.1	Section 4.5.4	Section 4.5.4
Reactor Building	Sections 3.8.1.2; 3.8.3; 3.8.1.4; 3.8.1.5	Section 3.8.1.6	Section 3.8.1.7
Engineered Safeguards System	Sections 6.2.2.2.2; 6.0; 6.3.2.4	Sections 6.0; 6.6	Sections 6.3.4; 6.4.3; 6.5.1.4; 6.2.2.4; 6.2.4
Electric Emergency Power Sources			Section 8.3.1.1.6

3.1.1.1 Oconee QA-1 Program

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

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

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

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

First Category, Original Oconee QA-1 Licensing Basis

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

Clarification regarding the six SSCs identified above is provided below.

1. Reactor Coolant System

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

2. Reactor Vessel Internals

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

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

3. Reactor Building

   The Reactor Building consists of the following:

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

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

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

   d. Access openings to the Reactor Building.

4. Engineered Safeguards System

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

   a. Provide emergency cooling to assure structural integrity of the core:

      High Pressure Injection System

      Low Pressure Injection System

      Core Flooding System

   b. Maintain the integrity of the Reactor Building

      Reactor Building Spray System

      Reactor Building Cooling System

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

   c. Provide for the collection and control of Reactor Building penetration leakage:

      Penetration Room Ventilation System

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

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

      1) Reactor Building Cooling Units

      2) Decay Heat Removal Coolers

      3) High Pressure Injection Pump Motors

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

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

5. Emergency Electric Power Sources

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

a. Keowee Hydroelectric Units 1 and 2, including:

   Keowee Hydro-Generator and Emergency Start Circuits,

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

   Keowee 125 VDC Power System.

   The following mechanical Keowee SSCs:

   1) Governor Oil System
   2) Governor Air System
   3) Guide Bearing Oil System
   4) Turbine Sump System
   5) Cooling Water System

b. Underground Emergency Power Path, including:

   Underground cable,

   Transformer CT4, and

   Standby Busses.

c. Overhead Emergency Power Path, including:

   Keowee Main Step-Up Transformer,

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

   230 KV Switchyard Yellow Bus,

   230 KV Switchyard 125 VDC Power System, and

   Unit Start-up Transformers (CT1, CT2, and CT3).

d. Unit Main Feeder Busses

e. 4160 VAC Safety Auxiliary Power System

f. 600/208 VAC Safety Auxiliary Power System

g. 120 VAC Vital I&C Power System

h. 125 VDC Vital I&C Power System

6. Reactor Protective System

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

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

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

These commitments are as follows:

1. The following portions of the emergency feedwater (EFW) systems are QA-1.
   a. the motor-driven (MD) EFW pumps
   b. the piping from the MD EFW pumps to the steam generators
   c. the EFW flow control valves (excluding the operators)
   d. the power supply to the MD EFW pumps and controls
   e. piping from the upper surge tanks (USTs) to the MD EFW pumps
   f. UST level monitoring circuitry and associated solenoid valves
   g. EFW flow transmitters upstream of the flow control valves
   h. MD and turbine-driven EFW low steam generator water level and Main Feedwater pump low hydraulic oil pressure pump initiation signals
2. The anticipatory reactor trips on (1) loss of main feedwater and (2) turbine trip are QA-1.
3. The following instruments are QA-1 per the Duke response to Regulatory Guide 1.97:
   a. Two channels of wide range Reactor Coolant System (RCS) pressure
   b. 24 core exit thermocouples (12 per train)
   c. Two channels of pressurizer level (one per train)
   d. Two channels of saturation margin (one monitoring loop A and the core, the other monitoring loop B and the core)
   e. Two channels of steam generator (SG) level per SG (O-388" range)
   f. Two channels of SG pressure per SG
   g. Three channels of borated water storage tank level
   h. Two channels of high pressure injection (HPI) flow
   i. Two channels of low pressure injection (LPI) flow
   j. Two channels of Reactor Building spray flow
   k. Two channels of Reactor Building hydrogen concentration
   l. Two channels of upper surge tank level (one per tank)
   m. Two channels of full range neutron flux
   n. Two channels of wide range RCS hot leg temperature (one per loop)
   o. Two channels of reactor vessel head level
   p. Two channels of hot leg level (one per loop)
   q. Two channels of wide range Reactor Building sump level
   r. Two channels of Reactor Building pressure
   s. One channel of valve position for each electrically-controlled Reactor Building isolation valve
   t. Two channels of high range Reactor Building radiation level
   u. Two channels of EFW flow per SG

v. One channel of low pressure service water (LPSW) flow to the LPI coolers (per cooler)

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

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

   a. SSF reactor coolant emergency makeup piping and components

   b. SSF auxiliary service water piping and components

   c. SSF cooling water piping for the diesel generator and HVAC

   The SSF equipment required for mitigation of a Turbine Building flood shall be QA-1, with the exception of plant equipment used for the SSF function that was not QA-1 prior to the construction of the SSF (e.g., pressurizer heaters) and the SSF Portable Pumping System.

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

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

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

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

10. The suction source for the Low Pressure Service Water (LPSW) System is QA-1. This includes:

    a. Emergency Condenser Circulating Water System first siphon which provides suction to the Low Pressure Service Water System following a LOOP event. This includes the pressure boundary of the Condenser Circulating Water pumps, pump discharge valves and piping from the intake up to and including the 42 inch crossover header

    b. Essential Siphon Vacuum System

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

12. The pressure transmitters, logic circuitry, and power sources for the Automatic Feedwater Isolation System (AFIS) and components used to terminate EFW flow to a faulted steam generator are QA-1.

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

14. The hydrogen recombiner interfacing piping systems shall be QA-1

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

16. The LPSW RB Waterhammer Prevention System

17. The Protected Service Water (PSW) System is QA-1 with limited exceptions, e.g., fire detection components, some HVAC equipment, building lighting, etc. (Note: Components that receive backup power from PSW or systems that connect to PSW retain their existing seismic and quality classifications).

3.1.1.2 Oconee QA-5 Program

The Oconee QA Condition 5 program is a voluntary program. Program elements have not been specified to the NRC and are not subjected to a formal NRC review and approval process. The QA-5 program itself is not a commitment to the NRC, but Duke has committed to the NRC to include certain equipment within the scope of this voluntary QA-5 program.

The QA-5 program was conceived in response to the recognition that there are some SSCs that were not covered in the original Oconee QA-1 Licensing Basis or deemed appropriate for the expanded QA-1 Licensing Basis that are however credited for prevention and mitigation of design basis and other selected events. The significance of these components warranted an augmented quality assurance program. To that end, Duke created the voluntary QA-5 Program, described in Attachments 4, 4a and 4b of Reference 1 in Section 3.1.1.3 and accepted by the NRC in Reference 2, to apply selected 10CFR 50 Appendix B criteria to such SSCs. The QA-5 classification designates those SSCs for testing and maintenance under selected Appendix B criteria while not requiring that they be procured per Appendix B criteria. Replacement parts for these SSCs will be procured "equal or better in quality" based on engineering judgment.

To determine the population of SSCs to which this classification would apply, a list of accidents and events in the Oconee licensing basis was made, excluding those accidents or events that did not require a safety-related function or were design criteria only. For each remaining accident or event, the primary critical safety functions and primary supporting functions were identified. The SSCs that performed those functions were then evaluated.

The QA-5 classification applies to that equipment which meets the criteria described in References 1 and 2 in Section 3.1.1.3.

The QA-5 program is not a design program or design criteria. Its inclusion here is to clarify the reasoning for excluding certain equipment from the QA-1 program. The QA-5 program clarifies the delineation between safety-related (QA-1) and non-safety-related equipment. Furthermore, the augmented maintenance and testing of the QA-5 Program improves equipment reliability for non-safety-related equipment.

The QA-5 Program was created by the Oconee Safety Related Designation Clarification (OSDRC) program. It addresses NRC concerns regarding a lack of quality assurance for non-safety components that were relied upon to mitigate design basis events. One such concern and how the QA-5 program addressed it are detailed in the (CLOSED) URI 50-269, 270, 287/98-03-09 in Reference 3 in Section 3.1.1.3.

3.1.1.3 Reference

1. Oconee QA-1 Licensing Basis and Generic Letter 83-28, Section 2.2.1, Subpart 1 Supplemental Response, submitted by J.W. Hampton (Duke) letter dated April 12, 1995 to Document Control Desk (NRC), Docket Nos. 50-269, -270, and -287.
2. GENERIC LETTER 83-28 SUPPLEMENTAL RESPONSE – OCONEE UNITS 1, 2, AND 3 (TAC NOS. M92023, M92024, AND M92025), submitted by Leonard A. Wiens (NRC, Office of Nuclear Reactor Regulation) dated August 3, 1995 to Mr. J.W. Hampton, Vice President, Oconee Site, Docket Nos. 50-269, 50-270 and 50-287.
3. OCONEE NUCLEAR STATION – NRC INSPECTION REPORT NOS. 50-269/00-12, 50-270/00-12, AND 50-287/00-12, submitted by Charles R. Ogle, Chief (NRC, Engineering Branch, Division of Reactor Safety) dated December 12, 2000 to Mr. W.R. McCollum, Vice President, Oconee Site, Docket Nos. 50-269, 50-270 and 50-287, License No: DPR-38 DPR-47, DPR-55.

3.1.2 Criterion 2 - Performance Standards (Category A)

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

Discussion

1. Essential Systems and Components

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

   a. Reactor Coolant System

   b. Reactor vessel internals

   c. Reactor Building

   d. Engineered Safeguards Systems

   e. Electric emergency power sources.

2. Natural Phenomena

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

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

   a. Earthquake

   b. Tornado - See details in Section 3.2.2

   c. Ground Water and Flood

   d. Wind and Hurricane

   e. Snow and Ice

   f. Other Local Site Effects

3.1.3 Criterion 3 - Fire Protection (Category A)

The reactor facility shall be designed: 1) to minimize the probability of events such as fires and explosions and, 2) to minimize the potential effects of such events to safety. Noncombustible and fire-resistant materials shall be used whenever practical throughout the facility, particularly

in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

Discussion

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

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

Electrical distribution equipment will be physically located to reduce vulnerability of vital circuits to physical damage as a result of accidents. Locations to achieve this result are described in Section 8.3.1.4.

3.1.4 Criterion 4 - Sharing of Systems (Category A)

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

Discussion

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

System	Shared by Units	Reference
Chemical Addition and Sampling	1, 2	9.3.2
Spent Fuel Cooling	1, 2	9.1.3
Reverse Osmosis System	1, 2	9.1.3
Liquid Waste Disposal	1, 2, 3	11.2.2
Gaseous Waste Disposal	1, 2	11.3.2
Solid Waste Disposal	1, 2, 3	11.4.1.2
Coolant Treatment	1, 2, 3	9.3.5
Recirculated Cooling Water	1, 2, 3	9.2.2.2.4
Low Pressure Service Water	1, 2	9.2.2.2.3
High Pressure Service Water	1, 2, 3	9.2.2.2.2
Control Room Ventilation	1, 2	9.4.1
Auxiliary Building Ventilation	1, 2	9.4.3
Turbine Building Ventilation	1, 2, 3	9.4.4
Area Radiation Monitoring	1, 2	12.3.3
Process Radiation Monitoring	1, 2	11.5
4.16 kV Standby Power Buses	1, 2, 3	8.3.1.1.3

System	Shared by Units	Reference
125/250 Volt DC Power System	1, 2, 3	8.3.2.1.2
120 Volt AC Vital Power System	1, 2, 3	8.3.2.1.4
120 Volt Regulated Power System	1, 2, 3	8.3.2.1.6
Auxiliary Steam System	1, 2, 3	10.3.2
Standby Shutdown Facility	1, 2, 3	9.6.1
Protected Service Water	1, 2, 3	9.7

3.1.5 Criterion 5 - Records Requirements (Category A)

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

Discussion

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

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

3.1.6 Criterion 6 - Reactor Core Design (Category A)

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

Discussion

The reactor is designed with the necessary margins to accommodate, without fuel damage, expected transients from steady-state operation including the transients given in the criterion. Fuel clad integrity is ensured under all normal and abnormal modes of anticipated operation by avoiding clad overstressing and overheating. The evaluation of clad stresses includes the effects of internal and external pressures, temperature gradients and changes, clad-fuel interactions, vibrations, and earthquake effects. Clad fatigue due to power and pressure cycling is minimized by pre-pressurizing with helium all fuel rods except those in the low burnup region of Core I, Oconee 1. The free-standing clad design prevents collapse at the end volume region

of the fuel rod and provides sufficient radial and end void volume to accommodate clad-fuel interactions and internal gas pressures (Section 4.2.2).

Clad overheating is prevented by satisfying the core thermal and hydraulic criteria shown in (Section 4.4.1).

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

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

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

The reactor coolant pumps are provided with sufficient inertia to maintain adequate flow to prevent fuel damage if power to all pumps is lost. Natural circulation coolant flow will provide adequate core cooling after the pump energy has been dissipated (Section 15.6).

3.1.7 Criterion 7 - Suppression of Power Oscillations (Category B)

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

Discussion

Power oscillations resulting from variations of coolant temperature are minimized by constant average coolant temperature when the reactor is operated above 15 percent power. Power oscillations from spatial xenon effects are minimized by the large negative power coefficient and axial power shaping rod assemblies.

The ability of the reactor control and protective system to control the oscillations resulting from variation of coolant temperature within the control system dead band and from spatial xenon oscillations has been analyzed. Variations in average coolant temperature provide negative feedback and enhance reactor stability during that portion of core life in which the moderator temperature coefficient is negative. When the moderator temperature coefficient is positive, rod motion will compensate for the positive feedback. The maximum rate of power change resulting from temperature oscillations within the control system dead band has been calculated to be less than 1 percent/minute. Since the unit has been designed to follow ramp load changes of 10 percent/minute, this is well within the capability of the control system (Section 7.6.1).

Control flexibility, with respect to xenon transients, is provided by the combination of control rods and nuclear instrumentation. Axial, radial, or azimuthal neutron flux changes will be

detected by the nuclear instrumentation. Individual control rods or groups of control rods can be positioned to suppress and/or correct flux changes (Section 4.3.2.2). The analysis of xenon-related power effects is presented in BAW-10010, "Stability Margin for Xenon Oscillation."

3.1.8 Criterion 8 - Overall Power Coefficient (Category B)

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

Discussion

The overall power coefficient is negative in the power operating range (Section 4.3.1).

3.1.9 Criterion 9 - Reactor Coolant Pressure Boundary (Category A)

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

Discussion

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

1. Material selection, design, fabrication, inspection, testing, and certification in accordance with ASME codes for all components excluding piping, which is done in accordance with the USAS B31.1 and B31.7 codes. The piping was redesigned to the 1983 ASME Code during the Steam Generator replacement project.
2. Manufacture and erection in accordance with approved procedures.
3. Inspection in accordance with code requirements plus additional requirements imposed by the manufacturer.
4. System analysis to account for cyclic effects of thermal transients, mechanical shock, seismic loadings, and vibratory loadings.
5. Selection of reactor vessel material properties to give due consideration to neutron flux effects and the resultant increase of the nil ductility transition temperature.

The materials, codes, cyclic loadings, and non-destructive testing are discussed further in Chapter 5.

3.1.10 Criterion 10 - Containment (Category A)

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

Discussion

Containment is provided by the Reactor Building. The Reactor Building has the capability to sustain, without loss of integrity, the effects of gross equipment failures, including the transient peak pressure associated with a hypothetical rupture of any pipe in the Reactor Coolant System including the effects of metal-water reactions described in Section 15.14.

The design parameters for the Reactor Building are tabulated in Section 3.8 and Engineered Safety Systems have been evaluated for various combinations of credible energy releases as

discussed in Section 15.14. Sufficient redundancy is provided both in equipment and control to ensure the functional availability and capability of systems required to protect the public.

3.1.11 Criterion 11 - Control Room (Category B)

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

Discussion

The reactors and associated equipment are controlled from panels located in the control rooms. The control rooms are designed to permit continuous occupancy following a maximum hypothetical accident (MHA) (Section 7.7.5).

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

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

The concrete Reactor Buildings and control room walls and roofs are designed to provide adequate protection against direct radiation to control room personnel at all times. Post-accident dose to control room personnel following the MHA is addressed in UFSAR Section 15.15.

The control rooms are provided with independent ventilation and filtration systems to minimize ingress of airborne radioactive contaminants escaping from the Reactor Building. The details of the control room ventilation system and its operation following an accident are described in Section 9.4.1.

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

Adequate instrumentation and controls are provided to maintain the reactor in Mode 3 (with T_{ave} ≥ 525°F) from outside the control room if access to the control room is lost or if the room must be evacuated temporarily in the unlikely event of a fire or other causes.

3.1.12 Criterion 12 - Instrumentation and Control Systems (Category B)

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

Discussion

Reactor regulation is based upon the use of movable control rods and a chemical neutron absorber (boron in the form of boric acid) dissolved in the reactor coolant. Input signals to the

reactor controls include reactor coolant average temperature, core thermal power demand, and reactor power. The reactor controls are designed to maintain a constant average reactor coolant temperature over the load range from approximately 15 to 100 percent of rated power. The steam system operates at constant pressure for all loads. Adequate instrumentation and controls are provided to maintain operating variables within their prescribed ranges (Section 7.7.2).

The non-nuclear instrumentation measures temperatures, pressures, flows, and levels in the Reactor Coolant System, Steam System, and Auxiliary Reactor Systems, and maintains these variables within prescribed limits (Section 7.4.2).

3.1.13 Criterion 13 - Fission Process Monitors and controls (Category B)

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

Discussion

This criterion is met by reactivity control means and control room display. Reactivity control is by movable control rods and by chemical neutron absorber (in the form of boric acid) dissolved in the reactor coolant. The position of each control rod will be displayed in the control room. Changes in the reactivity status due to soluble boron will be indicated by changes in the position of the control rods. Actual boron concentration in the reactor coolant is determined periodically by sampling and analysis (Sections 7.7.1 and 9.3.3.2).

3.1.14 Criterion 14 - Core Protection Systems (Category B)

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

Discussion

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

3.1.15 Criterion 15 - Engineered Safety Features Protection Systems (Category B)

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

Discussion

The Engineered Safeguards Protective System senses Reactor Coolant System pressure and Reactor Building pressure and initiates Emergency Core Cooling, Reactor Building isolation,

and Reactor Building cooling at the appropriate levels. It also initiates starting of the Standby Emergency Power Sources (Sections 6.3.2 and 8.3.1.1.3).

3.1.16 Criterion 16 - Monitoring Reactor Coolant Pressure Boundary (Category B)

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

Discussion

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

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

Gross leakage from the reactor coolant boundary will also be indicated by a decrease in pressurizer water level and a rapid increase in the Reactor Building sump water level (Section 5.2.3.8).

3.1.17 Criterion 17 - Monitoring Radioactivity Releases (Category B)

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

Discussion

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

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

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

3.1.18 Criterion 18 - Monitoring Fuel and Waste Storage (Category B)

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

Discussion

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

Level indicators are provided to alarm low water level in the spent fuel storage pool. Penetrations of the pool liner are arranged to prevent accidental drainage of the pool (Section 9.1.4.2.3)

Temperature sensors and flow monitors in the spent fuel pool cooling loop alarm on high temperature or loss of flow (Section 9.1.3).

Radiation monitors and alarms are provided in the Reactor Building, in all refueling areas, and in the waste storage and processing areas to warn operating personnel of excessive radiation levels (Section 12.3.3).

3.1.19 Criterion 19 - Protection Systems Reliability (Category B)

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

Discussion

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

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

Test connections and capabilities are built into the protective systems to provide for:

1. Pre-operational testing to give assurance that the protective systems can fulfill their required functions.
2. On-line testing to assure availability and operability (Section 7.1.2.1).

3.1.20 Criterion 20 - Protection Systems Redundancy and Independence (Category B)

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

Discussion

Reactor protection is by four channels with 2/4 coincidence, and engineered safeguards features are by three channels with 2/3 coincidence. All Protective System functions are

implemented by redundant sensors, instrument strings, logic, and action devices that combine to form the protective channels. Redundant protective channels and their associated elements are electrically independent and packaged to provide physical separation.

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The Reactor Protective System will determine action to be taken based on the type of module removed. These actions could range from indication of trouble within the system to a protective channel trip.

3.1.21 Criterion 21 - Single Failure Definition (Category B)

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

Discussion

The Protective Systems meet this criterion in that the instrumentation is designed so that a single event cannot result in multiple failures that would prevent the required protective action (Section 7.3).

3.1.22 Criterion 22 - Separation of Protection and Control Instrumentation Systems (Category B)

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

Discussion

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

3.1.23 Criterion 23 - Protection against Multiple Disability for Protection Systems (Category B)

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

Discussion

The Protective Systems are designed to extreme ambient conditions. The Protective Systems' instrumentation will operate from 40°F to 140°F and sustain the loss-of-coolant building environmental conditions, including 100 percent relative humidity, without loss of operability. Out-of-core neutron detectors, however, will withstand 90 percent relative humidity. The Protective Systems' instrumentation will be subject to environmental (qualification) testing as required by the proposed IEEE “Criteria for Nuclear Power Plant Protection Systems,” IEEE No. 279, dated August, 1968. Protective equipment outside the Reactor Building (control room and cable room) is designed for continuous operation in an ambient temperature and relative humidity representative of loss-of-coolant accident conditions (Section 7.1.2.1). The RPS / ESPS systems are also subject to IEEE Std 603-1998 “IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations”.

3.1.24 Criterion 24 - Emergency Power for Protection Systems (Category B)

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

Discussion

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

Redundant battery power is provided for vital instrumentation and control.

3.1.25 Criterion 25 - Demonstration of Functional Operability of Protection Systems (Category B)

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

Discussion

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

3.1.26 Criterion 26 - Protection Systems Fail-Safe Design (Category B)

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

Discussion

The Reactor Protective System will trip the reactor on loss of power. The Engineered Safeguards Protective System is supplied with multiple sources of electric power for control and valve action.

A total loss of power will not result in a trip condition. The loss of the affected channel signals will be indicated to the remaining channels and will not cause any trip condition on the system.

The system is designed for continuous operation under adverse environments, as described in the discussion of Criterion 23 (Sections 7.1.2.1 and 7.2).

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

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

3.1.27 Criterion 27 - Redundancy of Reactivity Control (Category A)

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

Discussion

This criterion is met by movable control rods Section 4.3.2, Section 7.6.1.1 and soluble boron poison (Section 4.3.2).

3.1.28 Criterion 28 - Reactivity Hot Shutdown Capability (Category A)

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

Discussion

A single reactivity control system consisting of 61 control rods is provided to rapidly make the core subcritical upon a trip signal. Trip levels are set to protect the core from damage due to the effects of any operating transient. The soluble absorber reactivity control system can add negative reactivity to make the reactor subcritical. However, its action is slow and its ability to protect the core from the damage, which might result from rapid load changes such as a full load turbine trip, is not a design criterion for this system. The high degree of redundancy in the Control Rod Drive System is considered sufficient to meet the intent of this criterion (Section 4.3.2 Section 7.6.1.1).

3.1.29 Criterion 29 - Reactivity Shutdown Capability (Category A)

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

Discussion

The reactor design meets this criterion both under normal operating conditions and under the accident conditions set forth in Chapter 15. The reactor is designed with the capability of providing a shutdown margin of at least 1 percent $\Delta k/k$ with the single most reactive control rod fully withdrawn at any point in core life with the reactor at a hot, zero power power condition. (Section 4.3.2.3). Table 4-6 illustrates a shutdown margin calculation for a sample Oconee fuel cycle.

3.1.30 Criterion 30 - Reactivity Holdown Capability (Category B)

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

Discussion

The reactor meets this criterion with control rods for hot shutdown under normal operating conditions and for shutdown under the accident conditions set forth in Chapter 15 except for the Steam Line Break Analysis. For details of this analysis refer to Section 15.13.

Reactor Shutdown margin is maintained during cooldown by increasing soluble boron concentration. The rate of reactivity compensation from boron addition is greater than the

reactivity change associated with the reactor cooldown rate of 100°F/hour. Thus, subcriticality can be maintained during cooldown with the most reactive control rod totally unavailable (Section 4.3.2).

3.1.31 Criterion 31 - Reactivity Control Systems Malfunction (Category B)

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

Discussion

The reactor design meets the intent of this criterion. A reactor trip will protect against any single malfunction of the reactivity control system. This conclusion is based on the analysis for a continuous rod group withdrawal accident (Section 15.3).

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

3.1.32 Criterion 32 - Maximum Reactivity Worth of Control Rods (Category A)

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

Discussion

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

3.1.33 Criterion 33 - Reactor Coolant Pressure Boundary Capability (Category A)

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

Discussion

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

3.1.34 Criterion 34 - Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention (Category A)

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

Discussion

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

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

3.1.35 Criterion 35 - Reactor Coolant Pressure Boundary Brittle Fracture Prevention (Category A)

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

Discussion

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

3.1.36 Criterion 36 - Reactor Coolant Pressure Boundary Surveillance (Category A)

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

Discussion

The reactor coolant pressure boundary components meet this criterion. Space is provided for non-destructive testing during plant shutdown. A reactor pressure vessel material surveillance program conforming to ASTM-E-185-66 has been established (Section 5.2.3.13).

3.1.37 Criterion 37 - Engineered Safety Features Basis for Design (Category A)

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

Discussion

The reactor design meets this criterion. The Emergency Core Cooling Systems can protect the reactor for any size leak up to and including the circumferential rupture of the largest reactor coolant pipe (Section 15.14).

3.1.38 Criterion 38 - Reliability and Testability of Engineered Safety Features (Category A)

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

Discussion

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

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

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

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

Testing and inspection of the Engineered Safeguards Systems is further described in Chapter 6.

3.1.39 Criterion 39 - Emergency Power for Engineered Safety Features (Category A)

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

Discussion

The electrical systems meet the intent of the criterion as discussed in Chapter 8.

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

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

3.1.40 Criterion 40 - Missile Protection (Category A)

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

Discussion

Engineered safety features are defined as Engineered Safeguards Systems. Engineered Safeguards System features are redundant. Engineered Safeguards Systems at Oconee are protected against dynamic effects and missiles resulting from hypothesized plant equipment failures. In general, missile protection for Oconee is described in Section 3.5. Two basic categories of plant equipment failure are hypothesized and considered in the Oconee design:

1. Missiles generated inside Containment - Assumptions and design requirments for missiles generated inside containment are described in Section 3.5.1.1.
2. Missiles generated by a main turbine failure - Assumptions and design requirements for missiles generated by a main turbine failure are described in Section 3.5.1.2.

3.1.41 Criterion 41 - Engineered Safety Features Performance Capability (Category A)

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

Discussion

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

3.1.42 Criterion 42 - Engineered Safety Features Components Capability (Category A)

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

Discussion

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

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

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

3.1.43 Criterion 43 - Accident Aggravation Prevention (Category A)

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

Discussion

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

3.1.44 Criterion 44 - Emergency Core Cooling Systems Capability (Category A)

At least two Emergency Core Cooling Systems, preferably of different design principles, each with a capability for accomplishing abundant emergency core cooling, shall be provided. Each Emergency Core Cooling System and the core shall be designed to prevent fuel and clad damage that would interfere with the emergency core cooling function and to limit the clad

metal-water reaction to negligible amounts for all sizes of breaks in the reactor coolant pressure boundary, including the double-ended rupture of the largest pipe. The performance of each Emergency Core Cooling System shall be evaluated conservatively in each area of uncertainty. The systems shall not share active components and shall not share other features or components unless it can be demonstrated that: a) the capability of the shared feature or component to perform its required function can be readily ascertained during reactor operation, b) failure of the shared feature or component does not initiate a loss-of-coolant accident, and c) capability of the shared feature or component to perform its required function is not impaired by the effects of a loss-of-coolant accident and is not lost during the entire period this function is required following the accident.

Discussion

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

3.1.45 Criterion 45 - Inspection of Emergency Core Cooling Systems (Category A)

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

Discussion

All critical parts of the Emergency Core Cooling Systems, including the reactor vessel internals, can be inspected during plant shutdown (Section 5.2.3.12).

3.1.46 Criterion 46 - Testing of Emergency Core Cooling Systems Components (Category A)

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

Discussion

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

3.1.47 Criterion 47 - Testing of Emergency Core Cooling Systems (Category A)

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

Discussion

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

3.1.48 Criterion 48 - Testing of Operational Sequence of Emergency Core Cooling Systems (Category A)

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

Discussion

The operational sequence that would bring the Emergency Core Cooling Systems into action, including transfer to alternate power sources, can be tested in parts (Sections 6.3.4 and 7.3).

3.1.49 Criterion 49 - Containment Design Basis (Category A)

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

Discussion

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

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

3.1.50 Criterion 50 - NDT Requirement for Containment Material (Category A)

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

Discussion

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

All principal load-carrying components of ferritic materials for the containment vessel exposed to the external environment have been selected and tested to confirm that their ductile-to-brittle-transition (NDT) temperature is at least 30°F below the minimum service metal temperature. The ferritic materials exposed to the external environment consist of the penetrations and large

openings (equipment access hatch and personnel locks), for which materials have been selected to conform with ASME Boiler and Pressure Vessel Code, Section III, for Class "B" Vessels. Material specifications for the penetrations are more completely described in Section 3.8.1.1.

3.1.51 Criterion 51 - Reactor Coolant Pressure Boundary outside Containment (Category A)

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

Discussion

The reactor coolant pressure boundary is defined as those piping systems or components which contain reactor coolant at high pressure and temperature. With the exception of the normal reactor coolant sampling line and the reactor coolant post accident liquid sample line, the reactor coolant pressure boundary, as defined above, is located entirely within the Reactor Building. These sampling lines are provided with remotely operated valves for isolation. The normal reactor coolant sampling line is used only during actual sampling operations. The reactor coolant post accident liquid sample line is used during performance testing of the post accident liquid sampling system and/or actual sampling operations (Section 9.3.6). No significant environmental dose would result from these sources (Sections 6.2.3, 11.2.2.)

3.1.52 Criterion 52 - Containment Heat Removal Systems (Category A)

Where active heat removal systems are needed under accident conditions to prevent exceeding containment design pressure, at least two systems, preferably of different principles, each with full capacity, shall be provided.

Discussion

Two systems of different principles are provided to remove heat from each Reactor Building following an accident. The systems are discussed in Sections 6.2 and 6.2.2. Analysis of peak accident pressure in containment following an accident is addressed in Sections 6.2.1.3 and 6.2.1.4 respectively. The analysis shows containment to be capable of withstanding peak accident pressure without the Reactor Building Spray System, or Reactor Building Cooling System.

The Reactor Building Cooling System removes heat by circulating building atmosphere over cooling coils.

The Reactor Building Spray System supplies droplets of cool, borated water which absorb sensible and latent heat from the containment atmosphere.

3.1.53 Criterion 53 - Containment Isolation Valves (Category A)

Penetrations that require closure for the containment function shall be protected with redundant valving and associated apparatus.

Discussion

Piping penetrations that require closure under accident conditions are provided with double barriers so that no single credible failure or malfunction could result in a loss of isolation. Valves

are manually, electrically or pneumatically operated. Check valves are used in certain applications. All isolation valves inside the Reactor Building requiring remote operation are electrically operated. As an alternative to valves, other types of apparatus which provide a suitable barrier for containment isolation may be utilized. Examples of such mechanisms include, but are not limited to flanges and closed loop piping systems that are designed to remain intact when containment isolation is required.

3.1.54 Criterion 54 - Containment Leakage Rate Testing (Category A)

Containment shall be designed so that an integrated leakage rate testing can be conducted at design pressure after completion and installation of all penetrations and the leakage rate measured over a sufficient period to verify its conformance with required performance.

Discussion

The Reactor Buildings are designed so that leakage rate can be determined at design pressure after completion and installation of all penetrations. The leak-rate test will verify that the maximum integrated leak rate does not exceed the design leakage rate (Section 3.8.1.7.3).

3.1.55 Criterion 55 - Containment Periodic Leakage Rate Testing (Category A)

The containment shall be designed so that integrated leakage rate testing can be done periodically at design pressure during plant lifetime.

Discussion

The Reactor Building has been structurally designed to permit integrated leakage rate testing at design pressure (Section 3.8.1.7.4).

3.1.56 Criterion 56 - Provisions for Testing of Penetrations (Category A)

Provisions shall be made for testing penetrations which have resilient seals or expansion bellows to permit leak tightness to be demonstrated at design pressure at any time.

Discussion

All Reactor Building penetrations with resilient seals or expansion bellows are constructed so that they may be pressurized to design pressure for leak tests at any time (Section 3.8.1.7.4 and Section 3.8.1.5.4).

3.1.57 Criterion 57 - Provisions for Testing of Isolation Valves (Category A)

Capability shall be provided for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valves leakage does not exceed acceptable limits.

Discussion

All remotely operated valves serving an Engineered Safeguards function have the capability for testing their functional operability. These tests can be conducted from the control rooms.

Isolation valves that are required to be closed from an Engineered Safeguards signal have test provisions for leak testing (Table 6-7).

3.1.58 Criterion 58 - Inspection of Containment Pressure - Reducing Systems (Category A)

Design provisions shall be made to facilitate the periodic physical inspection of all important components of the containment pressure-reducing systems, such as pumps, valves, spray nozzles, torus, and sumps.

Discussion

Provision is made to permit periodic physical inspection of components of the two containment pressure-reducing systems, the Reactor Building Spray System and the Reactor Building Cooling System. The Reactor Building spray pumps and the valves and operators associated with piping in each of these systems are located outside the Reactor Building, permitting the inspection of these components. The fan units of the Reactor Building cooling units are located so that physical inspection is possible during normal operation.

The cooling coils of the Reactor Building cooling units can be inspected during shutdown. The spray header and nozzles of the Reactor Building Spray System, located in the dome of the Reactor Building, can be inspected visually during shutdown. The sumps can be inspected and the strainers cleaned during shutdown.

3.1.59 Criterion 59 - Testing of Containment Pressure-Reducing System Components (Category A)

The containment pressure-reducing systems shall be designed so that active components, such as pumps and valves can be tested periodically for operability and required functional performance.

Discussion

The containment pressure-reducing systems have the capability of being periodically tested as follows:

1. Reactor Building Cooling Units
   a. The air fans can be individually tested for low speed operation.
   b. The cooling coil low pressure service water valves can be operated through their full travel with resulting flow alarm indication.
   c. The stand-by low pressure service water pumps can be tested for automatic starting.
2. Reactor Building Spray System
   a. The operation of the spray pumps can be tested by recirculating to the borated water storage tank through a test line.
   b. The building spray isolation valves can be operated through their full travel.

3.1.60 Criterion 60 - Testing of Containment Spray Systems (Category A)

A capability shall be provided to periodically test the delivery capability of the Containment Spray System at a position as close to the spray nozzles as is practical.

Discussion

The delivery capability of the spray nozzles will be tested by blowing low pressure air through the system and verifying flow through the nozzles.

The delivery capability of the pumps will be tested by recirculating to the borated water storage tank and monitoring the resultant flow.

3.1.61 Criterion 61 - Testing of Operational Sequence of Containment Pressure Reducing Systems

A capability shall be provided to test, under conditions as close to the design as practical, the full operational sequence that would bring the containment pressure-reducing systems into action including the transfer to alternate power sources.

Discussion

Each of the three redundant 4 kV switchgear buses supplying power to essential loads receives its power from two 4 kV main feeder buses. These main feeder buses are supplied by: 1) the main unit auxiliary transformers, 2) the startup transformer, and 3) the underground feeder from Keowee Hydro plant. Each main feeder bus is fed from each of the three sources above. In normal operation the two main feeders will be supplied through breakers from the unit auxiliary transformer and the breakers from the start-up transformer and the underground feeder will be open.

To test the transfer to alternate power source, the three breakers associated with one of the main feeders will be placed in test position with the normal breaker closed and the two alternate power sources breakers open. A low voltage simulation will be used to trip the normal breaker and close the start-up breaker. A low voltage and an Engineered Safeguards (ESG) simulation will be used to trip the start-up breaker and close the underground feeder breaker. In making these tests, the automatic dropping of load will not take place.

Testing the two independent channels for the Reactor Building Cooling System and the Building Spray System by inserting an analog signal can be accomplished without placing the systems in operation.

3.1.62 Criterion 62 - Inspection of Air Cleanup Systems

Design provisions shall be made to facilitate physical inspection of all critical parts of containment air cleanup systems such as ducts, filters, fans, and dampers.

Discussion

The Penetration Room Ventilation System (PRVS) was originally addressed by this design criterion. Due to the adoption of the Alternate Source Term, PRVS is no longer required and this design criterion no longer applies.

3.1.63 Criterion 63 - Testing of Air Cleanup System Components

Design provisions shall be made so that active components of the Air Cleanup Systems, such as fans and dampers, can be tested periodically for operability and required functional performance.

Discussion

The Penetration Room Ventilation System (PRVS) was originally addressed by this design criterion. Due to the adoption of the Alternate Source Term, PRVS is no longer required and this design criterion no longer applies.

The Control Room Ventilation System (CRVS) was not originally addressed by this design criterion. Operation and Maintenance of CRVS is addressed by the QA 5 program. Testing of CRVS is addressed by the Ventilation Filter Test Program.

3.1.64 Criterion 64 - Testing of Air Cleanup Systems

A capability shall be provided for in situ periodic testing and surveillance of the Air Cleanup Systems to ensure; a) filter bypass paths have not developed and, b) filter and trapping materials have not deteriorated beyond acceptable limits.

Discussion

The Penetration Room Ventilation System (PRVS) was originally addressed by this design criterion. Due to the adoption of the Alternate Source Term, PRVS is no longer required and this design criterion no longer applies.

The Control Room Ventilation System (CRVS) was not originally addressed by this design criterion. Operation and Maintenance of CRVS is addressed by the QA 5 program. Testing of CRVS is addressed by the Ventilation Filter Test Program.

3.1.65 Criterion 65 - Testing of Operational Sequence of Air Cleanup Systems (Category A)

A capability shall be provided to test under conditions as close to design as practical the full operational sequence that would bring the Air Cleanup Systems into action including the transfer to alternate power sources and the design air flow delivery capability.

Discussion

The Penetration Room Ventilation System (PRVS) was originally addressed by this design criterion. Due to the adoption of the Alternate Source Term, PRVS is no longer required and this design criterion no longer applies.

The Control Room Ventilation System (CRVS) was not originally addressed by this design criterion. Operation and Maintenance of CRVS is addressed by the QA 5 program. Testing of CRVS is addressed by the Ventilation Filter Test Program.

3.1.66 Criterion 66 - Prevention of Fuel Storage Criticality (Category B)

Criticality in new and spent fuel storage shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.

Discussion

Criticality of new or spent fuel is prevented by limiting the fuel assembly array size and limiting assembly interaction by fixing the mimimum separation between assemblies. Fuel assemblies cannot be placed in other than the prescribed locations (Section 9.1.2).

3.1.67 Criterion 67 - Fuel and Waste Storage Decay Heat (Category B)

Reliable Decay Heat Removal Systems shall be designed to prevent damage to the fuel in storage facilities that could result in radioactivity release to plant operating areas or the public environs.

Discussion

This criterion is met by the Spent Fuel Cooling System which incorporates provisions to maintain water cleanliness, temperature, and water level. Three pumps and three coolers will be adequate to maintain the spent fuel pool temperature within acceptable limits. The pumps in the system can be operated from the standby bus in case of loss of outside power to provide continuous cooling capability in the fuel storage facility (Section 9.1.3).

3.1.68 Criterion 68 - Fuel and Waste Storage Radiation Shielding (Category B)

Shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities to meet the requirements of 1OCFR20.

Discussion

Shielding meeting the requirements of 10CFR 20 is provided for protection of operating personnel:

1. During all phases of spent fuel removal and storage (Section 12.3.2).
2. From radioactive waste holdup tanks and other containers containing potentially radioactive solutions, resins, or gases (Section 12.3.2).

3.1.69 Criterion 69 - Protection against Radioactivity Release from Spent Fuel and Waste Storage (Category B)

Containment of fuel and waste storage shall be provided if accidents could lead to release of undue amounts of radioactivity to the public environs.

Discussion

Analyses in Chapter 15 have demonstrated that accidental release of the maximum activity content of a tank containing waste gases or liquids will not cause excessive off-site doses. The fuel handling accident, analyzed in Chapter 15 does not result in excessive off-site doses.

3.1.70 Criterion 70 - Control of Releases of Radioactivity to the Environment (Category B)

The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified: a) on the basis of 10CFR 20 requirements for normal operations and for any transient situation that might reasonably be anticipated to occur and b) on the basis of 10CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence except that reduction of the recommended dosage levels may be required where high population densities or very large cities can be affected by the radioactive effluents.

Discussion

The waste disposal system is designed to insure that station personnel and the general public are protected against excessive exposure to radioactive material in accordance with the regulations of 10CFR 20 and 10CFR 50 Appendix I.

The gaseous, liquid, and solid waste storage facilities are discussed in Chapter 11 where it is demonstrated that adequate holdup capacity is provided. Gaseous and liquid wastes will be sampled before release and will be monitored for activity level at all times during release, or

independent sampling and analysis will be performed prior to release when the appropriate monitor is out of service.

Control of leakage following a reactor accident is accomplished by the Reactor Building. Experience has shown that Reactor Building leakage is more likely at penetrations than in liner plates or weld joints. Prior to the adoption of the alternate source term, 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 processiong 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.

The release of radioactive materials produced by a reactor accident or waste gas tank failure are within the guidelines set by 10CFR 100.

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3.2 Classification of Structures, Components, and Systems

3.2.1 Seismic Classification

3.2.1.1 Structures

The design bases for normal operating conditions are governed by the applicable building design codes. The basic design criterion for the worst loss-of-coolant accident and seismic conditions is that there shall be no loss of function if that function is related to public safety.

AEC publication TID 7024, "Nuclear Reactors and Earthquake," as amplified in Chapter 3 is used as the basic design guide for seismic analysis.

The design basis earthquake ground acceleration at the site is 0.05g. The maximum hypothetical earthquake ground acceleration is 0.10g. and 0.15g for Class 1 structures founded on bedrock and overburden respectively.

The plant structures are classified as one of three classes according to their function and the degree of integrity required to protect the public.

3.2.1.1.1 Class 1

Class 1 structures are those which prevent uncontrolled release of radioactivity and are designed to withstand all loadings without loss of function. Class 1 structures include the following:

Portions of the Auxiliary Building that house engineered safeguards systems, control room, fuel storage facilities and radioactive materials.

Reactor Building and its penetrations.

CT4 Transformer and 4KV Switchgear Enclosures (Blockhouses) (Reference Section 8.3.1.4.1.)

Unit Vent.

Standby Shutdown Facility (SSF) (Reference Section 9.6.3.4.1.)

Protected Service Water (PSW) Building (Reference Section 9.7.3.5).

Note: From the license renewal review, it was determined that Class 1 civil structures are included in the scope for license renewal.

3.2.1.1.2 Class 2

Class 2 structures are those whose limited damage would not result in a release of radioactivity and would permit a controlled plant shutdown but could interrupt power generation. Class 2 structures include the following:

Oconee Intake Structure
Oconee Turbine and Auxiliary Buildings, except as included in Class 1
Oconee Intake Canal Dike
Oconee Intake Underwater Weir
Keowee Powerhouse
Keowee Spillway
Keowee Service Bay Substructure

Keowee Breaker Vault
Keowee Intake Structure
Keowee Power and Penstock Tunnels
Keowee Dam
CCW Intake Piping
CCW Discharge Piping
ECCW Piping (Structural Portion outside of Turbine Building)
Little River Dam and Dikes
Essential Siphon Vacuum System Intake Dike Trench
Essential Siphon Vacuum Cable Trench
Essential Siphon Vacuum Building

Note: From the license renewal, it was determined that Class 2 civil structures are included in the scope for license renewal.

3.2.1.1.3 Class 3

Class 3 structures are those whose failure could inconvenience operation, but which are not essential to power generation, orderly shutdown or maintenance of the reactor in a safe condition. They include all structures not included in Classes I and 2.

3.2.1.2 Components and Systems

Capability is provided to shutdown safely all three units in the event of a maximum hypothetical earthquake. Major equipment and portions of systems that can withstand the maximum hypothetical earthquake are identified in Section 3.2.2.

3.2.1.3 Seismic Loading Conditions

The design basis earthquake ground acceleration at the site is 0.05g. The maximum hypothetical earthquake ground acceleration is 0.10g. and 0.15g for Class 1 structures founded on bedrock and overburden respectively.

The terms Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) are sometimes referred to within the UFSAR.

Operating Basis Earthquake (OBE) is equivalent to Design Basis Earthquake (DBE).

Safe Shutdown Earthquake (SSE) is equivalent to Maximum Hypothetical Earthquake (MHE).

3.2.2 System Quality Group Classification

This section defines the design criteria used with respect to the loss-of-coolant accident (LOCA), and natural phenomena and also explains the division of components and piping into classifications related to design and function. These criteria are as follows:

A maximum hypothetical earthquake will not result in a loss-of-coolant accident (LOCA), but the simultaneous occurrence of these events will not result in loss of function to vital safety related components or systems. The simultaneous occurrence of the maximum hypothetical earthquake and a LOCA is only a design criteria. A LOCA is not postulated to occur simultaneously with a maximum hypothetical earthquake during accident analysis. In addition, pipe failures during a maximum hypothetical earthquake are not postulated as part of the accident analysis.

A tornado will not be allowed to cause a LOCA.

A tornado does not occur simultaneously with or following a LOCA.

A tornado and earthquake do not occur simultaneously.

An earthquake can occur simultaneously with a loss of offsite power.

A turbine missile will not be allowed to cause a LOCA.

A turbine missile does not occur simultaneously with a LOCA.

The following design objectives result from consideration of the design criteria:

1. Loss-of-Coolant Accident

   Capability is provided to assure necessary protective actions, including reactor trip and operation of the Emergency Core Cooling System, to protect the public during a LOCA, even in the event of a simultaneously occurring maximum hypothetical earthquake.

2. Turbine Missile Accident

   The Reactor Coolant System will not be damaged by a turbine missile. Capability is provided to safely shutdown the affected units.

3. Earthquake

   Major equipment and portions of systems that can withstand the maximum hypothetical earthquake include the following:

   a. Reactor Coolant System.

   b. Borated water storage tank and piping to high pressure and low pressure injection pumps and Reactor Building spray pumps.

   c. HP injection pumps and piping to Reactor Coolant System.

   d. LP injection pumps, LP injection coolers and piping to both Reactor Coolant System and Reactor Building spray pumps.

   e. Core flood tanks and piping to Reactor Coolant System.

   f. Reactor Building spray pumps, piping to spray headers, and the spray headers.

   g. Reactor Building coolers.

   h. Low pressure service water (LPSW) pumps, LPSW piping to LP injection coolers and Reactor Building coolers and LPSW piping from these coolers to the condenser circulating water (CCW) discharge.

   i. CCW intake structure, CCW pumps, pump motors, CCW intake piping to the LPSW pumps, also through the condenser and emergency CCW discharge piping and CCW discharge piping.

   j. Upper surge tanks, and piping to the emergency feedwater pump.

   k. Emergency feedwater pump and turbine and auxiliary feedwater piping to the steam generators.

   l. Main steam lines to and including turbine stop valves. Turbine bypass system up thru Main Steam System isolation valves, and steam supply lines to the emergency feedwater pump turbine.

   m. Penetration Room Ventilation System. (not required to operate for accident mitigation due to adoption of alternate source terms) (Reference 3)

n. Reactor Building penetrations and piping through isolation valves.

o. Siphon Seal Water System.

p. Essential Siphon Vacuum System.

q. Electric power for above.

r. Nitrogen supply to the EFW control valves FDW-315 and FDW-316.

Information relating to the seismic design of SSF systems and components is contained in Section 9.6.4.1 and 9.6.4.3. Information relating to the seismic design of the PSW System and its components are contained in Section 9.7.

4. Tornado

The Reactor Coolant System will not be damaged by a tornado. A loss of Reactor Coolant Pump (RCP) seal integrity was not postulated as part of the tornado design basis. Capability is provided to shutdown safely all three units.

The Reactor Coolant System, by virtue of its location within the Reactor Building, is protected from tornado damage. A sufficient supply of secondary side cooling water for safe shutdown is assured by Protected Service Water Pumps located in the Auxiliary Building and taking suction from Oconee 2 CCW intake piping. Redundant and diverse sources of secondary makeup water are credited for tornado mitigation. These include: 1) the other units' EFW Systems, 2) the PSW pumps, and 3) the SSF ASW pump.

Protected or physically separated lines are used to supply cooling water to each steam generator. The sources of power to the PSW pumps are the Keowee Hydro Station and the Central Tie Switchyard via a 100 kV transmission line to a 100/13.8 kV substation.

An external source of cooling water is not immediately required due to the large quantities of water stored underground in the intake and discharge CCW piping. The stored volume of water in the intake and discharge lines below elevation 791ft would provide sufficient cooling water for all three units for at least 30 days after trip of the three reactors.

Although not fully protected from tornadoes, the following sources provide reasonable assurance that a sufficient supply of primary side makeup water is available during a tornado initiated loss of offsite power.

a. The SSF Reactor Coolant Makeup Pump can take suction from the Spent Fuel Pool. The pump can be supplied power from the SSF Diesel.

b. A High Pressure Injection Pump can take suction from either the Borated Water Storage Tank or the Spent Fuel Pool. Either the “A” or “B” High Pressure Injection Pump can be powered from the PSW Switchgear.

Protection against tornado is an Oconee design criteria, similar to the criteria to protect against earthquakes, wind, snow, or other natural phenomena described in UFSAR section 3.1.2. A specific occurrence of these phenomena is not postulated, nor is all equipment that would be used to bring the plant to safe shutdown comprehensively listed. The statement, “Capability is provided to shutdown safely all three units” is intended to be a qualitative assessment that, after a tornado, normal shutdown systems will remain available or alternate systems will be available to allow shutdown of the plant. It was not intended to imply that specific systems should be tornado-proof. As part of the original FSAR development, specific accident analyses were not performed to prove this judgement, nor were they requested by the NRC. Subsequent probabilistic studies have confirmed that the original qualitative assessments were correct. The

risk of not being able to achieve safe shutdown after a tornado is sufficiently small that additional protection is not required.

In addition, there was considerable correspondence between Duke and NRC in the years post-TMI discussing Oconee's ability to survive tornado generated missiles. Based upon the probability of failure of the EFW and Station ASW systems combined with the protection against tornado missiles afforded by the SSF ASW system, the NRC concluded that the secondary side decay heat removal function complied with the criterion for protection against tornadoes.

3.2.2.1 System Classifications

Plant piping systems, or portions of systems, are classified according to their function in meeting design objectives. The systems are further segregated depending on the nature of the contained fluid. For those systems which normally contain radioactive fluids or gases, the Nuclear Power Piping Code, USAS B31.7 and Power Piping Code USAS, B31.1.0 are used to define material, fabrication, and inspection requirements.

Diagrams for each system are included in the FSAR sections where each system is described.

Fabrication and erection of piping, fittings, and valves are in accordance with the rules for their respective classes. Welds between classes of systems (Class I to II, I to III, or II to III) are performed and inspected in accordance with the rules for the higher class. This preceding sentence does not apply to valves where the class break has been determined to occur at the valve seat, and to pipe with 1" nominal diameter and less.

In-line instrument components such as turbine meters, flow nozzle assemblies, and control valves, etc. are classified with their associated piping unless their penetration area is equal to or less than that of a I inch i.d. pipe of appropriate schedule for the system design temperature and pressure, in which case they are placed in Class III. Definitions of the three classes are listed below:

Class I

This class is limited to the Reactor Coolant System (RCS) and Reactor Coolant Branch lines, as described herein. The Reactor Coolant Branch lines include connecting piping out to and including the first isolation valve. This section of piping is Class I in material, fabrication, erection, and supports and restraints. A Class I analysis of the piping to the first isolation valve has been completed for the following systems:

1. High Pressure Injection (Emergency Injection)
2. High Pressure Injection (Normal Injection)
3. High Pressure Injection (Letdown)
4. Low Pressure Injection (Decay Heat Removal Drop-line)
5. Low Pressure Injection (Core Flood)
6. Reactor Coolant Drain Lines
7. Pressurizer Spray
8. Pressurizer Relief Valve Nozzles

Modifications that affect the Reactor Coolant System and the Class I portion of the branch lines must demonstrate that the impact on the Class I piping is acceptable. The impact may be assessed by performing a Class I analysis or by other conservative techniques to assure Class I allowable limits are not exceeded. Isolation valves can be either stop, relief, or check valves. Piping 1 inch and less is excluded from Class I.

Class II

Class II systems, or portions of systems, are those whose loss or failure could cause a hazard to plant personnel but would represent no hazard to the public. Class II systems normally contain radioactive fluid whose temperature is above 212°F, and in addition, those portions of Engineered Safeguards Systems outside the Reactor Building which may see recirculated reactor building sump water following a LOCA. Piping I inch and less is excluded.

Class III

Class III systems, or portions of systems, are those which would normally be Class II except that the contained fluid is less than 212°F. Valves, piping, instrument fittings and thermowells with a penetration area equal to or less than a 1 inch i.d. pipe or less (all schedules) are placed in Class III regardless of system temperature or pressure, when such equipment is connected to Class I, II, or III systems.

3.2.2.2 System Piping Classifications

System piping is divided into eight classes, depending on the required function of the system or portion of a system. These eight piping classes result from the combination of the preceding system classifications with and without design for seismic loading, as indicated in Table 3-1. Piping classes A through C meet the intent of USAS B31.7 Nuclear Power Piping Code (February 1968) and Addenda (June 1968) with the exception of those portions of the code which lack adequate definition for complete application. The Class I RCS piping was redesigned to the 1983 ASME Code (No Addenda) during the Steam Generator replacement project.

Code Applicability: Due to the numerous code references located throughout this UFSAR, no attempt is made to revise these references as Codes are amended, superseded or substituted. Consequently, the station piping specifications should be relied upon to determine applicable codes. The existing Code references are the basis for design and materials; however, it is Duke Power Company's intent to comply with portions of, or all of, the latest versions of existing Codes unless material and/or design commitments have progressed to a stage of completion such that it is not practical to make a change. When only portions of Code Addenda are utilized, the appropriate engineering review of the entire addenda will be made to assure that the overall intent of the Code is still maintained. Detailed information for each station unit and code applicability with respect to design, material procurement, fabrication techniques, Nondestructive Testing (NDT) requirements and material traceability for each piping system class is described in the station piping specifications.

Table 3-1 applies uniformly to all piping except auxiliary systems in the Reactor Building. Due to schedule commitments, and concern over lack of definitive design guidance in B31.7, it was decided to use B31.1 and applicable nuclear cases in the Reactor Building, but the materials were bought, erected, and inspected to the standards set down in B31.7. The Reactor Coolant System was designed to B31.7, Class I. The Class I portion of the connecting piping to the RCS will have Class I analyses completed by August 31, 1999 (See Section 3.2.2.1). The Class I RCS piping was redesigned to the 1983 ASME Code (No Addenda) during the Steam Generator replacement project.

Oconee has a number of systems that were designed to USAS B31.7 Class II and Class III and to USAS B31.1.0 requirements [Reference Table 3-1]. Piping analyses for these systems include stress range reduction factors to provide conservatism in the design to account for thermal cyclic operations. Thermal fatigue of mechanical systems designed to USAS B31.7 Class II and Class III and to USAS B31.1 is considered to be a time-limited aging analysis because all six of the criteria contained in Section 54.3 of Reference 4 Section 3.12.1 are satisfied.

From the license renewal review, it was determined that the existing analyses of thermal fatigue of these mechanical systems are valid for the period of extended operation.

3.2.2.3 System Valve Classifications

In the absence of definitive codes, the non-destructive testing criteria applied to system valves are consistent with the intent of Par. 1-724 of USAS B31.7 Nuclear Power Piping Code (Feb. 1968) and the piping classification applicable to that portion of the system which includes the valve. On this basis, valves are grouped into the same eight classes as shown for piping in Table 3-1, and a valve is in the same class as the portion of system piping which includes the valve.

Code Applicability: Due to the numerous code references located throughout this UFSAR, no attempt is made to revise these references as Codes are amended, superseded, or substituted. Consequently, the station specifications applicable to a given valve should be relied upon to determine applicable codes.

3.2.2.4 System Component Classification

In the absence of definitive codes, the design criteria applied to pressure retaining system components are generally consistent with the intent of Sections III and VIII of the ASME Boiler and Pressure Vessel Code, the piping system classification applicable to that portion of the system which includes the component, and the required function of the component. Atmospheric water storage tanks important to safety conform to American Waterworks Association Standard for Steel Tanks, Standpipes, Reservoirs and Elevated Tanks for Water Storage, D100, or equivalent.

Components are listed by system in Table 3-2. This tabulation shows the code to which the component was designed, whether the component was designed to withstand the seismic load imposed by the maximum hypothetical earthquake, and the analytical technique employed in seismic analysis.

Code Applicability: Due to the numerous code references located throughout the UFSAR, no attempt is made to revise these references as codes are amended, superseded, or substituted. Consequently, the station specifications applicable to a given component should be relied upon to determine appplicable codes.

3.2.3 Reference

1. *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.
2. 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.
3. License Amendment No. 338, 339, and 339 (date of issuance - June 1, 2004); Adoption of Alternate Source Term.
4. License Amendment No. 386, 388, and 387 (date of issuance - August 13, 2014); Implementation of the Protected Service Water System.

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3.3 Wind and Tornado Loadings

All Class 1 structures, except those structures not exposed to wind, are designed to withstand the effects of wind and tornado loadings, without loss of capability of the systems to perform their safety functions.

3.3.1 Wind Loadings

3.3.1.1 Design Wind Velocity

The design wind velocity for all Class 1 structures is 95 mph. This is the largest wind velocity for a 100-year occurrence as shown in Figure 1(b) of Reference 1.

3.3.1.2 Determination of Applied Forces

The applied wind pressures are computed by the means outlined in ASCE Paper 3269 which states that the equivalent static force on a building is equal to the dynamic pressure (q) times the drag coefficient (C_D) multiplied by the elevation area. The dynamic pressure is the product of one-half the air density and the square of the velocity (the kinetic energy per unit volume of moving air). For air at 15° C at 760 mm Hg: $q = 0.002558\ V^2$ with q in psf and V in mph. The drag coefficient is based on test data and tabulated in Reference 1. For these high wind velocities, this equation may be excessively conservative, but no credit is taken for this possible pressure reduction.

3.3.2 Tornado Loadings

All Class 1 structures, except those structures not exposed to wind, are designed for tornado loads.

3.3.2.1 Applicable Design Parameters

Simultaneous external loadings used in the tornado design of Class I structures, with the exception of the Standby Shutdown Facility, are:

a. Differential pressure of 3 psi developed over 5 seconds.

b. External wind forces resulting from a tornado having a velocity of 300 mph.

The spectrum and characteristics of tornado-generated missiles is covered in Section 3.5.1.3.

Tornado loading parameters for the Standby Shutdown Facility are described in Section 9.6.3.1.

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 plant’s licensing basis in the 4th quarter of 2007. The design of new systems (and their associated components and/or structures) that are required to resist tornado loadings will conform to the tornado wind, differential pressure, and missile criteria specified in Regulatory Guide 1.76, Revision 1.

3.3.2.2 Determination of Forces on Structures

Tornado wind loadings are calculated in accordance with Section 3.3.1.2, using the tornado wind velocities given in Section 3.3.2.1. The tornado loading combination used for design of Class 1 structures is:

$$Y = \frac{1}{\phi}(1.0D + 1.0W_t + 1.0P_i)$$

Where Y, ϕ, and D are as defined in Table 3-14.

W_t	=	Stress induced by design tornado wind velocity (drag, lift and torsion)
P_i	=	Stress due to differential pressure

Shape factors will be applied in accordance with ASCE Paper 3269. No height or gust factors will be used with tornado loadings.

3.3.2.3 Effect of Failure of Structures or Components Not Designed for Tornado Loads

This information is described in Section 3.2.2

3.3.2.4 Wind Loading for Class 2 and 3 Structures

The wind loads are determined from the largest wind velocity for a 100-year occurrence as shown in Figure 1(b) of Reference 1. This is 95 mph at the site.

3.3.3 References

1. *Wind Forces on Structures*, Task Committee on Wind Forces, ASCE Paper No. 3269.
2. Regulatory Guide 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," Revision 1.

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3.4 Water Level (Flood) Design

3.4.1 Flood Protection

3.4.1.1 Flood Protection Measures for Seismic Class 1 Structures

The plant yard elevation is 796.0 ft. msl. All of the man-made dikes and dams forming the Keowee Reservoir rise to an elevation of 815.0 ft. msl with a full pond elevation of 800.0 ft. msl. See Section 2.4.2.2 for exceptions to the elevation of 815.0 ft. However, Class 1 structures and components are not subject to flooding since the Probable Maximum Flood (PMF) would be contained by the Keowee Reservoir. The minimum external access elevation for the Auxiliary, Turbine, and Service Buildings is 796.5 ft. msl which provides a 6 inch water sill. Also, the plant site is provided with a surface water drainage system that protects the plants facilities from local precipitation. As part of a long term compensatory action (Reference 5), stacked sandbags, drain plugs, and sheet metal have been installed or staged at various areas of the plant identified as being vulnerable to an updated Probable Maximum Precipitation rainfall height.

3.4.1.1.1 Current Flood Protection Measures for the Turbine and Auxiliary Buildings

The following information describes internal flood attributes beneficial to the management of flooding, but not required for the design basis flood. Many of these attributes have been considered in the Probabilistic Risk Assessment.

In the current Turbine Building flood handling analysis, it was found that there exists a remote possibility of flooding in the Turbine Building at the basement level due to failure of expansion joints in the Condenser Circulating Water System near the condenser water box inlet or outlet nozzles.

Condenser circulating water intake and discharge pipes are embedded in the Turbine Building substructure mat at points immediately below the inlet or outlet connections on the condenser inlet and outlet water boxes. At each waterbox connection, a 78 inch steel pipe is turned up and projected above the basement level and connected to a butterfly valve. A rubber expansion joint is located between each valve and waterbox connection. The rubber joint spans across a 4¼ inch physical gap in the 78 inch intake pipe and across a 2 inch physical gap in the 78 inch discharge. At maximum flow conditions through any condenser, a complete rupture of the 4¼ inch intake pipe point (all rubber removed) would result in a 235 cfs leak into the Turbine Building basement area. This is the worst case leak condition due to the higher head and wider possible gap situation that exists on the intake side of the condenser.

Each foot of depth in this 202 feet wide by 790 feet long structure contains a volume of 160,000 cubic feet. Therefore, a joint rupture would fill the Turbine Building at the rate of 0.088 feet per minute until the water surface reaches the height of the rupture and a reduced rate thereafter due to reduced differential head conditions, provided all flood water could be contained in the Turbine Building.

Deleted paragraph(s) per 2010 update.

Early licensing correspondences documented that possible Turbine Building floods could be isolated by the operators before safety related equipment was impacted. (See References 6 and 7.) ONS installed curbs and TB sump level alarms to provide operators adequate time to isolate the flood and contain the water in the TB. Subsequently, the Standby Shutdown Facility (SSF) was installed and became the licensed method for mitigating a TB Flood. See Section 9.6

To prevent transmission of flood water from the Turbine Building to the Auxiliary Building, the Turbine/Auxiliary Building wall along column line "N" is capable of withstanding a flood to a depth of 20 ft. above elevation 775 + 0. Six doors originally located on this wall have been made flood barriers. Three of the doors are permanently sealed while the remaining three have been replaced with "submarine type" flood doors. All other penetrations through the wall to elevation 795 + 0 have been sealed.

A Turbine Building Flood Statalarm is provided in the Unit 1 and 2 control room to indicate flood conditions in the Turbine Building basement. This alarm has a 2 out of 4 logic. The emergency procedure is entered immediately upon receipt of a turbine building flood "emergency high" alarm from the detectors mounted at elevation 775 ft. 6 inches (6 inches above the floor). Immediate actions include tripping all three units and stopping all CCW pumps.

A push button in each control room provides capability to close the Condenser Circulating Water (CCW) pump discharge valves to protect against CCW siphoning into the turbine building basement. This flood mitigation station modification has been installed pursuant to the recommendations made in the Oconee Probabilistic Risk Assessment Study.

It is desirable to allow a limited amount of backflow from the CCW discharge through the condensate coolers during a flood to provide suction for Low Pressure Service Water (LPSW) pumps and the Standby Shutdown Facility Auxiliary Service Water (SSF ASW) pump. Temperature control valves 2CCW-84 and 3CCW-84 have had their air supplies disconnected and clamps have been installed on the valves, effectively failing them in the open position (See Figure 9-9). This does not apply to Unit 1 since there are no service water or SSF suction requirements on the Unit 1 system.

The Auxiliary Building could be subject to flooding from a single break in any one of four non-seismic sources: the high pressure service water system (source for fire protection), the non-seismic portions of the low pressure service water system (the ventilation cooling water), the plant drinking water system, and the non-seismic portions of the Reverse Osmosis System when it is connected to the Unit 1 & 2 spent fuel pool or the Unit 1 or Unit 2 borated water storage tank. The high pressure service water Unit 1, 2, and 3 hatch and Unit 1 drumming station sprinkler systems are not considered flood sources based on the results of realistic seismic analyses that demonstrate the pipes and supports will not fail during a seismic event. The remaining portions of the non-seismic high pressure service system, the non-seismic portions of the low pressure service water system and the plant drinking water system are isolated or flow limited to allow operators sufficient time to identify and isolate the source. The Reverse Osmosis System can be isolated in time to prevent loss of safety-related equipment when aligned to the Unit 1 & 2 spent fuel pool or the Unit 1 or Unit 2 borated water storage tank. Operator actions are directed by abnormal operating procedures. Operator response times were tested to ensure flood mitigation can occur before safety related equipment is adversely affected. Flooding by these sources will be detected through the procedural response to a seismic event or high level alarm sensors (non-seismic) in the auxiliary building sumps.

3.4.1.1.2 Flood Protection Measures Inside Containment

The primary means for detecting leakage in the Reactor Building is the level indication for the normal sump. This indication has a range of 0-to-30 inches, with a statalarm occurring at 8 inches increasing level and a computer alarm at approximately 10 inches. These alarms would alert the operators in the control room such that appropriate actions could be taken. In addition to the alarms, sump level is input to the plant computer and is logged to the alarm log. Level is also recorded on a trend recorder in each control room. Safety related redundant level transmitters with a range of 3 inches to 24 inches are also provided in the normal sump. Both

transmitter levels are indicated in the control room on receiver gauges and one train is recorded. Thus, the operators have several methods for monitoring changes in sump level.

The sump fill rate is routinely measured to determine leakage rate. The sump capacity is 15 gallons per inch of height below embedded piping and each graduation on the indicator level indicates 1.5 gallons of leakage into the sump. A 1 gal/min leak would therefore be detectable within less than 10 minutes.

In addition to the normal sump level, indication of the emergency sump level is also provided by redundant safety related systems with a range of 0 to 3 feet. Both trains of instrumentation are indicated on receiver gauges in the control room and one train is recorded. This indication can be used in conjunction with the normal sump level indication to detect abnormal leakage in the Reactor Building. Two additional trains of containment level transmitters are installed in each Reactor Building to provide wide range level indication and recording with a range of 0 to 15 feet.

The normal sump is routinely pumped to the miscellaneous waste holdup tanks whenever the alarm point (8 inches) is reached. Pumping of the sump water is started manually, but terminates automatically when the sump level has dropped to 1 inch (which clears the statalarm). Each time the sump is pumped, it is recorded in the Unit Reactor Operator's Log Book. During pumping, a decreasing sump level indication and/or increasing miscellaneous waste holdup tank level indication can be used to verify flow from the normal sump. The flow rate from the sump can be determined using the rate of change in sump level.

In order to provide periodic monitoring of sump levels, the recording of normal and emergency sump levels is done daily. Daily monitoring of level indications is useful in confirming that level instrumentation are operable, while verifying the sump pumps are operable and maintaining the sump level at or below the alarm point. Calibration of the normal and emergency sump indications is performed during refueling.

In the event of increased leakage to the Reactor Building, sampling may be performed to determine the origin of the leakage (e.g., LPSW, feedwater, component cooling, or RC system).

Leakage from the LPSW system in containment can also be detected by the monitoring of other parameters. For example, the inlet and outlet LPSW flows for each Reactor Building Cooling Unit (RBCU) are monitored for any differences which could be indicative of a cooler leak. If a flow difference is detected, an alarm is provided to the control room. The operator can then promptly isolate the affected cooler by closing remote operated valves.

The Reactor Coolant Pump (RCP) motor parameters are also continuously monitored. A leak in the motor stator winding cooler would be alarmed in the control room. A leak in either of the motor bearing oil coolers could be detected by changing motor temperature in conjunction with increasing sump level. The pump could then be stopped and the cooling water isolated from the control room.

The component cooling system is designed to provide cooling water for various inside containment components. In-leakage of reactor coolant is detected by a radiation monitor and an increase in surge tank level which will be annunciated. Out-leakage from the system will result in a decreasing surge tank level which is annunciated. Volume of the surge tank is 50 ft^3 and allows relatively small volumes of in-leakage or out-leakage to be observed.

3.4.2 References

1. Elevations taken from Figure 2-2 of FSAR and Oconee FSAR 2.2.6.

2. Response to Question of Effects of Failure of Non-Category I Equipment, Oconee FSAR, Supplement 13 of January 29, 1973, Item No. 7347. Information received from Steam Department.

3. Response to Bulletin 80-24 on Cooling Systems Inside Containment, Attachment to Mr. W. O. Parker, Jr.'s letter of January 6, 1981, Item No. 760. Information received from Steam Department.

4. Deleted Per 2001 Update.

5. Engineering Change 108176, Install Sandbags and Other Devices as Protection Enhancement for a Probable Maximum Precipitation Rainfall Event.

6. Letter from AEC to Duke Energy, dated September 26, 1972, requesting that Oconee evaluate failures similar to Quad Cities expansion joint failure.

7. Letter from Duke Energy to AEC, dated October 24, 1972, responding to the Quad Cities expansion joint failure.

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3.5 Missile Protection

3.5.1 Missile Selection and Description

3.5.1.1 Internally Generated Missiles (Inside Containment)

The major components including reactor vessel, reactor coolant piping, reactor coolant pumps, steam generators, and the pressurizer are located within three shielded cubicles. Each of two cubicles contain one steam generator, two coolant pumps, and associated piping. One of the cubicles also contains the pressurizer. The reactor vessel is located within the third cubicle or primary shield. The reactor vessel head and control rod drives extend into the fuel transfer canal.

Penetrations in the generators, piping, and the pressurizer are located such that missiles which may be generated, such as valves, valve bonnets, valve stems, or reactor coolant temperature sensors will not escape the cubicles or possess sufficient energy to damage the Reactor Building liner plate.

Openings are provided in the lower shield walls to provide vent area. To assure that no missile will impact on the Reactor Building liner plate, concrete shielding is provided for the liner plate area opposite the openings. The shielding extends beyond the openings so that any missile will impact on the shields.

Pipe lines carrying high pressure injection water are routed outside the shield walls entering only when connecting to the loop. Missiles which may be generated in one cubicle cannot rupture high pressure injection lines for the other loop. Low pressure injection lines and core flooding lines are routed outside of the shield walls, behind missile shield walls, and through the primary shield where they enter the reactor vessel. They are, thus, protected from missiles which might be generated in either cubicle.

A concrete missile shield is located above the control rod drives to stop a control rod drive should it become a missile. The shield is removed during refueling.

The reactor cavity annulus seal ring and biological shield plugs are analyzed as potential missiles following a postulated Core Flood Line pipe rupture inside the reactor vessel cavity. The analysis indicates that the seal ring and plugs will not reach a sufficient height to become destructive missiles. Breaks of the RCS inside the reactor vessel cavity are not considered in the missile evaluation due to the successful application of a Leak-Before-Break analysis.

Items that could become missiles are oriented so they impinge on concrete surfaces.

Analysis of the missile penetration is based on the methods described in Nav. Docks P-51, Design of Protective Structures by Amirikan (Bureau of Yards and Docks, August 1950).

The penetration formulae are:

$$D = kApV'$$

where:

$$V' = \log_{10} 1 + \frac{V^2}{215000}$$

$$K = \frac{D^1}{D} = 1 + e^{-4(a^1 - 2)}$$

where:

$$a^1 = \frac{T}{D}$$

where:

D	=	Penetration in a slab of infinite thickness (ft.)
D^1	=	Penetration in a slab of thickness “T” (ft.)
T	=	Thickness of slab (ft.)
Ap	=	Sectional pressure, obtained by dividing the weight of missile by its cross sectional area (psf)
V	=	Velocity of missile (fps)
k	=	Material's coefficient, in our case, k=2.30 x 10^{-3} for reinforced concrete

Formulae for determining energy loss due to drag:

$$\frac{Ti}{Tc} = \frac{1}{1 + \frac{2Tc}{WL}};$$

$$L = 2\frac{W}{SAC_d}$$

Note: The above equation was revised in 1995 update.

where:

A	=	Average area
C_d	=	Drag coefficient (Cd=1.0 in our case)
Ti	=	Kinetic energy on impact
Tc	=	Kinetic energy after leaving casing
W	=	Weight in lbs.
S	=	Air density = 0.074 #/ft^3

In addition to the penetration calculation, the overall structural strength of the removable concrete slabs, its supports and anchors are analyzed based on the research paper “Impact Effect of Fragments Striking Structural Elements” by R. A. Williamson and R. R. Alvy.

The following three missiles are used to design the removable concrete slabs:

Description	Wt. Lbs.	Imp. Area In^2	Velocity FPS	Kin. Energy Ft-lbs.
C. R. Drive Assembly	1500	64.0	254	1.49 x 10^6
CRD Vent Cap w/Valve	55	13.4	546	0.12 x 10^6

Description	Wt. Lbs.	Imp. Area In^2	Velocity FPS	Kin. Energy Ft-lbs.
CRD Motor and Clutch Assem.	750	47.0	483	1.35×10^6

The properties of other missiles postulated by the Nuclear Steam System Supply (NSSS) vendor are given in Table 3-3 to Table 3-9.

Missile protection is provided to comply with the following criteria:

1. The Reactor Building and liner are protected from loss of function due to damage by such missiles as might be generated in a loss-of-coolant accident for break sizes up to and including the double-ended severance of a main coolant pipe.
2. The engineered safeguards system and components required to maintain Reactor Building integrity are protected against loss of function due to damage by the missiles defined below.

During the detailed plant design, the missile protection necessary to meet the above criteria was developed and implemented using the following methods:

1. Components of the Reactor Coolant System are examined to identify and to classify missiles according to size, shape and kinetic energy for purposes of analyzing their effects.
2. Missile velocities are calculated considering both fluid and mechanical driving forces which can act during missile generation.
3. The Reactor Coolant System is surrounded by reinforced concrete and steel structures designed to withstand the forces associated with double-ended rupture of a main coolant pipe and designed to stop missiles.
4. The structural design of the missile shielding takes into account both static and impact loads and is based upon the state of the art of missile penetration data.

The types of missiles for which missile protection is provided are:

1. Valve stems.
2. Valve bonnets.
3. Instrument thimbles.
4. Various types and sizes of nuts and bolts.

Protection is not provided for certain types of missiles for which postulated accidents are considered incredible because of the material characteristics, inspections, quality control during fabrication, and conservative design as applied to the particular component. Included in this category are missiles caused by massive, rapid failure of the reactor vessel, steam generator, pressurizer, main coolant pump casings and drives.

3.5.1.2 Turbine Missiles

The turbine-generator supplier has made a study of failure of rotating elements of steam turbines and generators. The postulated types of failures are: (1) failure of rotating components operating at or near normal operating speed and, (2) failure of components that control admission of steam to the turbine resulting in destructive shaft rotational speed.

3.5.1.2.1 Failure at or Near Operating Speed

All of the known turbine and generator rotor failures at near rated speed resulted from the combination of severe strain concentrations in relatively brittle materials. New alloys and processes have been developed and adopted to minimize the probability of brittle fracture in rotors, wheels, and shafts. Careful control of chemistry and detailed heat treating cycles have greatly improved the mechanical properties of all of these components. Transition temperatures (the temperature at which the character of the fracture in the steel changes from brittle to ductile, often identified as FATT) have been reduced on the low temperature wheel and rotor applications for nuclear units to well below startup temperatures. Improved steel mill practices in vacuum pouring and alloy addition have resulted in forgings which are much more uniform and defect free than ever before. More comprehensive vendor and manufacturer tests involving improved ultrasonic and magnetic particle testing techniques are better able to discover surface and internal defects than in the past. Laboratory investigation has revealed some of the basic relationships between structure strength, material strength, FATT and defect size, and location so that the reliability of the rotor as a structure has been significantly improved over the past few years.

New starting and loading instructions have been developed to reduce the severity of surface and bore thermal stress cycles incurred during service. The new practices include:

1. Better temperature sensors.
2. Better control devices for acceleration and loading.
3. Better guidance for station operators in the control speed, acceleration, and loading rates to minimize rotor stresses.

Progress in design, better materials and quality control, more rigorous acceptance criteria, and improved machine operation have substantially reduced the likelihood of burst failures of turbine-generator rotors operating at or near rated speed.

3.5.1.2.2 Failure at Destructive Shaft Rotational Speeds

Improvements of rotor quality discussed above, while reducing the chance of failures at operating speed, tend to increase the hazard level associated with unlimited overspeed because of higher bursting speed. Therefore, turbine overspeed protection systems have been evaluated as follows:

1. Main and secondary steam inlets have the following valves in series:
   a. Control valves - controlled by the speed governor and tripped closed by emergency governor and backup overspeed trip, thus providing three levels of control redundancy.
   b. Stop valves or trip throttle valve - actuated by the emergency governor and backup overspeed trip, thus providing two levels of control redundancy.

      Since 1948 there have been over 650 turbines, of over 10,000 kw each, placed in service by the Oconee turbine supplier with no report of main stop valves failing to close when required to protect the turbine. Impending sticking has been disclosed by means of the fully closed test feature so that a planned shutdown could be made to make the necessary correction. This almost always involves the removal of the oxide layer which builds up on the stem and bushing and which would not occur on a low temperature nuclear application.

c. Combined stop and intercept valves in cross around systems - these are actuated by the speed governor, emergency, and backup overspeed trips. These valves also include the testing features described above.

The speed sensing devices for the governor and emergency governor are separate from each other, thus providing two independent lines of defense.

2. Uncontrolled Extraction Lines to Feedwater Heaters

   If the energy stored in an uncontrolled extraction line is sufficient to cause a dangerous overspeed, two positive closing nonreturn valves are provided, to be actuated by the emergency governor and backup overspeed trip. These are designed for remote manual periodic tests to assure proper operation. The station piping, heater, and check valve system are reviewed during the design stages to make sure the entrained steam cannot overspeed the unit beyond safe limits.

Special field tests are made of new components to obtain design information and to confirm proper operation. These include the capability of controls to prevent excessive overspeed on loss of load.

Careful analysis of all past failures has led to design, inspection, and testing procedures to substantially eliminate destructive overspeed as a possible cause of failure in modern design units.

The study of postulated ruptures made by the turbine-generator supplier concludes that the missile having the highest combination of weight, size, and energy is the last stage wheel. The properties of this missile are summarized in Table 3-10. Initial velocities and energies shown in the table are based on 180 percent overspeed. As the missile penetrates the casing, 50 percent of the initial energy is considered absorbed in the casing.

Analysis of the above missile is based on calculations using methods presented in Reference 1 to determine the depth to which this missile would penetrate the concrete Reactor Building. Conservatively, no reduction of missile energy is made for penetration of the Turbine Building and/or impact with intervening equipment and structural components after leaving turbine shell. The energy loss from 23.25 x 10^6 ft-lbs to 18.0 x 10^6 ft-lbs is caused by air friction. This effect has been calculated by using a drag coefficient of 1.0. Since the offset between the Turbine and Reactor Buildings is relatively short, about 170 feet, no account has been taken for air friction losses for the case in which the missile is ejected nearly horizontally to strike the cylinder wall. Following are results of analysis:

Case I:

“Side on” impact. Missile could penetrate the concrete cylinder wall to a depth of approximately 6 inches and the dome to a depth of approximately 5½ inches. The tendons will not be damaged since they are protected to a depth of 7 3/4 inches in the cylinder wall and 8 inches in the dome.

Case II:

“End on” impact. In this case the missile could penetrate the concrete cylinder wall to a depth of approximately 13 3/4 inches and the dome to a depth of approximately 12¼ inches. The tendon arrangement is such that the missile could strike two adjacent tendons in the dome or a maximum of three horizontal and one vertical tendons in the cylinder wall. The local effect on the tendons could be one of either partial deflection or possible severance. However, analysis of the structure indicates that the structure can withstand the loss of three horizontal and three vertical tendons in the cylinder wall or five adjacent tendons in the dome without loss of function and a greater number of tendons without building failure.

Case III:

As a final analysis, an extreme case is considered in which none of the initial kinetic energy of the missile is absorbed by its penetration through the turbine casing. The total initial energy of 46.5×10^6 ft-lbs is available for penetration of the cylinder wall and 29.3×10^6 ft-lbs for penetration of dome where the reduction is due to air friction only. The maximum depth of penetration of cylinder wall is 35½ inches and the dome is 25 inches. The missile can strike five tendons in the dome or three horizontal and one vertical tendons in the cylinder wall. The local effect in the impact area would be as described in Case II above even though the depth of penetration is greater.

Depths of penetration of Reactor Building wall are summarized in Table 3-11.

Since the thicknesses of the cylinder wall and dome are 45 inches and 39 inches respectively, it can be seen that the turbine missile, even under extreme assumptions, does not penetrate the Reactor Building.

3.5.1.2.3 Application of Turbine Missile Design to Engineered Safeguards Systems

Low Trajectory Turbine Missiles

1. If the engineered safety feature is located outside of the missile strike zone as defined in Reg. Guide 1.115 Revision 1, no additional protection is required.
2. If the engineered safety feature is located within the missile strike zone, evaluate the probability of the engineered safety feature being struck and damaged by an equipment failure per Regulatory Guide 1.115 Revision 1, "Protection against Low-Trajectory Turbine Missiles", and NUREG 0800, Revision 2, "Standard Review Plan", Section 3.5.1.3. Should the probability of that particular engineered safety feature being struck and damaged be less than that specified, no protection would be required or provided.
3. Should the probability of the engineered safety feature being struck and damaged be greater than that specified, protection would be provided in the form of physical separation or shielding. A minimum of seven feet of separation, as viewed from the missile generation point on the turbine, constitutes adequate physical separation for low trajectory turbine missiles.

High Trajectory Turbine Missiles

High trajectory turbine missiles are characterized by their nearly vertical trajectories. Missiles ejected more than a few degrees from the vertical, either have sufficient speed such that they land offsite, or their speeds are low enough so that their impact on most plant structures is not a significant hazard.

1. The probability of a high trajectory turbine missile landing within a few hundred feet from the turbine is on the order of 10^{-7} per square foot of horizontal surface area. Consequently the risk from high trajectory turbine missiles is insignificant unless the vulnerable target area is on the order of 10^4 square feet or more.
2. Should the probability of the engineered safety feature being struck and damaged be greater than that specified, protection would be provided in the form of physical separation or shielding. A minimum of seven feet of separation, as shown in the plan view, constitutes adequate physical separation for high trajectory turbine missiles.

3.5.1.3 Missiles Generated by Natural Phenomena

For an analysis of missiles created by a tornado having maximum wind speeds of 300 mph, two missiles are considered. One is a missile equivalent to a 12 foot long piece of wood 8 inches in diameter traveling end on at a speed of 250 mph. The second is a 2000 pound automobile with a minimum impact area of 20 square feet traveling at a speed of 100 mph.

For the wood missile, calculations based on energy principle indicate that because the impact pressure exceeds the ultimate compressive strength of wood by a factor of about four, the wood would crush due to impact. However, this could cause a secondary source of missiles if the impact force is sufficiently large to cause spalling of the free (inside) face. The compressive shock wave which propagates inward from the impact area generates a tensile pulse, if it is large enough, will cause spalling of concrete as it moves back from the free (inside) surface. This spalled piece moves off with some velocity due to energy trapped in the material. Successive pieces will spall until a plane is reached where the tensile pulse becomes smaller than the tensile strength of concrete. From the effects of impact of the 8 inch diameter by 12 foot long wood missile, this plane in a conventionally reinforced concrete section would be located approximately 3 inches from the free (inside) surface. However, since the Reactor Building is prestressed, there will be residual compression in the free face, as the tensile pulse moves out and spalling will not occur. Calculations indicate that in the impact area a 2 inch or 3 inch deep crushing of concrete should be expected due to excessive bearing stress due to impact.

For the automobile missile, using the same methods as in the turbine failure analysis, the calculated depth of penetration is ¼ inch and for all practical purposes the effect of impact on the Reactor Building is negligible.

From the above, it can be seen that the tornado generated missiles neither penetrate the Reactor Building wall nor endanger the structural integrity of the Reactor Building or any components of the Reactor Coolant System.

Additional tornado missile requirements were subsequently imposed by NRC post-TMI on Emergency Feedwater Systems. ONS met these requirements based upon the probability of failure of the EFW and station ASW Systems combined with the protection against tornado missiles afforded the SSF ASW System. Subsequently, PSW replaced station ASW relative to this function. See UFSAR Sections 3.2.2 and 10.4.7.3.6 for additional information.

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 plant’s licensing basis in the 4th quarter of 2007. The design of new systems (and their associated components and/or structures) that are required to resist tornado loadings will conform to the tornado wind, differential pressure, and missile criteria specified in Requlatory Guide 1.76, Revision 1

3.5.2 Barrier Design Procedures

The Reactor Building and Engineered Safeguards Systems components are protected by barriers from all credible missiles which might be generated from the primary system. Local yielding or erosion of barriers is permissible due to jet or missile impact provided there is no general failure.

The final design of missile barrier and equipment support structures inside the Reactor Building is reviewed to assure that they can withstand applicable pressure loads, jet forces, pipe

reactions and earthquake loads without loss of function. The deflections or deformations of structures and supports are checked to assure that the functions of the Reactor Building and engineered safeguards equipment are not impaired. Missile barriers are designed on the basis of absorbing energy by plastic yielding.

3.5.3 References

1. Amirikian, A., Design of Protective Structures, Bureau of Yards and Docks, Department of the Navy, *NAVDOCKS P-51*, 1950.
2. Alvy, R. R., and Willimson, R. A., "Impact Effect of Fragments Striking Structural Elements."
3. Regulatory Guide 1.115, Revision 1, "Protection Against Low-Trajectory Turbine Missiles, dated July 1977.
4. Internal Duke Memorandum from Robert E. Miller to P.N. Hall et al, titled "Turbine Missile Properties", dated June 3, 1970.
5. NUREG 0800, Revision 2, "Standard Review Plan", Section 3.5.1.3, dated July 1981.
6. Letter from D. W. Montgomery (B&W) to W. H. Owen (Duke) regarding Potential Reactor Building Missiles, dated November 14, 1967.
7. Calculation BWC-006K-B932 (OSC-8433), Weight, Impact Area and Velocity, and Kinetic Energy of ROTSG Missiles, May 20, 2004, Rev.0.
8. Regulatory Guide 1.76, “Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants,” Revision 1.

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3.6 Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping

3.6.1 Postulated Piping Failures in Fluid Systems Inside and Outside Containment

3.6.1.1 Design Bases

The basic design criteria for pipe whip protection is as follows:

1. All penetrations are designed to maintain containment integrity for any loss of coolant accident combination of containment pressures and temperatures.
2. All penetrations are designed to withstand line rupture forces and moments generated by their own rupture as based on their respective design pressures and temperatures.
3. All primary penetrations, and all secondary penetrations that would be damaged by a primary break, are designed to maintain containment integrity.
4. All secondary lines whose break could damage a primary line and also breach containment are designed to maintain containment integrity.

Pipe whip restraints, jet impingement shields and other protective devices do not have to be installed to protect against an instantaneous double ended rupture of a large RCS pipe based on LBB analyses and technology. Per References 4 and 5, the NRC has approved the use of the LBB approach to eliminate the need to protect against the dynamic effects of large bore pipe breaks, as established in previous topical report submittals in References 6, 7, and 8.

3.6.1.2 Description

The major components including reactor vessel, reactor coolant piping, reactor coolant pumps, steam generators, and the pressurizer are located within three shielded cubicles. Each of two cubicles contain one steam generator, two coolant pumps, and associated piping. One of the cubicles also contains the pressurizer. The reactor vessel is located within the third cubicle or primary shield. The reactor vessel head and control rod drives extend into the fuel transfer canal.

Openings are provided in the lower shield walls to provide vent area. Pipe lines carrying high pressure injection water are routed outside the shield walls entering only when connecting to the loop.

3.6.1.2.1 Core Flood/Low Pressure Injection System

After implementation of the passive Low Pressure Injection (LPI) cross connect modification on each Oconee Unit, the pipe rupture design basis of Core Flood (CF) / LPI system inside containment is based on the system function during full power operations. The CF section (defined as the “A” and “B” train piping downstream of LP-176 and LP-177 respectively) qualifies as high energy during full power operations. For this CF piping, up to but not including the CF / Reactor Vessel nozzles, Leak Before Break technology was employed to eliminate the dynamic effects associated with postulated breaks (Refer to Section 5.2.1.9). For the LPI section of the system (defined as the “A” and “B” train piping upstream of LP-176 and LP-177 to their respective Reactor Building penetrations, and including the cross connect piping between

the “A” and “B” trains), USNRC Standard Review Plan Section 3.6.2 Branch Technical Position MEB 3-1 (Reference 3) was used for treatment of postulated pipe ruptures.

3.6.1.3 Protected Service Water (PSW) System

The PSW System is designed as a standby system for use under emergency conditions. With the exception of testing of the system, the system is not normally pressurized. Testing of the system is infrequent, typically every quarter. In addition, the duration of the test configuration is short, compared to the total plant (unit) operating time. Due to the combination of the infrequent testing and the short duration of the test, pipe ruptures are not postulated or evaluated for the PSW System.

3.6.1.4 Safety Evaluation

The analysis of effects resulting from postulated piping breaks outside containment is contained in Duke Power MDS Report No. OS-73.2, dated April 25, 1973 including revisions through supplement 2.

An evaluation of potential non-safety grade control system interactions during design basis high energy line break accidents is contained in the Duke Power/Babcock and Wilcox Report dated October 5, 1979.

An exception to report OS-73.2, extending the time allowed to align HPI after certain secondary piping breaks from 30 minutes to 1 hour, has been evaluated as acceptable.

The Reverse Osmosis Unit was added after MDS Report No. OS-73.2 was completed. It contains high energy piping that has been evaluated to have acceptable results.

3.6.2 References

1. Duke Power MDS Report No. OS-73.2, dated April 25, 1973 including revisions through supplement 2.
2. Duke Power/B&W Report, Oconee Nuclear Station, "Evaluation of Potentially Adverse Environmental Effects on Non-Safety Grade Control Systems", October 5, 1979.
3. USNRC Standard Review Plan (NUREG 0800) Section 3.6.2 Branch Technical Position MEB 3-1.
4. NRC Safety Evaluation of B&W Owners Group Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops, dated December 12, 1985.
5. NRC Safety Evaluation Relating To Elimination of Dynamic Effects of Postulated Primary Loop Pipe Ruptures from Design Basis in Regard to TMI-1, dated November 5, 1987.
6. B&W Topical Report BAW-1847, Revision 1, “Leak-Before-Break Evaluation of Margin Against Full Break for RCS Primary Piping of B&W Designed NSS,” September 1985.
7. B&W Topical Report BAW-1889P, “Piping Material Properties for Leak-Before-Break Analysis,” October 1985.
8. B&W Topical Report BAW-1999, “TMI-1 Nuclear Power Plant Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping”, April 1987.

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3.7 Seismic Design

3.7.1 Seismic Input

3.7.1.1 Design Response Spectra

The design response spectra curves for the 0.05g Design Base Earthquake (DBE), the 0.10g Maximum Hypothetical Earthquake (MHE) for Class 1 Structures founded on rock, and the 0.15g Maximum Hypothetical Earthquake (MHE) for structures founded on overburden are given in Figure 2-51, Figure 2-53, Figure 2-55, respectively.

3.7.1.2 Design Time History

The Time History record of the N-S, May 1940 El Centro earthquake is used (vertical and N-S horizontal components).

3.7.1.3 Critical Damping Values

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The following damping values are used for the seismic design of Class 1 structures:

	Percent of Critical Damping		
Item	**OBE**	**SSE**	**LOCA**
Welded carbon and stainless steel assemblies (This includes reactor internals, supports and similar weldments.)	1	1	4
Steel frame structures (Both welded and high strength bolted)	2	2	7
Reinforced concrete equipment supports	2	2	7
Reinforced concrete frames and buildings	5	5	7
Prestressed concrete structures under earthquake forces	2	5	5
Vital piping	0.5	0.5	3

3.7.1.4 Supporting Media for Seismic Class 1 Structures

The supporting media for each seismic Class 1 structure are defined in Section 2.5.

3.7.1.5 Response to Generic Letter 87-02

Generic Letter 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue (USI) A-46," was issued because the NRC concluded that the seismic adequacy of certain equipment in operating plants must be reviewed against seismic criteria developed during the resolution of Unresolved Safety Issue (USI) A-46. The concern was that equipment in nuclear plants with construction permit applications docket before 1972 may not be adequately qualified to ensure its survival and functionality in the event of a safe shutdown earthquake (SSE) because the equipment was not reviewed against current licensing criteria for seismic equalification of equipment (Regulatory Guide 1.100, IEEE 344-

1975, and NUREG-0800). This is a backfit seismic evaluation under 10 CFR.50.109 since the current criteria was not in use when the plants were licensed

The NRC determined that it is not feasible to require older operating plants to meet new licensing requirements that were not in use when plants were licensed. Threfore, an alternative method was selected to verify the seismic capability of equipment. This alternative method used a compilation of existing earthquake experience data supplemented by test data as the basis to verify the seismic capability of equipment. Generic Letter 87-02 allowed the seismic verification to be accomplished by utilities through a generic program, and the Seismic Qualification Utility Group (SQUG) was formed. The SQUG developed a Generic Implementation Procedure (GIP) that documents the seismic verification process, procedures, and methodologies for verifying the seismic qualification of equipment and resolving USI A-46. Supplement 1 of Generic Letter 87-02 (Reference 11) endorsed use of the GIP for the seismic qualification process and contained revised licensee actions.

Oconee performed the seismic qualification process in accordance with the NRC endorsed version of the GIP. In a Safety Evaluation Report (Reference 12), the NRC concluded that Oconee met the purpose and intent of the seismic qualification process and that the corrective actions and modifications provide sufficient basis to close the USI A-46 review at Oconee.

Additional discussions related to Generic Letter 87-02 can be found in seismic qualification discussions in Sections 3.9.2.2, 3.10.1, 3.10.2, 8.3.1.4.6.1, and 10.4.7.1. The seismic verification process is considered part of the seismic licensing basis for Oconee, so the seismic qualification criteria developed by the SQUG in response to Generic Letter 87-02 must be considered during mechanical and electrical equipment modifications

3.7.2 Seismic System Analysis

3.7.2.1 Seismic Analysis Methods

3.7.2.1.1 Reactor Building

Seismic loading of the structure controls in all cases over that of tornado or wind loading. The seismic analysis is conducted in the following manner: The loads on the Reactor Building caused by earthquake are determined by a dynamic analysis of the structure. The dynamic analysis is made on an idealized structure of lumped masses and weightless elastic columns acting as spring restraints. The analysis is performed in two stages: the determination of the natural frequencies of the structure and its mode shapes, and the response of these modes to the earthquake by the spectrum response method.

3.7.2.1.2 Auxiliary Building

In determining the response of the building to the earthquake the spectrum response technique is utilized. For this technique the earthquake is described by spectrum response curves presented in Figure 2-51 and Figure 2-53. From the curves, acceleration levels are determined as associated with the natural frequency and damping value of each mode. The standard spectrum response technique uses these values to determine inertial forces, shears, moments, and displacements per mode. These results are then combined on the basis of the absolute sum to obtain the structural response. The process is accomplished by the Bechtel computer program, CE641.

3.7.2.1.3 Turbine Building

Seismic analysis of Turbine Building is discussed in Section 3.8.5.4.

3.7.2.2 Natural Frequencies and Response Loads

3.7.2.2.1 Reactor Building

The natural frequencies and mode shapes are computed using the matrix equation of motion shown below for a lumped mass system. The form of the equation is:

$$(K)(\Delta) = \omega^2 (M)(\Delta)$$

K = matrix of stiffness coefficients including the combined effects of shear, flexure, rotation, and horizontal translation.

M = matrix of concentrated masses.

Δ = matrix of mode shape

ω = angular frequency of vibration.

The results of this computation are the several values of ω_n and mode shapes Δ_n for n = 1, 2, 3, . . . m, where m is the number or degrees of freedom (i.e., lumped masses) assumed in an idealized structure.

3.7.2.2.2 Auxiliary Building

The natural frequencies and mode shapes of the structure are obtained by the Bechtel computer program, CE617. This program utilizes the flexibility coefficients and lumped weights of the model. The flexibility coefficients are formulated into a matrix and inverted to form a stiffness matrix. The program then uses the technique of diagonalization by successive rotations to obtain the natural frequencies and mode shapes. The results are shown in Figure 3-1.

3.7.2.3 Procedure Used for Modeling

3.7.2.3.1 Reactor Building

The modeling of the Reactor Building is discussed in Section 3.7.2.4.

3.7.2.3.2 Auxiliary Building

The mathematical model of the structure is constructed in terms of lumped masses and stiffness coefficients. At appropriate locations within the building, points are chosen to lump the weights of the structure. Between these locations properties are calculated for moments of inertia, cross sectional areas, effective shear areas, and lengths. A sketch of the model is shown on Figure 3-3. The properties of the model are utilized in the IBM computer program, STRESS, along with unit loads to obtain the flexibility coefficients of the building at the mass locations. In Figure 3-4 are presented the moments, shears, displacements, and accelerations for the model subjected to 0.05 g ground motion and 5 percent damping.

3.7.2.3.3 Turbine Building

This information is outlined in Section 3.8.5.4.

3.7.2.4 Development of Floor Response Spectra

3.7.2.4.1 Reactor Building

The actual structural system is idealized as a mathematical model in form of a lumped mass system interconnected by elastic members. Lumped masses, which are a summation of structure and equipment masses, are located at pertinent floor levels and at other levels where response spectra are desired. These other levels would include equipment support elevations, pipe support elevations, etc.

In the case of the Reactor Building, two mathematical models are generated to describe the complete Reactor Building. The first model represents the Reactor Building shell, and the second model represents the internal structure. Modifications to each model are required to determine response from ground motions. After these models are developed, the following procedure is employed for all the models:

The flexibility matrix of the structural system is determined by using Bechtel program CE309. The procedure for each loading condition is to apply a unit load at each mass point and determine the deflection at all mass points.

The mode shapes and frequencies for the lumped mass systems are obtained by means of either of two Bechtel computer programs; CE548, “Symbolic Matric Interpretive System” or CE617, “Diagonalization Method for Eigenvalues and Eigenvectors”.

The Time-History record of the N-S, May 1940 El Centro earthquake is used (vertical and N-S horizontal components). Using the mode shapes and frequencies and the time history (time vs. acceleration record), properly scaled, the time history of the accelerations, velocities and displacements of the lumped masses are obtained. Bechtel program CE611 is utilized for this computation.

The acceleration time-history is applied at the base of a single degree of freedom system. Initially the system is set with specific values for its natural frequency and damping. The time-history response of the mass is determined and examined for the value of maximum acceleration. The same process is repeated over a range of natural frequencies. The resulting maximum G levels and frequencies are tabulated and plotted into the spectrum curve for a single structure elevation. The resulting curve is labeled with the damping value. The process is repeated for required structure elevations and damping values. Bechtel computer program CE591, “Spectral Analysis”, is used to obtain the acceleration and velocity response spectra at each floor for each percentage of damping required.

A sample of the acceleration spectrum curves at different floor levels of a building is shown in Figure 3-6. For these curves, the horizontal axis is logarithmic in cycles per second and the vertical axis is linear in G's. The curves are for 1½ percent of critical damping. The building has natural frequencies of 4.8 cps at the first mode and 10 cps at the second mode. Thus maximum accelerations occur between 1.0 cps and 10.0 cps. At the far right end, the curve converges on the peak value of the input earthquake as the single degree of freedom system becomes rigid, relative to the seismic excitation. At progressively higher locations, the building amplifies the input earthquake, especially in the vicinity of its natural frequencies. Note the sharp peak in each curve at the natural frequency of the building.

When using response curves for piping systems which are located at different elevations, it is necessary to superimpose several curves and plot the envelope curve for the system inputs. At the maximum acceleration peak of each specific curve used for the envelope curve, the envelope has a plateau of approximately ±10 percent to avoid the condition where a small change in frequency could result in a significant change in acceleration. Through the ME 601

program, the natural frequency and mode shapes of the pipe are found and combined with the spectrum curves to find the seismic forces on the pipe.

3.7.2.4.2 Auxiliary Building

The spectrum response curves for equipment inside the building are generated by the time history technique of seismic analysis. The sample earthquake utilized is that recorded at El Centro, California, N-S, May 18, 1940. Essentially the curves are generated by applying the recorded earthquake to the structure and obtaining the time history at selected mass points. Each of these time histories is then applied to a single degree of freedom system of which the values for damping and natural frequency are varied. The curves for Units 1 and 2 Auxiliary Buildings are accomplished by the Bechtel program, CE611. The curves for Unit 3 Auxiliary Building are generated by Duke. The spectrum curves were generated for both directions (East-West and North-South). At the high frequency end of the curve, the acceleration levels converge to the value of the location inside the building.

Digital computer program, CE 617, CE 641, CE 611, and CE 591 are proprietary programs of the Bechtel Corporation.

3.7.2.5 Components of Earthquake Motion

Seismic forces are applied in the vertical and in any horizontal direction. The horizontal and vertical components of ground motion are applied simultaneously.

3.7.2.6 Combination of Modal Responses

3.7.2.6.1 Reactor Building

The response of each mode of vibration to the design earthquake computed by the response spectrum technique, as follows:

1. The base shear contribution of the n^{th} mode $V_n = W_n S_{an}(\omega_n \Upsilon)$ where:

   W_n = effective weight of the structure in the n^{th} mode.

   $$W_n = \frac{\left(\sum_x \Delta_{xn} \, w_x\right)^2}{\sum_x \left(\Delta_{xn}\right)^2 w_x}$$

   where the subscript x refers to levels throughout the height of the structure, and w_x is the weight of the lumped mass at level x.

   ω_n = angular frequency of the n^{th} mode.

   $S_{an}(\omega_n \ \Upsilon)$ = spectral acceleration of a single degree of freedom system with a damping coefficient of Υ, obtained from the response system.

2. The horizontal load distribution for the n^{th} mode is computed as:

   $$F_x = V_n \frac{(\Delta_{xn} w_x)}{\sum_x \Delta_{xn} \, w_x}$$

   Note: The above equation was revised in 1995 update.

The several mode contributions are then combined to give the final response of the structure to the design earthquake.

3. The number of modes to be considered in the analysis is determined to adequately represent the structure being analyzed. The analytical model and results for the 0.05 g earthquake and 2 percent damping and for the 0.1 g earthquake and 5 percent damping are shown in Figure 3-7.

3.7.2.6.2 Auxiliary Building

For description of combining of modal responses, see Section 3.7.2.1.

3.7.2.7 Method Used to Account for Torsional Effects

Torsional modes are not considered in the seismic analysis. Insignificant torsional shear stresses exist, assuming a minimum of 10 percent eccentricity, based on "Torsion in Symmetrical Buildings," N. M. Newmark.

3.7.2.8 Methods for Seismic Analysis of Dams

The methods for the seismic analysis of dams are defined in Section 2.5.6.5.2.

3.7.2.9 Determination of Seismic Class 1 Structure Overturning Moments

The safety factor against overturning for the Reactor Building due to maximum hypothetical earthquake moment is 3.6.

3.7.2.10 Analysis Procedure for Damping

Damping values for the structural system are selected based upon evaluation of the materials and mode shapes. Appropriate damping values of individual materials are presented in Section 3.7.1.3. Evaluation of the mode shapes makes possible the selection of damping values to be associated with each mode.

3.7.3 Seismic Subsystem Analysis

3.7.3.1 Seismic Analysis Methods

The criteria for determining whether systems or portions of systems require a seismic analysis is defined in Section 3.2.1. Piping is further classified according to the required function of the system or portion of a system as shown in Table 3-1.

Two analytical techniques are employed in the seismic analyses: dynamic and static methods. The results obtained by the Section 3.7.3.3 static method are more conservative than the results calculated by the dynamic analysis. The use of the static analysis procedure is limited to piping systems which are not considered complex and where the anticipated seismic effects are minimal.

A special realistic seismic analysis has been used exclusively to qualify the auxiliary building HPSW sprinkler piping in the Units 1, 2 and 3 personnel hatch areas and Unit 1 drumming station to prevent auxiliary building flooding in the event of an earthquake. This method takes exception to some of the criteria specified in the UFSAR for seismic qualification such as the computer code used for analysis, damping values and allowable stresses. See Section 3.7.5, Reference 13

All seismically designed systems penetrating the Reactor Building wall are designed as follows: Within the Reactor Building, a dynamic analysis is performed except where noted below. As each penetration serves as an anchor to the system passing through the Reactor Building wall, a separate analysis is run on the piping outside the Reactor Building.

The design of the B, C, and F Systems outside the Reactor Building is based on a static analysis using a 0.5 g design acceleration. However, subsequent floor response spectra presented in Bechtel "Seismic Analysis Auxiliary Building" report dated January, 1970 and subsequent floor response spectra for Turbine Building developed by Duke Power Company show that there are peak accelerations greater than 0.5 g. Consequently, additional analysis is done to ensure that either (1) span lengths are reduced to avoid fundamental frequencies corresponding to accelerations above 0.5 g or (2) piping stresses and restraint load capabilities are reviewed for adequacy for the appropriate accelerations. Conservative manual methods will be used to determine span frequencies. Also, piping spans will be kept simple to avoid the necessity for modal analysis. Where this technique cannot be applied with confidence, a dynamic analysis will be performed.

Seismically designed systems which penetrate the Reactor Building with a very minor portion of the system inside the Reactor Building (i.e., from the penetration point to the inside isolation valve) are statically analyzed. These systems are as follows:

Reactor Building Purge System

Coolant Storage System

Liquid Waste Disposal System

Miscellaneous Non-Nuclear Service Systems; i.e., Service Air, Nitrogen, Demineralized Water, etc.

Although there is not a seismic classification type interface, the Reactor Coolant System is a B&W Duke system interface.

The scope of NRC Inspection and Enforcement Bulletin 79-14 was defined as all piping that was computer analyzed for seismic loadings and all piping greater than or equal to 2½" diameter that was seismically analyzed using criteria methods. The design inputs for the IEB 79-14 seismic analysis have been reconciled with the as-built. A rigorous computer analysis has been performed for all pipe reanalyzed for IEB 79-14.

Each pipe is idealized as a mathematical model consisting of lumped masses connected by elastic members. Lumped masses are located at carefully selected points in order to adequately represent the dynamic and elastic characteristics of the pipe system. Using the elastic properties of the pipe, the flexibility matrix for the pipe is determined. The flexibility calculations include the effects of torsional, bending, shear, and axial deformations. In addition, for curved members, the stiffness is decreased in accordance with USAS B31.1-1967, Code for Power Piping.

Once the flexibility and mass matrices of the mathematical model are calculated, the frequencies and mode shapes for all significant modes of vibration are determined. All modes having a period greater than 0.05 seconds are used in the analysis. The mode shapes and frequencies are solved in accordance with the following equation:

$$(K - w_n^2 M)\phi_n = 0$$

in which:

K = square stiffness matrix of the pipe

M = mass matrix for the pipe

w_n = frequency for the n^{th} mode

ϕ_n = mode shape matrix of the n^{th} mode

After the frequency is determined for each mode, the corresponding spectral acceleration is read from the appropriate response spectrum for the pipe. Using these spectral accelerations, the response for each mode is found by solving the following equation:

$$Y_n \max = \frac{R_n Sa_n D}{M_n w_n^2}$$

in which:

$Y_n max$ = response of the n^{th} mode

R_n = participation factor for the n^{th} mode = $\Sigma M_i \phi_{in}$

Sa_n = spectral acceleration for the n^{th} mode

D = earthquake direction matrix

M = generalized mass matrix for the n^{th} mode = $\Sigma M_i \phi^2{}_{in}$

Using these results, the maximum displacements for each mode are calculated for each mass point in accordance with the following equation:

$$V_{in} = \phi_{in} Y_n \max$$

in which:

V_{in} = maximum displacement of mass i for mode n

The total displacement for each mass is determined by taking the square root of the sum of the squares of the maximum deflection for each mode:

$$V_i = \sqrt{\Sigma V_{in}^2}$$

in which:

V_i = maximum displacement of mass i due to all modes calculated

The inertia forces for each direction of earthquake for each mode are then determined from:

$$Q_n = KV$$

in which:

Q_n = inertia force matrix for mode n

V = displacement matrix corresponding to Q_n

Each mode's contribution to the total displacements, internal forces, moments, and stresses are determined from standard structural analysis methods using the inertia forces for each mode as an external loading condition. The total combined results are obtained by taking the square root of the sum of the squares of each parameter under consideration, in a manner similar to that done for displacements.

The computer program PISOL, used for original Oconee piping analysis performed by Duke, was provided and maintained by EDS Nuclear of San Francisco, California. Subsequent revision piping analyses on Oconee have been performed by Duke using updated versions of PISOL and SUPERPIPE, by the NUS Corporation, and by Nuclear Power Services (NPS) of Secaucus, New Jersey using their proprietary program.

Both EDS and NPS have reviewed their programs and have verified that the algebraic summation methods were not used in either the earthquake co-directional responses or in the inter-modal responses per IE Bulletin 79-07.

Certain piping analyses on Oconee were performed by Bechtel Corporation, Gaithersburg, Maryland. Bechtel has verified that algebraic summation methods, as noted above, were not used in the piping analysis performed for Duke on Oconee. Bechtel's analysis was performed by EDS on EDS programs.

The verification of computer programs was done in a combination of ways. Due to the non-existence of the ASME benchmark problems during the time of the original analyses, original versions of programs were verified with hand calculated results. As more and more programs became commercially available, comparisons were made with these programs and with the ASME problems.

Specifically, EDS has used a combination of any or all of the following methods:

1. Comparison to ASME Benchmark Problem #1
2. Benchmark Problems Utilizing EDS Programs and Other Industry Programs (PIPESD, NUPIPE, ADLPIPE, ME-101).
3. Comparison to Hand Calculations.
4. Comparison Between EDS Programs and Updated Versions.

NPS has verified its program against PIPESD and ANSYS.

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3.7.3.1.1 Replacement Steam Generator Seismic Analysis

Framatome ANP analyzed the RCS loop model using the certified program BWSPAN. BWSPAN has been certified by comparison to STALUM and T3PIPE, as well as problems generated by Brookhaven National Laboratories and by hand calculations. BWSPAN was reviewed and approved by the NRC (Reference 7 and 8).

The model used to evaluate the effects of the RSG on the RCS contains the reactor vessel, steam generators, pump and pressurizer, as well as the system piping and the interior concrete structures.

Two seismic loadcases were analyzed:

1. Operating Basis Earthquake (OBE), also known as Design Basis Earthquake (DBE). OBE - X+Y Direction, OBE - Y+Z Direction
2. Safe Shutdown Earthquake (SSE), also known as Maximum Hypothetical Earthquake (MHE). SSE - X+Y Direction, SSE - Y+Z Direction

The seismic analyses consider modes up to the cutoff frequency of 33 Hertz. The contribution of those modes beyond the cutoff frequency is accounted for using the technique outlined in Standard Review Plan 3.7.2 of NUREG 0800.

Orthogonal modal damping was used in the seismic analyses. In this method, each section in the mathematical model is assigned a frequency dependent damping value based on information found in Section 3.7.1.3.

Composite damping for each mode is calculated by relating it to the bar strain energy of each of the model sections as described in Section N-1233.2 of Appendix N to the 1983 Section III Division 1 ASME B+PV Code. Once the composite damping for a mode is calculated, the spectral acceleration to be applied at the fixed support nodes is interpolated from the appropriate input spectra curves. Note that the seismic analyses are performed as enveloped spectra analyses with the basemat response spectra applied at all support points at the basemat elevation. The three response spectra acting in the directions of the global coordinate axes are input at several damping values. If the damping for a mode falls between two of the input spectra curves (as is the case most often) the bounding spectra curves are interpolated to get the correct spectral acceleration. The modal results are combined by taking the SRSS of all the modes per Section 3.7.3.1. The model responses to the directional earthquake input (applied response spectra) are combined two-dimensionally for the X+Y and Y+Z directions.

3.7.3.2 Procedure Used for Modeling

A general description of the modeling for the specific programs used for seismic analysis is included in Section 3.7.3.1. The following figures are isometric drawings of typical piping models:

1.	System 01A	Figure 3-9	Main Steam System – West Generator
	Problem #1-01-08		
2.	Systems 53A and 59	Figure 3-10	Core Flooding Tank 1A
	Problem #1-53-9	Figure 3-11	Low Pressure Injection System - West
3.	Systems 51A and 59	Figure 3-12	Reactor Coolant Pump Piping to High
	Problem #1-55-3	Figure 3-13	Pressure Injection Letdown Coolers
		Figure 3-14	
		Figure 3-15	

The practice of overlapping analysis problems was used in the original analytical work performed for the Oconee Nuclear Station piping systems. This approach was utilized to avoid erecting in-line pipe anchors for the sole purpose of defining piping analysis problems. In the reanalysis work required for IE Bulletin 79-14, every effort was made to reduce the number of problems with overlap regions. This was done by combining individual analysis problems into one larger problem. However, this could not be accomplished for all problems due to computer capacity limitations.

When necessary to separate analysis problems the models will be “overlapped” to obtain adequate boundary conditions. The overlap region (pipe modeled in both problems) shall be selected based on engineering judgement, considering the specific geometry to be modeled, to give acceptably accurate results at the problem boundary. As a minimum, the overlap region must include five effective restraints in each of three orthogonal directions. One axial restraint is

effective for the entire run between changes of direction. The overlap region should be located in the most rigid portion of the pipe to obtain maximum isolation between problems.

In the overlap region, S/R loads from both problems will be enveloped to obtain a conservative design load.

3.7.3.3 Use of Equivalent Static Load Method of Analysis of Piping Systems

3.7.3.3.1 Piping

Duke Engineering Design Report, Static Method of Seismic Analysis of Piping Systems for Oconee 1, 2, 3, File #OS-27-B, dated June 5, 1970, describes the approach and a sample problem for seismic piping.

The original method for determining seismic response based on static analysis for Reactor Building piping is as follows:

The envelope of response curve(s) developed for the dynamic analysis are used for the static analysis which is based on the assumption that the natural frequency of the piping system is at the critical frequency.

Static loads at points of support are determined by utilizing the computer program ME553-Piping Flexibility Analysis - to perform a modified weight analysis which is based on applying the maximum horizontal forces in the positive X or Z directions simultaneously with the maximum, vertical force.

The horizontal forces are obtained by using the maximum acceleration peak from the appropriate envelope curves as the multiplier to convert uniform pipe weight into forces. The vertical force is obtained from the pipe weight density multiplied by the vertical peak acceleration.

The valves and special fittings on the system are mathematically expressed in the analysis as equivalent pipe of the same weight as the valve or fitting.

The combination of all maximum forces in the positive directions produces resulting static loads of greater magnitude than the dynamic analysis.

Reactor Building Pipe Stress analyses revised subsequent to original analysis are based on the "Duke Engineering Design Report" discussed above.

3.7.3.3.2 Components

The seismic analysis of the component coolers shown in Figure 3-16, Figure 3-17, and Figure 3-18 is an example of the static analysis applied to components in Class B, C, and F Systems outside the Reactor Building.

3.7.3.4 Components of Earthquake Motion

Seismic forces are applied in the vertical and in any horizontal direction. The horizontal and vertical components of ground motion are applied simultaneously.

3.7.3.5 Combination of Modal Response

This information is addressed in Section 3.7.3.1.

3.7.3.6 Analytical Procedures for Piping

General Analytical Procedures are discussed in Section 3.7.3.1.

3.7.3.7 Multiple Supported Equipment and Components with Distinct Inputs

Floor response spectra developed as discussed in Section 3.7.2.5 are used as input for the piping analysis. When the pipe is supported from more than one elevation or structure, the response spectra for all support levels are enveloped and the envelope spectra are used in the analysis, except when the Independent Support Motion technique is used. In certain instances where one group of supports attach to a structure and another group of supports attach to a structure with a definite distinction in structural seismic response, ISM methods have been used in the qualification of existing pipe and supports during reanalysis for IEB 79-14 to mitigate the consequences and excessive conservatism of the total enveloped spectra method. Such consequences may include undue radiation exposure to personnel or undue hardship in implementing field modifications.

For piping passing from one building into another building, the maximum movements of the two buildings (deflections produced by earthquake) are summed absolutely and the piping system is subjected to these movements through the piping system restraints. The stresses produced in the piping by the building movements are considered additive to the stresses resulting from accelerations or thermal expansion.

Rocking of the turbine support structure has been considered with respect to the Main Steam System analysis and movements of the turbine support are negligible as compared to other design movements of the main turbine piping leads attached to the main steam stop valve and control valve assembly.

3.7.3.8 Buried Piping Tunnels Designed for Seismic Conditions

The CCW intake piping furnishing water to the LPSW pumps is placed on concrete bedding which rests on bed rock at the point of entry into the station. There will not be any differential movement at the piping-structural interface with the rock base, thereby precluding any stress problems. Except for the CCW piping described above, other seismically designed safety-related buried lines for the Oconee Project are the 48" emergency discharge CCW pipe and CCW discharge piping, and the SSF Auxiliary Service Water pump discharge line, the Siphon Seal Water line, and the Essential Siphon Vacuum lines.

3.7.3.9 Interaction of other Piping with Piping Designed for Seismic Conditions

The interaction between seismic/non-seismic lines are considered and safety system integrity is assured by the following methods:

Seismic/non-seismic lines are physically separated insofar as possible such that failure of a non-seismic line has no effect on safety-related piping.

Seismic/non-seismic boundaries are established by valves which are designed to meet the seismic design criteria. Failure in the non-seismic portion of the system cannot cause loss of function to the safety system in that automatic or remote manual-operated valves are used for valves normally open during Reactor Operation. A variation of this case is where the seismic/non-seismic boundary is beyond the automatic or remote-manual valve. (For example, LPSW piping to and from the Reactor Building Auxiliary Coolers have Seismic Category I/Seismic Category II boundaries inside containment at normally open manual valves. For containment isolation purposes in this particular case, the Seismic Category II piping, although it has structural seismic integrity, the piping is treated as non-seismic, from a piping pressure

boundary perspective, since it is non QA Condition 1. The containment isolation valves associated with the penetrations are located outside containment. Closure of the containment isolation valves mitigates the effects of failure of the non-seismic piping. The boundary is extended past the containment isolation valves since the seismic boundary is part of those particular containment penetrations. Seismic Category I and Seismic Category II are defined in Regulatory Guide 1.29, Seismic Design Classification.)

It is acceptable to open normally closed manual seismic boundary valves provided the opening and closing of these valves is controlled by approved plant procedures and the valve will be opened for a required operating evolution with a clearly definable beginning and end time. Examples include opening normally closed manual seismic valves to support testing, sampling, makeup, backwashing, etc. These type activities are inherent to the normal operation of a nuclear power plant. The expectation is that, when the associated safety system is required to be operable, the manual seismic boundary valve will be in the closed position much more than it is in the open position.

Automatic or remote manual-operated valves are not required for seismic/non-seismic boundaries that are normally open during reactor operation, provided that an analysis has demonstrated that a seismically-induced failure of the piping would not cause loss of system safety function. Such analysis shall assume only a single pipe break during a seismic event, and the analysis shall determine the effect on the safety-related portion of the system from the most limiting single pipe break.

Automatic or remote manual-operated valves are not required on the Spent Fuel (SF) Cooling system seismic/non-seismic boundary drain lines off the Fuel Transfer Canal and Incore Instrument Handling Tank that are normally open during reactor operation. Flooding is not a concern during reactor operation since these flow paths would channel flow from other systems to the Reactor Building sumps (i.e. act as funnels and are not sources of water). (Reference 6)

The seismic/non-seismic boundary valve is protected from seismic effects by restraining or anchoring the non-seismic portion of the system downstream of the valve.

3.7.3.10 Seismic Analysis of Reactor Internals

The core support structure is designed as a Class I structure, as defined in Section 3.2 to resist the effects of seismic disturbances. The basic design guide for the seismic analysis is AEC publication TID-7024, "Nuclear Reactors and Earthquakes."

Lateral deflection and torsional rotation of the lower end of the core support assembly is limited in order to prevent excessive deformation resulting from seismic disturbance thereby assuring insertion of control rod assemblies (CRA). Core drop in the event of failure of the normal supports is limited by guide lugs so that the CRA do not disengage from the fuel assembly guide tubes. Additional information on design of the Reactor Internals is included in Section 3.9.2.

3.7.3.11 Analysis Procedures for Damping

A 0.5 percent critical damping value is used for vital piping analysis (see Section 3.7.1.3).

3.7.4 Seismic Instrumentation Program

3.7.4.1 Location and Description of Instrumentation

Earthquake instrumentation being provided is a strong motion accelerograph designated SMA-3 and manufactured by Kinemetrics, Inc., of Pasadena, CA. This system consists of a central

recording system, control panel, one TS-3 triaxial seismic trigger package, and two force-balance triaxial accelerometer packages.

The operations sequence is as follows:

The seismic trigger senses the initial earthquake ground motion with a normal setting of 0.01g and actuates the SMA-3 to full operation in less than 0.1 second.

The SMA-3 operates for as long as the trigger detects the earthquake, plus an additional 10 seconds.

The accelerograph can thus record a single earthquake or a sequence of earthquakes and aftershocks lasting as long as 30 minutes.

The output of each triaxial sensor is recorded using frequency modulation on a single four track cassette tape. Three of the tracks on the tape are the data tracks; the fourth is a reference track used for tape speed and amplitude compensation.

The Seismic Trigger and one Force Balance accelerometer of the SMA-3 system are located in the Unit 1 Tendon Gallery. Also, a second Force Balance accelerometer is located directly above at elevation 797' + 6" in the Oconee 1 Reactor Building. The recorder for the system is located in the Unit 1 Cable Room.

Also, a seismic trigger/switch is located in the Unit 1 tendon gallery. The Kinemetrics Model TS-3A has a preset acceleration threshold of 0.05g which activates the statalarm in Units 1 and 3 control rooms, when design conditions occur.

Six 2g peak recording accelorometers, manufactured by Engdahl-Model PAR 400, are also installed at various locations within the Oconee 1 Reactor Building. The instruments will provide post-seismic data for the following locations or items:

1. Adjacent to the strong motion accelerograph located in Tendon Access Gallery.
2. Support of the pressurizer vessel.
3. Support of Core Flood Tank 1A.
4. Main steam line pipe hanger.
5. Feedwater line pipe hanger.
6. Core flood injection line pipe hanger.

The major Class 1 structures, Reactor Building and Auxiliary Buildings, will be founded on a common rock foundation and will have similar base motions. The dynamic structural properties and responses of these structures are generated using similar assumptions and analytical techniques. Therefore, the response of these structures can be determined based upon the instrumentation in one structure.

Top of soil (free field) responses will not provide useful analytical data for the evaluation of major Class 1 structures founded on rock. Therefore, it is felt that free field instrumentation will not contribute to the evaluation of these structures.

3.7.4.2 Comparison of Measured and Predicted Responses

In the event of an earthquake, the data will be analyzed to determine the magnitude of the earthquake. If the design earthquake is exceeded, the units would be shut down and structures, systems, and equipment thoroughly investigated. Responses from instruments located on selected structures, systems and components will be compared to calculated responses for

those structures, systems and components at the respective location when subjected to the same base response.

The recorded seismic data will be used for comparison and verification of seismic analysis assumptions, damping characteristics, and the analytical model used for the plant seismic design.

3.7.5 References

1. Bechtel Report, "Seismic Analysis Auxiliary Building", January, 1970.
2. Duke Power Engineering Design Report, "Static Method of Seismic Analysis of Piping Systems for Oconee 1, 2 and 3", File OS-27-B, June 6, 1970.
3. AEC Report TID-7024, "Nuclear Reactors and Earthquakes".
4. Newmark, N. M., "Torsion in Symmetrical Buildings".
5. Deleted Per 2000 Update
6. OSC-7462, Rev. 0, “Spent Fuel Cooling System – Normally Open Seismic Boundary Valves.”
7. Letter dated July 26, 2001 from W. R. McCollum, Jr. to NRC, requesting NRC review and approval of the methodology that will be used for the re-analysis of the reactor coolant loop as a part of steam generator replacement.
8. Letter dated September 6, 2001 from Dave E. LaBarge transmitting the NRC’s SER for the proposed methodology for the analysis of the reactor coolant loop in support of steam generator replacement.
9. Calculation OSC-7835, Steam Generator Replacement Project; ONS Unit 1 Reactor Coolant Structural Analysis for ROTSG’s.
10. Calculation OSC-7836, Steam Generator Replacement Project; ONS Units 2 and 3 Reactor Coolant Loop Structural Analysis for ROTSG’s.
11. Supplement No. 1 to Generic Letter 87-02 that Transmits Supplemental Safety Evaluation Report No. 2 (SSER No. 2) on SQUG Generic Implementation Procedure, Revision2, as Corrected On February 14, 1992 (GIP-2).
12. Letter dated September 9, 1999 from David E. Labarge to W. R. McCollum, Jr Transmitting Oconee Plant Specific Safety Evaluation Report for Unresolved Safety Issue A-46 Program Implementation, Including Keowee Hydro Station and Switchyard.
13. Letter dated November 14, 2007 from Leonard N. Olshan, Sr. transmitting the NRC's SER for the realistic seismic analysis of the auxiliary building HPSW sprinkler piping.

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3.8 Design of Structures

Class 1 structures are those which prevent uncontrolled release of radioactivity and are designed to withstand all loadings without loss of function.

Class 2 structures are those whose limited damage would not result in a release of radioactivity and would permit a controlled plant shutdown but could interrupt power generation.

Class 3 structures are those whose failure could inconvenience operation, but which are not essential to power generation, orderly shutdown or maintenance of the reactor in a safe condition.

Note: From the license renewal review, it was determined that Class 1 and Class 2 civil structures only are included in the scope for license renewal.

3.8.1 Concrete Containment

The concrete/steel containment is analyzed as a free standing structure and is referred to as the Reactor Building. It is constructed of reinforced concrete and structural liner plate steel with no separation between the two.

3.8.1.1 Description of the Containment

The structure consists of a post-tensioned reinforced concrete cylinder and dome connected to and supported by a massive reinforced concrete foundation slab as shown in Figure 3-19. The entire interior surface of the structure is lined with a ¼ inch thick welded ASTM A36 steel plate to assure a high degree of leak tightness. Numerous mechanical and electrical systems penetrate the Reactor Building wall through welded steel penetrations as shown in Figure 3-20 and Figure 3-21. The mechanical penetrations and access openings are design, fabricated, inspected, and installed in accordance with Subsection B, Section III, of the ASME Pressure Vessel Code.

Principal dimensions are as follows:

"These values are historical and for descriptive purposes only."

Inside Diameter	116 ft
Inside Height (Including Dome)	208½ ft
Vertical Wall Thickness	3-3/4 ft
Dome Thickness	3-¼ ft
Foundation Slab Thickness	8-½ ft
Liner Plate Thickness	¼ in.
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The Reactor Building is shown in Figure 1-2 through Figure 1-9.

In the concept of a post-tensioned Reactor Building, the internal pressure load is balanced by the application of an opposing external pressure type load on the structure. Sufficient post-tensioning is used on the cylinder and dome to more than balance the internal pressure so that a margin of external pressure exists beyond that required to resist the design accident pressure. Nominal, bonded reinforcing steel is also provided to distribute strains due to shrinkage and

temperature. Additional bonded reinforcing steel is used at penetrations and discontinuities to resist local moments and shears.

The internal pressure loads on the foundation slab are resisted by both the external bearing pressure due to dead load and the strength of the reinforced concrete slab. Thus, post-tensioning is not required to exert an external pressure for this portion of the structure.

The post-tensioning system consists of:

1. Three groups of 54 dome tendons oriented at 120° to each other for a total of 162 tendons anchored at the vertical face of the dome ring girder.
2. 176 vertical tendons anchored at the top surface of the ring girder and at the bottom of the base slab.
3. Six groups of 105 hoop tendons plus two additional tendons enclosing 120° of arc for a total of 632 tendons anchored at the six vertical buttresses.

Each tendon consists of ninety ¼ inch diameter wires with buttonheaded BBRV type anchorages, furnished by The Prescon Corporation. Replacement tendons installed during steam generator replacement were furnished by PSC. The tendons are housed in spiral wrapped corrugated thin wall sheathing. After fabrication, the tendon is shop dipped in a petroleum corrosion protection material, bagged and shipped. After installation, the tendon sheathing is filled with a corrosion preventive grease.

Ends of all tendons are covered with pressure tight grease filled caps for corrosion protection.

ASTM A615, Grade 60 reinforcing steel, mechanically spliced with T-series CADWELDS, is used throughout the foundation slab and around the large penetrations. A615, Grade 40 steel is used for the bonded reinforcing throughout the cylinder and dome as crack control reinforcing. At areas of discontinuities where additional steel is used, such steel is generally A615, Grade 60 to provide an additional margin of elastic strain capability. ASTM A615, Grade 60 was also used as necessary for the repaired area following steam generator replacement.

The ¼ inch thick liner plate is attached to the concrete by means of an angle grid system stitch welded to the liner plate and embedded in the concrete. The details of the anchoring system are provided in Figure 3-19. The frequent anchoring is designed to prevent significant distortion of the liner plate during accident conditions and to insure that the liner maintains its leak tight integrity. The design of the liner anchoring system also considers the various erection tolerances and their effect on its performance. The liner plate was coated during construction for corrosion protection. See Table 3-12 for Reactor Building coatings. There is no paint on the side in contact with concrete.

The concrete used in the original construction of the structure is made with crushed marble aggregate obtained from Blacksburg, South Carolina. Such aggregate produces an excellent high strength, dense, sound concrete. The design strengths are 5000 psi at 28 days for the shell and foundation slab. A 5000 psi high early strength, non-shrink or slightly expansive mix was used for repairing the temporary construction opening following steam generator replacement.

Personnel and equipment access to the structure is provided by a double door personnel hatch with double seals on the outer door and by a 19 ft. - 0 in. clear diameter double gasketed single door equipment hatch as shown in Figure 3-21. A double door emergency personnel escape hatch is also provided. These hatches are designed and fabricated of A516, Grade 70 firebox quality steel made to A3000 specification, Charpy V-notch impact tested to 0°F in accordance

with Section III of the ASME Pressure Vessel Code. All piping penetrations are furnished to the same requirements.

Structural brackets provided for the Reactor Building polar crane runway are fabricated of A36 steel shapes and A516, Grade 70 insert plates (Figure 3-19). Structural brackets and thickened plates are shop fabricated, stress relieved and shipped to the jobsite for welding into the ¼ inch liner plate similar to the penetration assemblies.

3.8.1.1.1 Coating Materials

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

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

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

3.8.1.2 Applicable Codes, Standards, and Specifications

"HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED"

The following codes, standards, and specifications were used during the design, construction, testing and inservice inspection of Class 1 Structures:

ACI 301	-	*Specification for Structural Concrete for Buildings*
ACI 318-63	-	*Building Code Requirements for Reinforced Concrete*
ACI 347	-	*Recommended Practice for Concrete Framework*
ACI 605	-	*Recommended Practice for Hot Weather Concreting*
ACI 613	-	*Recommended Practice for Selecting Proportions for Concrete*
ACI 614	-	*Recommended Practice for Measuring, Mixing and Placing Concrete*
ACI 315	-	*Manual of Standard Practice for Detailing Reinforced Concrete Structures*
ASME-1965	-	*Boiler and Pressure Vessel Code, Sections III, VIII, and IX*
AISC	-	*Steel Construction Manual, 6th ed* (1)
PCI	-	*Inspection Manual*
ACI 505	-	*Specification for Design and Construction of Reinforced Concrete Chimneys*
ACI	-	*American Concrete Institute*

ASME	-	*American Society of Mechanical Engineers*
AISC	-	*American Institute of Steel Construction*
PCI	-	*Prestressed Concrete Institute*

Notes:

1. For visual inspections of structural welds, reference the "Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants", NCIG-01, Rev. 2 dated 5/7/85.

3.8.1.3 Loads and Load Combinations

3.8.1.3.1 Loads Prior to Prestressing

Under this condition the structure is designed as a conventionally reinforced concrete structure. It is designed for dead load, live loads (including construction loads), and a reduced wind load. Allowable stresses are according to ACI 318-63.

3.8.1.3.2 Loads at Transfer of Prestress

The Reactor Building is checked for prestress loads and the stresses compared with those allowed by ACI 318-63 with the following exceptions: ACI 318-63, Chapter 26, allows concrete stress of $0.60f'_{ci}$ at initial transfer. In order to limit creep deformations, the membrane compression stress is limited to $0.30f'_{ci}$; whereas, in combination with flexural compression, the maximum allowable stress will be limited to $0.60f'_{ci}$ per ACI 318-63.

For local stress concentrations with nonlinear stress distribution as predicted by the finite element analysis, $0.75f'_{ci}$ is permitted when local reinforcing is included to distribute and control these localized strains. These high local stresses are present in every structure but they are seldom identified because of simplifications made in design analysis. These high stresses are allowed because they occur in a very small percentage of the cross section, are confined by material at lower stress and would have to be considerably greater than the values allowed before significant local plastic yielding would result. Bonded reinforcing is added to distribute and control these local strains.

Membrane tension and flexural tension are permitted provided they do not jeopardize the integrity of liner plate. Membrane tension is permitted to occur during the post-tensioning sequence but will be limited to $1.0\sqrt{f'_{ci}}$. When there is flexural tension but no membrane tension, the section is designed in accordance with Section 2605(a) of the ACI Code. The stress in the liner plate due to combined membrane tension and flexural tension is limited to 0.5 fy.

Shear criteria are in accordance with the ACI 318-63 Code, Chapter 26, as modified by the equations in Section 3.8.1.3.6 using a load factor of 1.5 for shear loads.

3.8.1.3.3 Loads Under Sustained Prestress

The conditions for design and the allowable stresses for this case are the same as above except that the allowable tensile stress in nonprestressed reinforcing is limited to 0.5 fy. ACI 318-63 limits the concrete compression to $0.45f'_c$ for sustained prestress load. Values of $0.30f'_c$ and $0.60f'_c$ are used as described above which bracket the ACI allowable value. However, with

these same limits for concrete stress at transfer of prestress, the stresses under sustained load are reduced due to creep.

3.8.1.3.4 Service Loads

This loading case is the basic “working stress” design. The Reactor Building is designed for the loading cases shown in Table 3-13.

Sufficient prestressing is provided in the cylindrical and dome portions of the vessel to eliminate membrane tensile stress (tensile stress across the entire wall thickness) under design loads. Flexural tensile cracking is permitted but is controlled by bonded reinforcing steel.

Under the design loads the same performance limits stated in Section 3.8.1.3.2 apply with the following exceptions:

1. If the net membrane compression is below 100 psi, it is neglected and a cracked section is assumed in the computation of flexural bonded reinforcing steel. The allowable tensile stresses in bonded reinforcing are 0.5 fy.
2. When the maximum flexural stress does not exceed $6\sqrt{f_c'}$ and the extent of the tensions zone is not more than 1/3 the depth of the section, bonded reinforcing steel is provided to carry the entire tension in the tension block. Otherwise, the bonded reinforcing steel is designed assuming a cracked section. When the bending moment tension is additive to the thermal tension, the allowable tensile stress in the bonded reinforcing steel is 0.5 fy minus the stress in reinforcing due to the thermal gradient as determined in accordance with the method of ACI-505.
3. The problem of shear and diagonal tension in a prestressed concrete structure should be considered in two parts: membrane principal tension and flexural principal tension. Since sufficient prestressing is used to eliminate membrane tensile stress, membrane principal tension is not critical at design loads. Membrane principal tension due to combined membrane tension and membrane shear is considered under Section 3.8.1.3.6.

   Flexural principal tension is the tension associated with bending in planes perpendicular to the surface of the shell and shear stress normal to the shell (radial shear stress). The present ACI 318-63 provisions of Chapter 26 for shear are adequate for design purposes with proper modifications as discussed under Section 3.8.1.3.6 using a load factor 1.5 for shear loads.

Crack control in the concrete is accomplished by adhering to the ACI and American Society of Civil Engineers Code Committee standards for the use of reinforcing steel. These criteria are based upon a recommendation of the Prestressed Concrete Institute and are as follows:

0.25 percent reinforcing shall be provided at the tension face for small members

0.20 percent for medium size members

0.15 percent for large members

A minimum of 0.15 percent bonded steel reinforcing is provided in two perpendicular directions on the exterior faces of the wall and dome for proper crack control.

The liner plate is attached on the inside faces of the wall and dome. Since, in general, there is no tensile stress due to temperature on the inside faces, bonded reinforcing steel is not necessary at the inside faces.

The Reactor Building shell is also designed for the following loads:

1. Dead load
2. Prestress forces
3. Live load including allowances for piping, ductwork and cable trays
4. Wind, including tornado
5. Earthquake
6. Thermal expansion of pipes attached to the Reactor Building wall

The external design pressure of the Reactor Building shell is 3 psig. This value is approximately 0.5 psig beyond the maximum external pressure that could be developed if the Reactor Building were sealed during a period of low barometric pressure and high temperature and, subsequently, the Reactor Building atmosphere were cooled with a concurrent rise in barometric pressure. Vacuum breakers are not provided.

3.8.1.3.5 Loadings Common to all Structures

Ice or Snow Loading

A uniform distributed live load of 20 pounds per square foot is considered for roofs as stated in Section 1203.2 of the Southern Standard Building Code.

3.8.1.3.6 Loads Necessary to Cause Structural Yielding

The structure is checked for the factored loads and load combinations that will cause structural yielding.

The load factors are the ratio by which loads will be multiplied for design purposes to assure that the load/deformation behavior of the structure is one of elastic, low-strain behavior. The load factor approach is being used in this design as a means of making a rational evaluation of the isolated factors which must be considered in assuring an adequate safety margin for the structure. This approach permits the designer to place the greatest conservatism on those loads most subject to variation and which most directly control the overall safety of the structure. It also places minimum emphasis on the fixed gravity loads and maximum emphasis on accident and earthquake or wind loads. The final design of the Reactor Building satisfies the loading combinations and factors tabulated in Table 3-14.

The load combinations, considering load factors referenced above, are less than the yield strength of the structure. The yield strength of the structure is defined as the upper limit of elastic behavior of the effective load carrying structural materials. For steels (both prestress and nonprestress), this limit is taken to be the guaranteed minimum yield given in the appropriate ASTM specification. For concrete, it is the ultimate values of shear (as a measure of diagonal tension) and bond per ACI 318-63 and the 28-day ultimate compressive strength for concrete in flexure (f'_c). The ultimate strength assumptions of the CI Code for concrete beams in flexure are not allowed; that is, the concrete stress is not allowed to go beyond yield and redistribute at a strain of three or four times that which causes yielding.

The maximum strain due to secondary moments, membrane loads and local loads exclusive of thermal loads is limited to that corresponding to the ultimate stress divided by the modulus of elasticity (f'_c/E_c) and a straight-line distribution from there to the neutral axis assumed.

For the loads combined with thermal loads, the peak strain is limited to 0.003 inch/inch. For concrete membrane compression, the yield strength is assumed to be $0.85f'_c$ to allow for local irregularities in accordance with the ACI approach. The reinforcing steel forming part of the load

carrying system is allowed to go to, but not to exceed, yield as is allowed for ACI ultimate strength design.

A further definition of yielding is the deformation of the structure which causes strains in the steel liner plate to exceed 0.005 inch/inch. The yielding of nonprestress reinforcing steel is allowed, either in tension or compression, if the above restrictions are not violated. Yielding of the prestress tendons is not allowed under any circumstances.

Principal concrete tension due to combined membrane tension and membrane shear, excluding flexural tension due to bending moments or thermal gradients, is limited to $3\sqrt{f'_c}$. Principal concrete tension due to combined membrane tension, membrane shear, and flexural tension due to bending moments or thermal gradients is limited to $6\sqrt{f'_c}$. When the principal concrete tension exceeds the limit of $6\sqrt{f'_c}$, bonded reinforcing steel is provided in the following manner:

1. Thermal Flexural Tension - Bonded reinforcing steel is provided in accordance with the methods of ACI-505. The minimum area of steel provided is 0.15 percent in each direction.
2. Bending Moment Tension - Sufficient bonded reinforcing steel is provided to resist the moment on the basis of cracked section theory using the yield stresses stated above with the following exception: When the bending moment tension is additive to the thermal tension, the allowable tensile stress in the reinforcing steel is fy minus the stress in reinforcing due to the thermal gradient as determined in accordance with the methods of ACI-505.

Shear stress limits and shear reinforcing for radial shear are in accordance with Chapter 26 of ACI 318-63 with the following exceptions: Formula 26-12 of the code shall be replaced by

$$V_{ci} = Kb'd\sqrt{f'_c} + M_{cr}\left(\frac{V}{M'}\right) + V_i$$

Where:

$$K = [1.75 - \frac{0.036}{np'} + 4.0\, np']$$

but not less than 0.6 for $p' \geq 0.003$.

For $p' < 0.003$, the value of K shall be zero.

$$M_{cr} = \frac{I}{Y}[6\sqrt{f'_c} + f_{pe} + f_n + f_i]$$

f_{pe}	=	Compressive stress in concrete due to prestress applied normal to the cross section after all losses (including the stress due to any secondary moment) at the extreme fiber of the section at which tension stresses are caused by live loads.
f_n	=	Stress due to axial applied loads (f_n shall be negative for tension stress and positive for compression stress).
f_i	=	Stress due to initial loads at the extreme fiber of a section at which tension stresses are caused by applied loads (including the stress due to any secondary moment). f_i shall be negative for tension stress and positive for compression stress.

n	=	$\dfrac{505}{\sqrt{f'_c}}$
p′	=	$\dfrac{A'_s}{bd}$
V	=	Shear at the section under consideration due to the applied loads.
M′	=	Moment at a distance d/2 from the section under consideration, measured in the direction of decreasing moment, due to applied loads.
V_i	=	Shear due to initial loads (positive when initial shear is in the same direction as the shear due to applied loads).

Lower limit placed by ACI 318-63 on V_{ci} as $1.7b'd\sqrt{f'_c}$ is not applied.

Formula 26-13 of the Code shall be replaced by

$$V_{cw} = 3.5b'd\sqrt{f'_c} \qquad \sqrt{1+\frac{f_{pc}+f_n}{3.5\sqrt{f'_c}}}$$

The term f_n is as defined on the previous page. All other notations are in accordance with Chapter 26, ACI 318-63.

1. This formula is based on the tests and work done by Dr. A. H. Mattock of the University of Washington.
2. This formula is based on the commentary for proposal redraft of Section 2610, ACI-318, by Dr. A. H. Mattock, dated December 1962.

When the above-mentioned equations show that allowable shear in concrete is zero, radial horizontal shear ties are provided to resist all the calculated shear.

3.8.1.4 Design and Analysis Procedures

The strength of the Reactor Building at working stress and overall yielding is compared to various loading combinations to assure safety. The Reactor Building is examined with respect to strength, the nature and the amount of cracking, the magnitude of deformation, and the extent of corrosion to assure proper performance. The structure is designed and constructed in accordance with design criteria based upon ACI 318-63, ACI 301, and ASME Pressure Vessel Code, Sections III, VIII, and IX to meet the performance and strength requirements prior to prestressing, at transfer of prestress, under sustained prestress, at design loads and at yield loads.

It is the intent of the criteria to provide a structure of unquestionable integrity that will meet the postulated design conditions with a low strain elastic response. The Oconee Reactor Building meets these criteria because:

1. The design criteria are, in general, based on the proven stress, strain, and minimum proportioning requirements of the ACI or ASME Codes. Where departures or additions from these codes have been made, they have been done in the following manner:

   a. The environmental conditions of severity of load cycling, weather, corrosion conditions, maintenance, and inspection for this structure have been compared and evaluated with those for code structures to determine the appropriateness of the modifications.

   b. The consultant firm of T. Y. Lin, Kulka, Yang and Associates was retained to assist in the development of the criteria. In addition to assisting with the criteria submitted in the PSAR, they have been involved in the continuing updating of the criteria and the review of design methods to assure that the criteria were being implemented as intended.

   c. Dr. Alan H. Mattock of the University of Washington was retained to assist in developing the proper design criteria from combined shear, bending, and axial load.

   d. All criteria, specifications, and details relating to the liner plate and penetrations, and corrosion protection have been referred to Bechtel's Metallurgy and Quality Control Department. This department maintains a staff to advise the corporation on problems of welding, quality control, metallurgy, and corrosion protection.

   e. The design of the Oconee Reactor Building was continually reviewed as the criteria were improved for successive license applications to assure that this structure does meet the latest criteria.

2. The primary membrane integrity of the structure is provided by the unbonded post-tensioning tendons, each one of which is stressed to 80 percent of ultimate strength during installation and performs at approximately 50 percent - 60 percent during the life of the structure. Thus, the main strength elements are individually proof-tested prior to operation of the plant.

3. 970 such post-tensioning elements have been provided, 162 in the dome, and 176 vertical and 632 hoop tendons in the cylinder. Any three adjacent tendons in any of these groups can be lost without significantly affecting the strength of the structure due to the load redistribution capabilities of the shell structure. The bonded reinforcing steel provided for crack control assures that this redistribution capability exists.

4. The unbonded tendons are continuous from anchorage to anchorage, being deflected around penetrations and isolated from secondary strains of the shell. Thus, the membrane integrity of the shell can be assured regardless of conditions of high local strains.

5. The unbonded tendons exist in the structure at a slightly ever-decreasing stress due to relaxation of the tendon and creep of the concrete and, even during pressurization, are subject to a stress change of very small magnitude (2 percent to 3 percent of ultimate strength). Thus, the main structural system is never subject to large changes in load, even during accident conditions.

6. The concrete portion of the structure, similar to the tendons, is subject to the highest state of stress during the initial post-tensioning. During pressurization, it is subject to a large change in load (or state of stress) but the change is, in general, a decrease in load. The large membrane compressive forces are diminished, and replaced, by relatively small radial pressures and stresses.

7. The deformations of the structure during plant operation, or due to accident conditions, are relatively minor due to the low strain behavior of the concrete. The largest deformations occur at the time of initial post-tensioning and shortly, thereafter, prior to operation. This low strain behavior, and the inherent strength of the structure, permit the anchoring of all piping penetrating the structure directly to the shell. Such details (see Figure 3-21) eliminate the use of expansion bellow seals and significantly reduce the likelihood of leaks developing at the penetrations.

The analysis for the Reactor Building falls into two parts, axisymmetric and nonaxisymmetric. The axisymmetric analysis is performed through the use of a finite element computer program for the individual loading cases of dead load, live load, temperature, prestress, and pressure, as described in Section 3.8.1.4.1. The axisymmetric finite element approximation of the Reactor Building shell does not consider the buttresses, penetrations, brackets, and anchors. These items of configuration, the lateral loads due to seismic or wind, and concentrated loads are considered in the nonaxisymmetric analysis described in Section 3.8.1.4.2.

This section discusses analytical techniques, references and design philosophy. The results of these analyses are discussed in Section 3.8.1.5. The design criteria and analysis have been reviewed by Bechtel's consultants, T. Y. Lin, Kulka, Yang and Associates.

3.8.1.4.1 Axisymmetric Techniques

The finite element technique is a general method of structural analysis in which the continuous structure is replaced by a system of elements (members) connected at a finite number of nodal points (joints). Conventional analysis of frames and trusses can be considered to be examples of the finite element method. In the application of the method to an axisymmetric solid (e.g., a concrete Reactor Building), the continuous structure is replaced by a system or rings of quadrilateral cross section which are interconnected along circumferential joints. Based on energy principles, work equilibrium equations are formed in which the radial and axial displacements at the circumferential joints are unknowns of the system. The results of the solution of this set of equations are the deformation of the structure under the given loading conditions. For the output, the stresses are computed knowing the strain and stiffness of each element.

The finite element mesh used to describe the structure is shown in Figure 3-22. The upper portion and lower portion of the structure were analyzed independently to permit a greater number of elements to be used for those areas of the structure of major interest such as the ring girder area and the base of the cylinder. The finite element mesh of the structure base slab was extended down into the foundation material to take into consideration the elastic nature of the foundation material and its effect upon the behavior of the base slab. The tendon access gallery is separated from the Reactor Building base slab by 3 in. compressible material. No moments or forces are transmitted from the base slab to the tendon access gallery. The maximum vertical elastic displacement of the base slab is one inch due to the maximum loading combinations. The tendon access gallery was designed as a separate structure with no reactions being generated from the bedrock to the ring shaped gallery structure.

The finite element mesh for the Reactor Building does not include the interior structure. The interior structure was included in the finite element input as a lump weight. The finite elements provide stresses for axisymmetric loads. The stresses from the eccentric interior structure loads and earthquake loads are superimposed analytically to the finite element stresses. The final algebraic summation of all stresses was used to design the base slab.

Stresses for Axisymmetric Loads	Stresses with Non-Axisymmetric Loads
11.0 kips/sq.ft.	26.0 kips/sq.ft.

The use of the finite element computer program permitted an accurate estimate of the stress pattern at various locations of the structure. The following material properties were used in the program for the various loading conditions:

	Load Conditions	
	D, F, T_O, T_A	P
$E_{concrete}$, Foundation (psi)	3.0×10^6	3.0×10^6
$E_{concrete}$, Shell (psi)	3.0×10^6	3.0×10^6
$\mu_{concrete}$ (Poisson's Ratio)	0.17	0.17
$\alpha_{concrete}$ (Coefficient of Expansion)	0.55×10^{-5}	–
$E_{subgrade}$ (psi)	4.5×10^6	4.5×10^6
E_{liner} (psi)	29×10^6	29×10^6
$f_{y\ liner}$ (psi)	36,000	36,000

The major benefit of the program is the capability to predict shears and moments due to internal restraint and the interaction of the foundation slab relative to the subgrade. The structure is analyzed assuming an uncracked homogeneous material. This is conservative because the decreased relative stiffness of a cracked section would result in smaller secondary shears and moments.

In arriving at the tabulated values of E, the effect of creep is included by using the following equation for long-term loads such as thermal load, dead load and prestress:

$$E_{cs} = E_{ci}(\varepsilon_i / (\varepsilon_s + \varepsilon_i))$$

Where:

E_{cs}	=	sustained modulus of elasticity of concrete.
E_{ci}	=	instantaneous modulus of elasticity of concrete.
ε_i	=	instantaneous strain, inch/inch per psi.
ε_s	=	creep strain, inch/inch per psi.

The thermal gradients used for design are shown in Figure 3-24. The gradients for both the design accident condition and the factored load condition are based on the temperature associated with the factored pressure (factored loads are described in Section 3.8.1.3.6). The design pressure and temperature of 59 psig and 286°F became 88.5 pisg and 286°F at factored conditions.

The upper stress limit for a linear stress-strain relationship was assumed to be 3000 psi (0.6 f'_c) for use with analyses made by the use of the axisymmetric finite element analytical method. (The analyses referred to considered the concrete as uncracked and the analytical model is the entire containment.) However, the maximum predicted compressive stress was about 2559 psi. The load combination considered was $0.95D+F+P+E'+T_A$ and the location for the predicted stress was for Section EF in ring girder (see Table 3-16). Therefore, only the linear portion of the stress strain curve was used in the analyses that used the entire containment structure as a model.

The compressive stress and strain level is the highest (after the LOCA when temperature is still relatively high, 200°F, and pressure is dropping rapidly) at the inside face of the concrete at the edge of openings and also under the liner plate anchors. Neither concentration is a result of what may be considered a real load. In the case of an opening, the real stress is a result of

prestress, reduced pressure and dead load. Applying stress concentration factors to these loads still keeps the concrete in essentially the elastic range. When the strain and resulting stress from the thermal gradient are also multiplied by a stress concentration factor, the total strain and resulting stress will be above the linear stress range determined as by a uniaxial compression test. The relatively high stress level is not of real concern due to the following:

1. The concrete affected is completely surrounded by either other concrete or the penetration nozzle and liner reinforcing plate. This confinement puts the concrete in triaxial compression and gives it the ability to resist forces far in excess of that indicated by a uniaxial compression test.
2. The high state of stress and strain exist at a very local area and really have no effect on the overall containment integrity.

However, to be conservative, reinforcing steel was placed in these areas, and also, the penetration nozzle will function as compressive reinforcement.

The concrete under the liner plate anchors has some limited yielding in order to get the necessary stress distribution required to resist the liner plate self-relieving loads.

The thermal loads are a result of the temperature differential within the structure.

The liner plate is not included in the model. The strains at the inner face of the concrete surface are taken as the strains in the liner plate.

Figure 3-22 shows the inclusion of the liner plate in the finite element mesh.

Under the design accident condition or factored load condition, cracking of the concrete at the outside face would be expected. The value of the sustained modulus of elasticity of concrete, E_{cs}, was used in ACI Code 505-54 to find the stresses in concrete, reinforcing steel and liner plate from the predicted design accident thermal loads and factored accident loads.

The isostress plots shown in Figure 3-25 and Figure 3-26 do not consider the concrete cracked. The thermal stresses are combined from the individual isostress output for the cases of D + F + T and D + F + 1.5P + T. The first case is critical for concrete stresses and occurs after depressurization of the Reactor Building; the second case is critical for the reinforcing stresses and it occurs when pressure and thermal loads are combined and cause cracking at the outside face. The loading cases for isostress plots shown in Figure 3-25 are D + F + 1.15P on Sheet 1, 0.95D + F + 1.5P + T on Sheet 2, D + F on Sheet 3, and T on Sheet 4. The loading cases for isostress Plots shown in Figure 3-26 are D on Sheet 1, F on Sheet 2, T on Sheet 3, 0.95D + T on Sheet 4, F + 1.15P on Sheet 5, and F + 1.5P on Sheet 6.

The general approach of determining stresses in the concrete and reinforcement required the evaluation of the stress blocks of the cross section being analyzed.

The value of stresses was taken from the computer output in case of axisymmetric loading and from analytical solutions in case of nonaxisymmetric loading. Both computations were based on homogeneous materials; therefore, some adjustment was necessary to evaluate the true stress-strain conditions when cracks develop in the tensile zone of the concrete.

An equilibrium equation can be written considering the tension force in the reinforcement, the compressive force in the concrete and the axial force acting on the section. In this manner, the neutral axis is shifted from the position defined by the computer analyses into a position which is the function of the amount of reinforcement, the modulus ratio, and the acting axial forces.

Large axial compressive force might prevent the existence of any tension stresses, as in the loading condition D + F + T; therefore, no self-relieving action exists; the stresses are taken

directly from the computer output, except at the buttresses where analysis showed tensile stress in the concrete exceeded the modulus of rapture.

In the case of D + F + 1.5P + T, the development of cracks in the concrete decreases the thermal moment and this effect was considered; but the self-relieving properties of other loadings were not taken into account, even in places where they do exist, such as at discontinuities, e.g., the cylinder-base slab connection. This means that in analyzing the section, a reduced thermal moment was added to the unreduced moment caused by other loadings.

The thermal stresses in the containment are comparable to those developed in a reinforced concrete slab, which is restrained from rotation. The temperature varies linearly across the slab. The concrete will crack in tension and the neutral axis will be shifted toward the compressive extreme fiber. The cracking will reduce the compression at the extreme fiber and increase the tensile stress in reinforcing steel.

The following analysis is based on the equilibrium of normal forces; therefore, any normal force acting on the section must be added to the normal forces resulting from the stress diagram. The effects of Poisson's ratio are considered while the reinforcement is considered to be identical in both directions.

Stress - Strain relationship in compressed region of concrete:

$$E_c \Sigma_x = \sigma_x - \nu_c \sigma_y \qquad \text{Equation 1}$$

$$E_c \Sigma_y = -\nu_c \sigma_x + \sigma_y \qquad \text{Equation 2}$$

From the above equations (1) and (2):

$$\sigma_x = E_c \frac{\Sigma_x + \Sigma_y \nu}{1 - \nu_c{}^2} \qquad \text{Equation 3}$$

$$\sigma_y = E_c \frac{\Sigma_y + \Sigma_x \nu}{1 - \nu_c{}^2} \qquad \text{Equation 4}$$

Substituting,

$\sigma_x = \sigma_y = \sigma_c \text{and}\, \Sigma_x = \Sigma_y = \Sigma_c$ into equations (3) and (4)

$$\sigma_c = E_c \Sigma_c \frac{1}{1 - \nu_c} = 1.205 E_c \Sigma_c (\text{if} \nu_c = .17)$$

The reinforcement is acting in one direction, independently from the reinforcement in the perpendicular direction.

Example: If $E_c = 3 \times 10^6$ and $E_s = 29 \times 10^6$

$$n_R = \frac{29}{1.205 \text{ x } 3} = 8.02$$

The liner plate is acting in two directions, similar to the concrete except for the difference caused by the Poisson's ratios:

$$\sigma_L = E_s \Sigma_s \frac{1}{1 - \nu_L} = 1.35 E_s \Sigma_s \qquad \nu_L = .25$$

$$n_L = \frac{1.35 \text{ x } 29}{1.205 \text{ x } 3} = 10.83 \qquad \nu_c = .17$$

The following is an example of the use of the analytical method derived for D + F + P + T_A + E (See Table 3-16).

The concrete and reinforcement stresses are calculated by conventional methods, from the moment caused by loading other than thermal. The analyses assume homogeneous concrete sections. Those concrete and reinforcing steel stresses are then added to the thermal stresses as obtained by the method described.

Data:

$E_c = 3 \text{ x } 10^6$ psi $\nu_L = 0.25$

$E_s = 29 \text{ x } 10^6$ psi $n_R = 8.02$

$\nu_c = 0.17$ $n_L = 10.83$

Notation:

E_c Modulus of elasticity of concrete.

E_s Modulus of elasticity of steel.

n_L Modular ratio of liner plate-concrete.

n_R Modular ratio of reinforcement-concrete.

$^{\Delta}\sigma c$ Reduction of concrete compressive stress, considering cracking.

Σ_c Concrete strain.

Σ_s Steel strain.

Σ_x Concrete strain in X direction.

Σ_y Concrete strain in Y direction.

ν_c Poisson's ratio of concrete.

ν_L Poisson's ratio of liner plate.

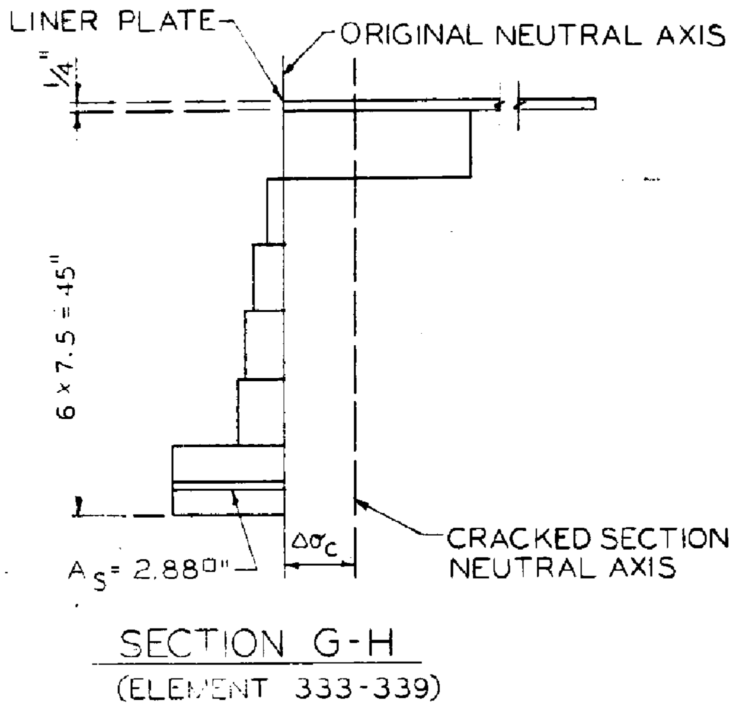
σ_c Stress in concrete.

σ_L Stress in liner plate.

σ_R Stress in reinforcement.

σ_x Stress in concrete in direction X.

STRESS BLOCK FROM THE COMPUTER OUTPUT



(Thermal) Stresses (Psi)	(Thermal) Forces (K/Ft.) Resultant
-36000	- 108.0
- 1997	- 179.7
+ 277	+ 24.9
+ 529	+47.6
+ 595	+ 53.5
+ 666	+ 60.0
+ 1467	+ 132.0

EQUILIBRIUM AFTER CRACKING

$2.88\ (1467+{}^{\Delta}\sigma c)\ 8.02 - (179.7+108)\ 1000 + {}^{\Delta}\sigma c\ (12x7.5+3x10.83) = N = -102{,}000$

$33884 + 23.1\ {}^{\Delta}\sigma c - 287700 + 124.5\ {}^{\Delta}\sigma c + 102{,}000 = 0$

$$147.6\ {}^{\Delta}\sigma c = 151{,}816$$

$${}^{\Delta}\sigma c = 1028.6$$

ASSUMED POSITION OF N. A. is O.K.

$${}^{\Delta}\sigma c = 1029 \text{ Psi}$$

σ_s(After Cracking) = (1467 + 1029) 8.02 = 20018 Psi

σ_c = -1997 + 1029 = -968 Psi

σ_R = $\sigma_{D+F+P} + \sigma_T + \sigma_E$

= -503 + 20018 ± 96 = 19611 (Tensile)

σ_c = $\sigma_{D+F+P} + \sigma_T + \sigma_E$

= 61 - 968 ± 11 = -918 (Compression)

3.8.1.4.2 Nonaxisymmetric Analysis

The nonaxisymmetric aspects of configuration or loading required various methods of analysis. The description of the methods used as applied to different parts of the containment is given below.

1. Buttresses

   The buttresses and tendon anchorage zones are defined as Class 1 elements and were designed in accordance with the general design criteria for the Reactor Building structure and with the applicable provisions of ACI 318-63, Chapter 26.

   The buttresses were analyzed for two effects, nonaxisymmetric and anchorage zone stresses. Both effects are shown in the results of a two-dimensional plane strain finite element analysis with loads acting in the plane of the coordinate system (Figure 3-27).

   At each buttress, the hoop tendons are alternately either continuous or spliced by being mutually anchored on the opposite faces of the buttress. Between the opposite anchorages, the compressive force exerted by the spliced tendon is twice as much as elsewhere. This value combined with the effect of the tendon which is not spliced will be 1.5 times the prestressing force acting outside of the buttresses. The cross-sectional area at the buttress is about 1.5 times that of the wall, so the hoop stresses as well as the hoop strains and radial displacements can be considered as being nearly constant all around the structure. Isostress plots of the plane strain analysis, Figure 3-28, confirm this.

   The vertical stresses and strains, caused by the vertical post-tensioning, become constant at a short distance away from the anchorages because of the stiffness of the cylindrical shell. Since the stresses and strains remain nearly axisymmetric despite the presence of the buttresses, their effect on the overall analysis is negligible when the structure is under dead load or prestressing loads.

   When an increasing internal pressure acts upon the structure, combined with a thermal gradient (Figure 3-29) such as at the design accident condition, the resultant forces being axisymmetric, the stiffness variation caused by the buttresses will decrease as the concrete develops cracks. The structure will then tend to shape itself to follow the direction of the acting axisymmetric resultant forces even more closely. Thus, the buttress effect is more axisymmetric at yield loads, which include factored pressure, than at design loads including pressure. This fact, combined with the design provision that alternate horizontal tendons terminate in a single buttress, indicates that the buttresses will not reduce the margins of safety available in the structure.

   The analysis of the anchorage zone stresses at the buttresses has been determined to be the most critical of all the various types of anchorage areas of the shell. The local stress distribution in the immediate vicinity of the bearing plates has been derived by the following three analysis procedures:

a. The Guyon equivalent prism method: This method is based on experimental photoelastic results as well as on equilibrium considerations of homogeneous and continuous media. It should be noted that the relative bearing plate dimensions are considered.

b. In order to include biaxial stress effects, use has been made of the experimental test results presented by S. J. Taylor at the March 1967 London Conference of the Institution of Civil Engineers (Group H, Paper 49). This paper compares test results with most of the currently used approaches (such as Guyon's equivalent prism method). It also investigates the effect of the rigid trumpet welded to the bearing plate.

c. The finite element method, assuming homogeneous and elastic material, was used in a plane strain analysis. The mesh and results are shown in Figure 3-27 and Figure 3-28.

The Guyon method yields the following results for a loading ratio (a′/a)[1] = 0.9 Maximum compressive stress under the bearing plate:

σ_c = -2400 psi

Maximum tensile stress in spalling zone:

σ spalling = +2400 psi = $-\sigma_c$

Maximum tensile stress in bursting zones:

σ maximum bursting = 0.04 P = +96 psi

S. J. Taylor's experimental results indicate that the anchor plate will give rise to a similar stress distribution pattern as Guyon's method; the main difference lies in the fact that the central bursting zone has a tensile stress peak of twice Guyon's value:

σ maximum bursting = +192 psi

By finite element analysis, the symmetric buttress loading yields a tensile peak stress in the bursting zone very close to S. J. Taylor's value:

s maximum bursting = +114 psi

A state of biaxial tension in the concrete will appear on the outside face under the loading case 1.05D + 1.5P + 1.0T_A + 1.0F. The superposition of the corresponding state of stress with the local anchor stresses reduces the load carrying capacity of the anchorage unit and caused a reduction in the maximum tensile strain to cracking.

On the other hand, the uniform compressive state of stress (vertical prestress) applied to the anchorage zone increases the load carrying capacity of the anchorage unit, with the maximum tensile strain to cracking being increased.

The design of the buttress anchor zones considered such additional vertical stresses, leading to a state of pseudo biaxial stress, the second direction being radial through the thickness.

For the above-mentioned case, 1.05D + 1.5P + 1.0T_A + 1.0F, the averaged vertical (meridional) stress component is:

$$f_a \simeq +400\text{psi}$$

[1] Ratio of width of bearing plate to width of concrete under bearing plate.

The compressive bearing plate stress at 10 inches depth below the bearing plate is:

$$f_c \simeq -1500\text{psi}$$

(Note: The steel trumpet carries 7.2 percent of the prestress force.)

Thus, the two values introduced in the biaxial stress envelopes proposed in S. J. Taylor's article:

$$f_c / f_c' = 1500/1500 = 0.3$$

$$f_c / f_c' = 400/5000 = 0.08$$

show that failure could occur if vertical reinforcing were not provided. In fact, the maximum allowable vertical averaged tensile stress according to Taylor's interaction curve is $f_a / f_c' = 0.03$; therefore, $f_a = +150$ psi.

The three dimensional stress distribution in the anchor zones was analyzed in sufficient detail to permit the rational evaluation of stress concentrations. A conical wedge segment was used as the basic design element and the radial splitting tension was determined as a tangential distribution function. The summation of splitting stresses through the entire volume of the lead-in zone established the value of the splitting force. This force is a function of the a/b ratio and the cone angle and/or, a/b and h. Several different combinations of the values were analyzed and the most critical values selected. A system analysis for the vertical splitting force was carried out based on statics and the magnitude of vertical and spalling forces were also determined.

The most unfavorable loads and load combinations were considered in the analysis of the anchorage zone and stresses based on transient thermal gradients were used in all cases where the use of a steady state gradient under-estimated the stresses and strains and were superimposed on the bursting stresses determined from the triaxial stress calculations. The computed stresses are less than the ACI allowable values. The design of the concrete reinforcement is based on this conservative analysis to provide a margin of safety similar to the other components of the Reactor Building structure and to control cracking in the anchorage zone. As a result, there is no danger of delayed rupture of the concrete under sustained load, due to local overstress and microcracking.

The reinforcing details, including the method for anchoring and splicing the reinforcing, are shown on Figure 3-30.

The reinforcement required has been designed primarily to resist tensile forces and has been located such that it will efficiently resist the tensile forces. The reinforcement was provided for load cases which create the maximum tensile forces and for other load cases the relevant shear forces or stresses were superimposed.

The possibility of the concrete breaking along shear planes was considered at the intersection of (1) the buttress with the cylinder and (2) the cylinder with the base slab.

a. Buttress - Cylinder Intersection

   An increase in the compression force at the buttress corresponds to an increase in the concrete area of the same magnitude.

b. Cylinder - Base Slab Intersection

   An analysis for the most critical radial shear conditions was performed. The difference in shear stiffness between the shell and the buttress and the remainder of the shell was

included as a shear amplification factor. The reinforcing required was less than the reinforcing provided.

The possibility of concrete breaking along a shear plane is excluded by providing ample reinforcing. In other locations, breakage along the shear plane has been excluded by the opposition of prestressing and anchor forces.

The following three sources of information were also considered in the design of the anchorage zone reinforcing:

a. Full-scale load tests of the anchorage on the same concrete mix used in the structure and review of prior uses of the anchorage.

b. The post-tensioning supplier's recommendations of anchorage reinforcing requirements.

c. Review of the final details of the combined reinforcing by the consulting firm of T. Y. Lin, Kulka, Yang, and Associates.

2. Large Opening (Equipment Hatch and Personnel Lock Opening)

The primary loads considered in the design of the equipment hatch and personnel lock opening, as for any part of the structure, were dead load, prestress, pressure, earthquake, and thermal loads. The secondary loads considered were the following effects caused by the above primary loads:

a. The deflection of tendons around the opening.

b. The curvature of the shell at the opening.

c. The thickening around the opening.

The primary loads listed are mainly membrane loads with exception of the thermal loads. In addition to membrane loads, accident pressure also produces punching shear around the edge of the opening. The values of these loads for design purposes were the magnitudes of these loads at the center of the opening. These are fairly simple to establish knowing the values of hoop and vertical prestressing, accident pressure, and the geometry and location of the opening.

Secondary loads were predicted by the following methods:

"HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED"

a. The membrane stress concentration factors and effect of the deflection of the tendons around the equipment hatch were analyzed for a flat plate by the finite element method. The stresses predicted by conventional stress concentration factors, compared with those values found from above-mentioned finite element computer program, demonstrated that the deflection of the tendons does not significantly affect the stress concentrations. This is a plane stress analysis and does not include the effect of the curvature of the shell. However, it gives an assurance of the correctness of the assumed membrane stress pattern caused by the prestressing around the opening. Results of this analysis are shown in Figure 3-31.

b. With the help of *Reference 1*, stress resultants around the large opening were found for various loading cases. Comparison of the results found from this reference, with the results of a flat plate of uniform thickness with a circular hole, showed the effect of the cylindrical curvature on stress concentrations around the opening.

Normal shear forces (relative to opening) were modified to account for the effect of twisting moments as shown in *Reference 1*. These modified shear forces are called

Kirschoff's shear forces. Horizontal wall ties were provided to resist a portion of these shear forces.

c. The effect of the thickening on the outside face around the large opening was considered using several methods. *Reference 2* was used to evaluate the effect of thickening on the stress concentration factors for membrane stress. A separate axisymmetric finite element computer analysis for a flat plate with anticipated thickening on the outside face was prepared to handle both axisymmetric and nonaxisymmetric loads to predict the effect of the concentration of hoop tendons, with respect to the Reactor Building at the top and bottom of the opening.

For the analysis of the thermal stresses around the opening, the same method was used as for the other loadings. At the edge of the opening, a uniformly distributed moment, equal but opposite to the thermal moment existing on the rest of the shell, was applied and evaluated using the methods of the preceding *Reference 1*. The effects were then superimposed on the stresses calculated for the other loads and effects.

In the case of accident temperature, after the accident pressure has already been decreased, very little or no tension develops on the outside, so thermal strains will exist without the relieving effect of the cracks. However, the liner plate will reach a high strain level and so will the concrete at the inside corner of the penetration, thereby relieving the very high stresses, but still carrying a high moment in the state of redistribution stresses.

In the case of 1.5P (prestress fully neutralized) + $1.0T_A$ (accident temperature), the cracked concrete with highly strained tension reinforcement constitutes a shell with stiffness decreased but still essentially constant in all directions. In order to control the increased hoop moment around the opening, the hoop reinforcement is about twice that of the radial reinforcement. See Figure 3-21.

The equipment hatch opening was thickened for the following reasons:

1) To reduce the larger than acceptable predicted membrane stresses around the opening.

2) To accommodate tendon placement.

3) To accommodate bonded steel reinforcing placement.

4) To compensate for the reduction in the overall shell stiffness due to the opening.

The working stress method (elastic analysis) was applied to both the load combinations for design loads, as well as for yield loads, for the analytical procedures described above. The only difference is the higher allowable stresses under yield conditions. The various factored load combinations and capacity reduction factors are specified in Section 3.8.1.3.6 and were used for the yield load combinations using the working stress design method. The design assumption of straight line variation of stresses was maintained under yield conditions.

The governing design condition for the sides of the equipment hatch opening at the outside edge of the opening is the accident condition. Under this condition, approximately 60 percent of the total bonded reinforcing steel needed at the edge of the opening at the outside face is required for the thermal load.

Excluding thermal load, the remaining stress (equivalent to approximately 40 percent of the total load including thermal) at the edge of the outside face is the sum of the following stress resultants:

1) Normal stresses resulting from membrane forces, including the effect of thickening, contribute approximately minus 35 percent (minus 14 percent of total).

2) Flexural stresses resulting from the moments caused by thickening on the outside face contribute approximately 150 percent (60 percent of total).

3) Normal and flexural stresses resulting from membrane forces and moments caused by the effect of cylindrical curvature contribute approximately minus 15 percent (minus 6 percent of total).

3. Penetrations

Analysis of the Reactor Building penetrations falls into three parts: (1) the concrete shell, (2) the liner plate reinforcement and closure to the pipe, and (3) the thermal gradients and protection requirements at the high-temperature penetrations. The three categories will be discussed separately.

a. Concrete Shell

In general, special design consideration is given to all openings in the Reactor Building. Analysis of the various openings has indicated that the degree of attention required depends upon the penetration size. Small penetrations are considered to be those with a diameter smaller than 2-½ times the shell thickness: i.e., approximately 8 feet in diameter or less. Reference 1 indicates that, for openings of 8-foot diameter or less, the curvature effect of the shell is negligible. In general, the typical concrete wall thickness has been found to be capable of taking the imposed stresses using bonded reinforcement, and the thickness is increased only as required to provide space requirements for radially deflected tendons. The induced stresses, due to normal thermal gradients and postulated rupture conditions, distribute rapidly and are of a minor nature compared to the numerous loading conditions for which the shell must be designed. The small penetrations are analyzed as holes in a plane sheet. Applied piping restraint loads due to thermal expansion or accident forces are assumed to distribute in the cylinder as stated in Reference 3. Typical details associated with these openings are indicated in Figure 3-20.

b. Liner Plate Closure

The stress concentrations around openings in the liner plate were calculated using the theory of elasticity. The stress concentrations were then reduced by the use of a thickened plate around the opening. In the case of a penetration with no appreciable external load, stud bolts are used to maintain strain compatibility between the liner plate and the concrete. Inward displacement of the liner plate at the penetration is also controlled by the stud bolts.

In the case of a pipe penetration in which significant external operating loads are imposed upon the penetration, the stress level from the external loads is limited to the design stress intensity values, S_m, given in the ASME Boiler and Pressure Vessel Code, Section III, Article 4. The stress level in the stud bolts from external loads is in accordance with the AISC Code.

The combining of stresses from all effects is performed using the methods outlined in the ASME Boiler and Pressure Vessel Code, Section III, Article 4, Figure N-414. The maximum stress intensity is the value from Figure N-415 (A) of the previously referenced code. Figure 3-32 shows a typical penetration and the applied loads.

Design stresses for the effects of pipe loads, pressure loads, dead load, and earthquake were calculated and the stress intensity kept below S_m.

The stresses from the remaining effects were combined with the above-calculated stresses and the stress intensity kept below S_a.

c. Thermal Gradient

The only high temperature lines penetrating the Reactor Building shell are the main steam and feedwater. Cooling fans and stacks are provided for the Main Steam penetrations. The feedwater penetrations rely on natural ventilation for cooling.

4. Liner Plate

There are no design conditions under which the liner plate is relied upon to assist the concrete in maintaining the integrity of the structure even though the liner will, at times, provide assistance in order to maintain deformation compatibility.

Loads are transmitted to the liner plate through the anchorage system and direct contact with the concrete and vice versa. Loads may be, at times, also transmitted by bond and/or friction with the concrete. These loads cause, or are caused by, liner strain. The liner is designed to withstand the predicted strains.

Possible cracking of concrete has been considered and reinforcing steel is provided to control the width and spacing of the cracks. In addition, the design is made such that total structural deformation remains small during the loading conditions, and that any cracking will be orders of magnitude less than that sustained in the repeated attempts to fail the prestressed concrete reactor vessel “Model 1,” and even smaller than the concrete strains of overpressure tests of “Model 2” (both at General Atomic). See Reference 4 and Reference 5.

As described, the structural integrity consequences of concrete cracking are limited by the bonded reinforcing and unbonded tendons provided in accordance with the design criteria. The effect of concrete cracking on the liner plate has also been considered. The anchor spacing and other design criteria are such that the liner will sustain orders of magnitude of strain, for example, less than did the liner of Model 1 at General Atomic (Reference 4) without tensile failure.

5. Liner Plate Anchors

The liner plate anchors were designed to preclude failure when subjected to the worst possible loading combinations. The anchors were also designed such that, in the event of a missing or failed anchor, the total integrity of the anchorage system would not be jeopardized by the failure of adjacent anchors.

The following loading conditions were considered in the design of the anchorage system:

a. Prestress

b. Internal Pressure

c. Shrinkage and Creep of Concrete

d. Thermal Gradients

e. Dead Load

f. Earthquake

g. Wind or Tornado

h. Vacuum

The following factors were considered in the design of the anchorage system:

a. Initial inward curvature of the liner plate between anchors due to fabrication and erection inaccuracies.

b. Variation of anchor spacing.

c. Misalignment of liner plate seams.

d. Variation of plate thickness.

e. Variation of liner plate material yield stress.

f. Variation of Poisson's ratio for liner plate material.

g. Cracking of concrete in anchor zone.

h. Variation of the anchor stiffness.

The anchorage system satisfies the following conditions:

a. The anchor has sufficient strength and ductility so that its energy absorbing capability is sufficient to restrain the maximum force and displacement resulting from the condition where a panel with initial outward curvature is adjacent to a panel with initial inward curvature.

b. The anchor has sufficient flexural strength to resist the bending moment which would result from Condition 5a.

c. The anchor has sufficient strength to resist radial pull-out force.

When the liner plate moves inward radially as shown in Figure 3-33, the sections will develop membrane stress due to the fact that the anchors have moved closer together. Due to initial inward curvature, the section between 1 and 4 will deflect inward giving a longer length than adjacent sections and some relaxation of membrane stress will occur. It should be noted here that section 1-4 cannot reach an unstable condition due to the manner in which it is loaded.

The first part of the solution for the liner plate and anchorage system is to calculate the amount of relaxation that occurs in section 1-4, since this value is also the force across anchor 1 if it is infinitely stiff. This solution was obtained by solving the general differential equation for beams and the use of calculus to simulate relaxation or the lengthening of section 1-4. Figure 3-33 shows the symbols for the forces that result from the first step in the solution.

Using the model shown in Figure 3-34 and evaluating the necessary spring constants, the anchor was allowed to displace.

The solution yielded a force and displacement at anchor 1, but the force in section 1-2 was (N) - $K_{R(Plate)}S_1$ and anchor 2 was no longer in force equilibrium.

The model shown in Figure 3-34 was used to allow anchor 2 to displace and then to evaluate the effects on anchor 1.

The displacement of anchor 1 was $S_1 + S'_1$ and the force on anchor 1 was $K_c(S_1 + S'_1)$. Then anchor 3 is not in force equilibrium and the solution continued to the next anchor.

After the solution was found for displacing anchor 2 and anchor 3, the pattern was established with respect to the effect on anchor 1 and by inspection, the solution considering an infinite amount of anchors was obtained in the form of a series solution.

The preceding solution yielded all necessary results. The most important results were the displacement and force on anchor 1.

Various patterns of welds attaching the angle anchors to the liner plate have been tested for ductility and strength when subjected to a transverse shear load such as N and are shown in Figure 3-35.

Using the results from these tests together with data from tests made for the Fort St. Vrain PSAR, Amendment No. 2 and Oldbury vessels, Reference 6, a range of possible spring constants was evaluated for the Oconee liner. By using the solution previously obtained together with a chosen spring constant, the amount of energy required to be absorbed by the anchor was evaluated.

By dividing the amount of energy that the system will absorb by the most probable maximum energy, the result then yielded the factor of safety.

By considering the worst possible loading condition which resulted from the listed loading conditions and conditions stated below, the results in Table 3-15 were obtained.

Case I	– Simulates a plate with a yield stress of 36 Ksi and no variation in other parameters.
Case II	– Simulates a 1.25 increase in yield stress and no variation in any other parameters.
Case III	– Simulates a 1.25 increase in yield stress, a 1.16 increase in plate thickness and a 1.08 increase for all other parameters.
Case IV	– Simulates a 1.88 increase in yield stress with no variation of any other parameters.
Case V	– Is the same as Case III except the anchor spacing has been doubled to simulate what happens if an anchor is missing or has failed.

6. Supports

In designing for structural bracket loads applied perpendicular to the plane of the liner plate, or loads transferred through the thickness of the liner plate, the following criteria and methods have been used:

a. The liner plate was thickened to reduce the predicted stress level in the plane of the liner plate. The thickened plate with the corresponding thicker weld attaching the bracket to the plate will also reduce the probability of the occurrence of a leak at this location.

b. Under the application of a real tensile load applied perpendicular to the plane of the liner plate, no yielding is to occur in the perpendicular direction. By limiting the predicted strain to 90 percent of the minimum guaranteed yield value, this criterion was satisfied.

c. The allowable stress in the perpendicular direction was calculated using the allowable predicted strain in the perpendicular direction together with the predicted stresses in the plane of the liner plate.

d. In setting the above criteria, the reduced strength and strain ability of the material perpendicular to the direction of rolling (in plane of plate) was also considered in the

bracket did not penetrate the liner thickened plate. In this case, the major stress is normal to the plane of the liner plate. The allowable stresses were reduced to 75 percent of the stress permitted in Item (3) above.

e. The necessary plate characteristics were assured by ultrasonic examination of the thickened plates for lamination defects.

3.8.1.4.3 Analysis of the Reactor Building for Steam Generator Replacement

Replacement of the steam generators required the creation of a construction opening in the shell wall of the reactor buildings. The structural analysis required to accomplish this task consisted of a finite element model which explicitly represented the vertical tendons, hoop tendons, and opening geometry. The model represented 180 degrees of the structure with the symmetry plan placed along the 0 to 180 degree azimuth of the building. The ANSYS computer program was used for this analysis.

The structure was analyzed for the load combinations given in the UFSAR and farther delineated by Oconee calculation OSC-6728. Additional load combinations were added, per ACI 318-63, that describe the structural loadings while the containment opening is in place. Each load combination was applied to the model in twelve load steps. Each step represents a significant point of change as the building is undergoing opening creation and repair.

3.8.1.5 Structural Acceptance Criteria

This section documents the manner in which the structural acceptance criteria were met by the designer.

Section 3.8.1.5.1 consists of isostress plots and tabulations of predicted stresses for the various materials. The isostress plots of the homogeneous uncracked concrete structure indicate the general stress pattern for the structure as a whole, under various loading conditions. More specific documentation is made of the predicted stresses for all materials in the structure. In these tabulations, the predicted stress is compared with the allowable to permit an easy comparison and evaluation of the adequacy of the design.

Sections 3.8.1.5.3 and 3.8.1.5.4 illustrate the actual details used in the design to implement the criteria.

3.8.1.5.1 Results of Analysis

The isostress plots, Figure 3-25 and Figure 3-26, show the three principal stresses and the direction of the principal stresses normal to the hoop direction. The principal stresses are the most significant information about the behavior of the structure under the various conditions and were a valuable aid for the final design.

The plots were prepared by a cathode-ray tube plotter. The data for plotting were taken from the stress output of the finite element computer program for the following design load cases:

$D + F$

$D + F + 1.15P$

$D + F + 1.5P + T_A$

$D + F + T_A$

The above axisymmetric loading conditions have been found to be governing in the design since they result in highest stresses at various locations in the structure.

The containment stress analysis results for structural concrete and liner plate, including shear stresses, are shown in Table 3-16.

3.8.1.5.2 Prestress Losses

In accordance with the ACI Code 318-63, the design provides for prestress losses caused by the following effects:

1. Seating of anchorage.
2. Elastic shortening of concrete.
3. Creep of concrete.
4. Shrinkage of concrete.
5. Relaxation of prestressing steel stress.
6. Frictional loss due to intended or unintended curvature in the tendons.

All of the above losses can be predicted with sufficient accuracy.

The environment of the prestress system and concrete is not appreciably different, in this case, from that found in numerous bridge and building applications. Considerable research has been done to evaluate the above items and is available to designers in assigning the allowances. Building code authorities consider it acceptable practice to develop permanent designs based on these allowances.

The following categories and values of prestress losses have been considered in the design:

Type of Loss	Assumed Value
Seating of Anchorage	None
Elastic Shortening	$\frac{f_{cpi}}{3.0\text{x}10^{6}}$ Inch/Inch
Creep of Concrete	$0.222 \times 10^{-7} \times \ln(t+1)$ Inch/Inch/psi
Shrinkage of Concrete	100×10^{-6} Inch/Inch
Relaxation of Prestressing Steel	
Hoop & Vertical	14.6% of $0.65f_s$ = 22.82 Ksi
Dome	16.04% of $0.65f_s$ = 25.06 Ksi
Frictional Loss	K = 0.0003, μ = 0.156

There is no allowance for the seating of the BBRV anchor since no slippage occurs in the anchor during transfer of the tendon load into the structure. Sample lift-off readings will be taken to confirm that any seating loss is negligible.

The loss of tendon stress due to elastic shortening was based on the change in the initial tendon relative to the last tendon stressed.

The concrete properties study conducted at Clemson University indicated an acutal creep value of 0.222×10^{-7} inch/inch/psi. Conversion of the unit creep data to hoop, vertical and dome stress gives these values of stress loss in the tendons:

	Hoop	-9.8 Ksi
	Vertical	-4.65 Ksi
	Dome	-5.60

The value used for shrinkage loss represents only that shrinkage that could occur after stressing. Since the concrete is, in general, well aged at the time of stress, little shrinkage is left to occur and add to prestress loss.

The value of relaxation loss is based on the information furnished by the tendon system vendor, The Prescon Corporation.

Frictional loss parameters for unintentional curvature (K) and intentional curvature (μ) are based on full-scale friction test data. This data indicates actual values of K = 0.0003 and μ = 0.125 versus the design values of K = 0.0003 and μ = 0.156.

Assuming that the jacking stress for tendons is 0.80 f'_s of 192,000 psi and using the above prestress loss parameters, the following tabulation shows the magnitude of the design losses and the final effective prestress at end of 40 years for a typical dome, hoop and vertical tendon.

	Dome (Ksi)	**Hoop (Ksi)**	**Vertical (Ksi)**
Jacking Stress	175.2	174.6	175.2
Friction Loss	13.3	12.26	10.4
Seating Loss	0	0	0
Elastic Loss	6.6	7.6	2.86
Creep Loss	5.6	9.8	4.65
Shrinkage Loss	2.9	2.9	2.9
Relaxation Loss	25.1	22.8	22.8
Final Effective Stress[1]	121.7	119.2	131.6

Note:

1. This force does not include the effect of pressurization which increases the prestress force.

To provide assurance of achievement of the desired level of Final Effective Prestress and that ACI 318-63 requirements are met, a written procedure was prepared for guidance of post-tensioning work. The procedures provided nominal values for end anchor forces in terms of pressure gauge readings for calibrated jack-gauge combinations. Force measurements were made at the end anchor, of course, since that is the only practical location for such measurements.

The procedure required the measured temporary jacking force, for a single tendon, to approach but not exceed 850 kips ($0.8f'_s$). Thus, the limits set by ACI 318-63 2606 (a) 1, and of the prestressing system supplier, were observed. Additionally, benefits were obtained by in place testing of the tendon to provide final assurance that the force capability exceeded that required by design. During the increase in force, measurements were required of elongation changes and force changes in order to allow documentation of compliance with ACI 318-63 2621 (a).

The procedures required that the prestressing steel be installed in the sheath before stressing for a sufficient time period that the temperatures of the prestressing steel and concrete reach essential equilibrium, to establish conformance with ACI 318-63 2621 (e). The jacking force of $0.8f'_s$ further provided for a means of equalizing the force in individual wires of a tendon to establish compliance with ACI 318-63 2621 (b). The procedures required compliance with ACI 318-63 such that, if broken wires resulted from the post-tensioning sequence, compliance with section 2621 (d) was documented. Each of the above procedures contributed to assurance that the desired level of Final Effective Prestress would be achieved.

The requirements of ACI 318-63 2606 (a) 2 state that f_s should not exceed $0.7f'_s$ for "post-tensioning" tendons immediately after anchoring.

Industry has been considering rewriting that requirement such that it has only one interpretation rather than the several now possible. Consideration is also being given to raising the value of $0.7f'_s$ or eliminating the requirement entirely and, instead, retaining the $0.8f'_s$ or some other limitation on temporary jacking force.

Paragraph 2606 (a) 2 of ACI 318-63 refers to "tendons" rather than to an individual tendon. Further, the paragraph does not refer to the location to be considered for the determination of f_s in the manner, for example, of the "temporary jacking force" referred to in 2606 (a) 1.

Two interpretations were therefore required. Both interpretations had to consider the effect of the resultant actions on both the prestressing system and structure.

The first interpretation was that the location for measurement of the seating force, used in calculating f'_s was at the end anchor and just subsequent to the measurement of the "temporary jacking force" referred to in ACI 318-63 2606 (a) 1. The advantages of this location are several. One is that it is a practical one and thus the possibility for achieving valid measurements is greater. The second is that it is the same location used for measuring the "temporary jacking force" and measurements could be made without the added complexity of additional measuring devices. The third advantage is that measurements at this location provide assurance that the calculated f'_s does not anywhere exceed the maximum f'_s to which that tendon has been subjected.

Several possible cases were considered for the second interpretation so as to allow anchoring of an individual tendon without exceeding the requirement stated for "tendons" collectively in ACI 318-63 2606 (a) 2. One such case assumed that the anchoring force for the typical tendon was that for a tendon anchored midway through the prestressing sequence. It further assumed that the losses to be assumed were one-half of the sum of elastic losses, and of the creep, shrinkage, and relaxation predicted to occur during the entire prestressing sequence. This interpretation, however, was not considered to be practical nor enforceable since it resulted in changing the seating forces as the actual (as compared to the schedule) time length of the prestressing period was dictated by weather and manpower availability.

Another case considered was that of anchoring each tendon at a measured force of 850 kips ($0.8f'_s$). Although there was no apparent detrimental effect to the prestressing system or structure, insertion of shims would be almost impossible. Further, it was concluded that this case would not establish compliance with ACI 318-63.

The case adopted was to seat each tendon with a measured "pressure" reading for the jack, at "lift-off" of the end anchor, of 775 kips (between 0.72 and 0.73 f'_s). This procedure has several advantages.

One advantage was that the force on the containment and the tendon was within the bounds of those for which it had been tested and resulted in no known detrimental effects. The second

advantage was that the stressing procedure was simplified since the stressing crews did not have to accommodate a large number of different anchoring force requirements. The third advantage was that, at the completion of stressing the last tendon, the expected losses were such that the average f'_s at the end anchors of the tendons would be less than 0.7 f'_s, thus establishing compliance with ACI 318-63 2606 (a) 1 and 2. The fourth advantage was that the percentage loss of prestressing force was less than would be the case if the tendons were anchored in such a manner the calculated value of f'_s nowhere exceeded 0.7 f'_s.

The latter advantage deserves special mention since it plays a strong role in assuring that the Final Effective Prestress equalled or exceeded the desired value. For example, if the f'_s at anchorage of the tendons were 0.1 f'_s, creep and shrinkage of concrete could result in the loss of almost all of the prestressing force. Assuming that the total losses due to creep, shrinkage, and elastic shortening equals 0.1 f'_s, then the Final Effective Prestress would be 20 percent percent of an initial prestress equivalent to 0.5 f'_s. If the initial prestress were equivalent to 0.7 f'_s, the Final Effective Prestress, neglecting relaxation for the moment, would be about 86 percent of the initial prestress. Clearly, the assurance (that the concrete creep and shrinkage losses have been properly accounted for) increases as the f'_s for the anchored tendons and tendon increases. However, this design was committed to meeting the ACI 318-63 requirement and the anchorage force for the tendons was kept at or below 0.7 f'_s in accordance with the interpretation described.

Loss of prestress in the post-tensioning system is due to material strain occurring under constant stress. Loss of prestress over time is accounted for in the design and is a time-limited aging analysis requiring review for license renewal.

In accordance with ACI 318-63 the design of the Oconee Containment post-tensioning system provides for prestress losses caused by the following:

1. Elastic shortening of concrete
2. Creep of concrete
3. Shrinkage of concrete
4. Relaxation of prestressing steel stress
5. Frictional loss due to curvature in the tendons and contact with tendon conduit.

No allowance is provided for seating of the anchor since no slippage occurs in the anchor during transfer of the tendon load into the structure.

By assuming an appropriate initial stress from tensile loading and using appropriate prestress loss parameters, the magnitude of the design losses and the final effective prestress at the end of 40 years for typical dome, vertical, and hoop tendons was calculated at the time of initial licensing.

Containment post-tensioning system surveillance will be performed in accordance with Oconee Improved Technical Specification SR 3.6.1.3. Acceptance criteria for tendon surveillance are given in terms of Prescribed Lower Limits and Minimum Required Values. Oconee Sele`cted Licensee Commitment, Oconee UFSAR, SLC 16.6.2 provides the required prescribed lower limits and minimum required values in Appendix 16.6-2, Figures 1, 2, and 3. Each prescribed lower limit line has been extended to 60 years of plant operation and remains above the minimum required values for all three tendon groups.

From the license renewal review, it was determined that the loss of prestress analysis is valid for the period of extended operation and will continue to be managed by the Containment Inservice Inspection Plan.

3.8.1.5.3 Liner Plate

The design criteria which are applied to the Reactor Building liner to assure that the specified leak rate is not exceeded under accident conditions are as follows:

1. That the liner be protected against damage by missiles (see Section 3.5.1.2).
2. That the liner plate strains be limited to allowable values that have been shown to result in leak tight vessels or pressure piping.
3. That the liner plate be prevented from developing significant distortion.
4. That all discontinuities and openings be well anchored to accommodate the forces exerted by the restrained liner plate, and that careful attention be paid to details of corners and connections to minimize the effects of discontinuities.

The most appropriate basis for establishing allowable liner plate strains is considered to be the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, Article 4. Specifically, the following sections have been adopted as guides in establishing allowable strain limits:

Paragraph N-412 (m)	Thermal Stress (2)
Paragraph N-414.5	Peak Stress Intensity Table N-413 Figure N-414, N-415 (A)
Paragraph N-412 (n)	
Paragraph N-415.1	

Implementation of the ASME design criteria requires that the liner material be prevented from experiencing significant distortion due to thermal load and that the stresses be considered from a fatigue standpoint (Paragraph N-412 (m) (2)).

The following fatigue loads are considered in the design on the liner plate:

1. Thermal cycling due to annual outdoor temperature variations. The number of cycles for this loading is 40 cycles for the plant life of 40 years.
2. Thermal cycling due to Reactor Building interior temperature varying during the startup and shutdown of the reactor system. The number of cycles for this loading is assumed to be 500 cycles.
3. Thermal cycling due to the loss-of-coolant accident will be assumed to be one cycle. Thermal load cycles in the piping systems are somewhat isolated from the liner plate penetrations by the concentric sleeves between the pipe and the liner plate. The attachment sleeve is designed in accordance with ASME Section III fatigue considerations. All penetrations are reviewed for a conservative number of cycles to be expected during the plant life.

The thermal stresses in the liner plate fall into the categories considered in Article 4, Section III, Nuclear Vessels of the ASME Boiler and Pressure Vessel Code. The allowable stresses in

Figure N-415 (A) are for alternating stress intensity for carbon steel and temperatures not exceeding 700°F.

In accordance with ASME Code, Paragraph 412 (m) (2), the liner plate is restrained against significant distortion by continuous angle anchors and never exceeds the temperature limitation of 700°F and also satisfies the criteria for limiting strains on the basis of fatigue consideration.

Paragraph 412 (n), Figure N-415 (A) of the ASME Code has been developed as a result of research, industry experience, and the proven performance of code vessels, and it is a part of a recognized design code. Figure N-415 (A) and its appropriate limitations have been used as a basis for establishing allowable liner plate strains. Since the graph in Figure N-415 (A) does not extend below ten cycles, ten cycles are being used for a loss-of-coolant accident instead of one cycle.

The maximum compressive strains are caused by accident pressure, thermal loading prestress, shrinkage and creep. The maximum strains do not exceed 0.0025 inch/inch and the liner plate always remains in a stable condition.

At all penetrations the liner plate is thickened to reduce stress concentrations in accordance with the ASME Boiler and Pressure Vessel Code 1965, Section III, Nuclear Vessels.

The liner plate is anchored as shown in Figure 3-19 with anchorage in both the longitudinal and hoop direction. The anchor spacing and welds were designed to preclude failure of an individual anchor. The load deformation tests referred to in Section 3.8.1.4.2 indicate that the alternate stitch fillet weld used to secure the anchor to the liner plate would first fail in the weld and not jeopardize the liner plate leak tight integrity.

Offsets at liner plate seams are controlled in accordance with ASME Section III Code, which allows 1/16 inch misalignment for ¼ inch plate. The flexural strains due to the moment resulting from the misalignment were added to calculate the total strain in the liner plate.

The liner plate plus structural shapes to support the liner are ASTM A36 or ASTM A516 steel. The selection of this material complies with “Safety Standard for Design, Fabrication and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors” prepared by Subcommittee N6.2, Containment, of ASA Sectional Committee N6, Reactor Safety Standards.

The interior surface of the Containment is lined with welded steel plate to provide an essentially leak tight barrier. At all penetrations, the liner plate is thickened to reduce stress concentrations. Design criteria are applied to the liner to assure that the specified leak rate is not exceeded under design basis accident conditions. The following fatigue loads were considered in the design of the liner plate and are considered to be time-limited aging analyses for the purposes of license renewal:

(a) Thermal cycling due to annual outdoor temperature variations. The number of cycles for this loading is 40 cycles for the plant life of 40 years.

(b) The combined loading of thermal cycling due to Reactor Building interior temperature varying during the startup and shutdown of the Reactor Coolant System and Type A integrated leak rate tests required by 10 CFR 50, Appendix J, including any Type A tests that may be performed if major modifications or repairs are made to the Containment pressure boundary. The number of cycles for this combined loading is assumed to be 500 cycles.

(c) Thermal cycling due to the loss-of-coolant accident will be assumed to be one cycle.

(d) Thermal load cycles in the piping systems are somewhat isolated from the liner plate penetrations by concentric sleeves between the pipe and the liner plate. The attachment sleeve is designed in accordance with ASME Section III considerations. All penetrations are reviewed for a conservative number of cycles to be expected during the plant life.

From the license renewal review, it was determined that the existing analyses of thermal fatigue for the Containment penetrations are valid for the period of extended operation.

3.8.1.5.4 Penetrations

Penetrations conform to the applicable sections of ASA N6.2-1965, "Safety Standard for the Design, Fabrication and Maintenance of Steel Containment Structures for Stationary Nuclear Power Reactors." Piping penetrations 25, 26, 27, 28, 63 and 64 conform to the requirements of ASME Section III, Subsections NE and NC, 1992 Edition, including all 1992 Addenda. Subsection NC applies only to the piping portion of the penetration. All personnel locks and any portion of the equipment access door extending beyond the concrete shell conform in all respects to the requirements of ASME Section III, Nuclear Vessels Code.

The basis for limiting strains in the penetration steel is the ASME Boiler and Pressure Vessel Code for Nuclear Vessels, Section III, Article 4, 1965, and therefore, the penetration structural and leak tightness integrity are maintained. Local heating of the concrete immediately around the penetration will develop compressive stress in the concrete adjacent to the penetration and a negligible amount of tensile stress over a large area. The mild steel reinforcing added around penetrations distributes local compressive stresses for overall structural integrity.

Horizontal and vertical bonded reinforcement is provided to help resist membrane and flexural loads at the penetrations. This reinforcement was located on both the inside and outside face of the concrete. Stirrups were also used to assist in resisting shear loads.

Local crushing of the concrete due to deflection of the reinforcing or tendons is precluded by the following details:

1. The surface reinforcements either have a very large radius such as hoop bars concentric with the penetration or are practically straight, having only standard hooks as anchorages where necessary.
2. The tendons are bent around penetrations at a minimum radius of approximately 20 feet. Maximum tendon force at initial prestress is 850 kips, which results in a bearing stress of about 880 psi on the concrete.

It is also important to note that the deflected tendons are continuous past the openings and are isolated from the local effects of stress concentrations by virtue of being unbonded.

In accordance with ASME Section III, piping penetration reinforcing plates and the weldment of the pipe closure to it are stress relieved. This code requirement and the grouping of penetrations into large shop assemblies permit a minimum of field welding at penetrations.

The personnel hatch consists of a steel cylinder with 3 ft-6 in. x 6 ft-8 in. doors at each end interlocked so that only one door can be open at any time. The hatch is designed to withstand all Reactor Building design conditions with either or both doors closed and locked. Doors open toward the center of the Reactor Building and are thus sealed under Reactor Building pressure. Design live load on the hatch floor is 200 psf.

Operation of the hatch is normally manual, that is, without power assist. Interlocks will prevent opening both doors at once.

Double gaskets are provided on the outer door to permit periodic pressurizing of the space between the gaskets from outside the Reactor Building. The hatch barrel may be pressurized to demonstrate its leak tightness without pressurizing the Reactor Building. Auxiliary restraint beams are attached to the inner door in this case to help the locking bars to resist internal lock pressure, which is greatly in excess of the Reactor Building design external pressure of 3 psig. The personnel hatch was pneumatically shop tested for pressure and leakage.

Figure 3-21 shows the principal features of the personnel hatch.

An emergency hatch is provided with 30 inch diameter doors. Its features are identical to the personnel hatch.

In order to support outage work activities during refueling operations, a temporary cover plate can be placed in the emergency hatch. The cover plate provides emergency hatch closure during refueling operations and is considered to be closed when a visual inspection shows no obvious leakage path.

The cover plate is approximately 36-inches in diameter and approximately 1-inch thick. The cover plate has multiple penetrations through it of various diameters. These penetrations have sleeves of varying lengths inserted through them and welded in place. The cover plate is installed and sealed against the inner emergency hatch door flange gasket. Positive sealing of the cover plate is accomplished by the use of RTV sealants. The cover plate is visually inspected to ensure that no gaps exist. All cables and hoses routed through the sleeves on the cover plate will also be installed and sealed. The sleeves will also be inspected to ensure that no gaps exist. Leak testing is not required prior to beginning fuel handling operations. Therefore, visual inspection of the cover plate over the emergency hatch satisfies the requirement that the emergency hatch be closed.

A 19-foot diameter equipment hatch opening to the outside provides the movement of large items into and out of the Reactor Building. The door is secured by bolts on the inside of the Reactor Building wall and can be opened only from inside the Reactor Building. It is opened only when the reactor is subcritical. Double gaskets on the door permit the seals to be pressurized from outside the Reactor Building to check the integrity of the seals. During operation, the space between the double gaskets is vented to the penetration room.

Figure 3-21 shows the principal features of the equipment hatch.

1. Piping and Ventilation Penetrations

   All piping and ventilation penetrations are of the rigid welded type and are solidly anchored to the Reactor Building wall or foundation slab, thus precluding any requirements for expansion bellows. All penetrations and anchorages are designed for the forces and moments resulting from operating conditions. External guides and stops are provided as required to limit motions, bending and torsional moments to prevent rupture of the penetrations and the adjacent liner plate for postulated pipe rupture. Piping and ventilation penetrations have no provision for individual testing since they are of all-welded construction.

   For typical details of piping penetrations, see Figure 3-20.

2. Electrical Penetrations

   Medium voltage penetrations for reactor coolant pump power shown on Figure 3-20 are canister type using glass sealed bushings for conductor seals. The canisters are filled to a positive pressure with an inert gas. The assemblies are bolted to mating flanges which

incorporate double “0” ring seals with a test port between as a means of verifying seal integrity.

Low voltage power, control and instrumentation assemblies are shown on Figure 3-20. These assemblies are designed to bolt to mating flanges mounted inside the Reactor Building. Electrical penetrations are designed to maintain containment integrity; thus, reliable environmental seals must be maintained. To accomplish the required reactor building environment seals, the interface between the mounting flange and the penetration header plate must be sealed and also the interfaces between the header plate and individual penetration feedthrough conductors must be sealed.

Dual “O” rings are used to complete the seal between the mating flange and the penetration header plate. The mating flange is welded to the penetration nozzle. The space between the “O” ring seals is charged with an inert gas. The charged gas space is piped to a charging valve located outside of the Reactor Building, which allows leakage around the “O” ring seals to be detected.

Depending upon the type of penetration utilized in a particular application, two different schemes are used to accomplish the seals between the header plate and the penetration feedthrough conductors. One scheme accomplishes the seal by utilizing two header plates to which are welded glass to metal sealed conductors. Another scheme accomplishes the feedthrough seals by use of polysulfide to metal sealed conductors. In both schemes, the space between the seals is also charged with an inert gas. The charged gas space is piped to a pressure gauge and a charging valve located outside of the Reactor Building, which allows leakage to be tested.

3.8.1.5.5 Miscellaneous Considerations

In various cases, it has been the designer's decision to provide structural adequacy beyond that required by the design criteria. Those cases are as follows:

1. Section 3.8.1.3.4 requires a minimum of 0.15 percent bonded reinforcing steel in two perpendicular directions on the exterior faces of the wall and dome for proper crack control. Due to the weather exposure, a minimum of approximately 0.5 percent was provided.
2. Section 3.8.1.3.4 requires a minimum of 0.15 percent bonded steel reinforcing (as stated above) for any location. At the base of the cylinder, the controlling design case requires 0.25 percent vertical reinforcing. As a result of pursuing the recommendation of the AEC Staff to further investigate current research on shear in concrete, several steps were taken:
   a. The work of Dr. Alan H. Mattock was reviewed and he was retained as a consultant on the implementation of the current research being conducted under his direction. The criteria has been updated in accordance with his recommendations.
   b. Concurrently with reviewing Dr. Mattock's work, the firm of T. Y. Lin, Kulka, Yang and Associates was consulted to review the detailed design of the cylinder to slab connection. It was their recommendation to use approximately 0.5 percent reinforcing rather than the 0.25 percent reinforcing indicated by the detailed design analysis for the vertical wall dowels. This increase would assure that there was sufficient flexural steel to place the section within the lower limits of Mattock's test data (approximately 0.3 percent) to prevent flexural cracking from adversely affecting the shear capability of the section.

   Additional information concerning structural acceptance criteria for liner plate, penetrations, supports, and buttresses can be found in Section 3.8.1.4.2.

3.8.1.6 Materials, Quality Control, and Special Construction Techniques

Test, code, and cleanliness requirements accompanied each specification or purchase order for materials and equipment. Hydrostatic, leak, metallurgical, electrical, and other tests to be performed by the supplying manufacturers are enumerated in the specifications together with the requirements, if any, for test witnessing by an inspector. Fabrication and cleanliness standards, including final cleaning and sealing, are described together with shipping procedures. Standards and tests are specified in accordance with applicable regulations, recognized technical society codes and current industrial practices. Inspection is performed in the shops of vendors and subcontractors as necessary to verify compliance with specifications.

3.8.1.6.1 Concrete

An experienced full-time concrete inspector continuously checked concrete batching and placing operations.

Concrete mixes were designed and the associated tests run by the concrete testing laboratory at Clemson University in accordance with ACI 613. During construction, the field inspection personnel made minor modifications that were necessitated by variations in aggregate gradation or moisture content.

In determining the design mixes; air content, slump, and bleeding tests were run in accordance with the appropriate ASTM Specifications.

The concrete ingredients consist of Type II Cement (ASTM C-150), Solar 25 air entraining agent (ASTM C-260), Plastiment water reducing agent (ASTM C-494), Aggregate (ASTM C-33), and water that was free from injurious amounts of chlorides, sulphates, oil, acid, alkali, organic matter, or other deleterious substances.

Fine aggregate consists of clean, sharp, washed sand of uniform gradation from Becker County Hagood Quarry. Coarse aggregate consists of washed crushed rock having hard, strong, durable pieces of Gaffney marble from Campbell Limestone Company. The acceptability of the aggregate was based on Los Angeles Abrasion, Clay Lumps Natural Aggregates, Material Finer Number 200 Sieve, Organic impurities effect on Mortar, Organic impurities - Sands, Potential Reactivity, Seive Analysis, Soundness, Specific Gravity and Absorption, and Petrographic tests based on the appropriate ASTM Specifications.

Acceptability of aggregates is based on the following ASTM tests. These are performed by a qualified testing laboratory.

Test	ASTM	
LA Abrasion	C131	
Clay Lumps Natural Aggregate	C142	
Material Finer No. 200 Sieve	C117	
Mortar making properties	C87	
Organic impurities	C40	
Potential Reactivity (chemical)	C289	
Potential Reactivity (mortar bar)	*C227*	"Historical Information Not Required to be Revised"
Sieve Analysis	C136	

Test	ASTM
Soundness	C88
Specific Gravity and Absorption Coarse	C127
Specific Gravity and Absorption Fine	C128

3.8.1.6.1.1 Cement

Cement conforms to ASTM C150 and tested to ASTM C114.

The manufacturer submits certified copies of mill test reports showing the chemical composition and certifying that the cement complies with the specification on each shipment delivered to the site. In addition to the manufacturer's tests, cement is sampled periodically at the site and tested to ascertain conformance with ASTM Specification C150.

3.8.1.6.1.2 Water

Water is potable and does not contain impurities in amounts that will cause a change of more than 25 percent in setting time for the Portland Cement, nor a reduction in the compressive strength of mortar of more than 5 percent as compared with results obtained using distilled water.

3.8.1.6.1.3 Admixtures

Admixtures, as to be determined by detailed mix design, conform to applicable ASTM Specification covering such materials and their testing.

3.8.1.6.1.4 Concrete Test Cylinders

Concrete cylinders for compression testing are made and stripped within 24 hours after casting, and marked and stored in the curing room. These cylinders are made in accordance with ASTM C31, “Making and Curing Concrete Compression and Flexure Test Specimens in the Field.”

Slump, air content, and temperature are taken when cylinders are cast and for each 35 yards of concrete placed. Slump tests are performed in accordance with ASTM C143, “Standard Method of Test for Slump of Portland Cement Concrete.” Air tests are performed in accordance with ASTM C231, “Standard Method of Test of Air Content of Freshly Mixed Concrete by the Pressure Method.” Compressive strength tests are made in accordance with ASTM C39, “Method of Test for Compressive Strength of Molded Concrete Cylinders.”

Six standard test cylinders are obtained and molded for concrete placed in excess of 10 cubic yards in any one day, with 6 additional cylinders for each successive 100 cubic yards placed. Two cylinders are tested at the age of 7, 28, and 90 days.

Concrete mixes are designed in accordance with “Recommended Practice for Selecting Proportions for Concrete” (ACI 613), using materials qualified and accepted for the work; and the strength, workability, and other characteristics of the mixes are ascertained before placement. Duke Power's concrete control laboratory is set up on the Oconee site. A batch-plant inspector is provided, and testing as shown below is performed. Field control is in accordance with the “Manual of Concrete Inspection” as reported by ACI Committee 611.

3.8.1.6.1.5 Mix Design

Only those mixes meeting the design requirements specified for Reactor Building concrete are used. Trial mixes are tested in accordance with the applicable ASTM Codes as follows:

Test	ASTM
Air Content	C231
Slump	C143
Bleeding	C232
Making and Curing Cylinders in Laboratory	C192
Compressive Strength Tests	C39

Six cylinders are cast from each design mix for two tests on each of the following days: 7, 28, and 90.

Test cylinders are cast from the mix proportions selected for construction and the following concrete properties determined:

Uniaxial creep

Modulus of elasticity and Poisson's Ratio

Autogenous shrinkage

Thermal diffusivity

Thermal coefficient of expansion

Compressive strength

3.8.1.6.1.6 Aggregates

Aggregate testing is performed as follows:

1. Sand sample for gradation (ASTM C33 Fine Aggregate)
2. Organic test on sand (ASTM C40)
3. 3/4" sample for gradation (ASTM C33, Size No. 67)
4. 1-½ inch sample for gradation (ASTM C33, Size No. 4)
5. Check for proportion of flat and elongated particles.

3.8.1.6.1.7 Concrete Construction

Cast-in-place concrete was used to construct the Reactor Building shell. The base slab construction was performed in seven pours utilizing large block pours. After the completion of the base slab steel liner erection and testing, an additional concrete slab was placed to provide protection for the floor liner.

The concrete placement in the walls was done in 10 ft high lifts with vertical joints at the radial center line of each of six buttresses. Cantilevered jump forms on the exterior face and interior steel wall liner served as the forms for the wall concrete.

The dome liner plate, temporarily supported by 18 radial steel trusses and purlins, served as an inner form for the initial 8 inch thick pour in the dome. The weight of the subsequent pour was

supported in turn by the initial 8 inch pour. The trusses were lowered away from the liner plate after the initial 8 inches of concrete had reached design strength, but prior to the placing of the balance of the dome concrete.

The standards or specifications on quality control and tests of concrete during construction are equal to or better than requirements of ACI 301. Some of the areas where quality control exceeds the requirements of ACI 301 are as follows:

1. Requirements for water quality.
2. Placing temperature of concrete.
3. Requirements for aggregate acceptability.
4. Requirements for test cylinders.

Horizontal construction joints are prepared for receiving the next lift by blasting with compressed air. Surface set retardant compounds are not used.

Horizontal surfaces are wetted and covered with a coating of mortar of the same cement-sand ratio as used in the concrete immediately before the concrete is placed.

Vertical joints are also blasted with compressed air, cleaned, and wetted before placing concrete.

Vertical joints are placed at the center of each buttress to take advantage of the 50 percent additional horizontal prestress due to the overlapping of the anchored hoop tendons.

Horizontal joints between buttresses are at the same elevation. These joints are prepared as stated above to provide maximum possible bond. Principal tension in the membrane is limited to $3\sqrt{f'_c}$.

3.8.1.6.1.8 Construction Opening for Steam Generator Replacement

Replacement concrete for the construction opening was developed through an exhaustive testing program developed especially for the purpose. The details are delineated in Specification SGRP-SPEC-C-003, Reactor Building – SGRP Construction Opening Concrete Work. The testing regiment covers all the original requirements for reactor building concrete plus testing to verify the shrinkage characteristics of the mix. The development efforts insure that the repair mix is compatible with the existing concrete and performs acceptability over the life of the building.

3.8.1.6.2 Prestressing

These instructions and methods describe the quality control standards and measures applied in the control, manufacture, and field installation of the prestressing phase of construction of the Reactor Building.

The BBRV post-tensioning system furnished by The Prescon Corporation was used. Tendons replaced during the steam generator replacement are by PSC. Each tendon consists of ninety ¼ inch diameter wires conforming with ASTM A-421-65T, two anchor heads and two sets of shims conforming with American Iron and Steel Institute (AISI) C-1045 HR. The tendon sheathing system consists of spirally wound carbon steel tubing connecting to a trumplate (bearing plate and trumpet) at each end. A513 Type 5 carbon steel tube was used for the replacement tendon sheathing in closing the construction opening following steam generator replacement. The bearing plates were fabricated from steel plate conforming with AISI C-1045 HR and the trumpets from AISI C-1010 HREW material.

The C-1045 HR material used for the stressing washers, dead-end washers, shims, and bearing plates was modified by the addition of silicon to obtain a finer grain structure and cleaner steel than unmodified C-1045. The average depth of the heat affected zone resulting from flame cutting is approximately 1/16 inch and the improved general ductility of modified C-1045 material should increase resistance to cracks starting in heat affected zones and decrease the probability of crack propagation. However, a cracked plate could continue to perform its function without loss of structural integrity and should be evaluated in terms of actual functional ability.

Flame cutting is limited to sizing the bearing plate and making the center hole. All other holes in the bearing plate are drilled. The dead-end washer is flame cut to size and drilled for the tendon wires. No flame cutting is performed on the stressing washer.

3.8.1.6.2.1 Control

Supervision

The subcontractor furnishes competent, experienced supervision of the tendon installation and tensioning operation until completion of post-tensioning. The above individual exercises a close check and rigid control of all post-tensioning operations, as necessary, for full compliance with specifications.

Inspection of Duke's Work

The subcontractor is responsible for the inspection of Duke's handling and installation of tendon sheaths and bearing plates. To this end, he provides a competent technical representative to check the installation of these items by Duke. If any of Duke's work or actions jeopardize the subcontractor's work, he notifies Duke's Resident Engineer in writing. Failure to do this constitutes acceptance of Duke's work as it affects subcontractor's responsibilities.

Arrangement of Prestressing Tendons

The configuration of the tendons in the dome is based on a three-way tendon system consisting of three groups of tendons oriented at 120 degrees with respect to each other. The vertical cylinder wall is provided with a system of vertical and horizontal (hoop) tendons. Hoop tendons are placed in a 120 degree system in which three tendons form a complete ring. Six buttresses are used as anchorages.

3.8.1.6.2.2 Detail Shop Drawings

“HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED”

Subcontractor

Upon award of the contract, Duke furnished engineering design drawings which were issued for construction of the prestressing work providing information required for the preparation of shop detail drawings by the subcontractor. The subcontractor furnished the following detail drawings and erection drawings to Duke:

1. *Outside dimensions of sheathing proposed for the tendon.*
2. *Complete details of the post-tensioned wall and dome including dimensional locations of the tendons and necessary equipment and materials to place the tendons.*
3. *Tendon characteristics indicating the A_s, F'_s, f_{sy}, and a typical stress-strain curve for the tendon used, as well as tendon force capability.*

4. *Details of anchorages, bearing plates, and other accessories pertinent to the post-tensioning system.*
5. *Erection drawings showing clearly the marking and positioning of tendons, anchorages, and sheaths, and details showing alignment and setting tolerances required.*
6. *Stressing sequence drawings.*

3.8.1.6.2.3 Prestressing Steel

Materials and Fabrication

High strength steel wires are in accordance with ASTM A416 or A421 as a minimum requirement.

Wire materials used for steam generator replacement: ASTM A421 Type BA, 0.25” diameter and guaranteed ultimate strength of 240 ksi.

Shim materials used for steam generator replacement are: ASTM A656, Type 7, Grade 80; ASTM A656, Type 7, Grade 70; ASTM A737, Grade C or ASTM A633, Grade E.

Wires are to be straightened if necessary to produce equal stress in all wires or wire groups that are to be stressed simultaneously or when necessary to insure proper positioning in sheaths. However, wires showing a permanent set are not to be straightened or installed if the bend exceeds 60 degrees and the radius is less than 1.25 inches.

Tests were made on wire bent to 30, 60, and 90 degrees with a bend radius of 1.25 inches (5 times wire diameter) and wire bent to 30 and 60 degrees with a zero radius. The test specimens were from two different heats of ¼ inch diameter wire. All specimens within one test series were from the same heat and coil. In the sequence of cutting, every sixth specimen fell into the same group. The first group consisted of straight specimens for comparison.

Specimens were cut to a length of 15-½ inches, bent to the prescribed angle and radius in a bend-tester, and straightened. The specimens were button headed on each end and tensile tested to failure. The test results presented in Table 3-22 show that the strength of prestressing wire is not affected by bending the wire 60 degrees around a 1.25 inch radius pin.

The button head is cold formed to a nominal diameter of 3/8 inch symmetrically about the axis of the wires. If splitting is consistent and appears in all heads or if there are more than two splits in which the opening exceeds 0.06 inch per head, the wire is rejected. No forming process is used that caused indentation in the wire. Wires showing indentations are rejected. Wires showing fabrication defects, wires having welds or joints made during manufacture, or broken wires are removed and replaced.

The BBR Bureau Standard for button head splits is a maximum number of two splits with a width of 0.06 inch. The Prescon Corporation has run tests on button heads with splits; and based on an evaluation of the test results, the BBR Bureau Standard is acceptable.

Protection

Prestressing steel is protected from mechanical damage and corrosion during shipment, storage, installation, and tensioning. A thin film of No-Ox-Id (R) 500, as manufactured by Dearborn Chemical Company or Visconorust 1601, manufactured by Viscosity Oil Company, is applied to the prestressing steel after fabrication in accordance with the manufacturer's instructions. The steel is then wrapped before shipment to the site. The steel is not handled, shipped, or stored in a manner that will cause a permanent set or notch, change its material

properties, or expose it to inclement weather or injurious agents such as chloride containing solutions. Damaged or corroded tendons are rejected.

Installation

The tendon installation prestressing procedure was carried out as follows:

1. To assure a clear passage for the tendons, a “sheathing Rabbit” was run through the sheathing both prior to and following placement of the concrete.
2. Tendons were uncoiled and pulled through the sheathing unfinished end first.
3. The unfinished end of the tendons was pulled out with enough length exposed so that field attachment of the anchor head and buttonheading could be performed. To allow this operation, trumplates on the opposite end had an enlarged diameter to permit pulling the shop finished ends with their anchor heads.
4. The anchor heads were attached and the tendon wires buttonheaded.
5. The shop finished end of the tendon was pulled back and the stressing jack attached.
6. The post-tensioning was done by jacking to the permissible overstressing force to compensate for friction and placing the shims precut to lengths corresponding to the calculated elongation. Proper tendon stress was achieved by comparing both jack pressure and tendon elongation against previously calculated values. The vertical tendons were prestressed from either one or both ends, while the horizontal and dome tendons were prestressed from both ends.
7. The grease caps were bolted onto anchorages at both ends and made ready for pumping the tendon sheathing filler material.
8. The tendon sheaths and grease caps were filled with sheathing filler and sealed. The sheathing filler material had limitations specified for deleterious water soluble salts.

   Corrosion protection of the tendons and interior surface of sheathing was applied prior to shipment.

   Tendon sheaths mark 24H34, 31H34 and 34V14 on Oconee 1, 31H21 on Oconee 2, 34V13 and 34V25 on Oconee 3 were plugged. The location of the plugged sheaths are shown in Figure 3-36.

3.8.1.6.2.4 Anchorages and Bearing Plates

Anchorages

Anchorages will develop the minimum guaranteed ultimate strength of the tendon and the minimum elongation of the tendon material as required by the applicable ASTM specification.

Bearing Plates

Bearing plates are capable of developing the ultimate strength of the tendon and distributing the bearing load over the bearing surface of the concrete. Bearing plates conform to the following requirements:

1. The transfer unit compressive stress on the concrete directly underneath the plate or assembly is in conformance with the ACI Code 318-63, latest edition.
2. Bending stresses in the plates induced by the pull of the prestressing steel shall not exceed 22,000 psi for structural steel and 15,000 psi for cast steel, except as experimental data may indicate that higher stresses are satisfactory.

3. Materials shall meet requirements of ASTM A36 for structural shapes or ASTM A148, Grade 80-40 for cast steel, or higher quality materials approved by Duke to meet strain requirements.
4. Design, fabrication, and erection shall meet the requirements of the latest AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings."

3.8.1.6.2.4.1 Tendons Installed During Steam Generator Replacement

New tendons installed during the SGR are of the BBRV system type currently existing in the structure, however they are manufactured in accordance with the Inryco design instead of the Prescon system currently used. The differences in these two systems are minor, head material is AISI 4140 and the wire button heads are slightly larger, which allows the use of the current maintenance equipment and ISI procedures.

Anchor Head Materials:

ITEM	VALUE
Material	AISI 4140
Yield	89 KSI
Ultimate	118 KSI
Elongation	12%
R/A	20%
Hardness	Rc 29 TO 33
Heat Treatment	As Needed for Performance to Spec.

Wire Materials:

ASTM A421 Type BA, 0.25" diameter, and Guaranteed ultimate strength of 240ksi

Shim Materials:

ASTM A656, type 7 Grade 80,

ASTM A656, type 7, Grade 70,

ASTM A737, Grade C, or

ASTM A633, Grade E.

3.8.1.6.2.5 Sheaths

Materials

Sheaths for post-tensioning tendons are ungalvanized corrugated articulated tubing and meet the following requirements:

1. The internal diameter is adequate to allow insertion of prestressing steel after concrete placement.
2. The sheaths will withstand the placing of concrete at a pour rate of two feet per hour (with mechanical vibration) without ovalling or changing alignment.
3. Sheaths are protected from corrosion during storage.

A513 Type 5 carbon steel tube was used for replacement tendon sheathing in closing the construction opening following steam generator replacement.

Sheath Fabrication

The sheaths are cut to length and bent to shape. The bending is accomplished without wrinkling the metal. Dented or wrinkled sheaths are replaced. Finished bent or straight dimensions are in accordance with approved drawings.

Installation (by Duke)

Sheaths are accurately installed in the forms at the location shown on the drawings to a tolerance of ± one-half (½) inch, except as otherwise indicated on the drawings. The sheaths are supported in such a manner as to prevent displacement during concrete placement. The sheath is supported at the ends and at such intervals as are in accordance with the drawings. Damaged or improperly bent sheaths are rejected.

Cleaning and Venting

Just prior to insertion of the tendon, the sheath is cleaned by the use of compressed air or other suitable means.

3.8.1.6.2.6 Corrosion Protective Grease

Corrosion protection is provided by grease injected into the sheaths under pressure. Grease will be Visconorust 2090P or 2090P-4 manufactured by Viscosity Oil Company.

The grease is sampled and laboratory tested for chemical analysis to establish conformance with specifications and for deleterious substances such as water soluble chlorides, nitrates, and sulfides.

Visconorust Casing Filler is a petroleum base corrosion preventive designed for bulk application and extended protection.

It has:

1. A three phase protective system starting with a polar agent preferentially wetting the wires and displacing any moisture, rust preventive additives molecularly attached to the wetting agent and a petroleum barrier completing the resistant coating.
2. The property to emulsify any moisture picked up in the system while being pumped through the casing and either carrying it out the other end or nullifying its rusting ability if the moisture is trapped in the casing.
3. Reserve Alkalinity - The basic formulation of Visconorust casing fillers are very stable and resistant to exterior moisture encroachment as well as mild acids and alkali. However, because of the probability of picking up moisture as the rust preventive is pumped through the tendons, an additional safety factor, besides the barrier action, is available to neutralize any acids that might form between the interface of the moisture and rust preventive.

   Tests have been run using volatile acids, such as Hydro Bromic Acid, in an attempt to penetrate the Visconorust casing filler film and cause corrosion without success.
4. Only a trace amount of water soluble chlorides, sulfides, or nitrates.
5. A plugging agent designed to supplement the natural tendency of the microwax crystals and amorphus solid components to form a filter cake bridging any hair line cracks in the concrete, with which the casing filler might come in contact.

6. Self-healing qualities at the ambient temperature expected during operation, to take care of any voids created by wire movement.
7. Thixotropic properties that provide pumpability below 50°F.
8. Radiation Resistance:

   Visconorust casing fillers have been subjected to 1 x 10^6 rads/(min). Results show that the Gamma rays did not have any material effect on either the physical or chemical structure (as noted by a negligible change in base number).

   Corroboration of the test results is readily noted in extensive literature on this subject, a few of which are listed below:

   Bibliography:

   a. The Lubrication of Nuclear Power Plants by R. S. Barnett - NLGI - October 1960.
   b. How Radiation Affects Petroleum Lubricants - Power, Vol. 100 December 1956, Page 164.
   c. Conventional Lubricants Are Sufficiently Radiation Resistant for Most Nuclear Power Reactor Applications by E. D. Reeves SAE Journal Vol. 66, May, 1956, Page 56-57.
   d. Organic Lubricants and Polymers for Nuclear Power Plants by Bolt and Carroll.

   The amount of nitrate found in the 90,000 gallons of Nuclear Grade material made for Palisades, Point Beach, and Turkey Point plants, so far, was “0” and practically, in order to keep the trace amounts allowed, be it 2 or 4, the amounts must be kept at zero. However, the refinery requires the use of 4 parts per million figure as a maximum.

   Infra-red spectographic analysis shows Visconorust 2090P and NO-OX-ID CM to be quite similar with approximately the same amounts of wetting agents and rust preventives in the petroleum carriers.

PERFORMANCE DATA

Item	NO-OX-ID	Visconorust 2090P	ASTM Method	2090P-4
Weight Per Gal.	7.2 - 7.5 lbs.	7.3 - 7.6 lbs.	--	7.3-7.4
Pour Point	110° - 120°F	--	D-97	
Flash Point (coc)	400°F	385°F	D-92	420°F
Viscosity @ 150°F	125 - 150 SSU	116 SSU	D-88	
Viscosity @ 210°F	55 - 75 SSU	59 SSU	D-88	150 - 300°F
Spec. Grav @ 60°F	0.88 - 0.90	0.88 - 0.91	D-287/1298	0.88 - 0.94
Pene. (cone) @ 77°F	325 – 370	370	D-937	170 - 200
Water Sol Chlorides	1 PPM	1 PPM	D-512	2 PPM

PERFORMANCE DATA				
Item	**NO-OX-ID**	**Visconorust 2090P**	**ASTM Method**	**2090P-4**
Water Sol Nitrates	2 PPM	4 PPM	D-1255/992-78	4 PPM
Water Sol Sulfides	1 PPM	1 PPM	D-992/APHA 4500S	2 PPM
Phenoloc Bodies (As Phenol)	1 PPM	1 PPM	--	--
Shrinkage Factor (150°F to 70°F)	3.5 - 4.5%	3.5 - 4.5%	--	--
Total Base Number	--	3	D-974	35

3.8.1.6.2.7 Tensioning Schedule

Prestressing begins after the concrete in the walls and the dome has reached the specified f′c. The dome and the hoop tendons are tensioned from both ends, and the vertical tendons are tensioned from either the top end or from both ends. Six jacks are used throughout the post-tensioning operations.

Phase 1

Twelve hoop tendons above elevation 943 feet + 6 inches on buttresses at 90 degrees, 210 degrees, and 330 degrees.

Phase 2

Thirty-six dome tendons in the periphery of the dome.

Phase 3

Twelve hoop tendons above elevation 943 feet + 6 inches on buttresses at 30 degrees, 150 degrees, and 270 degrees.

Phase 4

Remaining 126 dome tendons.

Phase 5

One hundred and forty-one hoop tendons from elevation 865 feet + 0 inches to elevation 943 feet + 6 inches on buttresses at 30 degrees, 150 degrees, and 270 degrees.

Phase 6

Close the construction opening if not closed prior to Phase 6.

Phase 7

One hundred and fifty-three hoop tendons from elevation 775 feet + 0 inches to elevation 865 feet + 0 inches on buttresses at 30 degrees, 150 degrees, and 270 degrees.

Phase 8

Forty-two hoop tendons from elevation 776 feet + 0 inches to elevation 801 feet + 6 inches on buttresses at 90 degrees, 210 degrees, and 330 degrees.

Phase 9

One hundred and seventy-six vertical tendons.

Phase 10

Two hundred and fifty-two hoop tendons from elevation 801 feet + 6 inches to elevation 943 feet + 6 inches on buttresses at 90 degrees, 210 degrees, and 330 degrees.

Phase 11

Ten hoop tendons above elevation 949 feet + 10-2/3 inches on buttresses at 90 degrees, 210 degrees, and 330 degrees.

Phase 12

Ten hoop tendons above elevation 949 feet + 10-2/3 inches on buttresses at 30 degrees, 150 degrees, and 270 degrees.

Force and Stress Measurements

Force and stress measurements are made by measurement of elongation of the prestressing steel after taking up initial slack and comparing it with the force indicated by the jack-dynamometer or pressure gauge. Force jack pressure gauge or dynamometer combinations are calibrated against known precise standards before application of prestressing force. All gauges are calibrated on a dead weight calibration apparatus. The presence of two gauges, one gauge on the pump and one gauge on the jack, provides a means to maintain a constant check of the calibration of the gauges. Based on the actual calibration tests of the stressing equipment, it was concluded that the pump efficiency does not influence the equipment accuracy and that the stressing accuracy depends only on the ram efficiency. Therefore, any combination of ram, gauge, and pump may be used interchangeably. During stressing, records are made of elongations as well as pressures obtained. Jack dynamometer or gauge combinations are checked against elongation of the tendon and any discrepancy exceeding plus or minus 5 percent will be evaluated by Design Engineering. The measured elongation will differ from the calculated elongation because of the following:

1. The statistical modulus of elasticity of 29.3 million psi for straight, untwisted wire.
2. The actual length and location of the tendon sheath will vary from the theoretical position due to approved placing tolerances.
3. All wires in a tendon are equal in length and the tendon is twisted to compensate for the difference in actual arc lengths. The twisting forms a wire cable configuration which does not follow the sheath centerline and which has a modified modulus of elasticity value.
4. The friction factor used in calculations is an average value based on experience. The true influence of friction on each tendon can be significantly different from the average value used in calculations.
5. The permissible tolerance in pressure gauge accuracy combined with the possible variables in stressing techniques such as reading the gauges and scales can constitute a significant difference.

Calibration of the pressure gauges are maintained accurate within the following limits:

0 to 2500 psi - Accuracy limit of the gauge, plus or minus 50 psi.

2500 to 7030 psi - Plus or minus 2 percent of gauge reading.

Pressure gauges are recalibrated after each stressing cycle on Oconee 3 and, as requested by Duke Power, during and at the end of the tensioning operations on Oconee 1 and 2.

Strain Gauge Installation and Protection

Strain or force gauging devices are installed on certain tendon areas prior to and/or during installation. These strain devices are monitored during the tensioning operation and used during subsequent pressure testing. Approximately 4 tendon sets are instrumented with load cells.

Tests, Samples, Inspections

Sampling and testing conforms to ASTM Standard A421 and as specified herein.

Each size of wire from each mill heat shipped to the site is assigned an individual lot number and tagged in such a manner that each such lot can be accurately identified at the job site. Anchorage assemblies are likewise identified. All unidentified prestressing steel or anchorage assemblies received at the job site are rejected.

Random samples as specified in the ASTM Standard stated above are taken from each lot of prestressing steel used in the work. With each sample of prestressing steel wire that is tested, there is submitted a certificate stating the manufacturer's minimum guaranteed ultimate tensile strength of the sample tested.

For the prefabricated tendons, one completely fabricated prestressing test specimen tendon 5 feet in length, including anchorage assemblies, is tested for each size of tendon contained in individual shipping release.

No prefabricated tendon is shipped to the site without first having been released by Duke, and each tendon is tagged before shipment for identification purposes. The release of any material by Duke does not preclude subsequent rejection if the material is damaged in transit or later damaged or found to be defective.

Duke shop inspects the prefabricated tendons prior to being shipped to the job site.

The anchorages and tendons are inspected at the job site for corrosion and mechanical damage during shipment, storage, installation, and tensioning. Damaged or corroded tendons and anchorages are rejected.

Acceptance

The Reactor Building has been analyzed based on missing tendons for the various loading conditions including missiles. The stresses for the various loading conditions were within the allowable design stresses. The missing tendons will not have any affect on the structure to withstand turbine and tornado generated missiles without loss of function. The missing tendons are located on the northwest face and shielded by location from a direct turbine missile strike. However, as stated in Section 3.5.1.3, the structure can withstand the loss of three horizontal and three vertical tendons in the cylinder wall without loss of function. The depth of penetration from tornado generated missiles as stated in Section 3.5.1.3 is less than the tendon concrete cover and will not endanger the structural integrity of the Reactor Building.

Final acceptance for warranty purposes is the successful completion of the pressure testing of the Reactor Building.

3.8.1.6.2.7.1 Steam Generator Replacement Tensioning Schedule

During steam generator replacement, tendons in the temporary construction opening were relaxed and/or removed. At the completion of the outage, the tendons were re-tensioned in accordance with specification SGRP-SPEC-C-002.

3.8.1.6.3 Reinforcing Steel

The concrete inspector visually inspected the shop fabricated reinforcing steel for compliance with drawings and specifications. Intermediate grade reinforcing steel conformed with ASTM A615, Grade 40 and high strength reinforcing steel conformed with ASTM A615, Grade 60. Mill test reports are submitted for engineering review and approval. Metallurgical inspection and testing of the reinforcing steel is done in accordance with the ACI Code 318-63, Chapter 8.

Reinforcing steel is inspected at delivery as well as at erection. The condition of the material must meet all of the requirements of ACI 318-63, as well as any additional requirements made by the inspector.

Number 14S and 18S reinforcing steel for which the ACI Code required welded or mechanical splices is spliced by the CADWELD process using full tensile strength “T” series connections. Quality control is maintained by qualification testing of the individual splicing crews, visual inspection of each completed connection, and random sampling and tensile testing of splices.

Prior to splicing operations, bar ends were inspected for damaged deformations and were power brushed to remove all loose mill scale, rust, and other foreign material. Immediately before the splice sleeve positioning, bar ends were preheated to assure complete absence of moisture.

Prior to making any production splices, each individual splicing crew prepares sample splices for tensile testing covering each bar size and position used in production to qualify. The sample splices must be properly filled, free of porous metal and meet the minimum requirement for tensile strength as stated below.

All splices are subjected to visual inspection and must meet the following standards:

1. Sound, nonporous filler metal must be visible at both ends of the splice sleeve and at the tap hole in the center of the splice sleeve. Filler metal is usually recessed ¼ inch from the end of the sleeve due to the packing material, and is not considered a poor fill.
2. Splices which contain slag or porous metal in the riser, tap hole, or at the ends of the sleeves (general porosity) are rejected. A single shrinkage bubble present below the riser is not detrimental and should be distinguished from general porosity as described above.

In addition to the above, random splices are subjected to mechanical tests and must meet the following standards:

1. The strength of 95 percent of the CADWELD splices tested will be greater than 125 percent of the specified minimum yield strength for the particular bar size and ASTM specification.
2. The strength of the average of all the splices tested will be equal to or greater than the minimum ultimate strength for the particular bar size and ASTM specification.
3. No failures of CADWELD splices below the required minimum yield strength are expected. In the unlikely event that one should occur, it would be sent to a testing laboratory for analysis of failure. Based on the testing laboratory's report, additional samples would be taken to insure that there are no other defective welds.

Tests are made in accordance with the following schedule for each position, bar size and grade of bar:

1 out of first 10 splices

3 out of next 100 splices

2 out of next 100 and each subsequent 100 splices

Test splices are made by having test bars of 3 feet length spliced in sequence with the production bars. In addition, two production splices are cut out and tested for each 100 test splices.

The inspections and tests are performed by individuals thoroughly trained by the CADWELD manufacturer.

For reinforcing steel of size 11 and under, lap splices are permitted in accordance with ACI 318-63, Chapter 8.

3.8.1.6.3.1 Steam Generator Replacement Reinforcing Steel

All new reinforcing steel, including replacement bars, are ASTM A615 Grade 60. The existing bars within the opening are A615 Grade 40. Mechanical splicing of bars will be accomplished through the use of BarSplice BPI XI swaged couplers. These devices are in compliance with ASME Section III, Division 2, Subsection CC and are capable of developing not less than of 125% of the specified yield strength of the bars in question.

Splice testing is in compliance with the UFSAR.

Where mechanical splices could not be used, direct-butt fusion welded splices were used. These splices were welded and inspected in accordance with AWS D1.4-98, Structural Welding Code – Reinforcing Steel.

3.8.1.6.4 Liner Plate

Construction of the liner plate conformed to the applicable portions of Part UW of Section VIII of the ASME Code. In addition, the qualification of all welding procedures and welders was performed in accordance with Part A of Section IX of the ASME Code. All liner angle welding was visually inspected prior to, during, and after welding to insure that quality and general workmanship met the requirements of the applicable welding procedure specification.

The erection of the liner plate was as follows:

After the floor plate embedments in the foundation slab had been placed and welded, and concrete was poured flush, the wall liner plates were erected in 60 degree segments and 10 feet high courses. This pattern was followed to the dome spring line and then the steel dome erection trusses were placed. During the period of erection o: wall liner plates, the floor liner plate was placed and welded.

The tolerances for liner plate erection were as follows:

1. The location of any point on the liner plate shall not vary from the design diameter by more than ± 3 inches.
2. Maximum inward deflection (toward the center of the structure) of the ¼ inch liner plate between the angle stiffeners of 1/8 inch, when measured with a 15 inch straightedge placed horizontally.

3.8.1.6.4.1 Steam Generator Replacement Liner Plate Repair and Fabrication

The liner plate and stiffeners removed to facilitate generator removal will be reused or replaced with new materials of the same grade as the existing. Fabrication of the new materials will be per ASME Boiler and Pressure Vessel Code, Section VIII, 1998 Edition with 1998 Addenda. Testing will be per ASME Boiler and Pressure Vessel Code, Section XI, Subsection IWL, IWE, and IWA of the 1992 Edition with 1992 Addenda. The actual repair was in accordance with the original liner plate specification.

3.8.1.6.5 Field Welding

This section outlines the general requirements for welding quality control to assure that all field welding is performed in full compliance with the applicable job specification. These requirements include the use of qualified welding inspectors and nondestructive testing technicians and the assurance that field welding is performed only by qualified welders using qualified procedures.

3.8.1.6.5.1 Qualifications for Welding Inspectors

Duke Power welding inspectors are qualified in accordance with the quality assurance requirements outlined in Chapter 17.

3.8.1.6.5.2 Instructions for Field Welding Inspectors

Quality Control procedures are in accordance with the quality assurance requirements outlined in Chapter 17. Visual inspection is performed after welding in accordance with Duke Power Company procedures, which reference ASME Sections III and VIII and NCIG-01.

3.8.1.6.5.3 Qualifications for Nondestructive Examination Inspectors.

Duke Power NDE inspectors are trained, qualified, and certified in accordance with the quality assurance requirements outlined in Chapter 17.

1. A technician will have a thorough knowledge of the type of testing he is to conduct. He will also be familiar with the welding procedure specification for the field welds he is inspecting.
2. The technician is properly certified in accordance with Duke Power Company procedures that incorporate the Society for Nondestructive Testing Recommended Practice No. SNT-TC.1A, as applicable.

3.8.1.6.5.4 Instructions for Nondestructive Examination Inspectors

NDE procedures are in accordance with the quality assurance requirements outlined in Chapter 17.

3.8.1.6.5.5 Welding Procedures

All welding is in strict accordance with approved welding procedure specifications.

Welder Qualification

All welders and welding operators who are to make welds under a code or standard which requires qualification of welders are tested and qualified accordingly before beginning production welding. Duke Power Company is responsible for testing and qualifying its own welders. The welding inspector is responsible in all cases for determining that the welders have passed the necessary qualification tests.

3.8.1.6.5.6 Steam Generator Replacement Field Welding of Liner Plate

Field welding, inspection, and welder qualifications are per ASME Boiler and Pressure Vessel Code, Section VIII, 1998 Edition with 1998 Addenda.

ASME Boiler and Pressure Vessel Code, Section VIII, 1998 Edition with 1998 Addenda.

3.8.1.7 Testing and Inservice Inspection Requirements

3.8.1.7.1 Structural Test

Each of the three Reactor Buildings will be pressurized to 115 percent of design pressure for one hour following completion of construction to establish the structural integrity of the building. The structural integrity test of each building will be conducted in accordance with a written procedure. Operating units will remain in operation during the structural test of another unit. Personnel access limitations included in the written procedures will designate areas of limited access during specific periods of the test. Except for personnel access restrictions, the operation of one unit will not be affected by a building being tested.

The structural integrity test of each building will verify the workmanship involved; in addition, the test of the Oconee 1 Reactor Building will verify the design and workmanship. The response of the Oconee 1 building will be compared with the calculated behavior to confirm the design by means of instrumentation.

3.8.1.7.1.1 Test Objectives

1. To provide direct verification that the structural integrity as a whole is equal to or greater than necessary to sustain the forces imposed by two different and large loading conditions.
2. To provide direct verification that the in-place tendons (the major strength elements) have a strength of at least 80 percent of guaranteed ultimate tensile strength and that the concrete has the strength needed to sustain a strain range from high initial average concrete compression when unpressurized to low average concrete compression when pressurized.
3. To acquire detailed strain data which will be compared with the analytical predictions.

To achieve objectives, data will be acquired and evaluated to determine the response of the structure during and immediately after post-tensioning to determine any indication of unanticipated and continued deformation under load. A quality assurance program was instituted. In addition, each individual tendon is tensioned in place to 80 percent of the guaranteed ultimate tensile strength and then anchored at a lower load that is still in excess of those predicted to exist at test pressure levels. During pressurization of the structure, the structure's response will be measured at selected pressure levels with the highest being 1.15 times the design pressure. An indication that the structure is capable of withstanding internal pressure will result from these tests. The strain measuring program is described in Section 3.8.1.7.2.

Individual test values which fall outside the predicted range will not be considered as necessarily indicative of a lack of adequate structural integrity. Structural integrity cannot be judged on the data acquired from only one sensor since such precise devices may malfunction.

3.8.1.7.1.2 Steam Generator Replacement Structural Testing

At the completion of the repair process the structure will undergo post modification testing. The building will be pressurized to the design pressure, Pa = 59 psig. This test will provide

verification of the integrity of the reactor building. The test will be performed in conjunction with a Type A Integrated Leak Rate Test.

3.8.1.7.2 Instrumentation

The structural response of the building will be assessed by comparing the theoretical analysis to test results of strains and deformations at boundaries, points of stress concentration, openings, areas of maximum creep, and at sections representing typical stress conditions.

The following instruments were installed in the first Reactor Building:

118	Two element strain rosette, waterproofed BLH Company designation FAET-12-12-S6, to be attached to the reinforcing bars.
9	Linear element, electric resistance strain gauges, BLH designation AS9-1 (Valore Type) to be attached to the surface of the concrete. Taut wire system for measuring building deformation.
6	Electric resistance strain gauge, Budd Company designation CP-1101 EX to be attached to the surface of the concrete for measuring crack propagation.
1	Cement Paint (Figure 3-37) to observe cracks in concrete.
7	Load cells each containing strain gauges to be attached to the tendons.
18	Three element rosette, electric resistance strain gauges BLH Company designation FAER-25-12-(60)S6, to be attached to the inside and outside face of the liner and penetration nozzles.
26	Two element strain rosette, BLH Company designation FAET-25-12-S6 to be attached to the inside face and outside face of the liner and penetration nozzles.

The instrument layout is shown on Figure 3-37, sheets 1, 2, and 3. The types and locations of the gauges are described in the legend on the figure. Because of the well-known vulnerability of the bonded resistance gauges to moisture, special care is taken in bonding and waterproofing of the gauges.

In order to reduce the possibilities of faulty preparation of the gauges in the field, the gauges are encapsulated and the wires soldered to the gauge leads and then waterproofed in the shop.

Bonding and waterproofing materials such as BLH EPY150 Cement, Epoxylite 222 and Microcrystalline Wax are used to install the gauges.

Gauges were calibrated in accordance with the manufacturer's instructions and set at zero reading during installation.

The final procedures in sequence of structural proof testing are as follows:

1. Test strain gauges immediately after installation.
2. Test strain gauges immediately after pouring concrete.
3. Record strains and deflections and observe cracking at three intervals suitably spaced during prestressing and immediately after all prestressing is completed.
4. After prestressing and before testing, a certain number of readings will be taken to determine the effects of creep and shrinkage.

5. Record measurements at increments of 10 psi up to 40 psi and then at increments of 5 psi up to proof-test pressure.
6. Record measurements at 15 psi increments during depressurization.
7. Observe the development of cracks during load application. Measurement of cracks with mechanical dial gauges will be made when deemed pertinent by the test engineer.

The Reactor Building air temperature is monitored by resistance thermometers and the dewpoint temperature is monitored by a dewpoint sensor. Using the Reactor Building coolers and electric heaters, the temperature is maintained between 60° and 100°F and above the dewpoint temperature.

The status of gauges on November 28, 1970 was as follows:

Gauge Mark	Number Inoperative	Number Operative	Number Being Replaced
SGA-1	114	4	(See 2 below)
SGE-2	7	2	(See 2 below)
SGC-3	0	6	–
SGR-4	7	11	6
SFT-5	7	19	6
LC (Load Cell)	1	6	(See 4 below)
Taut Wire System	0	–	–

Since a significant number of embedded gauges are inoperative, we believe it prudent to verify the design by (a) utilizing test results from Palisades and, (b) continuing with the Oconee Structural Test, as noted below:

1. The design and construction of Palisades and Oconee Reactor Buildings are very similar. The Palisades' structural instrumentation program was successful and permitted a detailed comparison between design calculations and observed response.
2. At Oconee, the taut wire system (building deformation) will permit verification that the structural response is consistent with the predicted behavior. In addition, twenty-six Carlson SAIOS strain gauges will be surface mounted on the Reactor Building to obtain concrete strains for comparison with Palisades and those predicted for Oconee as shown on Figure 3-37, Sheet 4.
3. Six inoperative gauges mark SGR-4 and SFT-5 are accessible and will be replaced to obtain data for comparison with Palisades and predicted strains for Oconee.
4. Load cells that are inoperative will be repaired or supplemented with prestress rams that have been modified with 20 psi division gauges to measure tendon forces. Prestress rams were used at Palisades and performed satisfactory. Results of measured forces can than be compared with those predicted.

The taut wire system consists of linear potentiometers (infinite resolution type) as the transducer element. Movement of the linear pontentiometers will be actuated by invar wires attached at one end to the point of measurement and at the other end to a reference point. Approximately 35 linear potentiometers will be used to measure building deformations during the structural test.

Oconee 2 and 3 Reactor Buildings are instrumented with the taut wire system for measuring building deformations as described above for Oconee 1. Displacement measurements are made at the following locations:

Dome	– Four points
Cylinder Wall	– Seven elevations at approximately 20 foot intervals at a buttress section and a wall section
Equipment Hatch	– Nine points with six of the points on the horizontal centerline and three of the points on the vertical centerline above the hatch
Vertical	– Two points

The above locations were selected so that deformation measurements could be compared with Oconee 1 measurements.

Concrete crack patterns are recorded at the base-wall intersection, cylinder wall mid-height, springline, equipment hatch opening, buttress-cylinder wall intersection, cylinder wall-ring girder intersection, and top of ring girder. Each inspection area consisted of approximately 40 square feet. Cracks that exceed 0.01 inch in width are mapped.

3.8.1.7.2.1 Reactor Building Structural Instrumentation for Steam Generator Replacement

Instrumentation will consist of a Laser Tracker Metrology System used to acquire the measurements on the outside of the Reactor Building by placing/adhering semi-permanent Spherical Mounted Retro-Reflector (SMR) Nests to the outside concrete in the area of the repair. The Laser Tracker combines the linear distance of the interferometer or Absolute Distance Measurement (ADM) with a position angle of the elevation and azimuth axes to derive a target's three dimensional (3-D) coordinate position. The 3-D coordinates are acquired by tracking the laser beam to SMR's and recording the data via wireless remote or keyboard entry. The expected accuracy in the volume of this scope is 0.006 of an inch. The Tracker will be positioned on a stable platform at ground level and the adhered targets will be acquired for a baseline. SMR's will be placed in each nest for continuous monitoring during the pressure test. A working coordinate system will be established to aid in interpretation of the displaced measurements.

3.8.1.7.3 Initial Leakage Tests

Following completion of the Reactor Buildings and prior to the hot functional tests and fueling of the reactors, integrated leakage rate tests will be performed on the containment systems. One test will be performed at or above the maximum calculated peak accident pressure. A second test will be performed at a pressure of not less than 50 percent of maximum calculated peak accident pressure.

The absolute pressure-temperature and/or the reference vessel method will be used for these tests. The objectives of these tests are:

1. To determine the initial integrated leakage rate for comparison with the design leakage rate.
2. To establish representative leakage characteristics of the containment system to permit retesting at reduced pressures.
3. To establish a performance history summary of the integrated leakage rate tests.

4. To establish a test method and the equipment to be used for subsequent retesting.

The leakage rate will be measured by integrating the leakage rate for a period of not less than 24 hours. This integrated leakage will be verified by the “pump-back” method and/or introduction of a known leak rate. The necessary instrumentation will be installed to provide accurate data for calculating the leakage rate. It will be demonstrated that the total Reactor Building leakage rate to the environment will maintain public exposure below 10CFR100 limits in the event of an accident.

3.8.1.7.3.1 Steam Generator Replacement Leakage Testing

Following the steam generator replacement a Type A Intergrated Leakage Test, (ILRT), will be performed in accordance with the requirements of 10 CFR 50.app J. This test will not be materially different from current station requirements.

3.8.1.7.4 Leakage Monitoring

A program of testing and surveillance of each of the three duplicate Reactor Buildings has been developed to provide assurance, during service, of the capability of each containment system to perform its intended safety function. This program consists of tests defined as follows:

Overall integrated leak rate tests of the Reactor Buildings and systems which under post accident conditions become an extension of the containment boundary.

Local leak detection tests of components having resilient seals, gaskets, or sealant compounds that penetrate or seal the boundary of the containment system. Components included in this category are:

1. Personnel Hatches
2. Emergency Hatches
3. Equipment Hatches
4. Fuel Transfer Tube Covers
5. Electrical Penetrations
6. Leak Rate Test Pressurization/Exhaust Penetration

Local leak detection and operability tests of containment isolation valves in systems that vent directly to the Reactor Building atmosphere or the Reactor Coolant system that must close upon receiving an isolation signal and seal the containment under accident conditions.

Operability tests of engineered safeguards systems which under post accident conditions are relied upon to limit or reduce leakage from the containment. Included in these tests are:

1. Reactor Building Spray Systems
2. Reactor Building Penetration Room Ventilation Systems (not required for accident mitigation due to adoption of alternate source term), Reference 34.
3. Reactor Building Cooling Systems
4. Reactor Building Isolation Valves not covered above

Following the integrated leakage rate tests, performed as a part of the preoperational testing, subsequent tests will be performed at the maximum calculated peak accident pressure or greater. The tests will be performed on schedule based on the following considerations:

1. There are three Reactor Buildings each having the same design. Information pertaining to deterioration in performance obtained in the testing of one Reactor Building is therefore applicable to the other Reactor Buildings.
2. Local leak detection tests will be performed on a more frequent basis than the integrated tests to detect and correct excessive leakage at containment penetrations. Where feasible, these tests will be performed during operation; otherwise, they will be performed during refueling outages and/or major maintenance outages. These tests will be performed at or above the maximum calculated peak accident pressure.
3. The engineered safeguards tests will also be performed at more frequent intervals than the integrated leak rate tests to verify the functional capability of these systems which are relied upon to limit or reduce leakage from the containment buildings in the case their service is required. These tests will be performed during outages for refueling and/or major maintenance outages.

The schedule of testing, type of test, and components to be tested are as follows:

Integrated Leak Rate Tests

Integrated leak rate tests shall be performed as follows:

1. Each Reactor Building shall be tested at the calculated peak accident pressure of 59 psig and at one-half this pressure prior to the initial fuel loading.
2. After the initial preoperational leakage rate test, integrated leakage rate tests shall be performed on each Reactor Building at intervals in accordance with 10CFR50 Appendix J Option B. These tests shall be conducted at or above peak accident pressure (P_t).

   Visual examinations of containment pressure retaining metallic surfaces shall be performed at least three times every 10 years and only those examinations performed in conjunction with each Type A test need to be performed during shutdown. When possible, these general visual examinations are to be performed concurrently with general visual examinations required by ASME Code Subsection IWE, Table IWE-2500-1, Examination Category E-A, Item 1.11 during each ISI interval.

Local Leak Detection and Operability Tests (Resilient Seals)

Local leak detection and operability tests shall be performed as required by the Technical Specifications.

The barrier to leakage in the Reactor Building is the one-quarter inch steel liner plate. All penetrations are continuously welded to the liner plate before the concrete in which they are embedded is placed. The penetrations, shown on Figure 3-20 and Figure 3-21, become an integral part of the liner and are so designed, installed, and tested.

The steel liner plate is securely attached to the prestressed concrete Reactor Building and is an integral part of this structure. This Reactor Building is conservatively designed and rigorously analyzed for the extreme loading conditions of a highly improbable hypothetical accident, as well as for all other types of loading conditions which could be experienced. Thorough control is maintained over the quality of all materials and workmanship during all stages of fabrication and erection of the liner plate and penetrations and during construction of the entire Reactor Building.

During construction, the entire length of every seam weld in the liner plate was leak tested. Individual penetration assemblies were shop tested. Welded connections between penetration assemblies and the liner plate were individually leak tested after installation. Following completion of construction, the entire Reactor Building, the liner, and all its penetrations were

tested at 115 percent of the design pressure to establish structural integrity. The initial leak rate tests of the entire Reactor Building were conducted at the maximum calculated peak accident pressure and one-half this pressure to demonstrate vapor tightness and to establish a reference for periodic leak testing for the life of the station. Multiple and redundant systems based on different engineering principles are provided as described in Chapter 6, to provide a very high degree of assurance that the accident conditions will never be exceeded and that the vapor barrier of the containment will never be jeopardized.

Under all normal operating conditions and under accidental conditions short of the worst loss-of-coolant accident, virtually no possibility exists that any leakage could occur or that the integrity of the vapor barrier could be violated in any way that would be significant to the public health and safety or to that of the station personnel. Adequate administrative controls are enforced to minimize the possibility of human error. Station operators are trained and licensed in accordance with regulations. Safety analyses are presented in Chapter 15.

Penetrations such as the personnel access and emergency hatches cannot be opened except by deliberate action and are interlocked and alarmed by failsafe devices such that the Reactor Building will not be breached unintentionally. The liner plate over the foundation slab is protected by cover concrete. Wherever access to the liner plate is blocked by interior concrete, means are provided so that weld seams can be tested for leakage. The liner plate is protected against corrosion by suitable coatings. Walls and floors for biological and missile shielding, and for access and operating purposes, also provide compartmentation which constitutes protection for the liner during operating as well as accident conditions.

Once the adequacy of the liner has been established initially, there is no reason to anticipate progressive deterioration during the life of the station which would reduce the effectiveness of the liner as a vapor barrier. Inside the Reactor Building, the atmosphere is subject to a high degree of temperature control. The outside of the liner is protected by 3-3/4 feet of prestressed concrete which is exceptionally resistant to all weather conditions.

Inspection on a periodic basis, as necessary, will be conducted in all spaces accessible under full power operation. Biological shielding is provided to reduce radiation to limits which make occupancy of spaces adjacent to the liner permissible.

All penetrations except those described in Section 6.5.1.2, are grouped within or vented to the penetration room. Any leakage that might occur from these penetrations will be collected and discharged through high efficiency particulare air (HEPA) filters and charcoal filters to the unit vent as described in Section 6.5. In this manner, leakage which might occur from these penetrations will be isolated from leakage which might occur through the Reactor Building itself.

Individual major penetrations or groups of penetrations will be tested by means of permanently installed pressure connections or temporarily installed pressure or vacuum boxes. If necessary, liner plate weld seams will be tested by the vacuum box soap bubble method, where accessible, or by means of the permanently installed backup channels and angles where inaccessible.

In any event, sources of excessive leakage will be located and such corrective action as necessary will be taken. This will consist of repair or replacement. Appropriate action will also be taken to minimize the possibility of recurrence of excessive leakage, including such redesign as might prove to be necessary to protect public health and safety. Leak testing will be continued until a satisfactory leak rate has again been demonstrated.

A considerable background of operation experience is being accumulated on containments and penetrations. Full advantage of this knowledge has been taken in all phases of design, fabrication, installation, inspection, and testing. Practical improvements in design and details have been incorporated as they are developed, where applicable.

The steel-lined Reactor Building is self-sufficient, and other than valves and hatch doors, there are no operating parts. The containment boundary is extended only by listed penetrations and further described and tabulated in Section 6.2.3.

3.8.1.7.5 Engineered Safeguards Tests

The Reactor Building Spray, Penetration Room Ventilation, Reactor Building Cooling Systems, and the Reactor Building Isolation Valves will be tested periodically to provide assurance of system reliability. These tests will include:

1. Reactor Building Cooling System.

   This system is operated periodically during normal operating periods. This normal operator initiated operation of this system provides verification of the operability. In addition to this normal operation, testing of this system in the engineered safeguards mode will be performed as indicated in the Improved Technical Specifications. This test will be initiated by inserting a simulated engineered safeguards signal as would occur during an accident situation. Verification of the proper operation of the components of this system will be determined and a record of the test results made a part of the permanent plant records.

2. Reactor Building Spray System.

   The Reactor Building Spray System will be tested in a similar manner as the system above, with some exceptions. The ES testing for this system will be performed on a refueling frequency. It will test only the initiation control circuitry, and will not actually start the Reactor Building Spray Pumps. The pump breaker will be positioned to "test" position, allowing the verification that the signal reaches the breaker and the breaker actuates, but the pump does not receive power. Separate testing is performed to verify the functional readiness of the pumps and valves on a quarterly frequency. For the pump tests, the headers are isolated to prevent spray water from entering the spray headers. A special test connection is provided ahead of the Reactor Building isolation valves so that the portion of the system outside the Reactor Building may be operated in recirculation alignment to the Borated Water Storage Tank. Following activities which could cause nozzle blockage, compressed air will be blown through each of the spray headers in the Reactor Building through special test connections to verify that spray water would be directed into the Reactor Building under accident conditions. Proper operation of the various components of this system will be verified and a record of the test results made a part of the plant records.

3. Penetration Room Ventilation System

   The Penetration Room Ventilation system is no longer required to operate for accident mitigation due to the adoption of the alternate source term, Reference 34. However, it will continue to be operated periodically during normal operation to verify the system functions properly. In addition to this normal operation, testing in the engineered safeguards mode will be performed by inserting a simulated engineered safeguards signal as would occur during an accident situation. Verification of proper system operation will be determined and record of the test results made a part of permanent plant records.

4. Reactor Building Isolation Valves

   Proper operability of the Reactor Building isolation valves not covered in the other tests will be verified by inserting a simulated engineered safeguards signal to initiate operation of these valves. The valves in the reactor building purge flow path are required to be maintained closed in Modes where the engineered safeguards system is required operable.

This is a requirement of NUREG 0737, Item II.E.4.2.6. Therefore Engineered Safeguards system testing of these reactor building purge valves is not required.

3.8.1.7.6 Post-Tensioning System

A surveillance program for the Reactor Building post-tensioning system, is executed in order to assure the continued quality of the system. The program consists of periodic inspections of randomly selected tendons - for symptoms of material deterioration or excessive pre-stress force reduction. The program assesses the condition and functional capability of the system and, therefore, verifies the adequacy of the system and provides an opportunity to take proper corrective action should adverse conditions be detected.

An end anchorage concrete surveillance program for the post-tensioning system is implemented to assure the continued structural integrity of the Reactor Buildings. The program consists of periodic inspections of end anchorages and adjacent concrete surfaces.

3.8.1.7.7 Liner Plate

"HISTORICAL INFORMATION NOT REQUIRED TO BE REVISED"

A surveillance program for the Reactor Building liner plate was implemented to assure continued integrity of the liner plate. The initial surveillance was conducted in conjunction with the initial Reactor Building Structural Integrity Test. The building was pressurized to 67.8 psig during the surveillance test to check the inward displacement of the liner plate. This program was completed for Oconee 1 on January 2, 1974.

Liner plate inspection now is conducted in accordance with 10CFR 50, Appendix J requirements.

3.8.2 Steel Containment

The Reactor Building does not have a steel containment vessel separate from the concrete shell. The description of the steel liner plate and all applicable supporting data is found in Section 3.8.1.

3.8.3 Concrete and Structural Steel Internal Structures of Containment

3.8.3.1 Description of the Internal Structures

The Reactor Building interior structure consists of (1) the reactor cavity, (2) two steam generator compartments, and (3) a refueling pool which is located between the steam generator compartments and above the reactor cavity.

The reactor cavity houses the reactor vessels and serves as a biological shield wall. The reactor cavity is also designed to contain core flooding water up to the level of the reactor nozzle.

Additional descriptive information can be found in Section 3.8.1.

3.8.3.2 Applicable Codes, Standards, and Specifications

The interior structures are designed in accordance with the applicable codes and specifications listed in Section 3.8.1.2.

3.8.3.3 Loads and Load Combinations

The loads and load combinations considered for the design of the interior structures are described in Section 3.8.1.3.

3.8.3.4 Design and Analysis Procedures

The Reactor Building interior structures are designed using conventional structural analytical techniques. Some of the provisions of the design are described below:

The primary functions of the steam generator compartment walls are to serve as secondary shield walls and to resist the pressure and jet loads.

The foundations for all NSSS equipment including the reactor vessel, the steam generators, and the pressurizer are designed to remain within the elastic range during rupture of any pipe combined with the "maximum earthquake".

The design pressure differential across walls and slabs of enclosed compartments in the internal structure are as follows:

Reactor Cavity–	208 psi
East Steam Generator Compartment–	15 psi
West Steam Generator Compartment–	15 psi

In addition to the peak differentials, the steam generator compartment walls are designed for simultaneous action of a single jet impingement load and the safe shutdown earthquake.

Pipe whipping restraints are provided for the main steam, feedwater, and other high-pressure piping.

3.8.3.5 Structural Acceptance Criteria

The Reactor Building interior structure (comprising all elements inside the Reactor Building shell) is a Seismic Class 1 structure and is designed on the following bases:

1. The stresses in any portion of the structure under the action of dead load, live load, and design seismic load will be below the allowable stresses given by either the ACI Building Code, ACI 318-1963 except as noted in Section 3.8.1.3.6, and the AISC Manual of Steel Construction, 6th Edition.
2. The stresses in any portion of the structure under the action of dead load, and thermal load will be below 133 percent of the allowable stresses given in (1).
3. The capability to safely shut down the plant will be maintained under the combined action of dead load, maximum seismic load, pressure and jet impingement load. The latter two loads are based on the rupture of one pipe in the primary loop. The deflections of structures and supports under these combined loads would be such that the functioning of engineered safeguards equipment would not be impaired. The yield load equations in Section 3.8.1.3.6 are adhered to except that local yielding is permitted for pipe, jet or missile barriers provided there is no general failure.
4. Under the combined action of dead load and maximum seismic load, reinforced concrete structures shall be designed in accordance with the requirements of Table 3-14. Structural steel structures shall be designed in accordance with the provisions of the AISC Manual of

Steel Construction except that normal allowable stresses may be increased by 150%, not to exceed 0.9 yield.

5. The maximum allowable concrete temperature at penetrations in the Primary Shield Wall shall not exceed 400°F.

3.8.3.6 Materials, Quality Control, and Special Construction Techniques

The materials used for the structural elements are as follows:

Structural Steel	– ASTM A36
Concrete	– f'_c = 4000 psi at 28 days
	– f'_c = 5000 psi at 28 days (for steam generator bases, reactor foundation, and primary shield wall).
Reinforcing	– ASTM A615, Grade 40 for Bars #11 and under ASTM A615, Grade 60 for Bars larger than #11.

Additional materials, quality control, and construction techniques and described in Section 3.8.1.6.

3.8.3.7 Testing and Inservice Surveillance Requirements

Testing and inservice surveillance requirements are outlined in Section 3.8.1.7.

3.8.4 AUXILIARY BUILDING

3.8.4.1 Description of the Structure

The Auxiliary Building was constructed on a 5.00 foot thick reinforced concrete mat foundation. Below grade, the building consists of reinforced concrete walls and slabs. Above grade, the building consists principally of reinforced concrete columns, beams, and slabs, with the slabs acting as diaphragms.

The following facilities related to the Nuclear Steam Supply System are located in the Auxiliary Building:

1. New and Spent Fuel Handling, Storage, and Shipment
2. Control Rooms
3. Waste Disposal System
4. Chemical Addition and Sampling System
5. Component Cooling System
6. Reactor Building Spray Systems
7. High and Low Pressure injection System
8. Spent Fuel Cooling System
9. Electrical Distribution System

3.8.4.2 Applicable Codes, Standards, and Specifications

The Class 1 Structure is designed according to the applicable codes and specifications listed in Section 3.8.1.2.

3.8.4.3 Loads and Load Combinations

The loads and load combinations considered for the design of the Auxiliary Building are listed in Table 3-23. The final design of the Auxiliary Building satisfies the loading combinations and factors tabulated in Table 3-14.

3.8.4.4 Design and Analysis Procedures

The design of the Auxiliary Building is performed using conventional structural analytical techniques. The provisions of the design for the Spent Fuel Pool are described below: The ultimate strength assumptions of the ACI Code for concrete beams in flexure are not allowed; that is, the concrete strain is not allowed to go beyond yield.

The Spent Fuel Pool Walls were analyzed for thermal loads in accordance with methods presented in ACI 505. The exterior wall temperature was assumed to be 60°F for areas enclosed by the Auxiliary Building and 0°F for exposed areas.

Under normal conditions, the interior wall temperature was 150°F and the maximum calculated thermal stress was 996 psi for concrete and 11,410 psi for reinforcing steel.

After prolonged outage of the cooling system, the interior wall temperature could reach 212°F and the maximum calculated thermal stress was 1681 psi for concrete and 25,600 psi for reinforcing steel. Reinforcing steel conforming with ASTM A516, Grade 60, was used.

A minimum of 0.30 percent reinforcing was used in the spent fuel pool walls to control concrete cracking. Also, a ¼ inch thick steel liner was used on the inside face of the pool for leak tightness.

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

The SFP concrete floor slab is designed to withstand the 100 ton cask drop. However, localized concrete could be crushed and the steel liner plate punctured in the area of 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.

3.8.4.5 Structural Acceptance Criteria

The areas of the Auxiliary Building housing the facilities listed in Section 3.8.4.1 have been designed for the loads and conditions as shown in Table 3-23 with maximum allowable stresses as follows:

Loading Condition	Maximum Allowable Stress
A	Stresses in accordance with ACI and AISC Codes
B, D	For Reinforced Concrete Design:

Loading Condition	Maximum Allowable Stress
	fc = 0.85 f'_c for Flexure
	fc = 0.70 f'_c for tied compression members
	Shear = $1.1\sqrt{f'_c}x1.33$ for beams with no web reinforcing
	fs = 0.90 fy for Flexure
	fs = 0.90 fy for web reinforcing
	fs = 0.85 fy for reinforcing steel with lap or mechanical splices
	Bond = $\frac{3.4\sqrt{f'_c}}{D}$ x 1.33 for top bars
	= $\frac{4.8\sqrt{f'_c}}{D}$ x 1.33 other than top bars
	For Structural Steel Design:
	AISC Code allowable stresses x 150%, not to exceed 0.9 Fy
C, E	Analyzed on basis of Reference 7

3.8.4.6 Materials, Quality Control, and Special Construction Techniques

This information is outlined in applicable portions of Section 3.8.1.6.

During the Unit 1 Steam Generator Replacement Outage, a portion of the Auxiliary Building roof was removed and repaired. The replacement reinforcing steel information is outlined in the applicable portions of Section 3.8.1.6.3.1.

3.8.4.7 Concrete Masonry Walls

The masonry walls are in-fill panels serving as partitions with some walls having pressure, fire and radiation barrier applications. The walls are single or multiple wythe and constructed of hollow or grouted concrete blocks or solid concrete blocks or bricks. All masonry walls are non-structural and constructed on a structural support system.

Pursuant to I.E. Bulletin 80-11, a safety re-evaluation of all masonry walls was undertaken by Duke. As a result of this reevaluation effort certain masonry walls were modified to meet minimum factors of safety.

Certain masonry walls that are part of the Units 1, 2, and 3 Auxiliary Buildings that house equipment needed to mitigate the adverse effects of a tornado were evaluated for tornado-induced differential pressure loading per Reference 41. Beginning in 2011, these walls were subsequently strengthened to meet these loads using a fiber-reinforced polymer (FRP) system. Per References 42 and 49, masonry walls constructed of concrete block and solid concrete brick strengthened by the FRP System have been found to be acceptable by the NRC Staff in resisting tornado-induced differential pressure.

3.8.4.7.1 Applicable Codes and Standards

The criteria used for the re-evaluation of masonry walls pursuant to I.E. Bulletin 80-11 are contained in Attachment 4 of Reference 14 and Reference 16. The criteria in Reference 14 use the American Concrete Institute "Building Code Requirements for Concrete Masonry Structures," ACI 531-79, as the governing code with supplemental allowables specified for cases not directly addressed in the code. The criterion in Reference 16 is the Arching Action Theory for Masonry Walls.

The criteria used for the re-evaluation of masonry walls to resist tornado-induced differential pressure loadings are contained in References 37, 38, 39, 40, 43, 44, 45, 46, 47, and 48 as approved by the NRC Staff in References 42, 49, 50 and 51. These criteria specify ACI 531-79 as the governing code for this evaluation with supplemental working stress allowables specified for the fiber-reinforced polymer (FRP) system.

3.8.4.7.2 Loads and Load Combinations

The design loadings for the masonry walls at Oconee are those specified in portions of Section 3.8.4. The only thermal effects which a masonry wall experiences are those pertinent to normal operation, and these are not considered a significant design consideration.

In addition, the design differential pressure for masonry walls evaluated for tornado-induced loadings is contained in Section 3.3.2.1. The load combinations for tornado-induced loadings, which include the differential pressure loading for which the fiber-reinforced polymer (FRP) system was used to mitigate, are contained in NUREG-0800, Standard Review Plan, Section 3.8.4, “Other Seismic Category I Structures”, Rev. 1 - July 1981.

3.8.4.7.3 Upgrade and Modification of Masonry Walls

A program of repairs was performed on selected masonry walls pursuant to I.E. Bulletin 80-11. The walls included in this program were not found to be unsafe in their original configuration; however, an added margin of safety was desired for these walls. The repairs provide increased factors of safety by either upgrading the walls to meet the allowable stresses set forth in the re-evaluation criteria or by shielding the safety related equipment located in proximity of the walls from damage, assuming the masonry walls were to collapse. References 12 through 24 and References 35 and 36 pertain to I.E. Bulletin 80-11.

Certain masonry walls that are part of the Units 1, 2, and 3 Auxiliary Buildings were modified beginning in 2011 to resist tornado-induced differential pressure loading (References 42 and 49). These walls were strengthened using a fiber-reinforced polymer (FRP) system.

3.8.5 Nonclass 1 Structures

The Turbine Building, the condenser circulating water structures, the Essential Siphon Vacuum System Intake Dike Trench, the Essential Siphon Vacuum Cable Trench, the Essential Siphon Vacuum Building, and the Keowee structures as listed in Section 3.2.1.1.2 are Class 2 structures.

Class 3 structures include all structures not included in Class 1 and 2.

3.8.5.1 Description of the Structures

1. Turbine Building

The building was constructed of reinforced concrete below grade consisting of substructure walls and a mat foundation. Above grade, the building consists of structural steel with metal siding.

2. Keowee Structures

   The Keowee Structures considered are Powerhouse, Power and Penstock Tunnels, Spillway, Service Bay Substructure, Breaker Vault, and Intake Structure.

3. Dams and Dikes

   The Keowee Dam, the Little River Dam and Dikes, and the Oconee Intake Canal Dike impound the waters of Lake Keowee to provide the source of flowing water for the Keowee hydroelectric power plant.

4. Oconee Intake Structure

   The intake structure supports the CCW pumps, intake screens, and inlets of the CCW pipes.

5. Oconee Intake Underwater Weir

   The underwater weir retains an emergency water supply in the event that the waters of Lake Keowee are released by the failure of a dam or dike.

6. CCW Intake Piping

   The CCW Intake Piping conveys water from the CCW pumps on the intake structure to the condenser, supplies water to the LPSW Pumps, and serves as the reservoir for the SSF Auxiliary Service Water System and the Protected Service Water System.

7. CCW Discharge Piping

   The CCW Discharge Piping conveys water from the condenser to the discharge structure and supplements the CCW intake piping as a reservoir for the Protected Service Water System.

8. ECCW Piping

   The ECCW Piping serves two different functions. 1) It can siphon the Condenser Circulating Water through the Condenser to be discharged at the treatment pond. 2) It can be used for recirculation of the Condenser Circulating Water back to the Intake Canal.

9. Essential Siphon Vacuum System Intake Dike Trench

   The Essential Siphon Vaccum (ESV) System Intake Dike Trench is constructed of reinforced concrete (bottom and walls). The covers for the trench are steel plate except at the roadway crossing. The covers at the roadway are removable reinforced concrete slabs.

   The Essential Siphon Vaccum (ESV) System Intake Dike Trench routes the ESV piping, the Siphon Seal Water (SSW) piping, electrical heat trace cables, and electrical instrumentation cables within the FERC boundary without reducing the integrity of the Oconee Intake Dike.

10. Essential Siphon Vacuum System Building

    The ESV Building is constructed of a reinforced concrete mat foundation and rigid structural steel frame with metal siding.

    The ESV Building encloses the ESV System's pumps, motors and associated equipment, providing protection (from weather & freezing) for that equipment and providing a suitable environment for maintenance activities.

11. Essential Siphon Vacuum System Cable Trench

The essential Siphon Vacuum (ESV) System Intake Cable Trench is constructed of reinforced concrete (bottom and walls). The covers for the trench are steel plate except at the traffic crossing. The covers at the crossing are removable reinforced concrete slabs.

The ESV System Cable Trench routes the cables associated with the ESV System and SSW System from the Radwaste Trench to the ESV Building.

3.8.5.2 Applicable Codes, Standards, and Specifications

Class 2 structures are designed in accordance with the following codes:

ACI 318-63 - Building Code Requirements for Reinforced Concrete

AISC - Steel Construction Manual, 6th edition and 9th edition (The 9th ed. is for the Essential Siphon Vacuum Building only)

The working stress design method will be used for normal and seismic conditions and stress will be in accordance with above codes, including the 33% increase for wind or earthquake leads. Class 2 structures are qualified for the Design Base Earthquake (DBE). All Keowee Structures necessary for Emergency Power Generation, the Oconee Turbine and Auxiliary Buildings (except as included in Class 1), the Oconee Intake Structure, the CCW Intake Piping, the CCW Discharge Piping, ECCW Piping (structural portion), the Oconee Intake Canal Dike, and the Essential Siphon Vacuum System Intake Dike Trench, the Essential Siphon Vacuum Cable Trench, and the Essential Siphon Vacuum Building are designed for Maximum Hypothetical Earthquake (MHE).

3.8.5.3 Loads and Load Combinations

3.8.5.3.1 Turbine Building

1. Transverse Loading

The loadings were applied as follows:

Dead Loads - Roof - 50 psf, reduced to 25 psf when the type of roof construction was finalized.

Floors - Grating Areas - 20 psf.

Concrete Areas

a. Operating Floor - 11-1/2 in. - 170 psf.

b. Mezzanine Floor - 8 in. - 115 psf.

c. Upper Surge Tank Floor - 4 in. - 65 psf plus tank at normal operating condition.

Crane Columns and Girders - Calculated weights.

Live Loads - Roof - 50 psf.

Grating Areas - 100 psf.

Operating a. Turbine Bay - 600 psf. *

Floor b. -Heater Bay - 400 psf. *

* Includes an allowance for undefined equipment and normal loads – known S/Rs (supports/restraints)

Mezzanine Floor
a. General Area - 250 psf *

b. Moisture Separator Tube Pull Area - 400 psf *

c. Moisture Separator Lay Down Area - 30 kip concentrated load @ c/l of collector beams

*Includes an allowance for undefined equipment and normal loads – known S/Rs (supports/restraints)

Upper Surge Tank Floor - 100 psf (all areas except those between column lines 28 & 29 and 44 & 45 - 250 psf) **

**Includes an allowance for normal tank reactions and values for normal loads – known S/Rs (supports/restraints)

Cranes - 180 Ton and 80 Ton Cranes fully loaded, lifted load and lateral force arranged to produce maximum stresses. The lateral forces were reduced to 15 percent of the sum of the weights of the lifted load and the crane trolleys.

Wind load - 30 psf.

Seismic Loading No. 1 - (Load Combinations)

a. Critical Damping - 2%
b. Maximum Ground Motion Acceleration - 5% of gravity
c. Maximum Acceleration for Design - 12% of gravity (This is the maximum value of the acceleration response curve for 2% damping.)
d. Loadings - Roof - 50 psf, reduced to 25 psf when the type of roof construction was finalized.
e. Operating Floor - dead load of floor plus equipment load. (Equipment load estimated at 250 psf.)
f. Mezzanine Floor - dead load of floor plus equipment load. (Equipment load estimated at 150 psf.)
g. Upper Surge Tank and Floor - 65 psf plus tank at normal operating condition.
h. Crane - 180 Ton Crane, fully loaded, at center of bay.
i. Crane Columns and Girders - Calculated weights.

Seismic Loading No. 2 - (Load Combinations)

a. Critical Damping = 2%
b. Maximum Ground Motion Acceleration - 10% of gravity

c. Maximum Acceleration for Design - 22% of gravity (This is the maximum value of the acceleration response curve for 2% damping.)

   Loadings - Roof - 25 psf.

d. Operating Floor - dead load of floor plus equipment load. (Equipment load estimated at 200 psf.)

e. Mezzanine Floor - dead load of floor plus equipment load. (Equipment load estimated at 125 psf.)

f. Upper Surge Tank and Floor - 65 psf plus tank at normal operating condition.

g. Cranes - 180 Ton Crane and 80 Ton Crane at rest and in unloaded condition.

h. Crane Columns and Girders - Calculated weights.

i. Loading for Dynamic Seismic Analysis - (Load Combinations)

j. Critical Damping = 2%

k. Maximum Ground Motion Acceleration - 10% of gravity.

l. Reference subsection "Dynamic Seismic Analysis" in Section 3.8.5.4.1 for design accelerations.

Loadings:

a. Roof - 25 psf

b. Operating Floor - dead load of floor plus equipment load. (Equipment load estimated at 150 psf)

c. Mezzanine Floor - dead load of floor plus equipment load. (Equipment load estimated at 150 psf)

d. Upper Surge Tank and Floor - 65 psf plus tank at normal operating condition.

e. Cranes - 180 Ton and 80 Ton Capacity Cranes at rest and in unloaded condition.

f. Crane Columns and Girders - calculated weights.

Seismic Loading No. 2 was introduced approximately six months after the building was analyzed for Seismic Loading No. 1. With more complete information, it was apparent that the equipment loads assumed for the Operating and Mezzanine Floors were too conservative. Therefore, the equipment loads were reduced for the analysis for Seismic Loading No. 2.

2. Longitudinal Loading

   The loadings were applied as follows:

Wind Load	- 30 psf.
Crane Load	- 10% of Maximum wheel load.
Seismic Loading No. 1	- Same as Seismic Loading No. 1 for Transverse Analysis with the following exceptions:
Loadings	- Operating Floor - Equipment load estimated at 130 psf.
	Mezzanine Floor - Equipment load estimated at 110 psf.

Seismic Loading No. 2 - Same as Seismic Loading No. 2 for Transverse Analysis with the following exceptions:

Loadings - Operating Floor - Equipment load estimated at 130 psf.

Mezzanine Floor - Equipment load estimated at 110 psf.

Loading for Dynamic Seismic Analysis - Same as loading for dynamic seismic analysis for transverse direction.

3. Loading Combinations and Factors

S = 1.0 D + 1.0 L

1.33S = 1.0 D + 1.0 L + 1.0 W

1.33S = 1.0 D + 1.0 E

1.64S = 1.0 D + 1.0 E′

S = Allowable stress due to normal loading - from AISC specifications

D = Dead Loads (Equipment loads included in the case of seismic loadings)

L = Live Loads

W = Wind Loads

E = Loads from Seismic Loading No. 1

E' = Loads from Seismic Loading No. 2

3.8.5.3.2 Keowee Structures

1. Powerhouse

A typical reinforced concrete frame was investigated for the following loading conditions using a static type analysis:

a. Dead load plus live load (1000 lbs per square foot) using allowable stresses in accordance with ACI Code. The maximum calculated stresses were f_s = 18,590 psi and f_c = 1122 psi.

b. Dead load plus live load (1000 lbs per square foot) plus seismic load equal to 0.10g times the dead load. The maximum calculated stresses were f_s = 19,120 psi and f_c = 1189 psi. Allowable stresses were $f_s = 0.9\ f_y$ = 36,000 psi and $f_c = 0.85\ f'c$ = 2550 psi.

c. Dead load plus live load (1000 lbs per square foot) plus seismic load equal to 0.20g times the dead load. The maximum calculated stresses were f_s = 19,700 psi and f_c = 1229 psi.

The large live loading of 1000 lbs per square foot was included to allow for heavy equipment loads expected during construction and maintenance. Therefore, to be conservative, the 1000 lbs per square foot was included to b and c above but with seismic loadings added as a function of dead load only.

2. Spillway

A typical spillway pier was investigated for the following loading conditions:

a. Dead load plus hydrostatic load with allowable stresses in accordance with ACI Code. The maximum calculated stresses were f_s = 0 and f_c = 61.7 psi.

b. Dead load plus hydrostatic load plus seismic load equal to 0.10 times dead load. The maximum calculated stresses were f_s = 7760 psi and f_c = 173 psi. The allowable stresses were f_s = 0.9 f_y = 36,000 psi and f_c = 0.85 f′c[2] = 3400 psi.

1) Same as b except seismic load equal to 0.20 times dead load. The maximum calculated stresses were f_s = 16,350 psi and f_c = 227 psi.

In addition, the taintor gate thrust girder was investigated for the following loading conditions:

a. Dead load plus hydrostatic load with allowable stresses in accordance with AISC Code. The maximum calculated stress was f_s = 23,300 psi.

b. Dead load plus hydrostatic load plus seismic load equal to 0.10 times dead load with allowable stress = 0.9 f_y = 32,500 psi. The maximu[2]m calculated stress was f_s = 25,000 psi.

c. Same as b except seismic load equal to 0.20 times dead load. The maximum calculated stress was f_s = 28,800 psi.

3. Service Bay Substructure

The Service Bay substructure contains the Control Room, Cable Room, Equipment Room, and Battery Room areas. The substructure was investigated for the following loading conditions:

a. Dead load plus live load with allowable stresses in accordance with ACI Code. The maximum calculated stresses were f_s = 19,700 psi and f_c = 1160 psi.

b. Dead load plus live load plus seismic load equal to 0.15 times the combined dead-live load. The allowable stresses were f_s = 0.9 f_y = 36,000 psi and f_c = 0.85 f′c = 2550 psi. The maximum calculated stresses were f_s = 24,000 psi and f_c = 1410 psi. It is apparent that the seismic loads could be substantially increased with resulting stresses being well below those allowable.

4. Breaker Vault

The Breaker Vault is located on the Operating Floor level of the Keowee Powerhouse and was designed primarily to afford tornado protection for electrical equipment. The controlling case was dead load plus equipment loads plus tornado wind and missile. Resulting stresses for this case were f_s = 38,000 psi and f_c = 2190 psi.

These compare to the allowable f_s = 0.9 f_y = 36,000 psi and f_c = 0.85 f′c = 2550 psi. The actual steel stresses were about 5-½ percent over the allowable stresses but 5-½ percent below the guaranteed minimum yield stress and are considered satisfactory for this severe loading combination.

A second case considered dead load plus seismic loads equal to 0.15 times the combined dead-live loads plus normal wind load. By inspection, it was found that this would result in

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[2] f′c = 4000 psi in piers.

substantially lower stresses than the loading combination above. Therefore, a detailed design check was not made.

5. Intake Structure

   Three design cases were considered:

   a. Construction condition (dead load plus wind load) with no water and allowable stresses being within the ACI and AISC Code. The resulting stresses were extremely low.

   b. Structure unwatered and stop logs in place. Allowable stresses were based on ACI and AISC Code. Calculated stresses were found to be well within the code limits.

   c. The third case considered the cylinder gate open, dead loads and seismic loads equal to 0.15 times the dead load. Maximum calculated stresses were f_s = 39,700 psi and f_c = 2050 psi.

   The resulting steel stresses are marginally below the guaranteed minimum yield stress and are considered satisfactory for the severe loading combination.

3.8.5.4 Design and Analysis Procedures

3.8.5.4.1 Turbine Building

Based on the basic criteria and general arrangement drawings of the Turbine Building, design studies were made to determine building dimensions, type of steel, member sizes, and shapes. A computer program, "Stress", was used in the analysis of the bents.

Transverse Analysis

Each bent consisted of the three main crane columns, on lines D, J, and M, the roof girders, the columns of lines K and L and the operating and mezzanine floor framing. Where continuity of framing was not interrupted by the turbine-generator support, the short columns and operating and mezzanine floor framing were included as a part of the rigid frame. See Figure 3-38 for typical Turbine Building cross-section.

Longitudinal Analysis

Column lines B, D, J, and M were braced with diagonal members. For lines D, J, and M, this bracing took the form of two members for each brace with batten plates and angle lacing tying them together.

Dynamic Seismic Analysis

A dynamic seismic analysis of the building was performed consisting of a three mass system. Section 3.8.5.3.1 describes loading conditions for the dynamic seismic analysis. Maximum accelerations in the transverse direction were taken as the absolute sum of the accelerations associated with the first three mode shapes and were 0.47 g at the roof, 0.20 g at the Operating floor, and 0.16 g at the Mezzanine floor. Maximum accelerations in the longitudinal direction were taken as the absolute sum of the accelerations associated with the first three mode shapes and were 0.57 g at the roof, 0.24 g at the Operating floor, and 0.18 g at the Mezzanine floor. It is considered that the absolute sum is a conservative value. The structure was analyzed using these accelerations and stresses were found to be within design criteria. Typical stress values, shown as percentage of allowable, are as follows:

Location	Normal Load	Seismic Load #2 (Static Analysis)	Seismic Load Dynamic Analysis)
Col. D at basement	83%	68%	44%
Col. D at roof	94%	36%	28%
Col. J below oper. floor	81%	51%	67%
Col. J above oper. floor	78%	37%	48%
Col. J at roof	88%	54%	59%
Col. M below oper. floor	89%	72%	75%
Col. M at oper. floor	84%	36%	47%
Col. M at roof	90%	56%	48%

3.8.5.4.2 Keowee Structures

The Keowee structures are designed using conventional structural analytical techniques.

3.8.5.4.3 Class 3 Structures

Class 3 structures are designed in accordance with design methods of accepted standards and codes insofar as they are applicable.

3.8.5.5 Structural Acceptance Criteria

The load combinations used in the design of the Turbine Building and Keowee structures and section strengths required to resist those load combinations are given in Section 3.8.5.3.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

Keowee Structures

All structures utilize concrete with a minimum compressive strength of 3000 psi, 40,000 psi reinforcing steel and A36 structural steel.

3.8.6 Foundations

The foundation for the Reactor Building is described in Section 3.8.1.1.

Foundation descriptions for Auxiliary and Turbine Buildings are given in Section 3.8.4.1 and Section 3.8.5.1, respectively.

3.8.7 References

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3.9 Mechanical Systems and Components

3.9.1 Special Topics for Mechanical Components

3.9.1.1 Design Transients

All Reactor Coolant System components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. Design transient cycles for the Reactor Coolant System are shown in Table 5-2 and the design transient cycles for the Pressurizer Surge Line Piping are shown in Table 5-23. Both sets of design cycles are documented in Reference 31.

As a result of NRC Bulletin 88-11, Pressurizer Surge Line Thermal Stratification, a reanalysis of the Pressurizer Surge Line (PSL) piping was performed in accordance with the 1986 ASME Code and documented in BAW-2127 (Ref. 34) and its Supplement 2 (Ref. 35). The new surge line design transients are based on actual operating transients and are used for the PSL piping analysis and are summarized in the Pressurizer Surge Line Functional Specification (BAW Document 18-1202139-000), Reference 22 of Section 5.2.4. The PSL Functional Specification is contained in Reference 31.

3.9.1.2 Computer Programs Used in Analysis1

Computer programs used to perform the code calculations on the casing for the reactor coolant pump are described in Section 5.4.1.2.

Additional computer programs used in analysis are given in Section 3.7.3.1.

3.9.1.3 Deleted Per 2004 Update

3.9.1.4 Considerations for the Evaluation of the Faulted Condition

The analytical method used for the evaluation of faulted conditions is elastic analysis. Stress limits for the faulted conditions are established in Section 3.9.3.1.

Faulted operating conditions were not applied to any components that were not a part of the reactor coolant pressure boundary.

The design stress limits for components comparable to the ASME Code Class 2 and 3 did not allow inelastic deformation.

3.9.2 Dynamic Testing and Analysis

3.9.2.1 Piping Vibration, Thermal Expansion, and Dynamic Effects

The following paragraphs describe the actions taken during the initial startup to address piping vibrations, thermal expansion and dynamic effects. It was Duke's normal practice and a startup procedure consideration to put essential and safety related systems through all of their normal and emergency modes of operation, visually observing the system for excessive movement and/or vibration. Based on operational reports indicating possible excessive movement and/or vibration, the Steam Production Department requested Design Engineering review of each case. Design Engineering observed the system making necessary measurements, readings, etc., as required to analyze the problem against existing design stress analysis and design

criteria. Based on this analysis, Design Engineering either approved the system as satisfactory or required additional design consideration. Additional supports or suppressors were designed to accommodate the effects of valve closures, pump trips, safety valve operations, and operational vibrations as required. Any problems defined for any unit were reviewed and corrected for all three units as required.

Although not required for the Oconee project, Duke conducted prior to initial station startup the following monitoring programs which are typical of Design Engineering reviews as discussed above for the purpose of comparing results with design analysis.

1. Thermal Movement Monitoring Program for the Reactor Coolant System Piping (Data was taken on Oconee 1 only; however, the report was qualified for all three units).
2. Thrust Movement Monitoring Program for the Main Steam Bypass to Condenser Piping (Data was taken on Oconee 1 only; however, the report was qualified for all three units).
3. Hanger and Restraint Setting Monitoring Program for the LP Injection System (Data was taken on Oconee 1 only; however, the report was qualified for all three units).

   Dynamic analysis is further described in Section 3.9.3.1.

3.9.2.2 Seismic Qualification Testing of Safety-Related Mechanical Equipment

When the response spectra at each elevation in the building has been determined, the G-loadings imposed on a component may then be determined. These loads are evaluated by the equipment supplier and in the case of complex components such as a heat exchanger, the design calculations performed by the supplier are reviewed by B&W Engineering or Duke, as applicable. The supplier has the freedom to use either of two alternate analytical methods to evaluate the equipment or he may choose to test it. Components may be tested by either shaker or impact tests and a certification of the test results are required. In a few cases, a manufacturer's certification that the equipment would withstand seismic conditions is acceptable based on tests of similar equipment, an example of this would be similar type pumps. Analytically the evaluation can be made by calculating the natural frequency of the component, entering the appropriate damping curve and determining the amplification factor from the response spectrum curve. The equipment is then evaluated using these G-loadings. As an alternate, the component may be evaluated without calculating the natural frequency by using the peak amplification factor from the appropriate damping curve to determine the equipment loads. This latter approach is conservative.

Special attention is given to foundation and nozzle loadings for equipment such as tanks, pumps, heat exchangers, demineralizers and filters. Loads imposed by connecting piping on a given component are included and in some cases, component nozzles have had to be reinforced to accommodate these loads. Components which are most likely to require special reinforcement due to seismic loads are long horizontal, saddle mounted tanks, vertical tanks mounted on legs, and stacked heat exchangers. These have all been evaluated and appropriately designed for the seismic conditions.

An alternate method of seismic qualification for mechanical equipment (within the applicable equipment classes) would be an experienced based approach. Seismic adequacy can be established using methods described in the Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment, Revision 3A, developed by the Seismic Qualification Utility Group (SQUG). This method is also commonly known as SQUG.

3.9.2.3 Pre-operational Flow-induced Vibration Testing of Reactor Internals

The test program developed to measure vibration of the reactor internals at Oconee 1 during hot functional testing is described in Topical Report *BAW-10038* (Reference 18).

The main objective of the testing program is to obtain measurements of flow-induced vibration to confirm the structural adequacy of the internals. The components to be instrumented were selected on the basis of an evaluation of the distressed areas in the Oconee 1 reactor internals during previous hot functional testing and of the regions with high flow velocities. Another objective of the program is to confirm current analytical methods.

BAW-10038 presents the documentation required by Safety Guide 20 to qualify the Oconee 1 internals as the prototype design for B&W's 177-fuel-assembly plant. Described therein are the instrumentation used on the internals and reactor vessel, the data acquisition system, the test conditions, the on-line analysis of data, the predicted component responses, the test acceptance criteria, and the inspection program.

3.9.2.3.1 Pre-Operational Testing

First-of-a-kind instrumentation which will measure flow induced vibrations at special locations during pre-operational testing was installed on the Oconee I internals. Confirmatory measurements were made on Oconee 2 and Oconee 3 internals.

<u>General</u>

The directions and velocities of the coolant flow are controlled by the design of the reactor internals and are primary criteria used to determine what internal components should not be measured. Consequently, a brief description of the coolant flow through the reactor as indicated in Figure 3-57 is given below.

Coolant for the core enters through the four reactor inlet nozzles. It is then directed downward in an outside annulus defined by the inside surface of the vessel, the core support shield, and the thermal shield. Approximately 99.6 percent of the downward flow enters an outside annulus at approximately 23 ft/sec. The remaining 0.4 percent enters an inside annulus between the inside surface of the thermal shield and the outside surface of the core barrel. The flow velocity in this annulus is limited to less than 1 ft/sec by orifices located in the bottom of the core barrel cylinder.

Flow in the outside annulus enters the plenum region in the bottom of the vessel, turns and then flows upward through the core. Approximately 1.5 percent of the upward flow passes through an annulus between the core barrel inside surface and the back side of the baffle plates. Velocity in this annulus is also limited to less than 1 ft/sec.

As the coolant exits from the core, it enters the plenum assembly. The plenum cylinder maintains the coolant flow parallel to the outside of the guide tube assemblies. Flow passes from the plenum to the two outlet nozzles through 34 inch and 22 inch diameter holes in the upper section of the plenum. The maximum flow velocity across the guide tube assemblies adjacent to the plenum outlets is approximately 19 ft/sec. At the two locations where a small amount of outlet flow passes through a cluster of twenty four 3-inch diameter holes, the flow across the adjacent guide tube assemblies is only 8 ft/sec.

The flow direction and velocity control where chosen to reduce the possibility of developing forces which would result in damaging vibrations in all regions of the core. The resulting velocities are low enough to preclude the necessity of measuring motions of the core barrel, control rod guide tube assembly (a part of the plenum assembly), and other upper plenum assembly components, as can be seen from the following:

1. The 19 ft/sec. flow velocity across the guide tube assemblies adjacent to the outlets in the plenum results in a vortex shedding frequency of only 6 cps. Since this shedding frequency is much lower than the 50 cps fundamental of the guide tube assembly, it was concluded that the assemblies will not have significant vibratory motion from the cross flow.
2. The flow velocity in the annulus between the core barrel and the thermal shield is less than 1 ft/sec. At this extremely small velocity, the vibratory motion of the shell modes will be negligible. Beam type motions of the core barrel can be measured by the upper accelerometer in the surveillance holder tube assembly. (The accelerometer instrumentation is described later.)
3. The plenum cover assembly is an extremely stiff assembly. Flow across the plenum cover occurs only at the outer edge of the assembly at a low velocity of 5 ft/sec. The force on the assembly due to flow is insignificant.
4. Since the coolant at 100 percent power operation is subcooled at the discharge of the fuel assembly, no steam bubbles exist which might induce vibration of the control rod guide tubes, plenum cylinder, or plenum cover assembly.

Pre-operational testing will yield results which are comparable to or more conservative than during operation for the following reasons:

1. The total flow is slightly greater during hot functional testing when the reactor core is not in place than during operation. This is particularly true for pump combinations of less than four pumps.
2. The velocities in areas of concern are not significantly influenced by the flow differences with or without the core.

Oconee 1

Instrumentation

The internal components which will be measured during pre-operational testing are the surveillance specimen holder tube, the thermal shield and the plenum cylinder. Details of the instrumentation follow.

A set of two accelerometer assemblies will be installed in each of two surveillance specimen holder tubes. The location of the holder tubes is shown in Figure 3-58. The accelerometer transducers will be located in the perforated section of the holder tube assembly as shown in Figure 3-59. In addition, two weights which simulate the surveillance capsules will be installed in each perforated tube.

The location of the lower accelerometer was selected to measure the midspan vibratory motions of the perforated tube. The perforated section of the surveillance holder tube is expected to have the largest flow induced vibratory amplitudes relative to the other sections of the holder tube assembly.

The upper end of the perforated tube is connected to the thermal shield. Consequently, the upper accelerometer will measure the thermal shield mid-plane vibratory amplitudes.

The 1-inch penetrations in the reactor vessel head permit the addition of three accelerometers to measure the shell mode vibrations of the plenum cylinder. One accelerometer will be located at the lower end of each of three tubes which are welded to the outside of the cylinder adjacent to the outlet holes as shown in Figure 3-60.

Each of the four accelerometers in the surveillance holder tube is biaxial. Therefore, there will be eight separate channels, four channels for measuring the acceleration amplitudes of the

lower section of the surveillance holder tube and four channels for detecting the accelerations of the thermal shield. The uniaxial accelerometers for the plenum cylinder will provide three channels for measuring the acceleration amplitudes of the cylinder.

The accelerometers, specially designed for the components, will be capable of measuring the frequency of the components over a range of 2 to 300 Hz at acceleration up to 30g's.

Analysis

The acceleration signals from the various components will be recorded on tape by a FM tape recorder. After the signals are recorded, the information on the tape will be digitized by use of a mini-computer which samples the data at preset time intervals. The digitized time history record will then be used as input to a computer program which will analyze the record.

A B&W proprietary computer program will be used to plot the time history of the fluctuating accelerations, determine the predominant frequencies, the autocorrelation of the signal and phase differences between signals.

Cyclic stress values will be determined from the measured acceleration amplitudes, frequency and mode shapes. These dynamic stresses will be combined with normal operational stresses. The combined stresses will be judged acceptable if they are less than the endurance limit for the materials used to manufacture the components.

Oconee 2 & 3

The reactor vessels and internals designed for Oconee 2 and 3 are essentially identical to Oconee 1. To confirm that the fabrication process has not altered the characteristics of the internals, one surveillance holder tube for Oconee 2 and one for Oconee 3 will be instrumented like Oconee 1. Measurements will be made as described for Oconee 1. The instrument cables will go through a control rod nozzle (requiring the removal of a control rod drive mechanism) because the reactor vessel heads for these units do not have the 1-inch penetrations. The results from each of these tests will be compared to those for Oconee I to confirm that the vibration characteristics are similar.

3.9.2.4 Dynamic System Analysis of the Reactor Internals Under Faulted Conditions (Reference 19)

FCF (AREVA) identified a calculation inconsistency between the Mark-B fuel assembly horizontal faulted condition analyses and Emergency Core Cooling System (ECCS) calculations specific to the requirements of 10CFR50.46 (Reference 20). FCF re-analyzed this condition and found all B&W designed plants (with skirt-supported and nozzle supported reactor vessels) fueled with Framatome Mark-B type fuel assemblies conform to the requirements of 10CFR50.46 with adequate margin of safety. Leak before-break (LBB) analyses are used to establish the design breaks and the resulting reactor internals loads and displacement time histories, and fuel assembly impact loads. The results are applicable to Oconee Nuclear Station for any Mark-B type fuel assembly.

3.9.2.4.1 Background

The existing Mark-B fuel assembly horizontal faulted analyses, i.e., Loss-of-Coolant Accident (LOCA) and combined LOCA and Safe Shutdown Earthquake (SSE), showed minor grid deformations on outer and inner fuel assemblies within the core. These deformations were shown for both the Mark-B and Mark-BZ fuel assemblies. (The Mark-B fuel assembly design comprises Inconel spacer grids; the Mark-BZ fuel assembly comprises zircaloy intermediate grids. Currently, Oconee Units are refueled with Mark-BZ type fuel assemblies). Review of the

existing ECCS calculation bases identified that no grid plastic deformation was considered for core interior fuel assemblies. Only grid plastic deformation for core periphery fuel assemblies was evaluated. Therefore, resolution of the inconsistency between the fuel assembly structural and ECCS calculation bases was necessary to ensure that the criteria set forth in 10CFR 50.46 are met.

The current Mark-B/BZ fuel assembly faulted condition analyses for horizontal fuel loadings embody substantial conservatism in the imposed forcing function (i.e. the pipe break selection). The current Mark-B/BZ fuel assembly faulted condition analyses apply reactor internals displacement time histories from large-bore pipe rupture loadings to determine fuel assembly grid loads and deformations. Highly conservative double-sided guillotine breaks at the hot leg and cold leg nozzle are evaluated in the existing Mark-B/BZ fuel assembly faulted condition analyses. Undue conservatisms are removed by using the LBB design breaks to establish the reactor internal structural loads and displacements for use in the fuel assembly structural evaluation.

Per References 21 and 27, the NRC staff has approved the use of the LBB to eliminate the large bore breaks from the design basis for structural evaluations of all BWOG plants, as established in previous topical report submittals per References 22, 23, and 28. Use of the remaining attachment line break loads is within the fuel assembly faulted structural guidelines set forth in NUREG-0800, Section 4.2, Appendix A. As such the RCS attachment line breaks are considered in the revised fuel assembly faulted structural analyses presented herein. Resulting fuel assembly spacer grid loads show that no plastic deformation occurs on any fuel assembly, thus the existing ECCS calculations conform to the requirements of 10CFR50.46.

3.9.2.4.1.1 Deleted per 1996 Revision

3.9.2.4.1.2 Deleted per 1996 Revision

3.9.2.4.1.3 Deleted per 1996 Revision

3.9.2.4.2 Postulated Loss-Of-Coolant Accidents

For reactor vessel skirt-supported plants, such as Oconee, the following postulated RCS attachment line LOCA breaks are evaluated: core flood line, decay heat line, and surge line. Note that the decay heat line break envelopes the surge line break as discussed in section 3.9.2.4.3.4.

3.9.2.4.3 Reactor Internals Analysis

The approach used to determine the core plate motions for the given attachment line breaks differs based on whether the plant is skirt-supported or nozzle-supported. Core plate motions are used as input for the fuel assembly faulted structural analysis.

3.9.2.4.3.1 RV Skirt Supported Plants

For skirt supported plants, such as the Oconee Units, the core plate motions are determined for the attachment break response by utilizing the same methodologies as used in BAW-1621 (Reference 24) for the large-bore pipe break analyses. The analyses are described below:

3.9.2.4.3.2 RV Internals Hydraulics Analysis

The hydraulic model of the RV internals and RCS loop representative of the RV skirt supported plants was retrieved and verified to be the same as that used in BAW-1621. The model was then modified to represent the three attachment line breaks (core flood line, decay heat line, and surge line). The analysis was executed using the same computer code as was done in BAW-1621 for the large-bore pipe breaks. The pressure output time-for-time was post-processed to determine the loadings on the RV and internals.

3.9.2.4.3.3 RV Asymmetric Cavity Pressure Analysis

For a postulated core flood line break (attached directly to the RV inside the RV cavity), mass and energy release from the break pressurizes the RV cavity and exerts asymmetric type loadings on the RV. Using the primary break cavity loadings as a guide, FCF determined that the Oconee cavity represented the bounding cavity for all of its RV skirt-supported plants. Hence, the Oconee RV cavity model was retrieved and verified to be the same as used in BAW-1621. Mass and energy data from the bounding core flood line break was in the computer code analysis of the cavity as was done in BAW-1621 for the large-bore pipe breaks. The pressure output time-for-time was post-processed to determine the loadings on the RV.

3.9.2.4.3.4 RV Internals Structural Analysis

The reactor vessel and internals model used for the RV skirt-supported plants in BAW-1621 was retrieved and verified. For the core flood line break, the RV internal hydraulics loadings, cavity pressure loadings, and thrust loadings were applied to the model. Upper and lower core plate motions were determined for use in the detailed fuel assembly model. For the surge line and decay heat line breaks, RV internal hydraulic loadings and thrust loads were applied to the model. These breaks have no cavity pressure, since they are attached to the primary piping outside the RV cavity. Comparison of results indicated that the decay heat line enveloped the surge line results. Core plate motions for the decay heat line break have been used in the detailed fuel assembly model.

3.9.2.4.4 Fuel Assembly Analysis

The horizontal displacement time histories for the reactor vessel lower core plate, upper core plate and the core baffle under the postulated loss-of-coolant loadings, as specified above, were used as input into the fuel assembly faulted structural analysis. A non-linear dynamic analysis was performed to calculate fuel assembly loadings and grid impact loads. The method and models of analysis were per NRC approved Topical Report BAW-10133, Revision 1 (Reference 25). For the horizontal analysis, two orthogonal directions were evaluated, namely X and Y. The detailed fuel assembly mathematical model used was applicable to the RV skirt-supported plants.

3.9.2.4.4.1 LOCA Analysis

Core Flood Line Guillotine - RV Skirt Supported Plant

FCF determined that the grid impact loads were less than the allowable spacer grid elastic load limit. This load limit is the 95/95 confidence level buckling load determined by impact tests performed on production grids at reactor operating temperatures. Tests were performed as described in BAW-10133P, Rev. 1. Therefore, the fuel assembly spacer grids will remain elastic, and the coolable geometry will be maintained during the postulated Core Flood Line break.

Decay Heat Line Guillotine - RV Skirt-Supported Plant

FCF results show no grid impact forces, i.e., no fuel assembly grid contact is made. Therefore, the coolable geometry will be maintained during the postulated Decay Heat Line break. These results are also applicable for the Surge Line break, since it is bounded by the Decay Heat Line break.

3.9.2.4.4.2 Combined Seismic and LOCA Analysis

The loads for LOCA and seismic conditions were combined by the square-root-of-sum-of-squares method (SRSS) as discussed and accepted by the NRC in NUREG-0800, Standard Review Plan 4.2 (Reference 26). The maximum grid impact forces for the seismic analyses were obtained from the existing seismic analysis performed for the Mark-BZ fuel assembly. The seismic input at the core supports, that envelope the seismic input for Oconee, was used in the fuel assembly seismic analysis.

FCF determined the maximum impact force for the Safe Shutdown Earthquake (SSE) condition and the combined SSE and LOCA condition. The impact force is less than the spacer grid elastic load limit. The empirical method for determining the spacer grid elastic load limit is as described in BAW-10133P, Rev. 1. Hence, the spacer grids will remain elastic for all loads from postulated breaks coupled with seismic excitation. Therefore, the requirement to maintain a coolable geometry is met for all fuel assemblies within the core.

3.9.2.4.5 Conclusion

The LBB licensing basis allows the most limiting breaks in RCS attachment lines to be used in evaluating RCS components for LOCA integrity. The use of the RCS attachment line breaks is thereby incorporated into the design basis for the evaluation of the dynamic effects of LOCA events on Mark-B/BZ fuel assemblies.

The spacer grid impact loads for all the faulted conditions with the LBB licensing basis are within the spacer grid elastic load limit. Therefore, no permanent grid deformation is predicted and the coolable geometry requirements are met for all fuel assemblies within the core.

FCF determined that substantial margin exists between the applied load and the grid elastic load limit (control rod insertion will not be hindered under any faulted condition); therefore, control rod insertability is ensured and the requirements of 10CFR50.46 are met.

3.9.2.4.5.1 Deleted per 1996 Revision

3.9.2.4.5.2 Deleted per 1996 Revision

3.9.2.4.5.2.1 Deleted per 1996 Revision

3.9.2.4.5.2.2 Deleted per 1996 Revision

3.9.2.4.5.2.3 Deleted per 1996 Revision

3.9.2.5 Deleted per 1996 Revision

3.9.2.5.1 Deleted per 1996 Revision

3.9.2.5.2 Deleted per 1996 Revision

3.9.2.5.3 Deleted per 1996 Revision

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

3.9.3.1 Load Combinations, Design Transients and Stress Limits

3.9.3.1.1 Reactor Coolant System

The Reactor Coolant System is designed structurally for 2,500 psig and 650°F. The system will normally operate at 2,155 psig and 604°F. The design transients are defined in Section 3.9.1.1.

The number of transient cycles specified in Table 5-2 and Table 5-23 for the fatigue analysis is conservative.

Reactor Coolant System components are designated as Class 1 equipment and are designed to maintain their functional integrity during an earthquake. Design is in accordance with the seismic design bases shown below. The loading combinations and corresponding design stress criteria for internals and pressure boundaries of vessels and piping are given in the section. These are summarized in Table 3-26. A discussion of each of the cases of loading combinations follows:

Case I - Design Loads Plus Operating Basis Earthquake (OBE) Loads - For this combination, the reactor must be capable of continued operation; therefore, all components excluding piping are designed to Section III of the ASME Code for Reactor Vessels. The primary piping is designed according to the requirements of USAS B31.1 and B31.7. The S_m values for all components, excluding bolting, are those specified in Table N-421 of the ASME Code. The S_m value for bolts are those specified in Table N-422 of the ASME Code. The primary system piping was redesigned to the 1983 ASME Code (No Addenda) during the steam generator replacement project.

Case II - Design Loads Plus Safe Shutdown Earthquake (SSE) Loads - In establishing stress levels for this case, a "no-loss-of-function" criterion applies, and higher stress values than in Case I can be allowed. The multiplying factor of (1.2) has been selected in order to increase the code-based stress limits and still insure that for the primary structural materials, i.e., 304 SST,

316 SST, SA302B, SA212B, and SA106C, an acceptable margin of safety will always exist. A more detailed discussion of the adequacy of these margins of safety is given in B&W Topical Report BAW-10008, Part 1, "Reactor Internals Stress & Deflection Due to LOCA and Maximum Hypothetical Earthquake (MHE)". Note that the MHE is equivalent to the SSE. The S_m values for all components are those specified in Table N-421 of the ASME Code.

The load cases for consideration of the faulted condition are defined below.

A loss-of-coolant accident coincident with a seismic disturbance has been analyzed to assure that no loss of function occurs. In this case, primary attention is focused on the ability to initiate and maintain reactor shutdown and emerg`ency core cooling. Two additional cases are considered as follows:

Case III - Design Loads Plus Pipe Rupture Loads - For this combination of loads, the stress limits for Case II are imposed for those components, systems, and equipment necessary for reactor shutdown and emergency core cooling.

Case IV - Design Loads Plus Safe Shutdown Earthquake (SSE) Loads Plus Pipe Rupture Loads - Two thirds of the ultimate strength has been selected as the stress limit for the simultaneous occurrence of safe shutdown earthquake and reactor coolant pipe rupture. As in Case III, the primary concern is to maintain the ability to shut the reactor down and to cool the reactor core. This limit assures that a materials strength margin of safety of 50 percent will always exist.

The design allowable stress of Case IV loads is given in B&W Topical Report BAW-10008 for 304 stainless steel. This curve is used for all reactor vessel internals including bolts. It is based on adjusting the ultimate strength curves published by U. S. Steel to minimum ultimate strength values by using the ratio of ultimate strength given by Table N-421 of Section III of the ASME Code at room temperature to the room temperature strength given by U. S. Steel.

In Cases II, III and IV, secondary stresses were neglected, since they are self-limiting. Design stress limits in most cases are in the plastic region, and local yielding would occur. Thus, the conditions that caused the stresses are assumed to have been satisfied. See B&W Topical Report BAW-10008, Part 1, for a more extensive discussion of the margin of safety, the effects of using elastic equations, and the use of limit design curves for reactor internals.

3.9.3.1.1.1 Deleted Per 2004 Update

3.9.3.1.1.1.1 Deleted Per 2004 Update

3.9.3.1.1.1.2 Deleted Per 2004 Update

3.9.3.1.1.1.3 Deleted Per 2004 Update

3.9.3.1.1.1.4 Deleted Per 2004 Updatel

3.9.3.1.1.1.5 Deleted Per 2004 Update

3.9.3.1.1.1.6 Deleted Per 2004 Update

3.9.3.1.1.1.7 Deleted Per 2004 Update

3.9.3.1.1.1.8 Deleted Per 2004 Update

3.9.3.1.1.1.9 Deleted Per 2004 Update

3.9.3.1.1.2 Steam Generator Replacement Analysis of the Reactor Coolant System

This section contains the following categories of information.

1. Pertinent information on the seismic design of the Reactor Coolant System.
2. A description of the type and location of each major component support analyzed, its design, and the seismic amplification associated with the location in the support building.
3. A correlation between a free-standing spacial analysis of the Nuclear Steam System and a planar analysis considering building-loop inter action.

3.9.3.1.1.2.1 Scope of Analysis

The Reactor Coolant System consists of the reactor vessel, coolant pumps, steam generators, pressurizer, and interconnecting piping. For the purpose of seismic analysis the Reactor Coolant System consists of all of the above components in addition to the pressurizer support steel, surge line snubbers, and the reactor coolant pump snubbers.

3.9.3.1.1.2.2 Description of Analytical Models

Seismic Analysis

See Section 3.7.3.1.1 for a description of the RCS seismic analysis for replacement steam generators.

Consistent mass is used to represent the Reactor Coolant Loop piping and the majority of the component weights. Lumped mass is used to represent the Control Rods, the Main and Auxiliary Feedwater Headers, the Pump assemblies, the Snubber's weight on the Pumps, the Pressurizer's whip restraint, and the Hot Legs' whip restraint. The torsional mass moment of inertia is included for the Reactor Vessel, Control Rod Drive mechanisms, Service Support structure, Pressurizer, and the Steam Generators since the analysis code does not calculate this term for those model sections having their cross sections modeled as pipe. Additionally, a portion of the "bending" mass moment of inertia, which represents the difference between a slender rod and a circular cylinder, is included for above components.

Dead Load Analysis

Input into the dead load analysis consists of three parts:

1. Material and Water Densities: These densities represent piping and support steel weight as well as piping water weight. Except for some cases where lumped masses are used to represent the weight.
2. Distributed (Linear) Weights: These weights represent component shells and heads, internals, entrained water and insulation and in the case of the Reactor Coolant Pumps they represent the motor and motor stand weight.
3. Applied Forces: These forces represent component head weights and attachment weights (supports and whip restraints, CRDM's, etc.). These applied forces are used to include the weights that are included as lumped masses.

Thermal Expansion

In order to consider all of the possible normal and upset operating conditions and perform a fatigue analysis, five thermal expansion load cases will be run. 0% power is used, as it is the highest power level where the hot leg and cold leg are at the same temperature (532^o F). It is used mostly in the fatigue analyses and is often subtracted from other power levels when considering an operating cycle. 8% power is used in the fatigue analyses, as it is the maximum power level for transients 1A and 1B (heatup/cooldown). 15% power is the maximum normal operating cold leg temperature (575^o F). This power level typically gives the maximum cold leg loads during the entire heatup. 100% power is used in the fatigue analysis as the maximum normal power level. The Trip 8/11 is a combined loadcase developed to determine the maximum hot and cold leg loads for any of the normal/upset transients included in the Reactor Coolant System Functional Spec (Reference 31).

Trip 8/11 Material Properties

The Trip 8/11 thermal condition is an envelop of all of the transients included in the RCS Functional Specification. It used in fatigue analysis and gives the worst case thermal expansion stresses for the piping and component nozzles. Review of that document showed that the highest hot leg temperature was 650^o F and occurred with transient 8A (Reference 31) (Figure 8-1). The highest cold leg temperatures was 592^o F and occurred with transient 11 (Reference 31) (Figure 11-1). Since the Steam Generator has a fixed base, the hot leg and cold leg are nearly isolated structurally. Therefore, choosing the highest temperature from all transients is a conservative method of enveloping the transients. It should be noted that the cold leg temperature used from transient 11 occurs over a very short period of time. Heat transfer analysis considering the actual temperature time history would show that the average through wall temperature of the cold leg would never reach this temperature during the actual transient. Therefore, this analysis is very conservative for the cold leg.

Transient 8A occurs at 100% power while transient 11 occurs at 15% power. Since the growth of the Steam Generator has more effect on the hot leg, the hot leg transient (Transient 8A) will be used for the Steam Generator. According to the Functional Spec for Transient 8A (Reference 31, Figure 8-1), the Steam Generator reaches a temperature of about 548^o F in about 1 minute and the feedwater and steam flow drop to about 0 lbm/sec. Therefore, 548^o F will be used for the Steam Generator shell, excluding the upper and lower heads, support stool, and tubesheets. The outlet nozzles will be considered at the same temperature as the rest of the cold leg (592^o F) as they do not affect the hot leg but have significant effect on the cold leg.

3.9.3.1.1.2.3 Stress Analysis of Reactor Coolant Piping

Stress calculations made at various locations throughout the piping system are done in accordance with subsection NB of Section III of the 1983 Edition of the ASME Boiler and Pressure Vessel Code, (no addenda). Stress calculations were performed using the pipe stress equations found in Article NB-3600 of the 19893 Code. Primary and primary plus secondary stresses were calculated at each location and comparison made to 1.5 S_m and 3 S_m respectively. The primary stresses are calculated using equation 9 in subsection NB-3652 of section III of the 1983 edition of the ASME Code, and the primary plus secondary stresses are calculated using equation 10 in subsection NB-3653 of the ASME code. The highest primary stress at any location was found to be 20,580 psi, which is below the allowable value of 58,200 psi. The loads used to calculate the faulted primary stresses are combined in the following manner. The seismic and LOCA conditions are combined using SRSS, then added to operating pressure and deadweight.

3.9.3.1.1.2.4 Stress Evaluation of the Reactor Vessel

Stress evaluation of the reactor vessel is discussed in Section 5.2.3.3.1.

3.9.3.1.1.2.5 Stress Evaluation of Steam Generators

The stress evaluation of the replacement steam generators is included in the Base Design Condition Report (BWC-006K-SR-01) (Reference 29) and the Transient Analysis Stress Report (BWC-006K-SR-02) (Reference 30).

3.9.3.1.1.2.6 Stress Evaluation of the Reactor Coolant Pumps

The reactor coolant pump casings are designed, fabricated, inspected and tested to meet the intent of the ASME Boiler and Pressure Vessel Code, Section III, for Class A vessels, but are not code stamped.

The reactor coolant pump casing has been completely analyzed including a dynamic analysis separately from the loop to insure that the stresses throughout the casing are below the allowable for all design conditions.

An analysis in accordance with Paragraph N-415.1 of the ASME Code was performed to determine if the pump casing required a fatigue analysis for the number of design cycles specified. This analysis showed that the pump casing bowl met all the requirements of Paragraph N-415.1. Thus a fatigue analysis was not required. However, a fatigue analysis was performed on the pump casing cover in which the worst possible stress combination was considered at the two most critical points in the cover. It was found from this analysis, with this very conservative approach, that the maximum cumulative usage factor is only 0.125 for design cycles specified for this plant.

See Section 5.4.1.2 for a discussion of the code allowables and maximum calculated stresses for the reactor coolant pumps.

The casing cover analysis indicates that the thermal stresses and pressure stresses on the cover are within the Section III code allowables.

There are no deviations from the applicable ASME Code requirements in the design and fabrication of the pump casings other than code stamping.

To accommodate for the different reactor coolant pumps, the reactor coolant loops for Unit 1 and Units 2, 3 were analyzed separately for Deadweight, Thermal, Seismic, and Loss of Coolant Accidents for the replacement steam generator analysis (Reference 32 and 33). The replacement steam generators do not change the overall structural performance of the reactor coolant pumps for Units 1, 2, and 3. The RCPs remain qualified using the applicable ASME Code requirements.

3.9.3.1.1.2.7 Stress Analysis of Pressurizer Surge Line Piping

Stress calculations made at various locations throughout the Surge Line were performed in accordance with the ASME Code, Section III, Subarticle NB-3600. Pursuant to the code, seismic, thermal, pressure and cyclic loadings were considered in the analysis. The results indicate that the subject pipe meets all design criteria.

Stress calculations were made at various locations throughout the surge line piping in the replacement steam generator analysis in accordance with applicable ASME Code requirements. The replacement steam generators do not impact the qualification of the surge line piping.

3.9.3.1.1.2.8 Summary and Conclusion

The replacement steam generator analysis for Units 1, 2, and 3 was used to generate forces and moments for normal, seismic and LOCA conditions at critical locations throughout the reactor coolant system. These loads or the stresses resulting from them were compared to the allowable loads and stresses for each of the individual component locations. The replacement steam generator analysis found that there is no impact to the qualification of the reactor coolant system.

3.9.3.1.2 Other Duke Class A, B, and C Piping

Piping which is Class A, B, or C is defined in Section 3.2. The applicable Code requirements are established in Section 3.2. The seismic requirements for this piping are defined in Section 3.2. The seismic analysis techniques are defined in Section 3.7.3.

3.9.3.1.3 Field Routed Piping and Instrumentation

Duke's practice is to detail the routing of all safety-related and non safety related process lines regardless of size, except as follows:

1. Process piping - All main run process piping in Duke System Classification A, B, C, D, and F is detailed on engineering drawings; however, items such as vents, drains, valve bypass warming lines, and pump seal water for all systems are “field run”. Class E piping not meeting the limitations of Specification OSS-0027.00-00-0003 is detailed on engineering drawings.

   Instrument impulse lines - end points and specific routing requirements of any safety-related and non-safety-related instrument tubing lines are established per instrumentation and Controls Field Installation Standards, Specification OSS-0060.00-00-0001.

   Class E, G & H piping can be field routed and supported with certain limitations per Specification OSS-0027.00-00-0003.

2. It is not practical to limit “field run” piping to an extent greater than this for the following reasons:

   a. Obstruction to desirable routing would be difficult to determine and documentation of a precisely designed path would be lengthy, difficult to prepare, and difficult to follow.

   b. Revision to major process piping would cause changes in routing of small lines, resulting in many drawing changes without significant improvement in the final result.

   c. Sloping of impulse lines would be difficult to accomplish and document.

   Thus, field routing of small lines results in a superior job since obstruction and other revisions are clearly visible and easier to consider while meeting design requirements as established by OSS-0060.00-00-0001.

3. The special rigorous quality assurance measures and performance tests that will be conducted to assure satisfactory installation of field run piping and instrument tubing lines are as follows:

   a. All field engineered lines are schematically shown either on a diagrammatic, an instrumentation detail or a piping drawing such that mistakes in valving, connection termination points and materials are virtually eliminated.

   b. Requirements for seismic design for field run piping and tubing is established prior installation.

c. Except for very low pressure lines downstream of vent and drain valves and instrument impulse lines, all “field run” piping is hydrostatically tested in accordance with the requirements of the main process system.

d. Prior to erection, engineering specifies requirements for interaction of seismically designed with non seismically designed structures. After erection, QA/QC reviews all safety-related and non-safety-related piping and tubing in the area to assure that appropriate criteria have been followed.

e. Instrument impulse line installation is inspected by site QA/QC prior to turnover of the system to operations.

f. Instrumentation testing programs are well defined by Duke test procedures. These tests document conclusively that the instrument loops are correctly installed and operate properly.

This practice of “controlled field routing” of small piping and instrument tubing lines produces the best possible overall results. It is not practical to limit “field run” piping to a greater extent.

3.9.3.2 Pump and Valve Operability Assurance

Equipment pre-operational test programs are described in Section 14.2.

The NRC issued IE Bullentin 88-04, "Potential Safety-Related Pump Loss," on May 5, 1988. The purpose of this bulletin was to request licensee investigation and correction, as applicable, of two miniflow design concerns for plant safety-related pumps. The first concern involved the potential for dead-heading of one or more pumps in safety-related systems that have a miniflow line common to two or more pumps or other piping configurations that do not precluded pump-to-pump interaction during miniflow operations. The second concern was whether or not the installed miniflow capacity is adequate for even a single safety-related pump in operation. Final evaluations and operability justifications per the requirements of this bulletin were presented in response to the NRC by letter on January 15, 1990 (letter form H.B. Tucker to the NRC, dated January 15, 1990). Futher programmatic enhancements and long-term corrective actions committed to in this response were verified complete/closed out in the letter from M.S. Tuckman to the NRC, dated January 10, 1991.

The NRC issued Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves," onAugust 17, 1995, to request that licensees take actions to identify safety-related power-operated gate valves that are susceptible to pressure locking or thermal binding, and ensure that they are capable of performing their safety functions. Evaluations of the valves within this category were completed with responses to the NRC submitted in References 37, 38, 39, and 40. The responses included commitments to replace specific valves, modify specific valves, and test specific valves during future outages. The NRC accepted the actions and closed this issue in Reference 41. Valve operators are designed to actuate the valves with the maximum system pressure drop across the valve plus packing friction forces and potential pressure locking and thermal binding forces. Where possible, power-operated gate valves are of the parallel disc design or the flexible wedge design, which release the mechanical holding force during the first increment of travel so that the operator works only against the frictional component of the hydraulic imbalance and the packing box friction.

3.9.3.3 Design and Installation Details for Mounting of Pressure Relief Devices

Design analysis and installation criteria for safety and relief valves located within the reactor coolant and main steam (thru main stop valves) pressure boundaries are as follows:

1. Piping and its Support-Restraint System are designed to accommodate and/or restrain the piping for both dynamic and static loadings as applicable such that stresses produced are within code allowables for the following:
   a. Dead weight effect
   b. Thermal loads and movements
   c. Seismic loads and deflections - movements
   d. Safety valve thrust and moment
   e. Maximum absolute differential movement between structures

   Applicable loadings are combined and considered as described in Section 3.2.
2. Nozzles are analyzed and appropriate reinforcement added such that code allowables stresses are maintained for:
   a. Internal pressure
   b. Safety valve thrust
   c. Safety valve moment

In particular, for the main steam lines outside the Containment, pressure relief is accomplished through the use of sufficient safety relief valves to meet code requirements. The safety valves are set for progressive relief in intermediate steps of pressure within the allowed range of pressure settings to prevent all valves actuating simultaneously. Valves are located on a horizontal run of pipe and are oriented in a manner that will produce torsion and bending in the main pipe during operation of the valves. The valves are staggered on opposite sides of the main steam line and set to relieve progressively to counterbalance the torque produced. But the valves could be within the allowed range and the maximum net torque on the piping system could result from four valves. The piping system is designed to accept the net torque resulting from four safety valves operating simultaneously on the same side of the line. The piping support and restraint system is designed using shock suppressors and rigid stops to limit piping system stresses within code allowables as discussed above.

Dynamic thrust effects were analyzed for the Reactor Coolant System pressurizer relief discharge line to the Quench Tank, constituting a closed system. Stresses produced by the thrust effects were within the established Code allowables for the station. No other safety-related closed systems exist for Oconee.

3.9.3.4 Component Supports

3.9.3.4.1 Reactor Coolant System Component Supports

3.9.3.4.1.1 Description of Supports

The reactor vessel is supported by a cylindrical skirt and the replacement steam generators by a conical stool. These supports are rigidly attached to the vessels and bolted to the foundation by means of an integral base plate. The skirts and stool are designed in accordance with ASME Section III and criteria stated in Section 3.9.3.1.1. Lateral support is provided for the steam generator at the upper tube sheet level by means of a structural tie to the secondary shield wall.

The pressurizer is supported by 8 support pads spaced symmetrically around the circumference of the vessel. The pads are designed in accordance with Section III and criteria stated in Section 3.9.3.1.1 of this report.

The reactor coolant piping is self-supporting with respect to dead weight, seismic, and thermal loading. The reactor coolant pumps are partially supported by hanger rods which are designed to support the dead weight of the pump motor, with the remainder of the dead weight of the pump being supported by the piping. To reduce seismic deflection, the pumps are supported laterally at the motor by means of hydraulic suppressors connected to the secondary shield wall.

3.9.3.4.1.2 Method of Analysis

3.9.3.4.1.2.1 Calculation of Foundation Loads for Reactor Vessel and Replacement Steam Generator

Deleted paragraph(s) per 2004 update

The reactor coolant system with the replacement steam generators was modeled using a three dimensional analysis. Consistent mass is used to represent the Reactor Coolant Loop piping and the majority of the component weights. Lumped mass is used to represent the Control Rods, the Main and Auxiliary Feedwater Headers, the Pump assemblies, the Snubber's weight on the Pumps, the Pressurizer's whip restraint, and the Hot Legs' whip restraint. The torsional mass moment of inertia is included for the Reactor Vessel, Control Rod Drive Mechanisms, Support Service Structure, Pressurizer, and the Steam Generators since the analysis code does not calculate this term for those model sections having their cross sections modeled as pipe. Additionally, a portion of the "bending" mass moment of inertia, which represents the difference between a slender rod and a circular cylinder, is included for above components. The reactor vessel support is modeled using rotational springs at the base to represent the flexibility of the anchor bolts and concrete foundation beneath the vessel. Similar modeling approach is completed for the replacement steam generator base support.

3.9.3.4.1.2.2 Calculation of Foundation Loads for Pressurizer

Deleted paragraph(s) per 2004 update

The pressurizer and its support frame were included in the reactor coolant loop model. Seismic and LOCA loads were generated by dynamic analysis. Loads due to the thermal expansion of the piping were included as well as the dead weight of the vessel at normal operating conditions.

The vessel wall was analyzed for local loading, from the attached support, by means of a method developed by P. P. Bijlaard. The resulting stress intensities were compared to stress allowables specified in ASME Section III and criteria stated in Section 3.9.3.1.1.

3.9.3.4.1.2.3 Analysis of Reactor Vessel and Steam Generator Supports

The reactor vessel support skirt and support skirt flange is designed and analyzed using procedures described in Chapter 10, Section 1, of Reference 15. That procedure is used to determine the tensile stress in the anchor bolts, the bearing stress on the support skirt flange and the location of the neutral axis of bending on the bolt-flange mechanism.

The skirt-flange mechanism was statically analyzed for the applied forces and moments due to seismic loading on the vessel, considering a free-standing vessel.

The support skirt flange and foundation is assumed to be rigid. In regard to the reactor vessel, effects of anchor bolt pretension on the bending moment capacity of the support skirt were evaluated. With no anchor bolt pretension, the location of the neutral axis is found by trial and error methods so that the difference between the first moment of the bolt tension area and first

moment of the flange compression area about the neutral axis is less than 5 percent of the smaller value. Increasing values of applied anchor bolt pretension result in less shift of the neutral axis.

The anchor pretension load necessary to prevent any separation of the support skirt flange from the foundation is the required load which will result in no shift of the neutral axis. In that case the neutral axis is located on the centerline of the vessel flange.

For a typical seismic load condition on the vessel, the support skirt flange was analyzed for flange bearing stress, anchor bolt loads, and location of neutral axis. Once the neutral axis was located, giving consideration to anchor bolt pretension loads, the flange, skirt, gusset mechanism was analyzed for applied tensile, compressive, and shear loads resulting from bending using methods from engineering mechanics.

The allowable stress criterion specified in Section 3.9.3.1.1 of this report was used where applicable. Finite element method is employed to analyze the replacement steam generator base support stool and flange for the applied forces and moments due to deadweight, thermal, seismic and LOCA. In the finite element model, the concrete foundation and the anchor bolts are represented by appropriate compressive-only and tensile-only elements, respectively, such that bearing stress on the concrete and tensile stress on the bolts can be calculated. The analysis is documented in the Base Design Condition Report (BWC-006K-SR-01) (Reference 29).

3.9.3.4.2 Supports for Other Duke Class A, B, C and F Piping

3.9.3.4.2.1 Allowable Stress Criteria

3.9.3.4.2.1.1 Structural Members

Allowable stresses are as follows:

1. Tension

Normal Ft = 0.6 Fy

Upset Ft = (1.33) (.6 Fy) = .8 Fy

Faulted Ft = Fy

2. Bending in Structural Members

(Laterally Supported to preclude local compressive instability)[2]

Normal Fb = 0.6 Fy

Upset Fb = (1.33) (0.6 Fy) = .8 Fy

Faulted Fb = Fy

3. Bending in Base Plates

Normal Fb = 0.75 Fy

Upset Fb = (1.33) (0.75 Fy) = Fy

Faulted Fb = Fy

4. Shear

Normal	Fv = 0.4 Fy
Upset	Fv = (1.33) (0.4 Fy) = 0.533 Fy
Faulted	Fv = (1.5) (0.4 Fy) = 0.6 Fy

5. Compression

Normal	Fc = Fa[1]
Upset	Fc = 1.33 Fa
Faulted	$Fc = [1.67 - (\frac{0.33(K1/r)}{Cc^1})]Fa,$
	For K1/r < Cc
	Fc = 1.33 Fa, for K1/r > Cc

Note:

1. See Section 1.5.1.3 AISC 6th Edition for definition.
2. See Section 1.5.1.4 AISC 6th Edition for allowable extreme fiber compressive stress in bending for rolled shapes, built-up members, channels, etc., when full lateral support is not provided.

3.9.3.4.2.1.2 Allowable Stresses for ASTM A36 Materials

Allowable Stress (Ksi)			
Loading	**Normal**	**Upset**	**Faulted**
Tension	21.6	28.8	36.0
Bending			
a. Members	21.6	28.8	36.0
b. Base Plates	27.0	36.0	36.0
Shear	14.4	19.2	21.6

Note:

1. Stress allowables for normal and upset load conditions are derived from the AISC Manual of Steel Construction, 6th Edition. Stress allowables for faulted load conditions are established by factoring AISC, 6th Edition allowables.
2. For stress conditions not covered in Section 3.9.3.4.2.1.1, the AISC allowables are utilized for normal loadings, and 133 percent of the AISC allowables shall be utilized for upset loadings.
3. No increase in stress allowables is permitted for material strain hardening and/or strain rate effects due to dynamic loadings.

The specified minimum yield stress (Fy) for ASTM A501 and ASTM A500, Grade B structural tubing is as defined in the 7th Edition of the AISC Manual of Steel Construction.

3.9.3.4.2.1.3 Weld Stresses

1. Tension, bending, compression, and shear on effective throat of complete penetration groove welds and normal compression and shear on effective throat of partial-penetration groove welds permissible allowables are the same as the base material.
2. All other shear:

   Normal Fv = 18.0 ksi

   Upset Fv = (1.33) (18.0) = 24.0 ksi

   Faulted Fv = (1.5) (18.0) = 27.0 ksi

Note:

1. The above shear allowables are based upon the use of E60XX electrodes. The normal allowable is taken from AWS Standard AWS D1.0-69 and Table 1.5.3 in the 7th Edition of the AISC Manual. The normal allowable is higher (18 ksi vs. 13.6 ksi) than the allowable given in the 6th Edition of the AISC Manual. For electrodes other than E60XX, the normal allowable is taken from Table 1.5.3 in the 7th Edition of the AISC Manual. The corresponding upset and faulted allowable is obtained by multiplying the normal allowable by 1.33 and 1.50 respectively.

   The normal weld allowables were increased 33 percent by AWS and AISC in 1969 to eliminate over conservatism required by the AISC 6th Edition. This over-conservatism was present due to design criteria for welds which were inconsistent with the remainder of the AISC Code and lack of test data. Changes in procedures or materials were not a consideration in this change. It is therefore considered appropriate to utilize the AISC 7th Edition weld allowables on Oconee Nuclear Station.

3.9.3.4.2.1.4 Standard Components

All standard components shall be limited by the recommended allowable load specified in either manufacturer's Hanger Standards, manufacturer's Load Capacity Data Sheets (LCD's) or Qualified Product Load Ratings.

3.9.3.4.2.1.5 Combined Stresses In Structural Members

Members subjected to both axial compression and uniaxial bending stresses shall comply with Section 1.6 of the AISC Manual, 6th Edition.

Members subjected to both axial tension and bending stresses shall comply with Section 1.6 of the AISC Manual, 6th Edition.

Members subjected to biaxial bending coincident with axial tension or compression shall be proportioned to satisfy the requirements of Section 1.6 of the AISC Manual, 6th Edition, in accordance with the guidance provided by Section 1.6 of the Commentary in the AISC Manual, 6th Edition.

3.9.3.4.2.1.6 Bolts and Threaded Parts

All allowable stresses are *based on unthreaded body area* of bolts and threaded parts.

A307 BOLTS

1. Tension

Normal = 14 ksi

Upset = 1.33 (14) = 18.6 ksi

Faulted = 1.67 (14) = 23.4 ksi

2. Shear

Normal = 10 ksi

Upset = 1.33 (10) = 13.3 ksi

Faulted = 1.5 (10) = 15 ksi

Threaded Parts of Other Steels

1. Tension

Normal = .4 Fy

Upset = 1.33 (.4 Fy) = .53 Fy

Faulted = 1.67 (.4 Fy) = .67 Fy

2. Shear

Normal = .3 Fy

Upset = 1.33 (.3 Fy) = .4 Fy

Faulted = 1.5 (.3 Fy) = .45 Fy

Stress allowables for other bolts are given in Section 1.5.2, Table 1.5.2.1 of the AISC Manual, 6th Edition. The following factors shall be applied to normal allowable stresses:

	Tension	**Shear**
Normal	1.0	1.0
Upset	1.33	1.33
Faulted	1.67	1.5

For combined shear and tension, refer to Section 1.6.3 of the AISC Manual, 6th Edition.

3.9.3.4.2.2 Snubbers

Piping systems designed to resist seismic forces have been restrained by steel supports capable of withstanding these seismic forces. Snubbers are used at locations where restraints are necessary based on piping stress analysis, but thermal movement of the pipe must not be constrained. Performance selection is based on manufacturer's load capacity data and the requirement that the allowable travel of the snubber exceed the calculated pipe thermal travel. The hot and cold settings on the snubber are established such that the pipe's calculated thermal travel will not exceed the snubbers travel range. In systems where it was necessary to use

hydraulic or mechanical snubbers to resist seismic forces, the mechanical action associated with the snubber makes it possible to consider them as restraints against pipe whipping (see Section 3.6).

Duke Power Company specifies a margin of zero between design requirements and purchase requirements because design loads are determined by detailed computerized piping analysis or other conservative analysis techniques. In most cases, a margin does exist between the design load and the maximum allowable design load of the suppresser supplied since:

1. Suppressers are manufactured for a relatively small number of load ranges; therefore, each suppresser size covers many possible loadings.
2. Suppressers supplied for the Oconee Nuclear Station clearly envelope the design load required for the particular restraint application.

Prior to at their installation at the Oconee Nuclear Station, all snubbers are functional tested on a specifically designed test stand to insure they meet design criteria. Hydraulic snubbers are tested for activation velocity and bleed rate. Mechanical snubbers are tested for drag and acceleration rate.

Visual inspections are performed on all hydraulic and mechanical snubbers on regular intervals to identify those that are damaged, degraded, or inoperable as caused by physical means, leakage, corrosion, or environmental exposure. The inspection interval is based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. The interval between inspections will not be greater than 48 months.

To verify that a snubber can operate within specific performance limits, Oconee Nuclear Station performs functional testing that involves removing the snubber and testing it on a specifically-designed test stand. As on installation, hydraulic snubbers are tested for activation velocity and bleed rate, and mechanical snubbers are tested for drag and acceleration rate. Ten percent of the total snubber population are tested during each refueling outage. Oconee Nuclear Station separates the snubber population into hydraulic and mechanical and pulls a minimum 10% sample from each group. For each snubber that does not meet the functional test acceptance criteria, an additional minimum 10% sample of that snubber type will be tested until none are found inoperable or all the snubbers of that type have been functionally tested. Functional testing in this manner provides a 95% confidence level that 90% to 100% of the snubbers operate within the specified acceptance limits.

3.9.3.4.2.2.1 Hydraulic Snubbers

When a seismic event acts on a system that uses a hydraulic snubber to resist the seismic forces, it causes the piston rod of the snubber to move faster than the activation threshold velocity of that snubber. When this happens, a differential pressure is generated on the valve that allows fluid to flow from one end of the snubber cylinder to the other and the valve closes. With this by-pass valve closed, the snubber acts as a near rigid structural member, thus limiting any further movement of the pipe at the point of attachment. A by-pass or bleed orifice between the two ends of the cylinder prevents the snubber from exceeding its' rated capacity and allows a gradual pressure drop even under sustaining load against the closed by-pass valve. A hydraulic snubber resists seismic forces by limiting velocity.

The design data for the hydraulic shock and sway suppressers used on Class I piping systems at the Oconee Nuclear Station are summarized in the charts below.

Grinnell Hydraulic Snubbers

Size Bore	(In.) Stroke	Acceleration	Activation Threshold Velocity In./Min	Activation Threshold Normal(1) Load(lbs)	One Time Load(lbs)(1)
1½	5	Not Applicable	8	3,000	4,000
1 ½	10		8	1,100	1,500
2 ½	5 and 10		8	12,500	25,700
3 ¼	5 and 10	Insensitive To Acceleration	8	21,000	43,500
4	5 and 10		8	32,000	66,000
5	5 and 10		8	50,000	103,000
6	5 and 10		5	72,000	148,000
8	5		3	128,000	264,000

Note:

1. Actual Allowable load may be less than specified depending on length of overall assembly.

LISEGA HYDRAULIC SNUBBERS

SNUBBER TYPE (1)	STROKE (2)	NORMAL LOAD (2)	REACTION VELOCITY	BYPASS VELOCITY
30185x	4 inches	675 lbs.	4.7 - 14.2 ipm	.47 - 4.7 ipm
30385x	4 inches	1800 lbs.	4.7 - 14.2 ipm	.47 - 4.7 ipm
30395x	8 inches	1800 lbs.	4.7 - 14.2 ipm	.47 - 4.7 ipm
30425x	5-7/8 inches	4000 lbs.	4.7 - 14.2 ipm	.47 - 4.7 ipm
30435x	11-3/4 inches	4000 lbs	4.7 - 14.2 ipm	.47 - 4.7 ipm
30525x	5-7/8 inches	10350 lbs	4.7 - 14.2 ipm	.47 - 4.7 ipm
30535x	11-3/4 inches	10350 lbs.	4.7 - 14.2 ipm	.47 - 4.7 ipm
30625x	5-7/8 inches	22450 lbs.	4.7 - 14.2 ipm	.47 - 4.7 ipm
30635x	11-3/4 inches	22450 lbs.	4.7 - 14.2 ipm	.47 - 4.7 ipm
30725x	5-7/8 inches	44900 lbs.	4.7 - 14.2 ipm	.47 - 4.7 ipm
30735x	11-3/4 inches	44900 lbs.	4.7.- 14.2 ipm	.47 - 4.7 ipm
30825x	5-7/8 inches	78600 lbs.	4.7 - 14.2 ipm	.47 - 4.7 ipm
30835x	11-3/4 inches	78600 lbs.	4.7 - 14.2 ipm	.47 - 4.7 ipm
30925x	5-7/8 inches	123500 lbs.	4.7 - 14.2 ipm	.47 - 4.7 ipm
30935x	11-3/4 inches	123500 lbs.	4.7 - 14.2 ipm	.47 - 4.7 ipm

SNUBBER TYPE (1)	STROKE (2)	NORMAL LOAD (2)	REACTION VELOCITY	BYPASS VELOCITY

Notes:

1. These are the model numbers for the stock Lisega snubbers. The model numbers used at the Oconee Nuclear Station are usually followed by a suffix showing that it is a replacement for a particular brand of snubber. (i.e. 303856RF1 is a type 3038 snubber, nuclear specification (5), design year 1986, that replaces a PSA-1 mechanical snubber with a flanged end.)
2. Snubber stroke and end attachments can be modified by the manufacturer, at the request of the purchaser, and this may effect the normal load of the snubber.
3. The 'x' on the end of the snubber type is an abbreviation of the design year (i.e. 6 = 1986)

Lisega has been audited by ASME to certify them to supply Component Standard Supports manufactured without welding in accordance with ASME Section III, Subsection NF.

3.9.3.4.2.2.2 Mechanical Snubbers

When seismic forces in a system are resisted using mechanical snubbers, the mechanical snubber translates linear movement between the system and the support structure into rotational motion within the snubber. The snubber's telescoping cylinder is attached to the fixed support cylinder by a screw and nut assembly. Relative motion between the two causes the screw shaft to turn which causes an inertia mass to turn. The torque required to start the rotary motion of the snubber internals limits the rate of acceleration of the attached pipe. A mechanical snubber resists forces by limiting acceleration.

The design data for mechanical snubbers used on Class I piping systems at the Oconee Nuclear Station is summarized in the chart below.

Pacific Scientific Mechanical Snubbers

Allowable Loads @ 300°F

	SIZE	STROKE (IN>)	ACCELERA-TION LIMIT	NORMAL[1] LOAD (LBS)	ONE TIME[1] LOAD(LBS.)
	1/4	4	0.64ft/sec^2	350	590
	1/2	2.5	0.64ft/sec^2	650	1,040
STD.	1	4	0.64ft/sec^2	1,500	2,300
STROKE	3	5	0.64ft/sec^2	6,000	11,520
	10	6	0.64ft/sec^2	15,000	23,600
	35	6	0.64ft/sec^2	50,000	91,000
	100	6	0.64ft/sec^2	120,000	180,000
EXT. STROKE	1	8	0.64ft/sec^2	1,487	2,200
	3	10	0.64ft/sec^2	6,000	11,520
	10	12	0.64ft/sec^2	14,400	22,032

	SIZE	STROKE (IN>)	ACCELERA-TION LIMIT	NORMAL[1] LOAD (LBS)	ONE TIME[1] LOAD(LBS.)

Note:

1. Actual allowable load may be less than specified depending on angular displacement of load path with end bracket.

3.9.4 Control Rod Drive Systems

The Control Rod Drive Mechanism is described in Section 4.5.3.

3.9.5 Reactor Pressure Vessel Internals

Reactor pressure vessel internals are described in Section 4.5.

3.9.6 References

1. Porse, L., "Reactor Vessel Design Considering Radiation Effects," *ASME Paper No. 63-WA-100*.
2. Pellini, W. S. and Puzak, P. P., "Fracture Analysis Diagram Procedures for the Fracture-Safe Engineering Design of Steel Structures," *Welding Research Council Bulletin 88, May 1963*.
3. Robertson, T. S., "Propagation of Brittle Fracture in Steel," *Journal of Iron and Steel Institute, Volume 175*, December 1953.
4. Kihara, H. and Masubichi, K., "Effects of Residual Stress on Brittle Fracture," *Welding Journal, Volume 38*, April 1959.
5. Hjarne, L., and Leimdorfer, M., "A Method for Predicting the Penetration and Slowing Down of Neutrons in Reactor Shields," *Nuclear Science and Engineering 24*, pp 165-174, 1966.
6. Cadwell, *et al*., "The PDQ-5 and PDQ-6 Programs for the Solution of the Two-Dimensional Neutron Diffusion-Depletion Problem," *WAPD-TM-477*, January 1965.
7. Aalto, *et al*., "Measured and Predicted Variations in Fast Neutron Spectrum in Massive Shields of Water and Concrete," *Nuclear Structural Engineering 2*, pp 233-242, August 1965.
8. Avery, A. F., "The Prediction of Neutron Attenuation in Iron-Water Shields," *AEEW-R 125*, April 1962.
9. Clark, R. H., and Baldwin, M. N., "Physics Verification Program, Part II," *BAW-3647-4*, June 1967.
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12. Chen, L. H., "Piping Flexibility Analysis by Stiffness Matrix," ASME *Journal of Applied Mechanics*, December 1959.
13. "Design of Piping System," The M. W. Kellogg Company, Second Edition, 1956.
14. "Nuclear Reactors and Earthquakes," *TID-7024*, Chapters 5 and Appendix E, August 1963.

15. Brownwell, L. E. and Young, E. H., *Process Equipment Design*, John Wiley and Sons, 1959.
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18. Thoren, D. E. and Harris, R. J., "Prototype Vibration Measurement Program for Reactor Internals, 177-Fuel Assembly Plant." Babcock and Wilcox. *BAW-10038*, September, 1972.
19. BAW-2292, Revision 0, Mark-B Fuel Assembly Spacer Grid Deformation in B&W Designed 177 Fuel Assembly Plants, Aug. 1997.
20. Letter, R. J. Schomaker, B&W Owners Group to Document Control Desk relating to the Interim Report of Potential Safety Concern on Mark-B Grid Deformation, Framatome Technologies PSC-21-96-5, May 24,1996.
21. NRC Safety Evaluation of B&W Owners Group Reports Dealing with Elimination of Postulated Pipe Breaks in PWR Primary Main Loops, dated December 12, 1985.
22. B&W Topical Report BAW-1847, Revision 1, "Leak-Before-Break Evaluation of Margin Against Full Break for RCS Primary Piping of B&W Designed NSS," September 1985.
23. B&W Topical Report BAW-1889P, "Piping Material Properties for Leak-Before-Break Analysis," October 1985.
24. B&W Topical Report BAW-1621, "B&W 177-FA Owners Group - Effects of Asymmetric LOCA Loadings - Phase 11 Analysis", July 1980.
25. B&W Topical Report BAW-10133P, Revision 1, "Mark C Fuel Assembly LOCA-Seismic Analyses," S. J. Shah and R. E. Lide, Revision 1, June 1986.
26. Standard Review Plan, Section 4.2, NUREG-0800, Rev. 2 U. S. Nuclear Regulatory Commission, July, 1981.
27. NRC Safety Evaluation Relating To Elimination of Dynamic Effects of Postulated Primary Loop Pipe Ruptures from Design Basis in Regard to TMI-1, dated November 5, 1987.
28. B&W Topical Report BAW-1999, "TMI-1 Nuclear Power Plant Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping", April 1987.
29. Babcock & Wilcox Canada Report No.: BWC-006K-SR-01, Revision 1, “Replacement Steam Generators Base Design Condition Report,” August 2003 (Proprietary) (OSC-8318).
30. Babcock & Wilcox Canada Report No.: BWC-006K-SR-02, Revision 1, “Replacement Steam Generators Transient Analysis Report,” August 2003 (Proprietary) (OSC-8319).
31. Calculation OSC-1862, Functional Specification (Transient Description) for Reactor Coolant System.
32. Calculation OSC-7835, Steam Generator Replacement Project: ONS Unit 1 Reactor Coolant Structural Analysis for ROTSG’s.
33. Calculation OSC-7836, Steam Generator Replacement Project: ONS Units 2 and 3 Reactor Coolant Loop Structural Analysis for ROTSG’s.
34. Nuclear Regulator Commission, Letter to All Holders of Operating Licenses or Construction Permits for Nuclear Power Reactors, from David B. Matthews, May 5, 1988, NRC Bulletin No. 88-04, "Potential Safety-Related Pump Loss."
35. Duke Power Company, Letter from H.B. Tucker to NRC, January 15, 1990, re: Response to NRC Bulletin No. 88-04, "Potential Safety-Related Pump Loss," Action 4 Final Report.

36. Duke Power Company, Letter from M.S. Tuckman to NRC, January 10, 1991, re: Response to NRC Bulletin No. 88-04, "Potential Safety-Related Pump Loss," Description of Actions Completed or in Progress.
37. Duke Power Company, Letter from M. S. Tuckman to NRC, February 13, 1996, "McGuire, Catawba, Oconee Response to Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves."
38. Duke Power Company, Letter from J. W. Hampton to NRC, July 18, 1996, "Oconee Nuclear Site, Docket Nos. 50-269, 270, 287, Generic Letter 95-07, "Response to Request for Additional Information."
39. Duke Power Company, Letter from W. R. McCollum to NRC, August 21, 1997, "Oconee Nuclear Site, Docket Nos. 50-269, 50-270, 50-287, "Supplemental Response to Generic Letter 95-07, "Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves."
40. Duke Power Company, Letter from W. R. McCollum to NRC, October 9, 1997, "Oconee Nuclear Site, Docket Nos. 50-269, 50-270, 50-287, "Supplemental Response to Generic Letter 95-07, Pressure Locking and Thermal Binding of Safety-Related Power Operated Gate Valves."
41. NRC, Letter to D. E. LaBarge to W. R. McCollum, November 6, 1997, "Evaluation of Generic Letter 95-07, Pressure Locking and Thermal Binding of Safety-Related Power-Operated Gate Valves, Response for Oconee Nuclear Station, Units 1, 2, and 3."
42. B&W Report BAW-2127, Final Submittal for Nuclear Regulatory Commission Bulletin 88-11, Pressurizer Surge Line Thermal Stratification, December 1990.
43. B&W Report BAW-2127 Supplement 2, “Pressurizer Surge Line Thermal Stratification for the B&W 177-FA Nuclear Plants, Summary Report, Fatigue Stress Analysis of the Surge Line Elbows,” May 1992.

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3.10 Seismic Qualification of Instrumentation and Electrical Equipment

This section describes the seismic considerations applied to instrumentation and electrical equipment during the original design of the Oconee Nuclear Station as well as in modifications to the station after issuance of the operating license.

3.10.1 Seismic Qualification Criteria

The seismic design basis for instrumentation and electrical equipment is that the electrical devices considered essential in performing Reactor Protection and Engineered Safeguards functions and in providing emergency power shall be designed to assure that they will not lose their capability to perform intended safety functions during and following the design basis event (MHE). This basic criteria has remained unchanged since the issuance of the operating license; however, the seismic qualification techniques and documentation requirements for various plant modifications have in many instances followed the advances in the state of the art.

The specific equipment included in the scope identified above including the associated seismic qualification documentation reference is provided in Table 3-68.

The seismic adequacy of all electrical cable tray supports is established by the methods and criteria established for cable tray supports in the Generic Implementation Procedure (GIP-3A) for Seismic Verification of Nuclear Plant Equipment, Rev 3A, developed by the Seismic Qualification Utility Group (SQUG).

3.10.2 Methods and Procedures for Qualifying Instrumentation and Electrical Equipment

In order to meet the seismic design objectives defined in Section 3.10.1, the following seismic evaluation methods were employed consistent with the applicable licensing commitment.

Testing

Devices may be qualified by either shaker or impact tests. A certification of the test results or a copy of the test results are required. Additionally, a manufacturer's certification that a certain type of equipment would withstand the seismic conditions is acceptable based on previous testing/experience with similar equipment.

Analysis

Devices may also be qualified by analytical methods. For example, one evaluation method involves calculating/determining the natural frequency of the device, entering the appropriate response spectra damping curves, and determining the corresponding amplification factor. The device is then evaluated using this "G" loading value. Alternatively, the devices may be evaluated without calculating/determining its natural frequency by using the peak amplification factor from the appropriate response spectra damping curve to determine the "G" loading.

An alternate method of seismic qualification for electrical equipment (within the applicable equipment classes) would be an experienced based approach. Seismic adequacy can be established using methods described in the Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment, Revision 3A, developed by the Seismic Qualification Utility Group (SQUG). This method is also commonly known as SQUG.

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3.11 Environmental Design of Mechanical and Electrical Equipment

3.11.1 Equipment Identification and Environmental Conditions

Duke has a program in place for environmental qualification of safety-related electrical equipment inclusive of equipment required to achieve a safe shutdown. The program is described in Duke Energy Procedures PD-EG-ALL-1612 and AD-EG-ALL-1612. Environmental effects resulting from the postulated design basis accidents documented in Chapter 15 have been considered in the qualification of electrical equipment which is covered by this program. This program has been reviewed and approved by NRC (Reference 2).

3.11.1.1 Equipment Identification

Safety-related electrical equipment that is required to perform a safety function(s) in a postulated harsh environment is identified in Oconee Nuclear Station Equipment Data Base.

Safety-related mechanical equipment including design information is identified in Section 3.2.2.

3.11.1.2 Environmental Conditions

The postulated harsh environmental conditions resulting from a LOCA or High Energy Line Break (HELB) inside the Reactor Building and a HELB outside the Reactor Building are identified and discussed in the Oconee Nuclear Station Environmental Qualification Criteria Manual.

The environmental parameters that compose the overall worst-case containment environment are as follows:

Containment Temperature: Time history as shown in Figure 6-37 for the Design Basis Accident (DBA), a 8.55 ft^2 cold leg break.

Containment Pressure: Time history as shown in Figure 6-36 for a 8.55 ft^2 cold leg break.

Relative Humidity: 100%

Radiation: Total integrated radiation dose for the equipment location includes the 60 year normal operating dose plus the appropriate accident dose based on equipment operability requirements. The bases for determining the containment radiation environment are discussed in Chapter 12.

Chemical Spray: Boric acid spray resulting from mixing in the containment sump with borated water from the borated water storage tank. Refer to Section 6.2.2 for additional information on chemical spray.

3.11.2 Qualification Test and Analysis

Safety-related equipment identified in Section 3.11.1.1 is qualified by test and/or analysis. The test report, which describes the method of qualification for this Class 1E equipment is identified in the Oconee Nuclear Station Environmental Qualification Maintenance Manual, EQMM-1393.01.

3.11.3 Qualification Test Results

The results of the qualification tests and/or analyses for the electrical equipment identified in Section 3.11.1.1 are presented in the qualification documentation references identified in the Oconee Nuclear Station Environmental Qualification Maintenance Manual, EQMM-1393.01.

3.11.4 Evaluation for License Renewal

Some qualification analyses for safety-related equipment identified in Section 3.11.1.1 were found to be a time-limited aging analyses for license renewal. Evaluations were performed for applicable electrical equipment with the results submitted in Reference 5.

3.11.5 Loss of Ventilation

The control area (control room, cable room and electrical equipment room) air conditioning and ventilation systems (Section 9.4.1) are conservatively designed to provide a suitable environment for the control and electrical equipment.

Deleted paragraph(s) per 2005 update.

Control area temperatures related to station blackout are addresssed by SLC 16.8.1.

3.11.6 Estimated Chemical and Radiation Environment

The estimated chemical and radiation environments at Oconee are discussed in Duke Power Company's response to NRC IE Bulletin 79.01B (Reference 1). Additional information regarding chemical and radiation conditions is presented in Section 6.5 and in Chapter 12, respectively.

3.11.7 References

1. Oconee Nuclear Station Response to IE Bulletin 79-OIB, as revised, including Response to NRC Equipment Qualification Safety Evaluation Report.
2. Letter from J. F. Stolz (NRC) to H. B. Tucker (Duke) dated March 20, 1985.

   Subject: Safety Evaluation Report on Environmental Qualification of Electrical Equipment Important to Safety.
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4. Deleted per 2008 Update.
5. 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.
6. 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.

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3.12 Cranes and Control of Heavy Loads

The load cycle limit of the Oconee Polar Cranes has been identified as a time-limited aging analysis by reviewing correspondence on the Oconee dockets associated with the control of heavy loads. In 1981, NRC issued Generic Letter 81-07 and NUREG-0612 [Reference 1]. NRC issued a letter [Reference 2] requesting additional information which Duke responded to by letter [Reference 3]. One of the concerns expressed in NUREG-0612 was the potential for fatigue of the crane due to frequent loadings at or near design conditions. Cranes at Oconee are not generally subjected to frequent loads at or near design conditions. The topic of lift cycles of cranes at or near rated load is considered to be a time-limited aging analysis for Oconee because the analysis meet all of the criteria contained in Section 54.3 [Reference 4].

From the license renewal review, the existing analyses addressing heavy load lifts of both the polar cranes and the spent fuel pool cranes were determined to be valid for the period of extended operation [Reference 5].

3.12.1 References

1. Generic Letter 81-07, *NUREG-0612, Control of Heavy Loads,* NRC, February 3, 1981.
2. J. F. Stolz (NRC) to W. O. Parker (Duke) letter dated February 18, 1982, Oconee Nuclear Station, Docket Numbers 50-269, 50-270, and 50-287.
3. W. O. Parker (Duke) letter to Document Control Desk (NRC) dated October 8, 1982, Oconee Nuclear Station, Docket Numbers 50-269, 50-270, and 50-287.
4. *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.
5. 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.

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3.13 Oconee Nuclear Station Response to Beyond-Design-Basis External Event Fukushima Related Required Action (FLEX)

3.13.1 Introduction

On March 11, 2011, an earthquake-induced tsunami caused Beyond-Design-Basis (BDB) flooding at the Fukushima Dai-ichi Nuclear Power Station in Japan. The flooding caused by the tsunami rendered the emergency power supplies and electrical distribution systems inoperable resulting in an extended loss of alternating current (AC) power (ELAP) in five of the six units on the site. The ELAP led to the loss of core cooling as well as spent fuel pool cooling capabilities and a significant challenge to containment. All direct current (DC) power was lost early in the event on Units 1 & 2 and after some period of time at the other units. Units 1, 2, and 3 were affected to such an extent that core damage occurred and radioactive material was released to the surrounding environment.

The US Nuclear Regulatory Commission (NRC) assembled a special task force, the Near-Term Task Force (NTTF) in order to advise the Commission on actions the US Nuclear Industry should undertake in order to preclude a release of radioactive material in response to a natural disaster such as that seen at Fukushima Dai-ichi. NTTF members created NRC Report "Recommendations for Enhancing Reactor Safety in the 21st Century: The Near-Term Task Force Review of Insights from the Fukushima Dai-ichi Accident," referred to as the "90-day Report," which contained a large number of recommendations for improving safety at US nuclear power sites.

Subsequently, the NRC issued Order EA-12-049, "Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events" (Agencywide Documents Access Management System (ADAMS) Package Accession No. ML12054A736) (Reference 1) and Order EA-12-051, "Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation" (ADAMS Package Accession No. ML12056A044) (Reference 2) to implement strategies for Beyond-Design-Basis External Events (BDBEE), and reliable spent fuel pool, respectively.

3.13.2 Order EA-12-049

NRC Order EA-12-049 was effective immediately and directed Oconee Nuclear Station to develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and spent fuel pool cooling in the event of a beyond-design-basis external event.

The Nuclear Energy Institute (NEI), working with the nuclear industry, developed guidelines for nuclear stations to implement the strategies specified in NRC Order EA-12-049. These guidelines were published in the NEI 12-06 document entitled "Diverse and Flexible Coping Strategies (FLEX) Implementation Guide" (Reference 3). This guideline was endorsed by the NRC in final interim staff guidance (ISG) document JLD-ISG-2012-01, Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design- Basis External Events, Revision 0, dated August 29, 2012 (ML12229A174) (Reference 4).

The NEI 12-06 FLEX implementation guide adopts a three-phase approach for coping with a BDB event.

- Phase 1 – the initial phase requires the use of installed equipment and resources to maintain or restore core cooling, containment, and SFP cooling capabilities.

- Phase 2 – The transition phase requires providing sufficient portable onsite equipment to maintain or restore these functions until resources can be brought from off site.
- Phase 3 – The final phase requires obtaining sufficient offsite resources to sustain these functions indefinitely.

This three-phase approach was utilized to develop the FLEX strategies for Oconee Nuclear Station.

3.13.3 Order EA-12-051

NRC Order EA-12-051 (Reference 2) states that procedures shall be established and maintained for the testing calibration and use of the primary and backup SFP instrument channels.

Duke developed procedures using guidelines and vendor instructions to address the maintenance, operation, and abnormal response issues associated with the SFP level instrumentation at ONS.

3.13.4 BDB Program

Strategies, details, and programmatic controls for mitigating beyond-design-basis external events are contained in a Duke program document (General Reference per NEI 98-03, Revision 1). Program changes are controlled in accordance with NEI 12-06, Section 11.8, as endorsed by the NRC.

A Duke program document (General Reference per NEI 98-03, Revision 1) also describes items such as a list of FLEX equipment, the BDB Storage Building, initial and periodic testing, FLEX equipment maintenance, and actions to be taken in the event of equipment unavailability.

A Duke program document (General Reference per NEI 98-03, Revision 1) also describes Spent Fuel Pool Instrumentation program requirements including procedures, testing and calibration, and quality assurance.

3.13.5 References

1. Order EA-12-049, "Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events" dated March 12, 2012 (ML12054A736).
2. Order EA-12-051, “Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation” dated March 12, 2012 (ML12056A044).
3. NEI 12-06, Diverse and Flexible Coping Strategies (FLEX) Implementation Guide, Revision 0, dated August 2012 (ML12242A378).
4. NRC Interim Staff Guidance JLD-ISG-2012-01, Compliance with Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events, Revision 0, dated August 29, 2012 (ML12229A174).
5. CSD-EG-ONS-1619.1000, “Diverse and Flexible Coping Strategies (FLEX) Program Document - Oconee Nuclear Station

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