TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

CHAPTER 4. REACTOR

4.1 SUMMARY DESCRIPTION

The reactor is a pressurized water reactor and is functionally comprised of the reactor internals, fuel system, and control rod drives. The fuel system consists of the fuel assemblies and control components.

The major functions of the reactor internals are to support the core, maintain fuel assembly alignment, and direct the flow of reactor coolant.

The fuel system is designed to operate at 2,568 MWt with sufficient design margins to accommodate transient operation and instrument error without damage to the core and without exceeding limits for the Reactor Coolant System (RCS). The fuel system is designed to meet the performance objectives within
6 the limits of design and operation specified in Section 4.2, "Fuel System Design," Section 4.3, "Nuclear
6 Design," and Section 4.4, "Thermal and Hydraulic Design."

The fuel assembly is designed for structural adequacy and reliable performance during core operation. This includes steady-state and transient conditions under the combined effects of pressure, temperature, hydraulic forces, and irradiation. The fuel assembly is mechanically compatible with the reactor internals
7 control rod assemblies and burnable poison rod assemblies. There are 2 axial power shaping rod (APSR)
7 designs. See Section 4.2.2, "Description - Fuel System Design" for information on compatibility of the
7 fuel assembly designs with the APSR designs. In addition to incore operation, the fuel assembly must be designed for handling, shipping, and storage to assure that the fuel assembly maintains its dimensional and structural integrity. Section III of the ASME Boiler and Pressure Vessel Code serves as a guide for fuel assembly and reactivity control component analysis.

The fuel assembly thermal-hydraulic operating characteristics have been determined and found to be compatible with design limits. Power peaks are controlled during transients so that no fuel melting
6 occurs. The minimum core DNB ratio at the design overpower is maintained above the design limit. Although net steam generation occurs in the hottest core channels at the design overpower, hydraulic stability analyses have shown that no flow oscillations will occur.

The control components (control rod assemblies, axial power shaping rod assemblies, and burnable poison rod assemblies) are designed to perform their functions in controlling the reactor.

Core reactivity is controlled by control rod assemblies (CRAs), axial power shaping rod assemblies (APSRAs), burnable poison rod assemblies (BPRAs) and soluble boron in the coolant. Sufficient CRA worth is available to shut the reactor down with at least 1% $\Delta k/k$ subcritical margin in the hot condition at any time during the life cycle with the most reactive CRA stuck in the fully withdrawn position. Equipment is provided to add soluble boron to the reactor coolant to ensure a similar shutdown capability when the reactor is cooled to ambient temperatures.

The reactivity worth of a CRA and the rate at which reactivity can be added are limited to ensure that credible reactivity accidents cannot cause a transient capable of damaging the RCS or causing significant fuel failure.

The control rod guide path is designed to ensure that the control assemblies will not disengage from the fuel assembly guide tubes during operation. Guidance is provided by close-tolerance indexing of the fuel assembly upper end fitting with the upper grid rib section.

4.2 FUEL SYSTEM DESIGN

The fuel system consists of fuel assemblies and control components which are designed to the bases described in Section 4.2.1, "Design Bases - Fuel System Design" and Section 4.2.2, "Description - Fuel System Design."

4.2.1 DESIGN BASES - FUEL SYSTEM DESIGN

The fuel is designed to meet the performance objectives specified in Section 4.2.1.1, "Fuel System Performance Objectives" without exceeding the limits of design and operation specified in Section 4.2.1.2, "Limits."

4.2.1.1 Fuel System Performance Objectives

The core is designed to operate at 2568 MWt (rated power) with sufficient design margins to
6 accommodate transient operation and instrument error without fuel damage.

The fuel rod cladding is designed to maintain its integrity for the anticipated operating transients
6 throughout the fuel assembly lifetime. The effects of gas release, fuel dimensional changes, and corrosion-
or irradiation-induced changes in the mechanical properties of cladding are considered in the design of fuel assemblies.

4.2.1.2 Limits

4.2.1.2.1 Nuclear Limits

5 The core has been designed to the following nuclear limits and capabilities, all of which are intended to
5 preserve the integrity of the fuel assemblies:

5 1. The core will have sufficient reactivity to produce the design power level and lifetime without
5 exceeding the control capacity or shutdown margin.

7 2. Fuel assemblies have been designed for the maximum burnups shown in Table 4-2.

5 3. Power histories must be bounded by those assumed within generic mechanical and thermal hydraulic
5 (fuel assembly) analyses. If they are not bounded, acceptable reanalyses shall be performed.

5 4. The maximum feed fuel enrichment is constrained by the maximum allowed in the Technical
5 Specifications (Spent Fuel Pool storage requirements).

5 5. Values of important core safety parameters predicted for the cycle have been verified to be
5 conservative with respect to their values assumed in the Chapter 15 safety/accident (and any other
5 pertinent) analyses. If they are not conservative, acceptable reanalyses shall be performed.

5 Controlled reactivity insertion rates due to a single CRA group withdrawal shall be limited to a
5 maximum value assumed within the Chapter 15 Rod Withdrawal Accident at Rated Power, and
5 within the Chapter 15 Startup Accident. Controlled reactivity insertion rates due to soluble boron
5 removal shall be limited to a maximum value assumed within the Chapter 15 Moderator Dilution
5 Accident.

5 The power Doppler and moderator temperature coefficients at power will be negative. However, as
5 described within Chapter 15, the control system is capable of compensating for reactivity changes
5 resulting from either positive or negative nuclear coefficients.

5 6. Reasonable and permissive reactor control and maneuvering procedures during nominal operation and
7 during transients will not produce unacceptable peak-to-average power distributions. This, along with
5 criteria 7 and 8, below, preserves the LOCA linear heat rate, linear heat rate to melt (LHRTM), and
5 DNBR limits.

5 7. Part length axial power shaping rods (APSRs) are to be utilized to allow the shaping of power axially
5 in the core, thereby thwarting any tendency towards axial instability resulting from a redistribution of
5 xenon.

5 To preclude the possiblity of azimuthal instability resulting from a redistribution of xenon, the highest
5 moderator temperature coefficient assumed within Chapter 15 safety/accident analyses must be
5 bounded by the threshold listed within Table 4-7.

5 8. Technical Specification limits of specified operating parameters (quadrant power tilt, power imbalance,
5 and control rod insertion), and on reactor protective system trip setpoints (power imbalance) after
5 allowance for appropriate measurement tolerances should have adequate margin from design limits of
5 these parameters during operational conditions throughout the cycle such that sufficient operating
5 flexibility is retained for the fuel cycle.

5 **4.2.1.2.2 Reactivity Control Limits**

5 The control system and operational procedures will provide adequate control of the core reactivity and
5 power distribution. The following control limits and capabilities shall be:

5 1. A control system consisting of part length axial power shaping rods (APSRs) shall be provided to
5 control the core axial power distribution.

5 2. A shutdown margin of at least 1.0% $\Delta\rho$ shall be maintained throughout core life with the most
5 reactive CRA stuck in the fully withdrawn position.

5 3. CRA withdrawal rate (as listed within Chapters 7 and 15) shall limit the maximum reactivity insertion
5 rate to that assumed within the Chapter 15 Rod Withdrawal Accident at Rated Power, and within the
5 Chapter 15 Startup Accident.

5 4. Boron dilution rate (as listed within Chapter 15) shall limit the maximum reactivity insertion rate to
5 that assumed within the Chapter 15 Moderator Dilution Accident.

5 5. A control rod shall not be misaligned from the group average by the value listed within the Technical
5 Specifications, and constrained within Chapters 7 and 15 (Control Rod Misalignment Accident).
5 Except during the startup physics test program, operating rod overlap shall be within the bounds
5 listed within the Technical Specifications, and constrained within Chapters 7 and 15 (Startup
5 Accident).

5 6. Maximum boron (hot full power, or otherwise) will be constrained by those assumed within Chapter
5 15 or Technical Specifications. Sufficient soluble boron shall be available within the control system
5 equipment (BWST, CBAST, and CFT) to ensure a 1.0% $\Delta\rho$ shutdown capability with the most
5 reactive CRA stuck in the fully withdrawn position when the reactor is cooled to ambient
5 temperatures.

5 7. There are no design constraints on BPRA poison enrichment or number of BPRA assemblies, except
5 for those inferred by the peak-to-average power distributions constraints listed within Table 4-1, by
5 Chapter 15 constraints, by Technical Specifications constraints (such as moderator temperature
5 coefficient), or by the limiting core bypass flow assumed within thermal hydraulic analyses.

5 For more detail, refer to Section 4.3, "Nuclear Design."

4.2.1.2.3 Thermal and Hydraulic Limits

The reactor core is designed to meet the following limiting thermal and hydraulic conditions:

5 1. The fuel pin must be designed so that the maximum fuel temperature does not exceed the fuel melting
9 limit at any time during core life. The TACO3 computer program is used to verify heat rate capacity
9 (Reference 2).

9 2. The minimum allowable DNBR during steady-state operation and anticipated transients is 1.18 with
5 the BWC correlation (Reference 6).

3. Although generation of net steam is allowed in the hottest core channels, flow stability is required during all steady-state and operational transient conditions.

By preventing a departure from nucleate boiling (DNB), neither the cladding nor the fuel is subjected to excessively high temperatures.

For more detail refer to Section 4.4, "Thermal and Hydraulic Design."

4.2.1.2.4 Mechanical Limits

Fuel assemblies are designed for structural adequacy and reliable performance during core operation, handling, and shipping. Design criteria for core operation include steady state and transient conditions under combined effects of flow induced vibration, temperature gradients, and seismic disturbances.

Spacer grids, located along the length of the fuel assembly, position fuel rods in a square array, and are designed to maintain fuel rod spacing during core operation, handling, and shipping. Spacer-grid to
6 fuel-rod contact loads are established to minimize fretting, but also allow axial relative motion resulting from fuel rod irradiation growth and differential thermal expansion.

The fuel assembly upper end fitting is indexed to the plenum assembly by the upper grid rib section immediately above the fuel assemblies to assure proper alignment of the fuel assembly guide tubes to the control rod guide tube. The guidance of the control rod assembly and axial power shaping rod assembly is designed such that these assemblies will never be disengaged from the fuel assembly guide tubes during operation.

1. Section III of the ASME Boiler and Pressure Vessel Code is used as a guide in classifying the stresses
7 into various categories and combining these stresses to determine stress intensities. Refer to Section
7 4.2.3.1.1, "Clad Stress and Strain" for the Duke clad stress and strain methodology.

7

6 2. Cyclic Strain limits for this stress condition are established based on low cycle fatigue techniques, not to exceed 90 percent of the material fatigue life. Evaluation of cyclic loading is based on conservative estimates of the number of cycles to be expected. An example of this type of stress is the thermal stress resulting from thermal gradients across the cladding thickness.

3. Cladding uniform strain is limited to a maximum of 1.0 percent.

4. Cladding Collapse

9 The digital computer code CROV (References 1 and 11) is used to demonstrate that the effective full
5 power hours (or equivalent burnup) to complete cladding collapse is greater than the incore residence
7 time. Refer to Section 4.2.3.1.2, "Cladding Collapse" for Duke's creep collapse methodology.

5. Fuel Thermal Analysis

9 The digital computer code TACO3 (Reference 2) is used to ensure that fuel performance is
satisfactory. Specifically the centerline temperature is maintained below fuel melt limits and end of life
5 pin pressure is maintained below the value which would cause clad lift off. Refer to Section 4.2.3.1.3,
5 "Fuel Thermal Analysis" for design evaluations of the fuel thermal analyses.

9 6. The cladding oxide thickness for the highest burnup rod in each sub-batch is limited to 100 μm as
9 calculated on a best estimate basis.

4.2.2 DESCRIPTION - FUEL SYSTEM DESIGN

The complete core has 177 fuel assemblies which are arranged in the approximate shape of a cylinder. All
fuel assemblies are similar in mechanical construction, and are mechanically interchangeable in any core
4 location. The reactivity of the core is controlled by 61 control rod assemblies (CRAs) and 8 axial power
shaping rod assemblies (APSRAs), a variable number of burnable poison rod assemblies (BPRAs), and
6 soluble boron in the coolant. APSRAs are similar in physical configuration to the CRAs but have
absorber material only in the lower portion of the rods. Burnable poison rod assemblies (Figure 4-1) are
installed in selected fuel assemblies not containing an APSRA or a CRA. The burnable poison rod
assemblies (BPRAs) assure a negative moderator temperature coefficient through core lifetime. The
4 mechanical and geometric configuration of the CRAs and BPRAs permit full interchangeability in any
6 fuel assembly.

6 There are 2 APSR designs which are not fully interchangeable between fuel assembly designs, because of
6 the difference in hold down spring designs and APSR drive mechanisms. Table 4-22 depicts the APSR
6 and fuel assembly compatibility for each unit.

Important core design, thermal, and hydraulic characteristics are tabulated in Table 4-1, and fuel assembly component materials are presented in Table 4-2.

4.2.2.1 Fuel Assemblies

4.2.2.1.1 General

9 Fuel assembly designs (References 17 and 18) are limited to those that have been analyzed with the
8 applicable NRC approved codes and methods. A limited number of lead test assemblies that have not
8 completed representative testing may be placed in non-limiting core locations. Detailed lead test assembly
8 design information is not included in the following descriptions because it is anticipated that new designs
8 will evolve as necessary.

9 The fuel assembly design shown in Figure 4-37 is typical of the designs used in Oconee 1, 2 and 3.

9 Cladding, fuel pellets, end caps, and fuel support components form a "fuel rod." Two hundred and eight
7 fuel rods, sixteen control rod guide tubes, one instrumentation tube assembly, seven segmented spacer
9 sleeves, eight spacer grids, and two end fittings make up the basic "Fuel Assembly" (Figure 4-37). The
guide tubes, spacer grids, and end fittings form a structural cage to arrange the rods and tubes in a 15 x 15
array. The center position in the assembly is reserved for instrumentation. Control rod guide tubes are
9 located in 16 locations of the array. Use of similar material in the guide tubes and fuel rods results in
9 minimum differential thermal expansion. Fuel assembly components, materials, and dimensions are
9 tabulated in Table 4-2. Fuel assembly design descriptions are depicted in Table 4-23.

9

9

4.2.2.1.2 Fuel Rod

The fuel rod consists of fuel pellets, cladding, fuel support components, and end caps. All fuel rods are internally pressurized with helium.

The pellets are manufactured by cold pressing enriched uranium dioxide powder into cylinders with edge
5 chamfers and dish at each end and then sintering to obtain the desired density and microstructure. After
sintering, the pellets are centerless ground to the required diametrical dimensions.

9 There are spring spacers located both above and below the pellet stack in the MK-B10D and MK-B10E
6 fuel assembly designs. Both springs are designed to accommodate maximum thermal expansion of the
fuel column without being deflected beyond solid height. The lower spring is much stiffer by design, so
the fuel column preload, thermal expansion and irradiation expansion principally compresses the upper
9 spring. The MK B-10F and higher fuel assembly designs do not contain a bottom plenum spring.

9

7 The fuel rods within an assembly may have differing enrichments radially. Axially, the fuel rods may be
7 of a constant enrichment, or they may be blanketed, which means that a portion of the top and bottom of
7 the fuel stack has a lower enrichment. The function, behavior, and analysis of fuel rods containing axial
7 blankets or radial zoning is the same as uniformly enriched fuel rods.

Fission gas generated in the fuel is released into pellet voids, the radial gap between the pellets and the
5 cladding, and into the plenum spring space. Fuel rod data are given in Table 4-2, and a typical fuel rod is
5 shown in Figure 4-4.

4.2.2.1.3 Spacer Grids

Spacer grids are constructed from strips which are slotted and fitted together in "egg crate" fashion. Each
grid has 32 strips, 16 perpendicular to 16, which form the 15 x 15 lattice. The square walls formed by the
interlaced strips provide support for the fuel rods in two perpendicular directions. Contact points on the
9 walls of each square opening are integrally punched in the strips.

4.2.2.1.4 Lower End Fittings

6 The lower end fitting positions the assembly in the lower grid rib section. During fabrication, the lower
6 ends of the fuel rods are seated on the grillage of the lower end fitting. Penetrations in the lower end
fitting are provided for attaching the control rod guide tubes and for access to the instrumentation tube
9 assembly. The lower end fittings are of an anti-straddle design which will prevent the fuel assembly from
9 being improperly seated on the lower grid assembly. The lower end fittings are removable to facilitate fuel
1 rod reconstitution from the bottom of the fuel assembly.

4.2.2.1.5 Upper End Fitting

The upper end fitting positions the upper end of the fuel assembly in the upper grid rib section and provides means for coupling the handling equipment. An identifying number on each upper end fitting provides positive identification.

Attached to the upper end fitting is a holddown spring. This spring provides a positive holddown margin to oppose hydraulic forces resulting from the flow of the primary coolant.

Penetrations in the upper end fitting grid are provided for the guide tubes.

9 The upper end fitting can be removed to perform fuel assembly reconstitution.

4.2.2.1.6 Guide Tubes

The Zircaloy guide tubes provide continuous guidance for the control rod assemblies when inserted in the
9 fuel assembly and provide the structural continuity for the fuel assembly. On the MK-B10D to
9 MK-B10F designs, the upper guide tube nut is held secure by a crimped locking cup. On the MK B-10G
9 and higher assemblies, each guide tube is designed to engage with a locking device on the upper end
6 fitting. Transverse location of the guide tubes is provided by the spacer grids. The guide tube hole size is
9 optimized so that more coolant flows alongside the fuel rods.

4.2.2.1.7 Instrumentation Tube Assembly

This assembly serves as a channel to guide, position, and contain the in-core instrumentation within the fuel assembly. The instrumentation probe is guided up through the lower end fitting to the desired core elevation. It is retained axially at the lower end fitting by a retainer sleeve.

4.2.2.1.8 Spacer Sleeves

The spacer sleeve fits around the instrument tube between spacer grids and prevents axial movement of the spacer grids during primary coolant flow through the fuel assembly.

4.2.3 DESIGN EVALUATION - FUEL SYSTEM DESIGN

This subsection contains a description of the fuel system design evaluation and is primarily a mechanical evaluation.

Nuclear design evaluation is contained within Section 4.3.3, "Nuclear Evaluation." Thermal hydraulic design evaluation is presented in Section 4.4.3, "Thermal and Hydraulic Evaluation."

4.2.3.1 Fuel Rod

The basis for the design of the fuel rod is discussed in Section 4.2.1, "Design Bases - Fuel System Design." Materials testing and actual operation in reactor service with Zircaloy cladding have demonstrated that Zircaloy-4 material has sufficient corrosion resistance and mechanical properties to maintain the integrity and serviceability required for design burnup.

7 If radiochemistry data indicates that there are fuel rods in the core with breached cladding, a campaign
7 may be scheduled for the next refueling outage to perform ultra-sonic testing of suspect fuel assemblies.

7 Fuel assemblies found with damaged or leaking fuel rods, can be reconstituted in order to replace
6 damaged rods. The typical replacement is a fuel rod that contains pellets of naturally enriched uranium
5 dioxide (UO_2). Aside from enrichment, this rod is similar in design and behavior as a standard fuel rod
5 and is analyzed using standard approved methods. If grid damage exists, solid filler rods made of stainless
5 steel or Zircaloy could be used as a replacement. A maximum of 10 such filler rods can be substituted
6 into a single fuel assembly. Fuel assemblies with severe structural damage or with failed pins that can not
6 be completely removed may be recaged or discharged. A recage operation entails transferring all of the
6 sound fuel rods from the damaged cage to a new fuel assembly cage. This new fuel assembly will function
6 the same as the assembly which it replaces. A safety evaluation (generic or core specific) is performed for
6 repaired fuel assemblies to ensure acceptable nuclear, mechanical, and thermal-hydraulic performance.

5 The NRC has approved Duke's reconstitution topical report (Reference 7). This report details the
5 methodology and guidelines Duke Power Company will use to support fuel assembly reconstitution with
5 filler rods. This methodology ensures acceptable nuclear, mechanical, and thermal-hydraulic performance
5 of reconstituted fuel assemblies.

4.2.3.1.1 Clad Stress and Strain

The following descriptions summarize the analyses of fuel rod cladding stress and strain for reload fuel
9 cycle designs, as performed by Duke. References 13, 14, 15 and 16 define the stress analysis methodology.
9 The strain methodology is defined in Reference 2.

1. Cladding Stress Analysis

   The cladding stresses for a new fuel cycle design are bounded by a conservative design analysis that uses Section III of the ASME Boiler and Pressure Vessel Code as a guide in classifying the stresses into various categories, assigning appropriate limits to these categories, and combining these stresses to determine stress intensity. Each new fuel cycle design is assessed to determine if reanalysis is required. The stress analysis is very conservative, and reanalysis should not be required for standard Mark B reloads.

7

7 • The static stress analysis uses design stress intensity limits on mechanical properties based on the
9 requirements of ASME code Article III-2000. The design stress intensity value for Zircaloy-4 is 2/3 of
9 the specified minimum non-irradiated yield strength at operating temperature.

In performing the stress analysis, all the loads are selected to represent the worst case loads and are then
combined. This represents a conservative approach since they cannot occur simultaneously. This insures
that the worst conditions for condition I and II events are satisfied. In addition, these input parameters
9 were chosen so that they conservatively envelope all Mk-B design conditions. The effects of corrosion are
9 accounted for in the stress analysis.

5 The primary membrane stresses result from pressure loading. Stresses resulting from creep ovalization are addressed in the creep collapse analysis.

8 Cladding tensile stresses are also addressed at cold (room temperature) conditions at BOL.

The minimum internal fuel rod pressure at HZP conditions is combined with the maximum design system pressure during a transient to simulate the maximum compressive pressure differential across the cladding. The worst case compressive pressure loads are combined with the other worst case loads. These are described below:

- The maximum grid loads will occur at BOL. During operation, the contact force will relax with time due to fuel rod creep-down and ovalization as well as grid spring relaxation.
- Conservative cladding dimensions with regard to stress.
- The maximum radial thermal stress will occur at the maximum rated power (power level corresponding to centerline fuel melt). This stress cannot physically occur at the same time the maximum pressure loading occurs, but is assumed to do so for conservatism. (Maximum cladding temperature gradient is combined with minimum pin pressure.)

6 • Ovality bending stresses are calculated at BOL conditions. A linear stress distribution is assumed.

- Flow induced vibration and differential fuel rod growth stresses are also addressed.

8 The resulting stresses meet the above criteria for both primary membrane and primary plus secondary stress intensities.

2. Cladding Strain Analysis

The limit on transient cladding strain is that uniform total strain of the cladding should not exceed 1.0%.

9 Duke performs a generic strain analysis using TACO3 to ensure that the strain criterion is not exceeded.
7 For each reload cycle, the generic strain power history is compared to the predicted power history in the
5 final fuel cycle design. If the generic power history is violated, cladding strain is re-analyzed using a new
5 bounding power history.

5 Maximum tensile elastic and plastic strain occurs at the clad inside diameter. Clad strain is calculated as:

$$\text{Clad Strain\%} = \frac{\text{Clad ID}_{\text{transient}} - \text{Clad ID}_{\text{transient beginning}}}{\text{Clad ID}_{\text{transient beginning}}} \text{x}100$$

5 where the clad ID prior to and after a power ramp (transient) is calculated by TACO3 using the
9 methodology explained in reference 2.

5 3. End of Life Pressures

5 An analysis is performed to demonstrate that the internal pin pressure does not exceed a value that would
5 cause: (1) the fuel-clad gap to increase due to outward cladding creep during steady-state operation and,
5 (2) extensive DNB propagation to occur. (Section 4.2.3.1.3, "Fuel Thermal Analysis.")

4.2.3.1.2 Cladding Collapse

Cladding creepdown under the influence of external (system) pressure is a phenomenon that must be evaluated during each reload fuel cycle design to ensure that the most limiting fuel rod does not exceed the cladding collapse exposure limit. Cladding creep is a function of neutron flux, cladding temperature, applied stress, cladding thickness, and initial ovality. Acceptability of a fuel cycle design is demonstrated by comparing the power histories of all the fuel assemblies against the generic assembly power history used in existing design analyses. Changes in pellet or cladding design are also evaluated against previously analyzed fuel rod geometries and a reanalysis is performed if necessary.

9 The CROV (References 1 and 11) computer code calculates ovality changes in the fuel rod cladding due to thermal and irradiation creep and is used to perform the fuel rod creep collapse analysis when required. CROV predicts the conditions necessary for collapse and the resultant time to collapse. Conservative inputs to the CROV cladding collapse analysis include the use of minimum cladding wall thickness and maximum initial ovality (conservatively assumed to be a uniform oval tube), as allowed by manufacturing specifications or batch specific as-built tolerance limits. Other conservatisms included are minimum backfill pressure and zero fission gas release. Internal pin pressure and cladding temperatures, input to
7 CROV (Reference 1), are calculated by TACO3 using a (conservative) generic radial power history, and a
9 typical axial flux shape.

The conservative fuel rod geometry and conservative power history are used to predict the number of
5 EFPH (or equivalent burnup) required for complete cladding collapse. To demonstrate acceptability, the
7 maximum cumulative residence time for the fuel is compared against the EFPH (or equivalent burnup) required for complete collapse. All operating cores must meet this criterion.

5

4.2.3.1.3 Fuel Thermal Analysis

9 Duke Power Company is performing its own reload design analyses per the approved methods in
9 Reference 2. Duke currently uses the TACO3 fuel pin performance codes. The following paragraphs
summarize the methods that are used by Duke in performing its Oconee reload fuel temperatures, end of
9 life pin pressure, and ECCS analysis interface criteria analyses.

1. Fuel Pin Pressure Analysis

5 The pin pressure limit is intended to preserve the fuel-clad heat transfer characteristics by preventing
5 clad liftoff. This limit provides reasonable assurance that: (1) excessive fuel temperatures, (2) excessive
5 internal gas pressures due to fission gas release, and (3) excessive cladding stresses and strains are
5 prevented.

5 The maximum allowable pin burnup is based on whichever of the following conditions occurs first:

5 A. <u>Maximum Internal Pin Pressure:</u> The fuel rod internal pressure is limited to a proprietary
5 value above the nominal system pressure.

5 B. <u>Clad Liftoff Limit:</u> Clad liftoff occurs when the clad's outward creep rate exceeds the
5 pellet's swelling rate. Clad liftoff is based on the ratio of cladding diametral strain rate
5 divided by the fuel diametral strain rate at each axial elevation. Fuel-clad liftoff occurs
5 when this ratio is ≥ 1.0 at any axial elevation where the local LHR is ≥ 3.0 kw/ft.

5 Duke performs a generic pin pressure analysis using the methodology described in Reference 2. For
5 each reload cycle, the generic power history is compared to the predicted power history in the final
5 fuel cycle design. If the generic power history is violated, the EOL pin pressure is re-calculated using
5 a new bounding power history.

5 2. Linear Heat Rate Capability

5 The fuel cannot exceed the temperature which would cause it to melt. Linear Heat Rate to Melt
5 (LHRTM) limits are used to determine core protection limits which ensure that fuel melting will not
5 occur. Duke performs a generic LHRTM analysis using the methodology described in Reference 2.

5 TACO3 reduces the best estimate fuel temperature by a proprietary value which is based on
5 comparison with measured data that inherently includes the effects of manufacturing variations, code
5 predictions, transient fission gas release, and cladding oxide formation.

5 For each reload cycle, the generic power history is compared to the predicted power history in the
5 final fuel cycle design. If the generic power history is violated, LHRTM is re-analyzed using a new
5 bounding power history.

3. ECCS Analysis Interface Criteria

Duke reviews each batch of fuel and the fuel cycle design for compatibility with the vendor's fuel rod thermal analysis inputs to the ECCS analysis. Review criteria have been developed by Duke and have been reviewed and approved by the vendor.

Should the fuel rod thermal analysis inputs for a specific cycle lie outside the vendor's generic analysis, Duke will reperform the fuel rod thermal analysis to ensure that the results remain bounded by the results of the vendor's generic analysis. In the unlikely event that the cycle specific thermal analysis results (fuel temperature and pin pressure) are more limiting than the vendor's generic analysis, either the fuel cycle design must be modified or the vendor must resolve the concern within the vendor's ECCS analysis. Responsibility for identification of incompatibility and resolution lies with Duke.

9 #### 4.2.3.1.4 Cladding Corrosion

9 The cladding oxide thickness, for the highest burnup rod in each sub-batch, is limited to 100 μm on a best
9 estimate basis. References 12 and 14 define the corrosion analysis methodology. If an assembly contains
9 a rod whose predicted oxide thickness is over 100 μm, it can be designated a lead corrosion assembly and
9 continue to operate. Corrosion measurements will be taken on these assemblies after they have been
9 discharged from the core. The total number of lead corrosion assemblies is limited to 8 per cycle. The
9 total number of lead corrosion and other demonstration assemblies is limited to 12 per cycle.

4.2.4 FUEL ASSEMBLY, CONTROL ROD ASSEMBLY, AND CONTROL ROD DRIVE MECHANICAL TESTS AND INSPECTION

To demonstrate the mechanical adequacy and safety of the fuel assembly, control rod assembly (CRA), and control rod drive, a number of functional tests have been performed.

4.2.4.1 Prototype Testing

A full-scale prototype fuel assembly, CRA, and control rod drive have been tested in the Control Rod Drive Line (CRDL) Facility located at the B&W Research Center, Alliance, Ohio (Reference 3). This full-sized loop is capable of simulating reactor environmental conditions of pressure, temperature, and coolant flow. To verify the mechanical design, operating compatibility, and characteristics of the entire
9 control rod drive fuel assembly system, the drive was stroked and tripped to duplicate the expected 20-year
9 operational life.

A portion of the testing was performed with maximum misalignment conditions. Equipment was available to record and verify data such as fuel assembly pressure drop, vibration characteristics, and hydraulic forces and to demonstrate control rod drive operation and verify scram times. All prototype components were examined periodically for signs of material fretting, wear, and vibration/ fatigue to insure that the mechanical design of the equipment met reactor operating requirements.

9 The Type C prototype drive mechanism used originally on Oconee 3 was tested at Diamond Power
Specialty Corporation, Lancaster, Ohio (Reference 3). This consisted of component testing, a 100 percent misalignment life test (equivalent to 20 year operation), and motor performance tests. Throughout these tests the drive components were examined for material fretting, wear and vibrational fatigue.

4.2.4.2 Model Testing

Many functional improvements have been incorporated in the design of the fuel assembly as a result of model tests. For example, the spacer grid to fuel rod contact area was fabricated to ten times reactor size and tested in a loop simulating the coolant flow Reynolds number of interest. Thus, visually, the shape of the fuel rod support areas was optimized with respect to minimizing the severity of flow vortices and pressure drop. A 9-rod (3 x 3) assembly using stainless steel spacer grid material has been tested at reactor conditions (640°F, 2,200 psi, 13 fps coolant flow) for 210 days. Two full sized canned fuel assemblies with stainless steel spacer grids have been tested at reactor conditions, one for 40 days and the other for 22 days. A prototype canless fuel assembly using Inconel 718 spacer grids has been tested for approximately 90 days, approximately half of that time at reactor conditions. The principal objectives of these tests were to evaluate fuel assembly and fuel rod vibration and/or fretting wear resulting from flow-induced vibration. Vibratory amplitudes have been found to be very small, and, with the exception of a few isolated instances which are attributed to pretest spacer grid damage, no unacceptable wear has been observed.

4.2.4.3 Component and/or Material Testing

4.2.4.3.1 Fuel Rod Cladding

5 Refer to Appendix B of Reference 5 for a detailed report of externally pressurized fuel rod creep collapse tests.

4.2.4.3.2 Fuel Assembly Structural Components

The structural characteristics of the fuel assemblies which are pertinent to loadings resulting from normal operation, handling, earthquake, and accident conditions are investigated experimentally in test facilities such as the CRDL Facility. Structural characteristics such as natural frequency and damping are determined at the relatively high (up to approximately 0.300 in.) amplitude of interest in the seismic and LOCA analyses. Natural frequencies and amplitudes resulting from flow-induced vibration are measured at various temperatures and flow velocities, up to reactor operating conditions.

4.2.4.3.3 B&W Fuel Surveillance Program

B&W conducts various test programs aimed at obtaining fundamental engineering data on fuel and control components for design, manufacturing, and licensing support. The extensive previous operating history and detailed fuel surveillance confirms the basic soundness of the B&W fuel design. The operation
8 of all B&W fuel will continue to be closely monitored using activities such as manufacturing reviews,
8 coolant chemistry monitoring, post irradiation examinations, etc. to ensure continued safe and reliable fuel
8 performance. Post irradiation examinations typically perform tests such as visual inspections, fuel
7 assembly growth measurements, spacer grid position determination, fuel assembly bow measurements,
8 shoulder gap measurement, water channel measurements, spring preload, verification of the quick
8 disconnect upper end fitting operation, and other non-destructive testing.

4.2.4.4 Control Rod Drive Tests and Inspection

4.2.4.4.1 Control Rod Drive Developmental Tests

The testing and development program for the roller nut drive has been completed. The prototype drive was tested at the B&W Research Center at Alliance, Ohio. Wear characteristics of critical components have indicated that material compatibility and structural design of these components would be adequate for the design life of the mechanism. The trip time for the mechanism as determined under test conditions of reactor temperature, pressure, and flow was well within the specification requirements.

The Type C prototype drive was tested at the Diamond Power Specialty Corporation, Lancaster, Ohio (Reference 3).

4.2.5 REFERENCES

5

5 1. T. Miles, D. Mitchell, G. Meyer, and L. Hassenpflug, Program to Determine In-Reactor Performance
5 of B&W Fuels - Cladding Creep Collapse, B&W, *BAW-10084P-A*, Rev. 3, Lynchburg, Va., July
5 1995.

7 2. DPC-NE-2008P-A, Duke Power Company Fuel Mechanical Reload Analysis Methodology using TACO3.

1 3. J. T. Williams, R. E. Harris, and John Ficor, Control Rod Drive Mechanism Test Program, Revision
1 3, B&W, *BAW-10029A*, Rev. 3, Lynchburg, Va., August 1976.

9 4. Deleted per 1999 Update.

9 5. Deleted per 1999 Update.

8 6. BWC Correlation of Critical Heat Flux, B&W, *BAW-10143P-A, Part 2,* Lynchburg, Va., April 1985.

5 7. DPC-NE-2007P-A, Duke Power Company Fuel Reconstitution Analysis Methodology, October
5 1995.

9 8. Deleted per 1999 Update.

9 9. Deleted per 1999 Update.

9 10. Deleted per 1999 Update.

9 11. Letter from H. N. Berkow (NRC) to M. S. Tuckman (DEC), Subject: Duke Power use of CROV
9 Computer Code, Dated: 19 June 1995.

9 12. Letter from D. LaBarge (NRC) to W. R. McCollum, Jr (DEC), Subject: Use of Framatome Cogema
9 Fuels Topical Report on High Burnup - Oconee Nuclear Station, Units 1, 2, and 3, (TAC Nos.
9 MA0405, MA0406, and MA0407), Dated: 1 March 1999.

9 13. Letter from M. S. Tuckman (DEC) to Document Control Desk (NRC), Subject: Duke Energy
9 Corporation's use of FCF's Extended Burnup Evaluation Topical Report BAW-10186P-A, Dated: 25
9 August 1999.

9 14. Letter from J. H. Taylor (FCF) to Document Control Desk (NRC), Subject: Application of
9 BAW-10186P-A, Extended Burnup Evaluation, Dated: 28 October 1997.

9 15. BAW-10186P-A, Extended Burnup Evaluation, June 1997.

9 16. BAW-10179-A, Safety Criteria and Methodology for Acceptable Cycle Reload Analyses, August 1993.

9 17. BAW-1781P-A, Mk-BZ Fuel Assembly Design Report, Apr. 1983.

9 18. BAW-10229P-A, Rev. 0, Mk-B11 Fuel Assembly Design Topical Report, Oct. 1999.

4.3 NUCLEAR DESIGN

The reactor core is designed to operate at 2568 MWt with sufficient nuclear design margins to accommodate transient operation without damage to the core. The core design characteristics are given in Table 4-1.

Core reactivity is controlled by control rod assemblies (CRA), soluble boron in the coolant, and burnable poison rod assemblies (BPRA). Sufficient CRA worth is available to shut down the reactor with at least a 1% $\Delta k/k$ subcritical margin in the hot condition at any time during the cycle with the most reactive CRA stuck in the fully withdrawn position. Equipment is provided to add soluble boron to the reactor coolant to ensure a similar shutdown capability when the reactor is cooled to ambient temperatures.

The reactivity worth of a CRA and the rate at which reactivity can be added are limited to ensure that credible reactivity accidents cannot cause a transient capable of damaging the RCS or causing significant fuel failure.

4.3.1 DESIGN BASES - NUCLEAR DESIGN

5 The core has been designed to the following nuclear limits and capabilities, all of which are intended to
5 preserve the integrity of the fuel assemblies:

5 1. The core will have sufficient reactivity to produce the design power level and lifetime without
5 exceeding the control capacity or shutdown margin.

7 2. Fuel assemblies have been designed for the maximum burnups shown in Table 4-2.

5 3. Power histories must be bounded by those assumed within generic mechanical and thermal hydraulic
5 (fuel assembly) analyses. If they are not bounded, acceptable reanalyses shall be performed.

5 4. The maximum feed fuel enrichment is constrained by the maximum allowed in the Technical
5 Specifications (Spent Fuel Pool storage requirements).

5 5. Values of important core safety parameters predicted for the cycle have been verified to be
5 conservative with respect to their values assumed in the Chapter 15 safety/accident (and any other
5 pertinent) analyses. If they are not conservative, acceptable reanalyses shall be performed.

5 Controlled reactivity insertion rates due to a single CRA group withdrawal shall be limited to a
5 maximum value assumed within the Chapter 15 Rod Withdrawal Accident at Rated Power, and
5 within the Chapter 15 Startup Accident. Controlled reactivity insertion rates due to soluble boron
5 removal shall be limited to a maximum value assumed within the Chapter 15 Moderator Dilution
5 Accident.

5 The power Doppler and moderator temperature coefficients at power will be negative. However, as
5 described within Chapter 15, the control system is capable of compensating for reactivity changes
5 resulting from either positive or negative nuclear coefficients.

6. Reasonable and permissive reactor control and maneuvering procedures during nominal operation and
7 during transients will not produce unacceptable peak-to-average power distributions. This, along with
5 criteria 7 and 8, below, preserves the LOCA linear heat rate, linear heat rate to melt (LHRTM), and
5 DNBR limits.

5 7. Part length axial power shaping rods (APSRs) are to be utilized to allow the shaping of power axially
5 in the core, thereby thwarting any tendency towards axial instability resulting from a redistribution of
5 xenon.

5 To preclude the possiblity of azimuthal instability resulting from a redistribution of xenon, the highest
5 moderator temperature coefficient assumed within the Chapter 15 safety/accident analyses must be
5 bounded by the threshold listed within Table 4-7.

5 8. Technical Specification limits of specified operating parameters (quadrant power tilt, power imbalance,
5 and control rod insertion), and on reactor protective system trip setpoints (power imbalance) after
5 allowance for appropriate measurement tolerances should have adequate margin from design limits of
5 these parameters during operational conditions throughout the cycle such that sufficient operating
5 flexibility is retained for the fuel cycle.

4.3.2 DESCRIPTION - NUCLEAR DESIGN

A summary of the nuclear characteristics of the core is given in Table 4-3.

4.3.2.1 Excess Reactivity

The Oconee reactor cores are designed with sufficient excess reactivity to yield the desired cycle length. This excess reactivity is controlled by soluble boron, burnable poison rod assemblies (BPRA), and control rod assemblies (CRA).

5 Generally, the nuclear designer makes an engineering trade-off between soluble boron and burnable
poison rods to assure that the BOC moderator coefficient for power levels above 95 percent Hot Full
Power (HFP) is nonpositive. Table 4-4 shows a typical eighteen month fuel cycle's excess reactivity at
various conditions.

Table 4-5 shows the k-effective calculated for a single fuel assembly. The minimum critical mass, with
and without xenon and samarium poisoning, may be specified as a single assembly or as multiple
assemblies in various geometric arrays. The unit fuel assembly has been investigated for comparative
7 purposes. A single cold, clean assembly containing an enrichment of 3.5 weight per cent is subcritical.
Two assemblies side-by-side are supercritical under these conditions.

4.3.2.2 Reactivity Control

The excess reactivity is controlled by a combination of soluble boron, lumped burnable poison, and control rods. Long term decreases in reactivity caused by fuel burnup are offset by decreases in soluble boron concentration and decreases in burnable poison worth. Short term reactivity effects are controlled by changes in control rod position.

<u>Soluble Boron</u>

Figure 4-5 illustrates a typical variation of soluble boron versus cycle length of an eighteen month fuel cycle. The change in boron concentration accounts for depletion of the fuel and is also a function of the BPRA loading and burnout.

<u>Burnable Poison Rod Assemblies (BPRAs)</u>

Figure 4-6 shows a typical burnable poison loading and enrichment scheme for an eighteen month fuel cycle. The BPRAs burnout as the fuel depletes and at end of cycle have a small residual reactivity effect caused by structural materials and water displacement effects.

The BPRA loadings and placement are chosen to shape radial power peaks and to decrease initial soluble
6 boron concentration to a level where the BOC moderator temperature coefficient is non-positive above

6 95% full power. Since the BPRA assemblies are located in the control rod guide tubes, they cannot be
placed in rodded locations. In addition, they will usually be in fresh fuel assemblies. See Section 4.5.2.4,
"Burnable Poison Rod Assembly (BPRA)" for a physical description of the BPRAs. See the appropriate
5 reload design change report for actual BPRA loadings for any particular cycle.

Control Rod Assemblies

Oconee has 61 full length control rods assigned to seven control rod groups (1 to 7). Groups 1 to 4 are
designated safety banks and are maintained out of the core above HZP. Groups 5 to 7 are designated
6 control banks and may be inserted to pre-established limits shown in the Core Operating Limits Report
between HZP and HFP.

A typical control rod pattern is shown in Figure 4-7. The groupings of control rods into the various rod
5 groups can vary with reload cycle and reference to the appropriate reload design change report should be
made for the particular pattern being used for a particular cycle. In addition to being able to shut the
reactor down, full length control rods are used to control reactivity changes caused by power level
changes, transient xenon, and small periodic boron dilution changes.

Oconee has 8 Axial Power Shaping Rods (APSRs) which are always assigned to Group 8. These rods do not insert upon reactor trip and are used for axial power shaping and can be used to damp axial xenon oscillations.

4.3.2.3 Reactivity Shutdown Analysis

5 The ability to shut down the core from any operating condition by 1% $\Delta\rho$ is a Technical Specification
requirement. This is accomplished by analytical calculations during the reload design and rod index limits
5 are set such that at least a 1% $\Delta\rho$ shutdown margin is available for a trip from any allowable operating
condition.

6 Table 4-6 illustrates a shutdown margin calculation for a sample Oconee fuel cycle. Conservatisms
include a worth reduction penalty for control rod burnup and a 10 percent rod worth uncertainty. The
5 flux redistribution effect is included if the power deficit was calculated with a two-dimensional code. This
5 item does not need to be shown in a shutdown margin table if a three-dimensional calculation of power
deficit was performed.

A detailed discussion of the calculation of the remaining parameters in Table 4-6 can be found in Reference 1 and Reference 2.

5 For the shutdown margin calculation for a particular reload cycle refer to the bases behind the appropriate
5 reload design change report.

4.3.2.4 Reactivity Coefficients

Reactivity coefficients form the basis for studies involving normal and abnormal reactor operating conditions. These coefficients have been investigated as part of the analysis of this core and are described below as to function and overall range of values.

4.3.2.4.1 Doppler Coefficient

The Doppler coefficient reflects the change in reactivity as a function of fuel temperature. The Doppler coefficient of reactivity is due primarily to Doppler broadening of the U-238 resonances with increasing fuel temperature. A rise in fuel temperature results in an increase in the effective absorption cross section

5 of the fuel and a corresponding reduction in neutron production. A typical range for the Doppler
5 coefficient under operating conditions would be -1.1 x 10^{-5} to -1.7 x 10^{-5} ($\Delta\rho$)/deg F.

4.3.2.4.2 Moderator Void Coefficient

The moderator void coefficient relates the change in neutron multiplication to the presence of voids in the moderator. The expected range for the void coefficient is shown in Figure 4-8.

4.3.2.4.3 Moderator Pressure Coefficient

The moderator pressure coefficient relates the change in moderator density, resulting from a reactor coolant pressure change, to the corresponding effect on neutron production. This coefficient is opposite in sign and considerably smaller when compared to the moderator temperature coefficient. A typical
5 range of pressure coefficients over a life cycle would be -1.4 x 10^{-7} to +3 x 10^{-6} ($\Delta\rho$)/psi.

4.3.2.4.4 Moderator Temperature Coefficient

The moderator temperature coefficient relates a change in neutron multiplication to the change in reactor coolant temperature. Reactors using soluble boron as a reactivity control have a less negative moderator temperature coefficient than do cores controlled solely by movable or fixed CRA. The major temperature effect on the coolant is a change in density. An increasing coolant temperature produces a decrease in water density and an equal percentage reduction in boron concentration. The boron concentration change results in a positive reactivity component by reducing the absorption in the coolant. The magnitude of this component is proportional to the total reactivity held by soluble boron. Distributed poisons (burnable poison rods or inserted control rods) have a negative effect on the moderator coefficient for a system with 1200 ppm boron and no rods inserted. Depending on the core size, core loading, and power density, a plant may or may not require additional distributed poisons to yield the appropriate moderator
5 temperature coefficient as determined by the safety analysis and the stability analysis of the core. An
5 example of this, as pertaining to the first cycle, is illustrated in Table 4-7.

Items 4d and 6 in Table 4-7 above reflect three dimensional calculations using thermal feedback. These coefficients are more negative than the two-dimensional isothermal values previously calculated and shown. It is seen from comparison (Table 4-7, Table 4-8, Table 4-9) that three-dimensional spatially distributed effects are important in the determination of reactivity coefficients.

The three-dimensional PDQ07 calculation with thermal feedback was also used to calculate for Oconee 1 Cycle 1 the change in spatially dependent moderator coefficient for changes in inlet, outlet, and core average moderator temperature (°F_m), as shown in Table 4-8.

7 The Oconee reactors operate above 15% of rated power on a constant core average moderator temperature with both inlet and outlet temperature changing with power level. The core average moderator temperature as seen by the control system is defined to be

$$T_m = \frac{T_{in} + T_{out}}{2}$$

5 The BOL distributed temperature moderator coefficients for different reactor power levels are presented in
5 Table 4-9 for Oconee 1, Cycle 1, and for a typical reload cycle with three dimensional codes PDQ07 and
5 SIMULATE-3P, respectively, and both with thermal feedback. These coefficients were found by
5 changing both inlet and outlet temperatures. Criticality in each case was attained by appropriate control
5 rod insertion for Oconee 1, Cycle 1, and by boron for the typical reload cycle.

The moderator temperature coefficient was also calculated for the equilibrium xenon condition at the
5 beginning of the fuel cycle. The calculation assumed 2.1% $\Delta\rho$ in control rods for Oconee 1, Cycle 1;
5 boron search was used for a typical reload cycle. The 100% power moderator coefficient varied in the
manner shown in Table 4-10.

The EOL coefficient was calculated for a change in both the inlet and outlet temperatures with a boron concentration of 17 ppm. The coefficient for 100% power was found to be:

$$\alpha_m = -2.8 \times 10^{-4} \frac{\Delta\rho}{°F_m}$$

This, then, is the "rods out" moderator coefficient at the end of the first fuel cycle for Oconee 1, Cycle 1.

5 The coefficients reported in Table 4-8 and Table 4-9 (Oconee 1, Cycle 1) are for a core containing 2.1
percent $\Delta\rho$ in control rods. A "rods out" calculation for the beginning of life moderator conditions in
5 Item 2, Table 4-8 was performed as a basis for comparison and the result was

$$\alpha_m = +0.52 \times 10^{-4} \Delta\rho/°F_m$$

An examination of the data in Table 4-7 shows that the limiting factor on a moderator coefficient is the
5 value used during Oconee 1, Cycle 1 safety analysis, i.e., $+0.9 \times 10^{-4}\ \Delta\rho/°F_m$. The margin between this
value and the nominal calculated value of $+0.27 \times 10^{-4}\ \Delta\rho/°F_m$ is considered adequate to cover
uncertainties.

4.3.2.4.5 Power Coefficient

The power coefficient, αp, is the fractional change in neutron multiplication per unit change in core power level. A number of factors contribute to αp, but only the moderator temperature coefficient and the Doppler coefficient contributions are significant. The power coefficient can be written as:

$$\alpha_p = \alpha_m \frac{\partial T_m}{\partial P} + \alpha_f \frac{\partial T_f}{\partial P}$$

where:

α_m = moderator temperature coefficient

α_f = fuel Doppler coefficient

$\frac{\partial T_m}{\partial P}, \frac{\partial T_f}{\partial P}$ = change in moderator and fuel temperature per unit change in core power.

5 Power coefficients were calculated for Oconee 1, Cycle 1 and for a typical reload cycle at BOL (time zero)
5 at various power levels. For Oconee 1, Cycle 1, a boron concentration of 1200 ppm was used for all
5 power levels, and criticality was achieved with control rods. For a typical reload cycle, boron search was
5 used for all power levels, and criticality was achieved by boron. The three-dimensional codes PDQ07 and
5 SIMULATE-3P, both with thermal feedback, were used to include the effects of spatially distributed fuel
5 and moderator temperatures.

5 The results are presented in Table 4-11.

4.3.2.4.6 pH Coefficient

Currently, there is no definite correlation which will permit prediction of pH reactivity effects. Some of the parameters needing correlation are the effects relating pH reactivity change for various operating reactors, pH effects versus reactor operating time at power, and changes in effects with varying clad, temperature, and water chemistry. Yankee, Saxton, and Indian Power Station 1 have experienced reactivity changes at the time of pH changes, but there is no clear-cut evidence that pH is the direct reactivity influencing variable without considering other items such as clad materials, fuel assembly crud deposition, system average temperature, and prior system water chemistry.

The pH characteristic of this design is shown below in Table 4-12 where the cold values are measured and the hot values are calculated.

Saxton experiments (Reference 3) have indicated a pH reactivity effect of 0.0016 $\Delta\rho/\Delta$pH unit change with and without local boiling in the core. Considering system makeup rate of 35,000 lb/h and the core in the hot condition with 1,200 ppm boron in the coolant, the corresponding changes in pH are 0.02 pH units per hour for boron dilution and 0.05 pH units per hour for ^{7}Li dilution (starting with 0.5 ppm ^{7}Li). Applying the pH worth value quoted above from Saxton, the total reactivity insertion rate for the hot condition is 3.1 x 10^{-8} $\Delta\rho$/sec. This insertion rate or reactivity can be easily compensated by the operator
6 or the Integrated Control System.

4.3.2.5 Reactivity Insertion Rates

6 Figure 4-10 displays a typical integrated rod worth of three overlapping rod banks as a function of
5 distance withdrawn. The indicated groups are those used in the core during power operation. Using an
5 assumed nominal of 1.2% $\Delta\rho$ CRA groups and an assumed 30 in./min CRA drive speed in conjunction
5 with the reactivity response given in Figure 4-10 yields a maximum reactivity insertion rate of 1.09 x 10^{-4}
5 ($\Delta\rho$)/sec. The maximum reactivity insertion rate for soluble boron removal, using an assumed boron
5 dilution rate of 500 GPM, is 0.16 x 10^{-4} ($\Delta\rho$)/sec.

4.3.2.6 Power Decay Curves

Figure 4-11 displays the beginning-of-life power decay curves for the CRA worths corresponding to the 1 percent hot shutdown margin with and without a stuck rod. The power decay is initiated by the trip of the CRA with a 300 msec delay from initiation to start of CRA motion. The time required for insertion of a CRA 2/3 of the distance into the core is 1.4 sec.

4.3.3 NUCLEAR EVALUATION

5 The nuclear evaluation for a fuel cycle design is composed of the preliminary fuel cycle design, the final
5 fuel cycle design, safety analysis physics parameters, maneuvering analysis, core operating limits (Technical
6 Specifications and Core Operating Limits Report) calculation, final core loading map calculation, and core
5 monitoring parameters calculation.

5 The preliminary fuel cycle design determines the number and enrichment of the fresh fuel to be inserted
5 for a given cycle.

5 The final fuel cycle design uses the models discussed in Section 4.3.3.1, "Analytical Models" to optimize
5 the placement of fresh and burned fuel assemblies, control rod groupings, and BPRA (if any) to result in
5 an acceptable fuel design. It must meet the following current design criteria with appropriate reductions
5 to account for calculational uncertainties:

5 1. Operate to the scheduled end-of-cycle (EOC) plus ten days, with a minimum boron (typically 10
5 ppmb) remaining at the EOC midpoint.

5 2. The U235 fuel enrichment must be bounded by that listed within the Technical Specifications (Spent
5 Fuel Pool storage requirements).

5 3. Maximum pin burnup must be bounded by the appropriate limit for a fuel type.

5 4. Maximum assembly average burnup must be bounded by the appropriate limit for a fuel type.

7 5. The power histories must be bounded by those used in generic analyses, or provide acceptable results
7 when specifically analyzed.

5 6. For the current bypass flow assumptions, the typical number of 44 BPRAs gives sufficient margin.

5 During the safety analysis physics parameters, a number of physics parameters are calculated and are
5 verified as conservative with respect to those assumed within the Chapter 15 safety/accident analyses.
5 These include, but are not limited to, the following:

5 1. Moderator temperature coefficient

5 2. Doppler coefficient

5 3. Ejected rod worth

5 4. Dropped rod worth

5 5. Total/maximum CRA group worth

5 6. Kinetics parameters

5 7. Shutdown margin

5 8. Maximum reactivity insertion rates (due to controlled rod withdrawal and boron dilution)

5 9. Differential boron worth

5 10. Boron concentrations

5 The purpose of a maneuvering analysis is to generate three dimensional power distributions, rod positions,
5 and imbalances for a variety of reasonable and permissive rod positions, xenon distributions, and power
5 levels. The maneuvering analysis can be described as four discrete phases. The first is the nominal fuel
5 cycle depletion performed at a nominal rod index (typically, rod index = 292 and APSRs at 35%
5 withdrawn) to establish a fuel depletion history. The second is the power maneuver performed at BOC (4
5 EPFD), at EOC (with appropriate adjustments to ensure critical conditions), and at least one other point
5 in between; APSRs are positioned as necessary to maintain xenon control and to maintain predetermined
5 imbalance limits. The third is to perform control rod and APSR scans at the most severe times of the
5 power maneuver. The fourth step is to perform selected control rod and APSR scans at various nominal
5 depletion steps. Each of these phases involves running multiple three dimensional cases and generation of
5 three dimensional power distributions, rod positions, and imbalances for each case. The data is processed
5 by utility codes to calculate margins to LHRTM, DNBR, and LOCA limiting criteria, and to produce
7 'fly-speck' plots. Application of appropriate calculational conservatisms are described within References 2,
7 4, and 18. Note that the derivations of the LHRTM, DNBR, and LOCA limiting criteria have been
5 bounded by limiting power distribution listed within Table 4-1.

5 In addition, the initial rod positions assumed within the following safety parameters must be bounded by
5 the rod insertion limits determined during the maneuvering analysis:

5 1. Shutdown margin at HZP, BOC to EOC $\geq$ 1.0% $\Delta\rho$ (with the most reactive CRA stuck in the fully
5 withdrawn position).

5 2. Maximum ejected rod worth at HZP, NoXe, BOC and EOC, as bounded by that assumed within the
5 Chapter 15 Rod Ejection Accident.

5 3. Maximum ejected rod worth at HFP, EqXe, BOC and EOC, as bounded by that assumed within the
5 Chapter 15 Rod Ejection Accident.

5 4. Maximim dropped rod worth at HFP, NoXe, EOC, as bounded by that assumed within the Chapter
5 15 Control Rod Misalignment Accident.

5 5. Maximum dropped rod worth HFP, EqXe, EOC, as bounded by that assumed within the Chapter 15
5 Control Rod Misalignment Accident.

5 During core operating limits calculation, data from 'fly-speck' plots generated during the maneuvering
5 analysis are used to set limits on operational alarm setpoints (quadrant power tilt, control rod insertion,
5 power-imbalance), and reactor protective system trip setpoints (power-imbalance). In addition, limits on
5 control rod insertion based on the shutdown margin and required boron concentrations within the control
5 system equipment are developed (or retrieved from appropriate sources). These limits are chosen such
5 that sufficient operating flexibility is retained for the fuel cycle, while maintaining sufficient margin to
5 design and safety criteria. Thse limits are set according to the allowances for appropriate measurement
5 tolerances and uncertainties, which include, but are not limited to the following:

5 1. In-core detector system (observability and variability) and in-core monitoring software uncertainties

5 2. Out-of-core to In-core calibration/correlation uncertainty

5 3. Control rod position uncertainties

5 4. Flux-flow ratio adjustment

5 5. Reactor protective system hardware uncertainties

5 6. Boron concentration and volume uncertainties

5 The following Technical Specification limit is presumed as being met by the startup physics test program
5 criteria for moderator temperature coefficient (which must be less than $+0.5 \times 10^{-4}\ \Delta\rho$/deg F):

5 1. Moderator temperature coefficient $\leq$ 0.0 at > 95% hot full power.

5 Table 4-9 for a typical reload design, generated with SIMULATE-3P, shows that this presumption is
5 valid. The moderator temperature coefficient (MTC), when changing conditions from BOC, HZP, NoXe,
5 and ARO to BOC, 95%fp, NoXe, and ARO, becomes negative; note that the 95% no xenon condition is
5 conservative. Given the change, and startup physics test program criteria of $+0.5 \times 10^{-4}\ \Delta\rho$/deg F, the
5 MTC at 95%fp is negative (approximately $-0.20 \times 10^{-4}\ \Delta\rho$/deg F).

5 During final core loading map calculation, placement of fuel assemblies and related core components in a
5 reload core are determined. Special considerations such as even distributions of fresh fuel loadings and
5 BPRA poison loadings are taken into account to minimize the possibility of an asymmetric or tilted core
5 which would perturb the assumptions and predictions made during the fuel cycle design process.

5 During core monitoring parameters calculation, certain physics parameters are calculated to enable an
5 orderly and safe startup of the cycle, to perform the startup physics test program, and to perform
5 corefollow calculations. Other physics parameters are used to update the in-core monitoring software
5 residing within the plant process computer. The in-core monitoring software monitors the quadrant tilt,
5 power imbalance, and rod positions, and actuates alarms if these parameters violate the operational limits.
5 Periodic calculations are also done to verify the existence of the 1.0% $\Delta\rho$ shutdown margin, and to check
5 predictions versus measured data. As such, monitoring of core performance during cycle operation
5 confirms the validity of predictions and ensures that design and safety criteria are satisfied.

5 The analytical models and their applications are discussed in this section as well as core instabilities
5 associated with xenon oscillations.

4.3.3.1 Analytical Models

5 Reactor design calculations are made using a large number of computer codes. The following section
5 describes the major analytical models employed by DUKE in the design of Oconee reload cores.
Table 4-13 specifies the cycle of each unit when these methodologies were first applied. The
5 methodology used in a particular reload design is stated in the bases behind appropriate reload design
5 change report.

5 **4.3.3.1.1 CASMO-3/SIMULATE-3P-Based Methodology**

5 The CASMO-3/SIMULATE-3P-based calculational methods for nuclear design have been reviewed and
5 approved by the NRC in Reference18 . This methodology was first applied during the reload design
5 analysis of Unit 1 Cycle 16.

Verification to Measured Data

5 The verification of the CASMO-3/SIMULATE-3P-based methods for nuclear design is documented in
5 Reference 18.

5

4.3.3.1.2 Control of Power Distributions

The reactors are designed to permit power maneuvering on control rods. Various calculations are
5 performed during the maneuvering analysis to develop operational power-imbalance, RPS power
imbalance, and rod insertion limits. These three-dimensional calculations account for effects of rod insertion, xenon distribution, and power level on the power distribution. A more detailed discussion of these calculations can be found in Reference 2.

During startup testing an out-of-core detector correlation test is performed to calibrate the imbalance as
9 measured by the out-of-core detectors (NI-5, -6, -7, and -8) to that measured by in-core detectors.
Uncertainties in the measurement of imbalance and power level are accounted for to assure that the reactor trips before any DNBR or fuel melt limit is reached.

9 The out-of-core neutron flux detectors, except NI-9 on Unit 1 which is a three section detector, each
9 consist functionally of two nominally 70 inch sections of uncompensated ion chambers placed opposite
the top and bottom halves of the core. Comparison of the signals from the two detectors gives an indication of the core axial offset or imbalance. This imbalance signal (top core power minus bottom core power) is monitored in the control room. When an imbalance is indicated, the operator will move the APSR's in the direction of the imbalance to reduce the axial offset, i.e.,

positive offset - move APSR's toward top;

negative offset - move APSR's toward bottom.

The integrated control system will automatically compensate for reactivity changes and consequent power swings caused by the part length control rod movement.

4.3.3.1.3 Nuclear Design Uncertainty (Reliability) Factors

In various calculations additional conservatism is applied to the calculated parameters. The factors
5 sometimes are analysis dependent and are tabulated in Reference 18.

4.3.3.1.4 Power Maldistributions

Misaligned Control Rods

The reactor has a control function to protect against a rod out of step with its group. The position of
5 each rod is compared to the average of the group. If an assymetric fault is detected at power levels greater
than 60% of rated power, a rod withdrawal inhibit is activated and the Integrated Control System (ICS)
7 runs the plant back to 55% of rated power. If a rod is dropped, the Integrated Control System (ICS)
cannot maintain core power to match demand by withdrawal of other rods, and the plant is run back to
7 less than 60 percent of rated power. Several cases were also analyzed for BOL for Oconee 1, Cycle 1,
with single dropped rods. The calculations were performed with half-core X-Y geometry in PDQ07 at
rated power without thermal feedback. The results are given in Figure 4-13.

The maximum radial-local power peak is 1.92. The original FSAR design limit is a 2.1 radial-local at rated power with a 1.5 cosine yielding a 1.3 DNBR based on the W-3 correlation. At a 114 percent overpower condition the design limit can also be expressed as a 1.9 radial-local with a 1.5 cosine yielding a 1.3 DNBR. The dropped rods illustrated in Figure 4-13 do not represent violations of the thermal limits of the design.

5 Several dropped rod cases run with SIMULATE-3P and current core design models indicate less severe
5 radial-local power peaks; primarily because the current cores operate in a feed and bleed mode and the
5 Oconee 1, Cycle 1 core was a rodded core. It should also be noted that dropped rod accidents are
5 analyzed within Chapter 15 (Control Rod Misalignment Accident), and that this analysis showed that the
5 consequence of a dropped rod is minimal such that the core and RCS pressure boundary are preserved,
5 even when the worst assumed safety parameters are used and no credit is taken for ICS action.

5 Radial power tilts can be detected with the out-of-core and in-core instrumentation, and the operator has
5 the flexibility to monitor the upper or the lower out-of-core detectors to determine the X-Y power
symmetry condition at any time.

5 For the assumed case where one CRA is left out of the core while the remainder of the group is fully
5 inserted, this condition would not occur except with regard to rod "swaps." Since rod swaps are
performed at reduced power, and since the operator can monitor the out-of-core detectors, an X-Y tilt
resulting from such a condition could be detected and appropriate action taken before the approach to
thermal limits could be realized.

The APSR drives are also equipped with the position monitors and the alarm function for a rod out of step with the group average. These drives, however, do not permit rod drops. With the power removed from the rod drive windings of the APSR, the roller nut will not disengage and the rod remains in its position. Since the APSR's are made of low-absorbing (gray) material, it is not likely that thermal limits will be exceeded if one of the rods were stuck and the rest of the group were moved.

Azimuthal Xenon Oscillations

The Oconee reactors are predicted to have a substantial margin to threshold for azimuthal xenon oscillations. Therefore, this mode is not considered to be likely to produce a power peaking problem.

Fuel Misloading

7 Assurance of the proper loading of fuel rods into assemblies is provided through fuel vendor loading
7 controls and procedures. Fuel rods are mechanically identified so that traceability and accountability of
7 each rod exists. The manufacturing process relies on administrative procedures and quality control
7 independent verification to assure that fuel rods are placed in the proper assembly location.

Gross fuel assembly misplacement in the core is prevented by administrative core loading procedures and the prominent display of fuel assembly identification markings on the upper end fitting of each assembly. After the core is loaded, an independent check is performed to verify the core loading.

During startup physics testing, misloaded fuel may be discovered by unexpected quadrant power tilt or differences between predicted and measured power distributions.

4.3.3.2 Xenon Stability Analysis and Control

5 Modal and digital analysis of the Oconee 1, Cycle 1 core indicated that a tendency toward xenon
instability in the axial mode would exist for a given combination of events (BOL, rodded core).
Therefore, eight part-length Axial Power Shaping Rod Assemblies (APSRA) have been included in the
design. They will be positioned during operation to maintain an acceptable distribution of power for any
particular operating condition in the core, thereby reducing the tendency for axial oscillations. Similar
5 analysis which was performed on the Oconee 2, Cycle 1 core indicated that it would be stable with regard
5 to axial oscillations. Oconee 3, Cycle 1 was assumed to have characteristics similar to those of Oconee 1.

The azimuthal stability of the cores are dependent upon core loadings, power densities, and moderator
temperature coefficients. In any event, the cores will not be susceptible to diverging azimuthal
oscillations. If the loadings and power densities are low enough, the core will be inherently stable
5 (Oconee 1, Cycle 1). If not, then burnable poison is added in the amount necessary to provide a
5 moderator temperature coefficient that will result in azimuthal stability (Oconee 2&3, Cycle 1). A detailed
description of the xenon analyses performed on Unit 1 and 2 cores may be found in Reference 5.

The first two parts of Reference 5, which considered modal and one-dimensional digital analyses, pointed out the need for multi-dimensional calculations regarding xenon stability. The reactor core designs for Oconee Units 1 and 2, Cycle 1, have been analyzed in three dimensions with thermal feedback. For the Unit 1 operating core at beginning of life, the predicted azimuthal stability index is -0.07 hr^{-1}. Using modal analysis with the three-dimensional results shows that the shape factor must be approximately 50 percent flat for the power coefficient of -5.05×10^{-6} as calculated by previously described methods. Since the curves in Part 1 of Reference 5 were generated for a power coefficient of -3.92×10^{-6} $\Delta\rho$/MWt, it was necessary to generate two new curves for azimuthal stability. These curves are shown in Figure 4-14 and Figure 4-15. From Figure 4-14 the threshold (i.e., stability index = 0) moderator coefficient for the nominal case is approximately $+3 \times 10^{-4}$ $\Delta\rho/°F_m$. Including compounded errors from Figure 4-15, the threshold moderator coefficient is approximately $+1 \times 10^{-4}$ $\Delta\rho/°F$. Using the least favorable predictions of the Doppler and moderator coefficients, a stability index of -0.067 hr^{-1} is obtained. This corresponds to a power coefficient of -4.73×10^{-6} $\Delta\rho$/MWt. For the Unit 2 operating core at beginning of life (96 FPH), the predicted azimuthal stability index is -.085 hr^{-1}. Again, using modal analysis combined with three-dimensional results shows the shape factor to be approximately 40 percent flat for the calculated power coefficient of -4.67×10^{-6} $\Delta\rho$/MWt. Azimuthal stability curves for the nominal and compounded error cases are shown in Figure 4-16 and Figure 4-17 respectively. From Figure 4-16 the nominal threshold moderator coefficient extrapolates to approximately $+5 \times 10^{-4}$ $\Delta\rho/°F_m$. When compounded errors are considered as in Figure 4-17 the threshold moderator coefficient is approximately $+2.5 \times 10^{-4}$ $\Delta\rho/°F_m$.

5 This analysis is considered to be valid and bounding for the current core designs for the following reasons:

5 1. The minimum moderator temperature coefficient (MTC) threshold value, as listed within Table 4-7,
5 is $+1.0 \times 10^{-4}$ $\Delta\rho$/degF. The most positive moderator temperature coefficient assumed within
5 Chapter 15 safety/accident analysis is less than the threshold value.

5 2. There is considerable margin for a BPRA core (i.e., Oconee 2, Cycle 1 within Reference 5) between
5 the Table 4-7 threshold MTC and the calculated threshold MTC, even when compounded errors are
5 taken into account.

5 3. Current nuclear design bases require that the MTC and the power Doppler coefficient be negative at
5 power. As such, any azimuthal oscillations within current cores are self-damping by virtue of
5 reactivity feedback effects.

Operating procedures are in effect which allow the reactor operator to damp out any axial xenon oscillation if it should occur.

4.3.4 NUCLEAR TESTS AND INSPECTIONS

Nuclear Testing and Inspection can be divided into two areas:

1. Initial Core
2. Startup Testing for Reload Cores.

4.3.4.1 Initial Core Testing

The startup testing performed on Oconee 1, 2, and 3 initial cores was an extensive program to verify both calculational methods and proper behavior of the core. The results of this testing was reported in References 6, 7, and 8.

4.3.4.2 Zero Power, Power Escalation, and Power Testing For Reload Cores

3 The Startup Physics Test Program for Oconee Nuclear Station, or OSPTP, is structured to provide
3 assurance that the installed reactor core following each reload conforms to the design core. This
3 document provides the minimum test program which will be conducted on each Oconee unit. Additional
3 tests may be performed during a specific startup test program as conditions warrant. However, in all
3 cases, the following tests will be performed:

3 1. Pre-critical Test Phase

3 a. Control Rod Drop Time

3 2. Zero Power Physics Test Phase

3 a. Critical Boron Concentration

3 b. Moderator Temperature Coefficient

3 c. Control Rod Worth

3 3. Power Escalation Test Phase

3 a. Low Power Testing (5-30% FP)

3 b. Intermediate Power Testing (40-75% FP)

3 c. Full Power Testing (90-100% FP)

3 In addition to the above tests, which comprise the basic Startup Physics Test Program, a separate test, the
8 Reactivity Anomaly at Full Power is performed during steady-state operation pursuant to Technical
8 Specification SR 3.1.2.1, "Reactivity Balance". This procedure is used to verify that the measured
3 "all-rods-out" (ARO) hot full power (FP) critical boron concentration is in agreement with the predicted
3 value. The test conditions, procedure descriptions, acceptance criteria, and review requirements for each
3 of the above are provided in this document.

3 For all these tests, specific acceptance criteria are provided (see OSPTP Summary). Upon completion of
3 each test, the results are reviewed by a designated individual. If the results meet the specific acceptance
3 criteria, then the test is considered to be satisfactorily completed. However, if the results exceed the
3 specific acceptance criteria, an extensive review is performed by cognizant engineers from within Duke
3 Power Company or from outside organizations, as appropriate, to identify and correct the cause of the
3 discrepancy. Continuation of the test program, including any power escalations, will be dependent upon
3 satisfactory resolution of any unacceptable test result. Representatives from Oconee Nuclear Station
3 Nuclear Engineering and Regulatory Compliance, and General Office Nuclear Engineering will approve
3 actions under the conditions stated for each test.

3 The current Startup Physics Test Program for Oconee Nuclear Station was submitted by References 9 and
3 13, approved by Reference 14, and subsequently modified by References 10 and 16, and approved by
3 References 15 and 17.

3 4.3.5 PRE-CRITICAL TEST PHASE

3 4.3.5.1 Control Rod Drop Time

3 4.3.5.1.1 Plant Conditions

3 Full reactor coolant system (RCS) flow (4 pumps).

3 4.3.5.1.2 Procedure

3 The control rod drop time for each full-length control rod assembly (CRA) to fall from the fully
4 withdrawn position to the 25% withdrawn position is measured. The sequence of events recorder is
4 normally used to record the time interval between initiation and termination of the event. The test may
3 be performed either by dropping all full-length CRAs simultaneously from the fully withdrawn position,
6 or by dropping one full length CRA group at a time. In either case, the sequence of events recorder
3 records the drop time of each CRA individually.

3 The results are reviewed by the Test Coordinator and compared with the acceptance criterion, 1.66
4 seconds. The accuracy of the measurement of control rod drop time as performed by the sequence of
4 events recorder is approximately ± 0.005 seconds.

9 The use of Type C Control rod drive mechanisms requires the use of a slightly higher trip delay time.
6 This difference is accounted for in the affected safety analysis.

3 4.3.5.1.3 Follow-Up Actions

3 If any measured control rod drop time is greater than 1.40 seconds but less than 1.66 seconds, then the
3 results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to
3 resolve the discrepancy. This review will be completed prior to 100% FP.

3 If any control rod drop time exceeds 1.66 sec., then the results will be reviewed by cognizant engineers to
3 determine the appropriate corrective actions required to resolve the discrepancy. Also, the actions
8 specified by Technical Specifications 3.1.4, "Control Rod Group Alignment Limits", will be taken.

3 4.3.6 ZERO POWER PHYSICS TEST PHASE

3 4.3.6.1 Critical Boron Concentration

3 4.3.6.1.1 Plant Conditions

3 Hot Zero Power, ~532°F, ~2155 psig, steady RCS flow (3 or 4 pumps).

3 4.3.6.1.2 Procedure

3 The ARO critical boron concentration is measured by establishing an equilibrium RCS boron
3 concentration near the predicted ARO critical boron concentration. Control Rod Groups 1 through 7 are
3 fully withdrawn. Control Rod Group 8 is maintained at the nominal designed position. A sample of the
3 equilibrium boron concentration is then taken and analyzed to determine the critical boron concentration.
3 Since it may not be practical to establish critical equilibrium conditions with Group 7 fully withdrawn, the
3 small amount of inserted worth of Group 7 or worth of Group 8 (from its nominal designed position) is
3 measured by a reactivity calculation or Reactimeter. This reactivity is then used to adjust the boron
3 concentration to obtain the measured ARO boron concentration.

The results are reviewed by the Test Coordinator and compared with the predicted boron concentration. If the difference between the measured and predicted values does not exceed 50 ppm Boron, the results are acceptable.

4.3.6.1.3 Follow-Up Actions

If the acceptance criterion (± 50 ppmb) between measured and predicted ARO critical boron concentration is not met, the results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy. This review will be completed with the results and recommended corrective actions approved by representatives from Oconee Nuclear Station Nuclear Engineering and Regulatory Compliance, and General Office Nuclear Engineering prior to 100% FP.

If the difference between measured and predicted ARO critical boron concentration is greater than 100 ppm Boron, the results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy. This review will be completed with the results and recommended corrective actions approved by representatives from Oconee Nuclear Station Nuclear Engineering and Regulatory Compliance, and General Office Nuclear Engineering prior to exceeding 15% FP.

4.3.6.2 Moderator Temperature Coefficient

4.3.6.2.1 Plant Conditions

Hot Zero Power, ~532°F, ~2155 psig, steady RCS flow (3 or 4 pumps).

4.3.6.2.2 Procedure

The moderator temperature coefficient (MTC) test begins with the reactor at critical equilibrium conditions. This test is performed by executing a change in RCS average temperature of approximately ± 5°F while data are taken. Stability in RCS temperature is necessary at this first plateau. The hold time at each RCS temperature plateau during the test is approximately five minutes. After data are taken at the first RCS temperature plateau, the RCS average temperature is changed approximately 10°F in the opposite direction and allowed to stabilize. Changes in reactivity associated with the induced RCS temperature transient are measured by a reactivity calculation or Reactimeter. This overall temperature coefficient is corrected for the contribution of the isothermal doppler coefficient or reactivity to give the moderator coefficient of reactivity. The measurement is also corrected to an average temperature of 532°F.

The results are reviewed by the Test Coordinator and compared with the predicted MTC. If the difference between the measured and predicted values does not exceed 0.3×10^{-4} $\Delta k/k/°F$, then the results are acceptable.

4.3.6.2.3 Follow-Up Actions

If the measured maximum positive MTC exceeds 0.5×10^{-4} $\Delta k/k/°F$, the results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy. This review will be completed with the results and recommended actions approved by representatives from Oconee Nuclear Station Nuclear Engineering and Regulatory Compliance, and General Office Nuclear Engineering prior to exceeding 15% FP.

If the 0.3×10^{-4} $\Delta k/k/°F$ acceptance criterion is exceeded and the maxium positive MTC is less than 0.5×10^{-4} $\Delta k/k/°F$, the results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy. This review will be completed with the results and

3 recommended corrective actions approved by representatives from Oconee Nuclear Station Nuclear
3 Engineering and Regulatory Compliance, and General Office Nuclear Engineering prior to 100% FP.

3 4.3.6.3 Control Rod Worth

3 4.3.6.3.1 Plant Conditions

3 Hot Zero Power, ~532°F, ~2155 psig, steady RCS flow (3 or 4 pumps).

3 4.3.6.3.2 Procedure

3 The measurement of regulating rod group worths begin from a critical steady state condition with all
3 regulating rod groups withdrawn as far as possible (i.e., within 0.12% $\Delta k/k$ of ARO). From this point a
3 boron concentration necessary to deborate control rod Groups 7, 6 and 5 to fully inserted is calculated.
3 The resulting reactivity change during deboration is compensated for by discrete insertion of control rods
3 with both signals being recorded by a reactivity calculation or Reactimeter. Integral rod worths are
3 calculated by summing the differential rod worths for each control rod group.

3 The results are reviewed by the Test Coordinator and compared with the predicted control rod group
3 worths. If the difference between the measured and predicted individual rod group worths does not
3 exceed 15%, and the difference between the measured and predicted total worth of control rod Groups 5,
3 6 and 7 does not exceed 10%, then the results are acceptable.

3 4.3.6.3.3 Follow-Up Actions

3 If the difference between the measured and predicted total worth of control rod Groups 5, 6, and 7
3 exceeds 10%, then, following calculation of the minimum control rod position for which the worth of the
3 control rods withdrawn would equal 1% $\Delta k/k$, additional control rod group worths will be measured.
3 The worths of safety control rod groups will be measured in sequence from Group 4 to Group 2, until
3 either the difference between the measured and predicted total worth of all control rod groups measured
3 does not exceed 10%, or the calculated minimum control rod position is reached. In the latter case,
3 control rod worth testing will halt. The results will be reviewed by cognizant engineers to determine the
3 appropriate additional corrective actions required to resolve the discrepancy. This review will be
3 completed with the results and the recommended actions approved by representatives from Oconee
3 Nuclear Station Nuclear Engineering and Regulatory Compliance, and General Office Nuclear
3 Engineering prior to exceeding 15% FP.

3 If the difference between the measured and predicted control rod worths of any of the individual control
3 rod groups exceeds 15%, the results will be reviewed by cognizant engineers to determine the appropriate
3 corrective actions required to resolve the discrepancy. This review will be completed prior to reaching
3 100% FP.

3 4.3.7 POWER ESCALATION TEST PHASE

3 4.3.7.1 Low Power Testing

3 4.3.7.1.1 Plant Conditions

3 5 to 30% FP, ~579°F, ~2155 psig, full RCS flow (4 pumps).

3 4.3.7.1.2 Procedure

3 Once the unit is between 5 and 30% FP, the output of the plant OAC reactor calculations program is
3 analyzed. This program processes the signals from fixed incore detectors and provides a relative core
3 power distribution as output. The incore detector outputs are checked in order to identify malfunctioning
3 detectors. After these have been eliminated, the results for corrected assembly power in functioning
3 instrumented symmetric core locations are compared.

3 The results are reviewed by the Test Coordinator. If the reactor calculations outputs appear normal, and
3 the deviation between the highest and lowest corrected assembly power for symmetric core locations is
3 less than ±10%, then the results are acceptable.

3 4.3.7.1.3 Follow-Up Actions

3 If the reactor calculations outputs appear abnormal, the raw detector signals are evaluated to determine if
3 a significant core asymmetry exists. If no significant asymmetry exists, power escalation is continued. If
3 an asymmetry exists, the Site Nuclear Engineering Supervisor is contacted to initiate a program of testing
3 and evaluation before further power increase. The problem with the reactor calculations program is
3 investigated and corrected, but this is not a prerequisite for power increase if no significant asymmetry
3 exists.

3 If the reactor calculations outputs appear normal and the deviation between corrected assembly powers
3 for symmetric core locations is greater than ±10%, the cause of the indicated deviation is investigated. If
3 the deviation is due to identifiable reactor calculations program problems, it is corrected per normal
3 procedures and power escalation testing may continue. If the cause of the deviation cannot be identified,
3 the Site Nuclear Engineering Supervisor is contacted to initiate a program of testing and evaluation.

3 The results will be reviewed by cognizant engineers to determine the appropriate corrective actions
3 required to resolve the deviation. This review will be completed with the results and the recommended
3 corrective actions approved by representatives from Oconee Nuclear Station Nuclear Engineering and
3 Regulatory Compliance, and General Office Nuclear Engineering prior to any further escalation of power.

3 4.3.7.2 Intermediate Power Testing

3 4.3.7.2.1 Plant Conditions

3 40 to 75% FP, ~579°F, ~2155 psig, full RCS flow (4 pumps).

3 4.3.7.2.2 Procedure

3 Once the unit is between 40 and 75% FP, the output of the plant OAC reactor calculations program is
3 analyzed. This program processes the signals from fixed incore detectors and provides a relative core
3 power distribution as output. The incore detector outputs are checked, in order to identify
3 malfunctioning detectors. After these have been eliminated, the radial and total peaking factors obtained
3 from the plant OAC are compared with the values calculated using the computer codes utilized during the
3 reload design process on an eighth-core basis.

3 The results are reviewed by the Test Coordinator. If the highest measured radial peaking factor does not
3 exceed the highest predicted radial peaking factor by more than 5.0% of the highest measured radial
3 peaking factor, and if the highest measured total peaking factor does not exceed the highest predicted total
3 peaking factor by more than 7.5% of the highest measured total peaking factor, and if the RMS difference
3 between predicted and measured radial peaking factors is less than 7.5%, then the results are acceptable.

3 4.3.7.2.3 Follow-Up Actions

3 If any observed parameter exceeds its specified values in the Technical Specifications, actions will be taken
3 as required by the Technical Specifications.

3 Also, the observed parameter will be reviewed by cognizant engineers to determine the appropriate
3 corrective actions required to resolve the discrepancy. This review will be completed with the results and
3 recommended corrective actions approved by representatives from Oconee Nuclear Station Nuclear
3 Engineering and Regulatory Compliance, and General Office Nuclear Engineering prior to any further
3 escalation of power.

3 If any acceptance criteria are exceeded, the results will be reviewed by cognizant engineers to determine
3 the appropriate corrective actions required to resolve the discrepancy. This review will be completed with
3 the results and recommended corrective actions approved by representatives from Oconee Nuclear Station
3 Nuclear Engineering and Regulatory Compliance, and General Office Nuclear Engineering prior to
3 escalation to 100% FP.

3 4.3.7.3 Full Power Testing

3 4.3.7.3.1 Plant conditions

3 90 to 100% FP, ~579°F, ~2155 psig, full RCS flow (4 pumps).

3 4.3.7.3.2 Procedure

3 Once the unit is between 90 and 100% FP with Xenon equilibrium, the output of the plant OAC reactor
3 calculations program is analyzed. This program processes the signals from fixed incore detectors and
3 provides a relative core power distribution as output. The incore detector outputs are checked, in order to
3 identify malfunctioning detectors. After these have been eliminated, the radial and total peaking factors
3 obtained from the OAC are compared with the values calculated as part of the reload design process on
3 an eighth-core basis. The results are reviewed by the Test Coordinator. If the highest measured radial
3 peaking factor does not exceed the highest predicted radial peaking factor by more than 5.0% of the
3 highest measured radial peaking factor, and if the highest measured total peaking factor does not exceed
3 the highest predicted total peaking factor by more than 7.5% of the highest measured total peaking factor,
3 and if the RMS difference between predicted and measured radial peaking factors is less than 7.5%, then
3 the results are acceptable.

3 4.3.7.3.3 Follow-Up Actions

3 If any observed parameter exceeds its specified values in the Technical Specifications, actions will be taken
3 as required by the Technical Specifications.

3 Also, the observed parameter will be reviewed by cognizant engineers to determine the appropriate
3 corrective actions required to resolve the discrepancy. This review will be completed with the results and
3 the recommended corrective actions approved by representatives from Oconee Nuclear Station Nuclear
3 Engineering and Regulatory Compliance, and General Office Nuclear Engineering prior to any escalation
3 of power.

3 If any acceptance criteria are exceeded, the results will be reviewed by cognizant engineers to determine
3 the appropriate corrective actions required to resolve the discrepancy. This review will be completed with
3 the results and recommended corrective actions approved by representatives from Oconee Nuclear Station
3 Nuclear Engineering and Regulatory Compliance, and General Office Nuclear Engineering prior to any
3 escalation of power.

3 4.3.7.4 Reactivity Anomaly

3 4.3.7.4.1 Plant Conditions

3 Hot Full Power, ~579°F, ~2155 psig, full RCS flow.

3 4.3.7.4.2 Procedure

3 As a part of the periodic testing program and separate from the startup testing program, the ARO critical
8 boron concentration at power is checked against normalized predicted values approximately each 31
3 EFPD of steady-state operation. With the reactor at steady-state conditions, as near a practical to full
3 power ARO conditions, a sample of the RCS is taken an analyzed for boron concentration. This value of
3 boron concentration is then adjusted to account for the reactivity worth of regulating control rod
3 assemblies in the core at the time of the measurement, and any other minor variations from designed
3 conditions.

3 The results are reviewed by the Site Nuclear Engineering Supervisor and are compared with the
3 normalized predicted ARO boron concentration for the time in the cycle at which the measurement was
3 taken. If the difference between the measured and predicted ARO boron concentration values does not
3 exceed 50 ppm Boron, then the results are acceptable.

3 4.3.7.4.3 Follow-Up Actions

3 If the acceptance criterion (±50 ppmb) is not met and the difference between measured and predicted
3 ARO boron concentration is less than 100 ppm Boron, the results will be reviewed by cognizant engineers
3 to determine the appropriate corrective action required to resolve the discrepancy. This review will be
3 completed with the results and recommended corrective actions approved by representatives from Oconee
3 Nuclear Station Nuclear Engineering and Regulatory Compliance, and General Office Nuclear
3 Engineering within 14 days.

3 If the acceptance criterion (±50 ppmb) is not met and the difference between measured and predicted
3 ARO boron concentration is greater than 100 ppm Boron, then the results will be reviewed by cognizant
3 engineers to determine the appropriate corrective actions required to resolve the discrepancy pursuant to
8 Technical Specification 3.1.2. "Reactivity Balance".

3 Oconee Startup Physics Test Program (OSPTP) Summary

	TEST	PLANT CONDITIONS	ACCEPTANCE CRITERIA
3 3			
3	1. Control Rod Trip		
3	Time Test	RCS Full Flow	1.66 seconds
3			
3	2. All Rods Out Critical		
3	Boron	HZP	± 50 ppmB
3			
3	3. Moderator Temperature		
3	Coefficient	HZP	$\pm\ 0.3 \times 10^{-4}\ \Delta\rho/°F$
3			

TEST	PLANT CONDITIONS	ACCEPTANCE CRITERIA
4. Control Rod Worth	HZP	Individual Groups ± 15% Sum of Groups ± 10%
5. Low Power Testing	5 - 30 %FP	Relative Core Power Distribution ± 10%
6. Intermediate Power Testing	40 - 75%FP	Total/Radial Peaking ±7.5%/±5.0% RMS (Radial) < 7.5%
7. Full Power Testing	90 - 100 %FP	Total/Radial Peaking ±7.5%/±5.0% RMS (Radial) < 7.5%
8. Reactivity Anomaly	HFP	All Rods Out Critical Boron ± 50 ppmB

4.3.8 REFERENCES

1. J. J. Romano, Core Calculational Techniques and Procedures, *BAW - 110118A*, Babcock & Wilcox, Lynchburg, Virginia, October 1977.
2. Oconee Nuclear Station Reload Design Methodology, NFS-1001A Duke Power, Charlotte, North Carolina, April 1984.
3. Saxton, Large Closed-Cycle Water Research and Development Work Program for the Period July 1 to December 31, 1964, *WCAP-3269-4.*
4. Oconee Nuclear Station Reload Design Methodology II, DPC-NE-1002A, Duke Power, Charlotte, North Carolina, October 1985.
5. Stability Margin for Xenon Oscillations, Two- and Three-Dimensional Digital Analyses, *BAW-10010*, Babcock & Wilcox, Lynchburg, Virginia, June 1971.
6. Oconee Nuclear Station Unit 1 Startup Report DPR-38, Docket No. 50-269, November 16, 1973.
7. Oconee Nuclear Station Unit 2 Startup Report DPR-47, Docket No. 50-270, July 12, 1974.
8. Oconee Nuclear Station Unit 3 Startup Report DPR-55, Docket No. 50-278, March 14, 1975.
9. Letter W. O. Parker, Jr. to H. R. Denton, Oconee Nuclear Station Generic Startup Physics Test Program, Docket Nos. 50-269, -270, -287, July 11, 1980.

3 10. Letter A. C. Thies to H. R. Denton, Oconee Nuclear Station 1, 2, and 3, Docket Nos. 50-269, -270, -287, May 29, 1981.

11. R. A. Turner, Fuel Densification Report, *BAW-10054, Rev. 2*, Babcock & Wilcox, Lynchburg, Virginia, May 1973.
12. Oconee Unit 1, Cycle 7, Reload Report, *BAW-1660*, Babcock & Wilcox, Lynchburg, Virginia, March 1981.

3 13. Letter W. O. Parker, Jr. to H. R. Denton, Oconee Nuclear Station 1, 2, and 3, Docket Nos. 50-269,
3 -270, -287, August 15, 1980.

3 14. Letter J. F. Stolz to W. O. Parker, Jr., Oconee Nuclear Station, Generic Startup Physics Test
3 Program, March 23, 1981.

3 15. Letter P. C. Wagner to W. O. Parker, Jr., Oconee Nuclear Station 1, 2 and 3, Docket Nos. 50-269,
3 -270, -287, November 30, 1981.

3 16. Letter H. B. Tucker to H. R. Denton, Oconee Nuclear Station 1, 2 and 3, Docket Nos. 50-269, -270,
3 -287, September 2, 1986.

3 17. Letter J. F. Stolz to H. B. Tucker, Oconee Nuclear Station 1, 2 and 3, Revisions to the Startup
3 Physics Testing Program, October 7, 1986.

5 18. Nuclear Design Methodology using CASMO-3/SIMULATE-3P, DPC-NE-1004A, Duke Power,
5 Charlotte, North Carolina, November 1992

4.4 THERMAL AND HYDRAULIC DESIGN

4.4.1 DESIGN BASES

The bases for the thermal and hydraulic design have been established to enable the reactor to operate at 2,568 MWt rated power with sufficient design margins to accommodate both steady-state and transient operation without damage to the core and without exceeding the design pressure limits for the reactor coolant system. The thermal-hydraulic design bases also help to ensure that the fuel rod cladding will maintain its integrity during steady-state operation, design overpower, and anticipated operational transients occurring throughout core life.

Fuel cladding integrity is ensured by limiting the core to the following thermal-hydraulic boundaries during steady-state operation at power levels up to and including the design overpower, and during anticipated transient operation.

1. The fuel pin cladding, fuel pellets, and fuel pin internals must be designed so that the fuel-to-clad gap characteristics ensure that the maximum fuel temperature does not exceed the fuel melting limit at the 112 percent design overpower at any time during core life. See Section 4.2.3.1.3, "Fuel Thermal Analysis" for a discussion of fuel melting temperature.

0 2. The minimum allowable DNBR during steady-state operation and anticipated transients for Mark-BZ
9 and Mark-B11 fuel are:

9 a. Mark-BZ fuel is established as 1.18 with the BWC correlation (Reference 1) for non-SCD
9 analyses and 1.43 for SCD analyses (Reference 13).

9 b. MK-B11 fuel is established as 1.19 with the BWU-Z correlation with the FB11 multiplicative
9 factor for non-SCD analyses and 1.33 for SCD analyses (Reference 13).

9 These limits on MDNBR ensure a 95 percent confidence level that there is a 95 percent probability
9 DNB will not occur.

3. Although generation of net steam is allowed in the hottest core channels, flow stability is required during all steady-state and operational transient conditions.

By preventing a departure from nucleate boiling (DNB), neither the cladding nor the fuel is subjected to excessively high temperatures.

The core flow distribution and coolant velocities have been set to provide adequate cooling capability to the hottest core channels and to maintain minimum DNB ratios greater than the design limit. Fuel assembly design and cladding integrity criteria are discussed in Section 4.2.1.2.4, "Mechanical Limits."

4.4.2 DESCRIPTION OF THERMAL AND HYDRAULIC DESIGN OF THE REACTOR CORE

9 Table 4-1 and Table 4-2 depicts typical thermal-hydraulic design conditions.

4.4.2.1 CORE DESIGN ANALYSIS DESCRIPTION

The methodology of the analysis used with the design bases criterion is fully described in
6 DPC-NE-2003P-A (Reference 2) and DPC-NE-2005P-A (Reference 13).

The input information and analytical tools for the thermal hydraulic design and for the evaluation of individual hot channels is as follows:

9 1. Heat transfer, critical heat flux equations, and data correlations.

2. Nuclear peaking factors.

3. Engineering hot channel factors.

4. Core flow distribution hot channel factors.

9 5. Design reactor power.

9 6. Thermal hydraulic analysis computer codes.

These inputs have been derived from test data, physical measurements and calculations.

9 Critical heat flux (CHF) calculations are performed with the Framatome Cogema Fuel (FCF) BWC
9 correlation. Items 1 through 5 on the above list are explained in Chapters 5 and 6 of DPC-NE-2003P-A
9 (Reference 2). VIPRE-01 (Reference 3) is the computer code used in these analyses (Item 6).

The design overpower is the highest credible reactor operating power permitted by the Reactor Protective System including maximum instrumentation errors. Normally, trip on overpower will occur at a
0 significantly lower power than the design overpower.

6 The Statistical Core Design, or SCD, methodology described in Reference 13 allows for the statistical
6 combination of the variables that directly affect the DNB performance of the fuel. The key DNB
8 parameters include: reactor power, core inlet temperature, core flow rate, core exit pressure, and three
6 dimensional power distribution. This statistical combination takes into account the probability of each
6 key DNB parameter being within a specified uncertainty distribution at any given point in time. The
6 result is the ability to input nominal values of these parameters into any analysis and still maintain the
6 95% probability with 95% confidence that DNB will not occur.

4.4.3 THERMAL AND HYDRAULIC EVALUATION

4.4.3.1 Introduction

0 A summary of the characteristics of the reactor core design is given in Section 4.1, "Summary
0 Description." The methodology of the thermal and hydraulic design analysis is presented in
6 DPC-NE-2003P-A (Reference 2) and DPC-NE-2005P-A (Reference 13).

0 ### 4.4.3.2 Deleted Per 1990 Update

0

4.4.3.3 Evaluation of the Thermal and Hydraulic Design

4.4.3.3.1 Hot Channel Coolant Conditions

0 The NRC approved VIPRE-01 code is used to calculate the reactor coolant enthalpy, mass flow, vapor
0 void, and DNBR distributions within the core for all expected operating conditions. The VIPRE-01 code
0 is described in detail in (Reference 3), and the models and empirical correlations that are used are
0 discussed in (Reference 2).

0 Steady-state analyses yield the MDNBR and quality in the hot channel at nominal and maximum design
9 overpower conditions. Table 4-1 contains a typical hot channel MDNBR value at nominal reactor
5 conditions.

0

4.4.3.3.2 Coolant Channel Hydraulic Stability

Flow regime maps of mass flow rate and quality were constructed in order to evaluate channel hydraulic stability. The confidence in the design is based on a review of both analytical evaluations (References 4 through 8) and experimental results obtained in multiple rod bundle burnout tests. Bubble-to-annular
9 and bubble-to-slug flow limits proposed by Baker (Reference 4) are consistent with the FCF experimental data in the range of interest. The analytical limits and experimental data points have been plotted to obtain the maps for the four different types of cells in the reactor core. These are shown in Figure 4-21, Figure 4-22, Figure 4-23, and Figure 4-24. The experimental data points represent the exit conditions in the various types of channels just previous to the burnout for a representative sample of the data points obtained at design operating conditions in the nine rod burnout test assemblies. In all of the bundle tests, the pressure drop, flow rate, and rod temperature traces were repeatable and steady, and did not exhibit any of the characteristics associated with flow instability.

Values of hot channel mass velocity and quality at 114 percent and 130 percent power for both nominal and design conditions are shown on the maps. The potential operating points are within the bounds suggested by Baker. Experimental data points for the reactor geometry with much higher qualities then the operating conditions have not exhibited unstable characteristics (Reference 9).

0 4.4.3.3.3 Reactor Coolant Flow System

0 Another significant variable to be considered in evaluating the design is the total reactor coolant system
0 (RCS) flow. Conservative values for system and reactor pressure drop have been determined to insure
9 that the required system flow is obtained in the as-built plant. Measured RCS flow is above the design
flow used in the core reload thermal hydraulic analyses.

0 The difference between the RCS flow and the reactor core flow is the core bypass flow. The core bypass
8 flow is defined as that part of the flow that does not contact the active heat transfer surface area. The
8 bypass flow paths are (1) core shroud, (2) core barrel annulus, (3) the control rod guide tubes and
8 instrument tubes, and (4) all interfaces separating the inlet and outlet regions of the reactor vessel. The
0 core bypass flow is generally less than 9%; however, the bypass flow rate is dependent on the number of
0 assemblies not containing control rods, burnable poison rods, or source rods in each cycle as explained in
0 Reference 2.

0

0 4.4.3.3.4 Deleted Per 1990 Update

0

0 4.4.3.3.5 Core Flow Distribution

0 Inlet plenum effects have been determined from a 1/6 scale model flow test. The isothermal flow test data
0 has shown that the hot bundle receives average or better flow. It is conservatively assumed in all DNB
7 analysis (assuming 4 operating RC pumps) that the inlet flow in the hot bundle is 5 percent less than the
7 average bundle flow (Reference 2). A more restrictive inlet flow maldistribution factor is assumed for 3
4 pump operation analyses.

0 Flow redistribution accounts for the reduction in flow in the hot channel resulting from the high flow
0 resistance due to the local or bulk boiling in the hot channel. The effect on flow of the non-uniform
0 design power distribution is inherently considered in the VIPRE-01 code for all of the conditions
0 analyzed.

0 #### 4.4.3.3.6 Mixing Coefficient

0 The flow distribution within the hot assembly is calculated using the VIPRE-01 code which allows for the
0 interchange of momentum and heat between channels. The turbulent mixing model incorporated in the
0 VIPRE-01 code and used for all core thermal-hydraulic analyses is discussed in Reference 2. A
9 conservative mixing coefficient of 0.01, based on predictions of mixing tests, is used for DNB analyses for
9 Mark-BZ fuel assembly design. A conservative mixing coefficient of 0.038, per Reference 13, is used for
9 DNB analyses for Mark-B11 fuel assembly design due to the presence of mixing vane grids.

0 #### 4.4.3.3.7 Deleted Per 1990 Update.

0

0 #### 4.4.3.3.8 Hot Channel Factors

0 Hot channel factors are applied to the hot channel in the core to conservatively compensate for possible
0 deviations of several parameters from their design values. The power hot channel factor, F_q, accounts for
0 variations in average pin power caused by differences in the absolute number of grams of U_{235} per rod. F_q
5 is applied to the heat generation rate of the hot pin of the hot subchannel. The value of F_q used is given
5 in Reference 2. F_q is increased when calculating Maximum Allowable Peaking (MAP) limits to account
8 for the axial nuclear uncertainty. References 14 and 15 have shown that small local heat flux spikes (which
8 result from power spikes due to flux depressions at the spacer grids and local variations in pellet
8 enrichment/weight) have no effect on the critical heat flux. The hot channel flow area is also reduced as
0 discussed in Reference 2 to account for manufacturing tolerances.

4.4.3.3.9 Rod Bow Effects and Penalty

9 The mechanisms and resulting effects of fuel rod bow are discussed in FCF topical report
9 BAW-10147P-A (Reference 10) and BAW-10186P-A (Reference 17). The topical report concludes that
the DNB penalty due to rod bow is insignificant and unnecessary because the power production capability
of the fuel decreases with irradiation. The rod bow correlation developed in Reference 10 also
9 conservatively predicts the rod bow behavior of Mark-BZ fuel and Mark-B11 fuel.

4.4.4 THERMAL AND HYDRAULIC TESTS AND INSPECTION

4.4.4.1 Reactor Vessel Flow Distribution and Pressure Drop Test

A 1/6-scale model of the reactor vessel and internals has been tested to evaluate:

1. The flow distribution to each fuel assembly of the reactor core and to develop any necessary modifications to produce the desired flow distribution.
2. Fluid mixing between the vessel inlet nozzle and the core inlet, and between the inlet and outlet of the core.
3. The overall pressure drop between the vessel inlet and outlet nozzles, and the pressure drop between various points in the reactor vessel flow circuit.

4. The internals vent valves for closing behavior and for the effect on core flow with valves in the open position.

The reactor vessel, flow baffle, and core barrel were made of clear plastic to allow use of visual flow study techniques. All parts of the model except the core are geometrically similar to those in the production
0 reactor. The simulated core was designed to maintain dynamic similarity between the model and production reactor.

Each of the 177 simulated fuel assemblies contained a calibrated flow nozzle. The test loop is capable of
9 supplying cold water (75°F) to three inlet nozzles and hot water (140°F) to the fourth. Temperature was measured in the inlet and outlet nozzles of the reactor model and at the inlet and outlet of each of the fuel assemblies. Static pressure taps were located at suitable points along the flow path through the vessel. This instrumentation provided the data necessary to accomplish the objectives set forth for the tests. The tests are summarized in BAW-10037 (Reference 9).

4.4.4.2 Fuel Assembly Heat Transfer and Fluid Flow Tests

9 Although the original design of the reactor is based on the W-3 heat transfer correlation, FCF has conducted a continuous research and development program for fuel assembly heat transfer and fluid flow applicable to the design of the reactor. Single-channel tubular and annular test sections and multiple rod
9 assemblies have been tested at the Alliance Research Center. Also, 5x5 rod bundle sections have been
9 tested at the Columbia University Heat Transfer Laboratory. This test work substantiates the thermal
0 design of the reactor core. The multiple rod CHF tests are briefly discussed below.

0 **4.4.4.2.1 Deleted Per 1990 Update**

0

4.4.4.2.2 Multiple-Rod Fuel Assembly Heat Transfer Tests

9 The following sections discuss the fuel assembly heat transfer tests for the BWC and BWU-Z with FB11
9 multiplicative factor CHF correlations.

9 <u>BWC CHF Correlation</u>

0 As a part of the development of the 15 x 15 Zircaloy grid Mark-BZ fuel assembly design, a series of CHF
9 tests were run at FCF's Alliance Research Center heat transfer facility. The tests were performed for 15 x
0 15 geometry with Zircaloy grids and full length non-uniform axial flux shapes. A total of 211 data points
0 were obtained covering the following conditions:

8 **Note:** The following conditions were revised in 1998 update.

9	Pressure	1,600 < P < 2,600 psia
9	Local Mass Velocity	0.43 < G < 3.8 -Mlbm/hr-ft^2
9	Local Quality	-0.20 < X_{loc} < 0.26

9

0 The BWC correlation was developed from 17 x 17 Mark-C CHF data. The BWC correlation was shown
0 to conservatively represent the Mark-BZ CHF data with a 95/95 DNBR limit of 1.18 (Reference 1).

9 The BWC correlation was developed by FCF using the LYNX2 computer code (Reference 12). To verify
0 use of the BWC correlation with the VIPRE-01 code, the Mark-BZ CHF data was predicted and

9 compared with FCF's LYNX2 results. As discussed in Reference 2, the VIPRE-01 BWC results show
0 that a DNBR limit of 1.18 will provide 95% probability of precluding DNB at a 95% confidence level.

9 <u>BWU-Z CHF Correlation, With FB11 Multiplicative Factor</u>

9 As part of the development of a 15x15 mixing vane grid design, critical heat flux tests have been
9 performed at Columbia University Heat Transfer Research Facility (HTRF) for Mark-B11 fuel. The tests
9 were performed for a 15x15 geometry with Zircaloy mixing vane grids and full length non-uniform axial
9 flux shape. The BWU-Z CHF correlation with the FB11 multiplier, Reference 16, was developed based
9 on Mark-B11 15x15 mixing vane CHF data. The FB11 multiplier of 0.98 on the BWU-Z CHF
9 correlation is based on a total of 216 data points. The BWU-Z CHF correlation was developed by FCF
9 from a data base of 530 data points on fuel with Zircaloy mixing vane spacer grid designated Mark BW17.
9 The Mark-B11 spacer grid design is a 15x15 version of FCF's 17x17 Mark-BW17 design. The BWU-Z
9 CHF correlation with the FB11 multiplier is applicable to the following range of variables:

9	Pressure	$400 \leq P \leq 2465$ psia
9	Local Mass Velocity	$0.36 \leq G_{loc} \leq 3.55$ Mlbm/ft^2-hr
9	Local Quality	$X_{loc} \leq 0.74$

9 The BWU-Z correlation with the Mark-B11 multiplier of 0.98 was shown to conservatively represent the
9 Mark-B11 CHF data with a 95/95 DNBR limit of 1.19 (Reference 16).

9 The BWU-Z correlation with Mark-B11 multiplier was developed by FCF using the LYNX2 computer
9 code (Reference 12). To verify use of the BWU-Z correlation with the Mark-B11 multiplier with the
9 VIPRE-01 code, the Mark-B11 CHF data was predicted and compared with FCF's LYNX2 results. As
9 discussed in Reference 13, the VIPRE-01 BWU-Z with Mark-B11 multiplier results show that a DNBR
9 limit of 1.19 will provide 95% probability of precluding DNB at a 95% confidence level.

4.4.4.2.3 Fuel Assembly Flow Distribution, Mixing and Pressure Drop Tests

Flow visualization and pressure drop data have been obtained from a ten-times-full-scale (10X) model of a single rod in a square flow channel. These data have been used to refine the spacer grid designs with respect to mixing turbulence and pressure drop. Additional pressure drop testing has been conducted using 4-rod (5X), 4-rod (1X), 1-rod (1X), and 9-rod (1X) models.

Testing to determine the extent of interchannel mixing and flow distribution has also been conducted. Flow distribution in a square 4-rod test assembly has been measured. A salt solution injection technique was used to determine the average flow rates in the simulated reactor assembly corner cells, wall cells, and unit cells. Interchannel mixing data were obtained for the same assembly. These data have been used to confirm the flow distribution and mixing relationships employed in the core thermal and hydraulic design. Flow tests on a mockup of two adjacent fuel assemblies have been conducted. Additional mixing, flow distribution, and pressure drop data will be obtained to improve future core power capability. The following fuel assembly geometries have been tested to provide additional data:

1. A 9-rod (3 x 3 array) mixing test assembly, to determine flow pressure drop, flow distribution, and degree of mixing.

2. A 64-rod assembly simulating larger regions and various mechanical arrangements within a 15 x 15 fuel assembly and between adjacent fuel assemblies to determine flow distribution in the assembly and between adjacent assemblies.

9 <u>Mark-B11 Fuel Assembly Flow Tests</u>

9 The flow-induced vibration (FIV) tests, pressure drop tests, Laser Doppler Velocimeter (LDV) tests, and
9 critical heat flux (CHF) tests were conducted on the Mark-B11 fuel assembly design per Reference 18.
9 The FIV tests were performed to examine the vibrational response of the Mark-B11 fuel assembly and to
9 verify that there were no flow related phenomena that would adversely affect fuel integrity. The pressure
9 drop tests were conducted to determine form loss coefficients for the Mark-B11 components. The LDV
9 tests were conducted to characterize the subchannel flow distribution within the Mark-B11 fuel assembly
9 design. The CHF tests were conducted to develop a CHF correlation that would accurately represent the
9 CHF performance of the Mark-B11 mixing vane grid.

4.4.5 REFERENCES

8 1. BWC Correlation of Critical Heat Flux, Babcock & Wilcox, *BAW-10143P-A*, *Part 2*, Lynchburg, Va.,
8 April 1985.

0

0 2. Oconee Nuclear Station Core Thermal Hydraulic Methodology, Duke Power Company,
0 *DPC-NE-2003P-A*, Charlotte, N. C., October 1989.

0 3. Stewart, C. W., et al. VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores. 5 vols. Battelle,
0 Pacific Northwest Laboratories, *EPRI NP-2511-CCM-A, Rev. 3*, Richland, Washington, August 1989.

0

0 4. Baker, 0., "Simultaneous Flow of Oil and Gas," *Oil and Gas Journal: 53* pp 185-195 (1954).

0 5. Rose, S. C., Jr., and Griffith, P., Flow Properties of Bubbly Mixtures, ASME Paper *No. 65-HT-38*
0 (1965).

0 6. Haberstroh, R. D. and Griffith P., The Transition From the Annular to the Slug Flow Regime in
0 Two-Phase Flow, *MIT TR-5003-28*, Department of Mechanical Engineering, MIT, June 1964.

0 7. Bergles, A. E., and Suo, M., Investigation of Boiler Water Flow Regimes at High Pressure,
0 *NYO-3304-8*, February 1, 1966.

0 8. Kao, H. S., Cardwell, W. R., Morgan, C. D., HYTRAN - Hydraulic Transient Code for Investigating
0 Channel Flow Stability, Babcock & Wilcox, *BAW-10109*, Lynchburg, Va., January 1976.

0 9. Mullinax, B. S., Walker, R. J., and Karrasch, B. A., Reactor Vessel Model Flow Tests, Babcock &
0 Wilcox, *BAW-10037 Rev. 2*, Lynchburg, Va., November 1972.

0

0 10. Fuel Rod Bowing in Babcock and Wilcox Fuel Designs, Babcock & Wilcox, *BAW-10147P-A*, *Rev. 1*,
0 Lynchburg, Va., May 1983.

0 11. C. E. Barksdale (B&W) to R. Powers (SMUD), Letter, Rancho Seco Unit 1 Response to NRC
0 Questions on Mark-BZ Fuel, March 5, 1984.

0

0 12. LYNX2: Subchannel Thermal-Hydraulic Analysis Program, Babcock and Wilcox, *BAW-10130-A*,
0 Lynchburg, VA, July 1985.

9 13. DPC-NE-2005P-A Rev. 2, Duke Power Company Thermal-Hydraulic Statistical Core Design
9 Methodology, June 1999.

8 14. Westinghouse Topical Report WCAP-8202, Effect of Local Heat Flux Spikes on DNB in
8 Non-Uniformly Heated Bundles, K. W. Hill, F. E. Motely, and F. F. Cadek, August 1973.

8 15. Combustion Engineering Topical Report CENPD-207, CE Critical Heat Flux, Critical Heat Flux
8 Correlation for CE Fuel Assemblies with Standard Spacer Grids - Part 2, Non-Uniform Axial Power
8 Distribution, June 1976.

9 16. Addendum 1 to BAW-10199P-A, The BWU Critical Heal Flux Correlations, Framatome Cogema
9 Fuels, April 6, 2000.

9 17. BAW-10186P-A, Extended Burnup Evaluation, Framatome Cogema Fuels, June 1997.

9 18. BAW-10229P-A, Mark-B11 Fuel Assembly Design Topical Report, Framatome Cogema Fuels,
9 October 1999.

4.5 REACTOR MATERIALS

9 4.5.1 REACTOR VESSEL INTERNALS

4.5.1.1 Reactor Internal Materials

Reactor internals are fabricated primarily from SA-240 (Type 304) material and designed within the allowable stress levels permitted by the ASME Code, Section III, for normal reactor operation and transients. Structural integrity of all core support assembly circumferential welds is assured by compliance with ASME Code Sections III and IX, radiographic inspection acceptance standards, and welding qualification.

4.5.1.2 Design Bases

The reactor internal components are designed to withstand the stresses resulting from startup; steady state operation with one or more reactor coolant pumps running; and shutdown conditions. No damage to the reactor internals will occur as a result of loss of pumping power.

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

Lateral deflection and torsional rotation of the lower end of the core support assembly is limited in order to prevent excessive deformation resulting from seismic disturbance thereby assuring insertion of control rod assemblies (CRAs). Core drop in the event of failure of the normal supports is limited by guide lugs so that CRAs do not disengage from the fuel assembly guide tubes (Section 4.5.1.3, "Description - Reactor Internals").

The structural internals are designed to maintain their functional integrity in the event of any major loss-of-coolant accident. The dynamic loading resulting from the pressure oscillations because of a loss-of-coolant accident will not prevent CRA insertion.

Internals vent valves are provided to relieve pressure resulting from steam generation in the core following a postulated reactor coolant inlet pipe rupture, so that the core will be rapidly re-covered by coolant.

Allowable Stresses

6 Section 3.9.2.4, "Dynamic System Analysis of the Reactor Internals Under Faulted Conditions (Ref. 19)"
6 describes the stress analysis for fuel assemblies under faulted conditions. Section 3.9.3.1, "Load
7 Combinations, Design Transients and Stress Limits" describes the analysis of the reactor internals.
Additional criteria for stresses due to flow-induced vibratory loads are given in B&W Topical Report "Design of Reactor Internals and Incore Instrument Nozzles for Flow Induced Vibrations," (Reference 1).

Methods of Load Analysis to be Employed for Reactor Internals and Fuel Assembly.

6 Section 3.9.2.4, "Dynamic System Analysis of the Reactor Internals Under Faulted Conditions (Ref. 19)"
6 describes the methods used to analyze fuel assemblies under faulted conditions. Section 3.9.3.1, "Load
7 Combinations, Design Transients and Stress Limits" describes the analysis of the reactor internals.

9 Duke actively participated in a B&W Owners Group effort that developed a series of technical reports
9 whose purpose was to demonstrate that the aging effects for reactor coolant system components are
9 adequately managed for the period of extended operation for license renewal. One of the B&W Owners
9 Group topical reports that was submitted is BAW-2248A [Reference 6] which addresses the reactor vessel
9 internals. Time-limited aging analyses applicable to the Oconee reactor vessel internals are addressed
9 within BAW-2248A. This report was incorporated by reference onto the Oconee dockets [Reference 7].

9 Time-limited aging analyses applicable to the Oconee reactor vessel internals, along with the results of
9 their review for license renewal, are as follows: (1) flow-induced vibration endurance limit assumptions -
9 A review of the existing analysis showed conservatism in the original design, and no further action is
9 needed in the period of extended operation to assure validity of the design; (2) transient cycle count
9 assumptions for the replacement bolting - The ongoing programmatic actions under the Thermal Fatigue
9 Management Program (See Section 5.2.1.4, "Cyclic Loads") will assure the validity of the design
9 assumptions in the period of extended operation; and (3) reduction in fracture toughness - The actions
9 developed as a part of the Reactor Vessel Internals Inspection (See Section 18.3.20, "Reactor Vessel
9 Internals Inspection") will assure the validity of the design assumptions in the period of extended
9 operation. [Reference 8]

4.5.1.3 Description - Reactor Internals

Reactor internal components include the plenum assembly and the core support assembly. The core support assembly consists of the core support shield, vent valves, core barrel, lower grid, flow distributor, incore instrument guide tubes, and thermal shield. The plenum assembly consists of the upper grid plate, the control rod guide assemblies, and a plenum cylinder. Figure 4-26 shows the reactor vessel, reactor vessel internals arrangement, and the reactor coolant flow path. Figure 4-27 shows a cross section through the reactor vessel, and Figure 4-28 shows the core flooding arrangement.

Reactor internal components do not include fuel assemblies, control rod assemblies (CRAs), or incore instrumentation. Fuel assemblies and control rod assemblies are described in Section 4.2.2, "Description - Fuel System Design," control rod drives in Section 4.5.3, "Control Rod Drives," and core instrumentation in Section 7.7.3, "Summary of Alarms."

The reactor internals are designed to support the core, maintain fuel assembly alignment, limit fuel assembly movement, and maintain CRA guide tube alignment between fuel assemblies and control rod drives. They also direct the flow of reactor coolant, provide gamma and neutron shielding, provide guides for in core instrumentation between the reactor vessel lower head and the fuel assemblies, and support the internals vent valves. The vent valves are designed to vent the stream generated within the core, thereby permitting the rapid re-covering of the core by coolant following a reactor coolant inlet pipe rupture. All reactor internal components can be removed from the reactor vessel to allow inspection of the reactor internals and the reactor vessel internal surface.

A shop fitup and checkout of the internal components for Oconee 1 in an as-built reactor vessel mockup insured proper alignment of mating parts before shipment. Dummy fuel assemblies and control rod assemblies were used to check fuel assembly clearances and CRA free movement.

To minimize lateral deflection of the lower end of the core support assembly as a result of horizontal seismic loading, integral weld-attached, deflection-limiting guide lugs are welded on the reactor vessel inside wall. These blocks also limit the rotation of the lower end of the core support assembly which could result from flow-induced torsional loadings. The lugs allow free vertical movement of the lower end of the internals for thermal expansion throughout all ranges of reactor operating conditions. In the unlikely event that a flange, circumferential weld, or bolted joint might fail, the lugs limit the possible core drop to 1/2 in. or less. The elevation plane of these lugs was established near the elevation of the vessel support skirt attachment to minimize dynamic loading effects on the vessel shell or bottom head. A 1/2 in. core drop does not allow the lower end of the CRA rods to disengage from their respective fuel assembly guide tubes, even if the CRAs are in the full-out position. In this rod position, approximately 6-1/2 in. of rod length remains in the fuel assembly guide tubes. A core drop of 1/2 in. does not result in a significant reactivity change. The core cannot rotate and bind the drive lines, because rotation of the core support assembly is prevented by the guide lugs.

The core internals are designed to meet the stress requirements of the ASME Code, Section III, during normal operation and transients. Additional criteria and analysis are given in Reference 1. A detailed stress analysis of the internals under accident conditions has been completed and is reported in B&W Topical Report No. 10008, Part 1 (Reference 2). This report analyzes the internals in the event of a major loss-of-coolant accident (LOCA) and for the combination of LOCA and seismic loadings. It is shown that although there is some internals deflection, failure of the internals does not occur because the stresses are within established limits. These deflections would not prevent CRA insertion because the control rods are guided throughout their travel, and the guide-to-fuel assembly alignment cannot change because positive alignment features are provided between them and the deflections do not exceed allowable values. All core support circumferential weld joints in the internals shells are inspected to the requirements of the ASME Code, Section III.

4.5.1.3.1 Plenum Assembly

The plenum assembly is located directly above the reactor core and is removed as a single component before refueling. It consists of a plenum cover, upper grid, CRA guide tube assemblies, and a flanged plenum cylinder with openings for reactor coolant outlet flow. The plenum cover is constructed of a series of parallel flat plates intersecting to form square lattices and has a perforated top plate and an integral flange at its periphery. The cover assembly is attached to the plenum cylinder top flange. The perforated top plate has matching holes to position the upper end of the CRA guide tubes. The plenum cover is attached to the top flange of the plenum cylinder by a flange. Lifting lugs are provided for remote handling of the plenum assembly. These lifting lugs are welded to the cover grid. The CRA guide tubes are welded to the plenum cover top plate and bolted to the upper grid. CRA guide assemblies provide CRA guidance, protect the CRA from the effects of coolant cross-flow, and provide structural attachment of the grid assembly to the plenum cover.

Each CRA guide assembly consists of an outer tube housing, a mounting flange, 12 perforated slotted tubes and four sets of tube segments which are oriented and attached to a series of castings so as to provide continuous guidance for the CRA full stroke travel. The outer tube housing is welded to a mounting flange, which is bolted to the upper grid. Design clearances in the guide tube accommodate misalignment between the CRA guide tubes and the fuel assemblies. Final design clearances are established by tolerance studies and Control Rod Drive Line Facility (CRDL) prototype test results. The test results are described in Section 4.2.4.4, "Control Rod Drive Tests and Inspection."

The plenum cylinder consists of a large cylindrical section with flanges on both ends to connect the cylinder to the plenum cover and the upper grid. Holes in the plenum cylinder provide a flow path for the coolant water. The upper grid consists of a perforated plate which locates the lower end of the individual CRA guide tube assembly relative to the upper end of a corresponding fuel assembly. The grid is bolted to the plenum cylinder lower flange. Locating keyways in the plenum assembly cover flange engage the reactor vessel flange locating keys to align the plenum assembly with the reactor vessel, the reactor closure head control rod drive penetrations, and the core support assembly. The bottom of the plenum assembly is guided by the inside surface of the lower flange of the core support shield.

4.5.1.3.2 Core Support Assembly

The core support assembly consists of the core support shield, core barrel, lower grid assembly, flow distributor, thermal shield, incore instrument guide tubes, and internals vent valves. Static loads from the assembled components and fuel assemblies, and dynamic loads from CRA trip, hydraulic flow, thermal expansion, seismic disturbances, and loss-of-coolant accident loads are all carried by the core support assembly.

The core support assembly components are described as follows:

1. Core Support Shield

   The core support shield is a flanged cylinder which mates with the reactor vessel opening. The forged top flange rests on a circumferential ledge in the reactor vessel closure flange. The core support shield lower flange is bolted to the core barrel. The inside surface of the lower flange guides and aligns the plenum assembly relative to the core support shield. The cylinder wall has two nozzle openings for coolant flow. These openings are formed by two forged rings, which seal to the reactor vessel outlet nozzles by the differential thermal expansion between the stainless steel core support shield and the carbon steel reactor vessel. The nozzle seal surfaces are finished and fitted to a predetermined cold gap providing clearance for core support assembly installation and removal. At reactor operating temperature, the mating metal surfaces are in contact to make a seal without exceeding allowable stresses in either the reactor vessel or internals. Eight vent valve mounting rings are welded in the cylinder wall for internals vent valves.

2. Core Barrel

   The core barrel supports the fuel assemblies, lower grid, flow distributor, and incore instrument guide tubes. The core barrel consists of a flanged cylinder, a series of internal horizontal former plates bolted to the cylinder, and a series of vertical baffle plates bolted to the inner surfaces of the horizontal formers to produce an inner wall enclosing the fuel assemblies. The core barrel cylinder is flanged on both ends. The upper flange of the core barrel cylinder is bolted to the mating lower flange of the core support shield assembly and the lower flange is bolted to the lower grid assembly. All bolts are lock welded after final assembly. Coolant flow is downward along the outside of the core barrel cylinder and upward through the fuel assemblies contained in the core barrel. A small portion of the coolant flows upward through the space between the core barrel outer cylinder and the inner baffle plate wall. Coolant pressure in this space is maintained lower than the core coolant pressure to avoid tension loads on the bolts attaching the plates to the horizontal formers.

3. Lower Grid Assembly

   The lower grid assembly provides alignment and support for the fuel assemblies, supports the thermal shield and flow distributor, and aligns the incore instrument guide tubes with the fuel assembly instrument tubes. The lower grid consists of two lattice type grid structures, separated by short tubular columns, and surrounded by a forged flanged cylinder. The upper structure is a perforated plate, while the lower structure consists of intersecting plates welded to form a grid. The top flange of the forged cylinder is bolted to the lower flange of the core barrel.

   A perforated flat plate located midway between the two lattice structures aids in distributing coolant flow prior to entrance into the core. Alignment between fuel assemblies and incore instruments is provided by pads bolted to the upper perforated plate.

4. Flow Distributor

   The flow distributor is a perforated dished head with an external flange which is bolted to the bottom flange of the lower grid. The flow distributor supports the incore instrument guide tubes and distributes the inlet coolant entering the bottom of the core.

5. Thermal Shield

   A cylindrical stainless steel thermal shield is installed in the annulus between the core barrel cylinder and reactor vessel inner wall. The thermal shield reduces the incident gamma absorption internal heat generation in the reactor vessel wall and thereby reduces the resulting thermal stresses. The thermal shield upper end is restrained against inward and outward vibratory motion by restraints bolted to the core barrel cylinder. The lower end of the thermal shield is shrunk fit on the lower grid flange and secured by 96 high strength bolts.

6. Incore Instrument Guide Tube Assembly

   The incore instrument guide tube assemblies guide the incore instrument assemblies from the instrument penetrations in the reactor vessel bottom head to the instrument tubes in the fuel assemblies. Horizontal clearances are provided between the reactor vessel instrument penetrations and the instrument guide tubes in the flow distributor to accommodate misalignment. Fifty-two incore instrument guide tubes are provided and are designed so they will not be affected by the core drop described in Section 4.5.1.3, "Description - Reactor Internals."

7. Internals Vent Valves

   Internals vent valves are installed in the core support shield to prevent a pressure imbalance which might interfere with core cooling following a postulated inlet pipe rupture. Under all normal operating conditions, the vent valve will be closed. In the event of the pipe rupture in the cold leg of the reactor loop, the valve will open to permit steam generated in the core to flow directly to the leak, and will permit the core to be rapidly recovered and adequately cooled after emergency core coolant has been supplied to the reactor vessel. The design of the internals vent valve is shown in Figure 4-29 and Figure 4-30.

   Each valve assembly consists of a hinged disc, valve body with sealing surfaces, split-retaining ring, and fasteners. Each valve assembly is installed into a machined mounting ring integrally welded in the core support shield wall. The mounting ring contains the necessary features to retain and seal the perimeter of the valve assembly. Also, the mounting ring includes an alignment device to maintain the correct orientation of the valve assembly for hinged-disc operation. Each valve assembly will be remotely handled as a unit for removal or installation. Valve component parts, including the disc, are of captured design to minimize the possibility of loss of parts to the coolant system, and all operating fasteners include a positive locking device. The hinged-disc includes a device for remote inspection of disc function. Vent valve materials are listed in Table 4-16.

   The vent valve materials were selected on the basis of their corrosion resistance, surface hardness, antigalling characteristics, and compatibility with mating materials in the reactor coolant environment.

   The arrangement consists of eight 14-in. inside diameter vent valve assemblies installed in the cylindrical wall of the internals core support shield (refer to Figure 4-26). The valve centers are coplanar and are 42 in. above the plane of the reactor vessel coolant nozzle centers. In cross section, the valves are spaced around the circumference of the core support shield wall.

   The hinge assembly provides eight loose rotational clearances to minimize any possibility of impairment of disc-free motion in service. In the event that one rotational clearance should bind in service, seven loose rotational clearances would remain to allow unhampered disc free motion. In the worst case, at least four clearances must bind or seize solidly to adversely affect the valve disc free motion.

   In addition, the valve disc hinge loose clearances permit disc self-alignment so that the external differential pressure adjusts the disc seal face to the valve body seal face. This feature minimizes the possibility of increased leakage and pressure-induced deflection loadings on the hinge parts in service.

The external side of the disc is contoured to absorb the impact load of the disc on the reactor vessel inside wall without transmitting excessive impact loads to the hinge parts as a result of a loss-of-coolant accident.

4.5.1.4 Evaluation of Internals Vent Valve

A vapor lock problem could arise if water is trapped in the steam generator blocking the flow of steam from the top of the reactor vessel to a cold leg leak. Under this condition, the steam pressure at the top of the reactor would rise and force the steam bubbles through the water leg in the bottom of the steam generator. This same differential pressure that develops a water leg in the steam generator will develop a water leg in the reactor vessel which could lead to uncovering of the core.

The most direct solution to this problem is to equalize the pressure across the core support shield, thus eliminating the depression of the water level in the core. This was accomplished by installing vent valves in the core support shield to provide direct communication between the top of the core and the coolant inlet annulus. These vent valves open on a very low-pressure differential to allow steam generated in the core to flow directly to the leak from the reactor vessel. Although the flow path in the steam generator is blocked, this is of no consequence since there is an adequate flow path to remove the steam being generated in the core.

During the vent valve conceptual design phase, criteria were established for valves for this service. The design criteria were (1) functional integrity, (2) structural integrity, (3) remote handling capability, (4) individual part capture capability, (5) functional reliability, (6) structural reliability, and (7) leak integrity throughout the design life. The design criteria resulted in the selection of the hinged-disc (swing-disc) check valve, which was considered suitable for further development.

Because of the unique purpose and application of this valve, B&W recognized the need for a complete detailed design and development program to determine valve performance under nuclear service conditions. This program included both analytical and experimental methods of developing data. It was performed primarily by B&W and the selected valve vendor or his subcontractors.

Vent valve preliminary design drawings were prepared and analyzed both by B&W and the vendor/subcontractor. Specifications and drawings were prepared, and orders were placed with the vendor for the design, development, fabrication, and test of a full-size prototype vent valve. The prototype valve was completed and subjected to the tests described in Section 4.5.4, "Internals Tests and Inspections." All testing was successfully completed and minor problems encountered during valve assembly handling or use were corrected to arrive at the final design for the production valve (Reference 4).

The only significant problem encountered during test was seizing of one jack screw. This was attributable to an excessive thickness of "Electrolyze" which spalled off the screw threads. This problem was corrected by reducing the specified "Electrolyze" thickness from 0.0015 in. to 0.0004 in. max. and no further galling was encountered. To further enhance resistance to galling, the final design jackscrew has a 1-1/8 in.-8 Acme thread form instead of a 1 in.-12 UNF and the material is an age hardened corrosion resistant alloy instead of 410 SS.

No further jackscrew problems have occurred or are anticipated on the basis that the surfaces are separated by the low friction "Electrolyze," different materials of different hardnesses are used, loose fits are employed, and thread contact stresses are low (3775 psi).

The final design of this valve is shown in Figure 4-29. The valve disc hangs closed in its natural position to seal against a flat, stainless steel seat inclined 5 degrees from vertical to prevent flow from the inlet coolant annulus to the plenum assembly above the core. In the event of LOCA, the reverse pressure differential will open the valve. At all times during normal reactor operation, the pressure in the coolant annulus on the outside of the core support shield is greater than the pressure in the plenum assembly on the inside of the core support shield. Accordingly, the vent valve will be held closed during normal operation. With four reactor coolant pumps operating, the pressure differential is 42 psi resulting in a several-thousand pound closing force on the vent valve.

Under accident conditions, the valve will begin to open when a pressure differential of less than 0.15 psi develops in a direction opposite to the normal pressure differential. At this point, the opening force on
4 the valve counteracts the natural closing force of the valve. With an opening pressure differential of no greater than 0.3 psi, the valve would be fully open. With this pressure differential, the water level in the core would be above the top of the core. In order for the core to be half uncovered, assuming solid water in the bottom half of the core, a pressure differential of 3.7 psi would have to be developed. This would provide an opening force of about 10 times that required to open the valve completely. This is a conservative limit since it assumes equal density in the core and the annulus surrounding the core. The hot, steam-water mixture in the core will have a density much less than that of the cold water in the annulus, and somewhat greater pressure differentials could be tolerated before the core is more than half uncovered.

An analog computer simulation was developed to evaluate the performance of the vent valves in the core support shield. This analysis demonstrated that adequate steam relief exists so that core cooling will be accomplished.

The behavior of the valve disc during LOCA conditions was investigated and the rather complex dynamic behavior of the disc during LOCA was analyzed as a series of simpler models which provide conservative predictions of peak stresses and deflections.

The valve disc remains closed initially for the LOCA hot leg (36 in. pipe) case and the disc opening on subsequent differential pulses is less than one-half of the initial disc to vessel wall impact velocity for the LOCA cold leg (28 in. pipe) case. Therefore, the disc motion and initial impact with the vessel inside wall was chosen as the worst case and the only one requiring consideration. The cold-leg LOCA pressure time history acting on the disc was approximated by a piecewise linear time function. The momemt due to pressure was equated to the rotary inertia of the disc to determine the velocity of impact with the vessel inside wall.

The model chosen for the initial impact consisted of three effective springs and two masses to represent the disc with its lug, the compliance of the disc, and the vessel inside wall.

Loads generated on impact were based on the conservation of energy. The stresses obtained for these loads indicated that the elastic model assuming conservation of energy was not valid and that the impact must assume plastic deformation. The locations and modes of plastic deformation are illustrated in BAW-10005 (Reference 4).

The plastic analysis provided the following information:

1. Crush deformation of lug after disc corner contacts the vessel wall is predicted to be 0.165 inches.
2. The total deformation of lug from contact with the vessel wall until disc assembly motion is arrested is predicted to be 0.483 inches.
3. The total angular deformation at the plastic hinge is predicted to be 0.016 radians.
4. An analysis was performed on the reactor vessel wall for disc assembly impact and the results indicate that while the stainless steel cladding is deformed locally, the reactor maintains its structural and pressure boundary integrity.

Because of conservative assumptions used in the plastic analysis, actual deformations will be considerably less than the above predicted values. Although plastic deformation may occur as predicted above on impact, the disc will retain its structural integrity. Plastic deformation of the disc dissipates the stored kinetic energy stored in the disc effectively; thus the energy available for rebound is less than 1 percent of the initial impact energy and is too low to overcome the pressure differential and cause impact on the valve body. Disc and body hinge components were analyzed for worst case disc impact loadings and the resulting stresses were found to be less than the allowable limits; therefore, the valve disc free-motion (venting) function will be unaffected.

From the above, it is concluded that vent valve performance will not be impaired during the course of an accident because disc free-motion part stresses remain within allowable limits, disc structural integrity is maintained, vessel pressure boundary integrity is maintained, and plastic deformation of the disc seating surface improves the venting function.

With reference to Figure 4-30, each jackscrew assembly consists of a jackscrew, internally splined mating nut ring, nut ring spring, capture cover and cover attachment fasteners (socket head cap screws). In the figure, the splined nut ring and its spring are hidden from view by the capture cover. The potential for loss of jackscrew assembly parts during the plant lifetime is considered remote on the basis that the jackscrews and capture parts are accessible for visual inspection during scheduled refueling outages. A jackscrew loss is considered remote because a failure in service is highly improbable with the low compressive load (1000 psi) involved and the jack screw is retained in the valve body by a central shoulder and the ends are threaded into the retaining rings. An in-service failure of the splined nut ring and its spring is remote because these parts are subjected to little or no load and even if they did fail all parts would be retained within the capture cover. Capture cover failure and loss is highly improbable on the same basis that is it not loaded in service. The capture cover is attached to the upper retaining ring by socket head cap screws which are lock welded to the cover at installation. By design, these screws are retention rather structural devices and are not loaded in service. These screws do not require a pre-load to hold the formed cover in place; therefore, a loss of pre-load by lock welding would not jeopardize the cover or screw installation or structural integrity. Two fillet welds 180° apart are used to lock weld each screw head to the capture cover and in the absence of loads on both the cover and screws, the likelihood of lock weld failure and loss of screw heads is considered remote. With the capability to inventory these cap screw heads visually at scheduled refuelings, any problem related to the loss of these screws would be apparent early in the plant life and the valve assemblies could be removed for corrective action.

The internals vent valves are described, including materials and hinge part loose clearances in Table 4-17.

The internals vent valves have been tested for ability to withstand the effects of vibratory excitations and for other functional characteristics as described in Section 4.5.4, "Internals Tests and Inspections."

4.5.2 CORE COMPONENTS

This section addresses core components that are not an integral part of the fuel assembly itself. Specifically addressed are the following: control rod assembly, axial power shaping rod assembly, and burnable poison rod assembly.

4.5.2.1 Fuel Assemblies

The fuel system (fuel assembly and its components) is addressed in Section 4.2, "Fuel System Design."

4.5.2.2 Control Rod Assembly (CRA)

Each control rod assembly (Figure 4-31) has 16 control rods, a stainless steel spider, and a female coupling. The 16 control rods are attached to the spider by means of a nut threaded to the upper shank of each rod. After assembly, all nuts are lock welded. The control rod drive is coupled to the CRA by a bayonet type connection. Full length guidance for the CRA is provided by the control rod guide tube of the upper plenum assembly and by the fuel assembly guide tubes. The CRAs and guide tubes are designed with adequate flexibility and clearances to permit freedom of motion within the fuel assembly guide tubes throughout the stroke.

Oconee 3, Cycle 8 introduced a new long life control rod assembly design. Future replacement CRAs for all units will be of this type. The extended life control rod assembly (CRA) is nearly identical to B&W's standard design. The present designed spider/coupling arrangement is retained as are all other envelope dimensions. Reference to Table 4-18, demonstrates the differences between the standard and the plant-life CRA design. The major differences are found in the slight reduction in the absorber OD and the use of Inconel 625 clad (as compared to the standard SS 304 material). Inconel 625 CRA cladding was selected because of its added creep and corrosion resistance. In addition, the rodlets are prepressurized with helium, and the cladding is slightly thicker to retard creepdown and ovalization.

Each control rod has a section of neutron absorber material. The absorber material is an alloy of silver-indium-cadmium. End pieces are welded to the tubing to form a water-tight and pressure-tight container for the absorber material.

Both the inconel and the stainless steel tubing provide the structural strength of the control rods and
8 prevents corrosion of the absorber material. A tube spacer similar to the type used in fuel assemblies is used to prevent absorber motion within the cladding during shipping and handling, and to permit differential expansion in service.

These control rods are designed to withstand all operating loads including those resulting from hydraulic force, thermal gradients, and reactor trip deceleration. The ability of the control rod clad to resist collapse has been established in a test program on cold-worked stainless steel tubing. Because the Ag-In-Cd alloy poison does not yield a gaseous product under irradiation, internal pressure and swelling of the absorber material does not cause excessive stressing or stretching of the clad.

Because of their length and the possible lack of straightness over the entire length of the rod, some interference between control rods and the fuel assembly guide tubes is expected. However, the parts involved, especially the control rods, are flexible and only small friction drag loads result. Similarly, thermal distortions of the control rods are small because of the low heat generation and adequate cooling. Consequently, control rod assemblies do not encounter significant frictional resistance to their motion in the guide tubes.

4.5.2.3 Axial Power Shaping Rod Assembly (APSRA)

Gray APSR's are provided for additional control of axial power distribution. Each axial power shaping rod assembly (Figure 4-32) has 16 axial power shaping rods, a stainless steel spider, and a female coupling. The 16 rods are attached to the spider by means of a nut threaded to the upper shank of each rod. After assembly all nuts are lock welded. The axial power shaping rod drive is coupled to the APSRA by a bayonet connection. The female couplings of the APSRA and CRA have slight dimensional differences to ensure that each type of rod can only be coupled to the correct type of drive mechanism.

6 There are 2 APSR designs which are not fully interchangeable between fuel assembly designs, because of
6 the difference in hold down spring designs and APSR drive mechanisms. Table 4-22 depicts the APSR
6 and fuel assembly compatibility for each unit.

When the APSRA is inserted into the fuel assembly it is guided by the guide tubes of the fuel assembly. Full length guidance of the APSRA is provided by the control rod guide tube of the upper plenum assembly. At the full out position of the control rod drive stroke, the lower end of the APSRA remains within the fuel assembly guide tube to maintain the continuity of guidance throughout the rod travel length. The APSRAs are designed to permit maximum conformity with the fuel assembly guide tube throughout travel.

Each axial power shaping rod has a section of neutron absorber material. For these gray APSRs, this absorber material is Inconel 600, and the clad is coldworked, Type 304 stainless steel tubing. The tubing provides the structural strength of the axial power shaping rods and prevents corrosion of the absorber material.

Gray APSRs are designed with improved creep life. Cladding thickness and rod ovality control, which are the primary factors controlling the creep life of a stainless steel material, have been improved to extend the
5 creep life of the gray APSR. Minimum design cladding thickness is 25 mils.

5 The gray APSRs are prepressurized to extend their lifespan.

Pertinent data on gray APSRs is shown in Table 4-19.

These axial power shaping rods are designed to withstand all operating loads including those resulting from hydraulic forces and thermal gradients. The ability of the axial power shaping rod clad to resist collapse due to the system pressure has been established in a test program on cold worked stainless steel tubing. The absorber material does not yield gaseous products under irradiation, therefore, internal pressure is not generated within the clad. Swelling of the absorber material is negligible, and does not cause unacceptable clad strain.

Because of their great length and unavoidable lack of straightness, some slight mechanical interference between axial power shaping rods and the fuel assembly guide tubes must be expected. However, the parts involved are flexible and result in very small friction drag loads. Similarly, thermal distortions of the rods are small because of the low generation and adequate cooling. Consequently, the APSRAs do not encounter significant frictional resistance to their motion in the guide tubes.

4.5.2.4 Burnable Poison Rod Assembly (BPRA)

Each BPRA (Figure 4-1) has 16 burnable poison rods, a stainless steel spider, and a coupling mechanism. The coupling mechanism and the 16 rods are attached to the spider. The BPRA is inserted into the fuel assembly guide tubes through the upper end fitting. Retention is provided by the feet on the BPRA spider, which rest upon the fuel assembly holddown spring retainer ring. Thus the BPRA is pinned between this retainer ring and the reactor's upper grid pads. All Oconee fuel which is of the Mk B5 (or later) design, uses this BPRA design.

The burnable poison rod is clad in cold-worked Zircaloy-4 tubing and Zircaloy-4 upper and lower end pieces. The end pieces are welded to the tubing to form a water and pressure-tight container for the absorber material. The Zircaloy-4 tubing provides the structural strength of the burnable poison rods.

In addition to their nuclear function, the BPRA also serve to minimize guide tube bypass coolant flow. Pertinent data on the BPRA is shown in Table 4-20.

The burnable poison rods are designed to withstand all operating loads including those resulting from hydraulic forces and thermal gradients. The ability of the burnable poison rod clad to resist collapse due to the system pressure and internal pressure has been demonstrated by an extensive test program on cold-worked Zircaloy-4 tubing (Section 4.2.4.3.1, "Fuel Rod Cladding").

9 A tubular spacer is used in conjunction with a spacer spring at the top of the poison stack to control the
9 poison pellet motion with the cladding during shipping and handling and to allow for thermal expansion
9 and swelling during cycle operation.

4.5.3 CONTROL ROD DRIVES

9 Oconee 3 uses the Type C control rod drive mechanism. Oconee Units 1 & 2 use Type A and Type C mechanisms. Both types are sealed, reluctance motor-driven screw units and the design requirements are identical. Section 4.5.3.1, "Type A Mechanisms" describes Type A mechanisms, and Section 4.5.3.2, "Type C Mechanisms" describes Type C mechanisms.

4.5.3.1 Type A Mechanisms

The control rod drive mechanism (CRDM) positions the control rod within the reactor core, provides for controlled withdrawal or insertion of the control rod assemblies, is capable of rapid insertion or trip, and indicates the location of the control rod with respect to the reactor core. The speed at which the control rod is inserted or withdrawn from the core is consistent with the reactivity change requirements during reactor operation. For conditions that require a rapid shutdown of the reactor, the shim safety drive mechanism releases the CRA and supporting CRDM components permitting the CRA to move by gravity into the core. The reactivity is reduced during such a rod insertion at a rate sufficient to control the core under any operating transient or accident condition. The control rod is decelerated at the end of the rod trip insertion by a buffer assembly in the CRDM upper housing. The buffer assembly supports the control rod in the fully inserted position. The CRDM data is listed in Table 4-21, and criteria applicable to drive mechanisms for both control shim rod assemblies and axial power shaping rod assemblies are given below. Additional requirements for the mechanisms which actuate only control shim rod assemblies are also given below.

4.5.3.1.1 General Design Criteria

1. Single Failure

   No single failure shall inhibit the protective action of the control rod drive system. The effect of a single failure shall be limited to one CRDM.

2. Uncontrolled Withdrawal

   No single failure or sequence of dependent failures shall cause uncontrolled withdrawal of any control rod assembly (CRA).

3. Equipment Removal

   The disconnection of plug-in connectors, modules, and subassemblies from the protective circuits shall be annunciated or shall cause a reactor trip.

4. Position Indication

   Continuous position indication, as well as an upper and lower position limit indication, shall be provided for each CRDM. The accuracy of the position indicators shall be consistent with the tolerance set by reactor safety analysis.

5. Drive Speed

9 The control rod drive control system shall provide two uniform mechanism speeds. The drive
controls, or mechanism and motor combination, shall have an inherent speed limiting feature. The
9 speed of the mechanism shall be 30 in./min for both insertion and withdrawal in the "Run" mode of
9 control. The withdrawal speed shall be limited to not exceed 25 percent overspeed in the event of
9 speed control fault. The speed of the mechanism shall be 3 in./min for both insertion and withdrawal
9 in the "Jog" mode of control.

6. Mechanical Stops

   Each CRDM shall have positive mechanical stops at both ends of the stroke or travel. The stops shall be capable of receiving the full operating force of the mechanisms without failure.

7. Control Rod Positioning

   The control rod drives shall provide for controlled withdrawal or insertion of the control rods out of, or into, the reactor core to establish and hold the power level required.

4.5.3.1.2 Additional Design Criteria

The following criterion is applicable only to the mechanisms which actuate control rod assemblies: The shim safety drives are capable of rapid insertion or trip for emergency reactor conditions.

4.5.3.1.3 Shim Safety Drive Mechanism

The shim safety drive mechanism consists of a motor tube which houses a lead screw and its rotor assembly, and a buffer. The top end of the motor tube is closed by a closure and vent assembly. An external motor stator surrounds the motor tube (a pressure housing) and position indication switches are arranged outside the motor tube extension.

The control rod drive output element is a non-rotating translating lead screw coupled to the control rod. The screw is driven by separating anti-friction roller nut assemblies which are rotated magnetically by a motor stator located outside the pressure boundary. Current impressed on the stator causes the separating roller nut assembly halves to close and engage the lead screw. Mechanical springs disengage the roller nut halves from the screw in the absence of a current. For rapid insertion, the nut halves separate to release the screw and control rod, which move into the core by gravity. A hydraulic buffer assembly within the upper housing decelerates the moving CRA to a low speed a short distance above the CRA full-in position. The final CRA deceleration energy is absorbed by the down-stop buffer spring. The CRDM is a totally sealed unit with the roller nut assemblies magnetically driven by the stator coil through the motor tube pressure housing wall. The lead screw assembly is connected to the control rod by a bayonet type coupling. An anti-rotation device (torque taker) prevents rotation of the lead screw while the drive is in service. A closure and vent assembly is provided at the top of the motor tube housing to permit access to couple and release the lead screw assembly from the control rod. The top end of the lead screw assembly is guided by the buffer piston and its guide. Two of the six phase stator housing windings are energized to maintain the control rod position when the drive is in the holding mode.

4.5.3.1.4 CRDM Subassemblies

9 The CRDM is shown in Figure 4-34. Subassemblies of the CRDM are described as follows:

1. Motor Tube

   The motor tube is a three-piece welded assembly designed and manufactured in accordance with the requirements of the ASME Code, Section III, for Class A nuclear pressure vessel. Materials conform to ASTM or ASME, Section II, Material Specifications. All welding shall be performed by personnel qualified under ASME Code, Section IX, Welding Qualifications. The motor tube wall between the rotor assembly and the stator is constructed of magnetic material to present a small air gap to the motor. This region of the motor tube is of low alloy steel clad on the inside diameter with stainless steel or with Inconel. The upper end of the motor tube functions only as a pressurized enclosure for the withdrawn lead screw and is made of stainless steel transition welded to the upper end of the low alloy steel motor section. The lower end of the low alloy steel tube section is welded to a stainless steel machined forging which is flanged at the face which contacts the vessel control rod nozzle. Double gaskets, which are separated by a ported test annulus, seal the flanged connection between the motor tube and the reactor vessel.

2. Motor

The motor is a synchronous reluctance unit with a slip-on stator. The rotor assembly is described in 6 below. The stator is a 48-slot four-pole arrangement with water cooling coils wound on the outside of its casing. The stator is encapsulated after winding to establish a sealed unit. It is six phase star-connected for operation in a pulse-stepping mode and advances 15 mechanical degrees per step. The stator assembly is mounted over the motor tube housing as shown in Figure 4-34.

3. Plug and Vent Valve

The upper end of the motor tube is closed by a closure insert assembly containing a vapor bleed port and vent valve. The vent valve and insert closure have double seals. The insert closure is retained by a closure nut which is threaded to the inside of the motor tube. The sealing for the closure is applied by jackscrews threaded through the closure nut.

4. Actuator

The actuator consists of the translating lead screw, its rotating nut assembly, and the torque taker assembly on the screw. The actuator lead screw travel is 139 inches.

5. Lead Screw

The lead screw has a lead of 0.750 in. The thread is double lead with a single pitch spacing of 0.375 in. Thread lead error is held to close tolerances for uniform loading with the roller nut assemblies. The thread form is a modified ACME with a blank angle that allows the roller nut disengage without lifting the screw.

6. Rotor Assembly

The rotor assembly consists of a ball bearing supported rotor tube carrying and limiting the travel of a pair of scissors arms. Each of the two arms carry a pair of ball bearing supported roller (nut) assemblies which are skewed at the lead screw helix angle for engagement with the lead screw. The current in the motor stator (two of a six winding stator) causes the arms that are pivoted in the rotor tube to move radially toward the motor tube wall to the limit provided thereby engaging the four roller nuts with the centrally located lead screw. Also, four separating springs mounted in the scissor arms keep the rollers disengaged when the power is removed from the stator coils. A second radial bearing mounted to the upper end of the rotor tube has its outer race pinned to both scissor arms thereby synchronizing their motion during engagement and disengagement. When a three phase rotating magnetic field is applied to the motor stator, the resulting force produces rotor assembly rotation.

7. Torque Extension Tube and Torque Taker

The torque extension tube is a separate tubular assembly containing a keyway that extends the full length of the lead screw travel. The tube assembly is supported against rotation and in elevation by the upper end of the motor tube extension. The lower end of the tube assembly supports the buffer and is the down stop. A set of indexing serrations mate to prevent rotation and orient the torque extension tube with the motor tube below the cap and vent valve assembly. An integral shoulder at the top of the tube rests against a step in the motor tube inside diameter to provide a vertical support.

The torque taker assembly consists of the position indicator permanent magnet, the buffer piston, and a positioning key. The torque taker key fixed at the top of the lead screw is mated with the torque extension tube keyway to provide both radial and tangential positioning of the lead screw.

8. Buffer

The buffer assembly is capable of decelerating the translating mass from the unpressurized terminal velocity to zero velocity without applying greater than ten times the gravitational force on the control rod. The water buffer consists of a piston fixed to the top end of the screw shaft and a cylinder which is fixed to the lower end of the torque extension tube. Twelve inches above the bottom stop, the piston at the top of the screw enters the cylinder. Guiding is accomplished because the piston and torque key are in a single part, and cylinder and keyway are in a single mating part. As the piston travels into the cylinder, water is driven into the center of the lead screw through holes in the upper section which produce the damping pressure drop. The number of holes presented to the buffer chamber is reduced as the rod moves into the core, so that the damping coefficient increases as the velocity reduces, thereby providing an approximately uniform deceleration. A large helical buffer spring is used to take the kinetic energy of the drive line at the end of the water buffer stroke. The buffer spring accepts a five foot per second impact velocity of the drive line and control rod with an instantaneous overtravel of one inch past the normal down stop. The inclusion of this buffer spring permits practical clearances in the water buffer.

9. Lead Screw Guide

The lead screw guide bushing acts as a primary thermal barrier and as a guide for the screw shaft. As a primary thermal barrier, the bushing allows only a small path for free convection of water between the mechanism and the closure head nozzle. Fluid temperature in the mechanism is largely governed by the flow of water up and down through this bushing. The diametrical clearance between screw shaft and bushing is large enough to preclude jamming the screw shaft and small enough to hold the free convection to an acceptable value. In order to obtain trip travel times of acceptably small values, it is necessary to provide an auxiliary flow path around the guide bushing. The larger area path is necessary to reduce the pressure differential required to drive water into the mechanism to equal the screw displacement. The auxiliary flow paths are closed for small pressure differentials (several inches of water) by ball check valves which prevent the convection flow but, open fully during trip.

10. Position Indications

Two methods of position indication are provided: an absolute position indicator and a relative position indicator. The absolute position transducer consists of a series of magnetically operated reed switches mounted in a tube parallel to the motor tube extension. Each switch is hermetically sealed. Switch contacts close when a permanent magnet mounted on the upper end of the lead screw extension comes in close proximity. As the lead screw (and the control rod assembly) moves,
2 switches operate sequentially producing an analog voltage proportional to position. Additional reed
switches are included in the same tube with the absolute position transducer to provide full
withdrawal and insertion signals. The relative position indicator consists of a small pulse-stepping
2 motor driving a potentiometer that generates a signal proportional to the position demand for the rod
2 as indicated by the number of power pulses received by the rod drive motor.

11. Motor Tube Design Criteria

The motor tube design complies with Section III of the ASME Boiler and Pressure Vessel Code for a Class A vessel. The operating transient cycles, which are considered for the stress analysis of the reactor pressure vessel, are also considered in the motor tube design.

Quality standards relative to material selection, fabrication, and inspection are specified to insure safety function of the housings essential to accident prevention. Materials conform to ASTM or ASME, Section II, Material Specifications. All welding shall be performed by personnel qualified under ASME Code, Section IX, Welding Qualifications. These design and fabrication procedures establish quality assurance of the assemblies to contain the reactor coolant safely at operating temperature and pressure.

In the highly unlikely event that a pressure barrier component or the control rod drive assembly does fail catastrophically, i.e., ruptured completely, the following results would ensue:

a. Control Rod Drive Nozzle

The assembly would be ejected upward as a missile until it was stopped by the missile shield over the reactor. This upward motion would have no adverse effect on adjacent assemblies.

b. Motor Tube

The failure of this component anywhere above the lower flange would result in a missile-like ejection into the missile shielding over the reactor. This upward motion would have no adverse effect on adjacent mechanisms.

12. Axial Power Shaping Rod Drive

For actuating the partial length control rods which maintain their set position during a reactor-trip of the shim safety drive, the CRDM is modified so that the roller nut assembly will not disengage from the lead screw on a loss of power to the stator. Except for this modification, the shim drives and the axial power shaping rod drives are identical.

4.5.3.2 Type C Mechanisms

4.5.3.2.1 Shim Safety Drive Mechanism

The Type C shim safety drive mechanism consists of a motor tube which houses a torque tube, a leadscrew, its rotor assembly, and a snubber assembly. The top end of the motor tube is closed by a closure and vent assembly. An external motor stator surrounds the motor tube (a pressure housing) and position indication switches are arranged outside the motor tube extension.

4.5.3.2.2 CRDM Subassemblies

Those parts of the Type C CRDM subassemblies which are different from the Type A CRDM (Section 4.5.3.1.4, "CRDM Subassemblies") are described below:

1. Motor Tube

That portion of the motor tube wall between the rotor assembly and the stator is constructed of martensitic stainless steel.

2. Motor

The stator is a 48-slot four-pole arrangement with water cooling coils in the outside casing. The stator is varnish impregnated after winding to establish a sealed unit.

3. Plug and Vent Valve

9 For the Units 1 & 2 Type C CRDMs the upper end of the motor tube is closed by a closure insert
9 assembly containing a vapor bleed port and vent valve. The vent valve and closure have double seals.
9 The insert closure is retained by a closure nut which is threaded to the inside of the motor tube. The
9 sealing for the closure is applied by hydraulically tensioning the closure insert and is retained by the
9 closure nut.

9 For the Unit 3 Type C CRDMs refer to Section 4.5.3.1.4, "CRDM Subassemblies."

4. Actuator

Refer to Section 4.5.3.1.4, "CRDM Subassemblies."

5. Lead Screw

   Refer to Section 4.5.3.1.4, "CRDM Subassemblies."

6. Rotor Assembly

   Refer to Section 4.5.3.1.4, "CRDM Subassemblies."

7. Torque Tube and Torque Taker

   The torque tube is a separate tubular assembly containing a key that extends the full length of the leadscrew travel. The tube assembly is secured in elevation and against rotation at the lower end of the closure assembly by a retaining ring, keys and the insert closure. The lower end of the torque tube houses the snubber assembly and is the down stop. The leadscrew contacts the insert closure assembly for the upper mechanical stop.

   The torque taker assembly consists of the position indicator permanent magnet, the snubber piston and a positioning keyway. The torque taker assembly is attached to the top of the leadscrew and has a keyway that mates with the key in the torque tube to provide both radial and tangential positioning of the leadscrew.

8. Snubber Assembly

   The total snubber assembly is composed of a piston that is the lower end of the torque taker assembly and a snubber cylinder and belleville spring assembly which is attached to the lower end of the torque tube. The snubber cylinder is closed at the bottom by the snubber bushing and leadscrew. The snubber cylinder has a twelve-inch active length in which the free-fall tripped leadscrew and control rod assembly is decelerated without applying greater than ten times gravitational force on the control rod. The damping characteristics of the snubber is determined by the size and position of a number of holes in the snubber cylinder wall and the leakage at the snubber piston and bushing. Leakage reduction at the snubber piston and bushing can only be reduced to a minimum amount caused by practical operating clearances. Therefore, at the end of the snubbing stroke, there is kinetic energy from a five foot per second impact velocity that is absorbed by the belleville spring assembly by a slight instantaneous overtravel past the normal down stop.

9. Lead Screw Guide

   Refer to Section 4.5.3.1.4, "CRDM Subassemblies."

10. Position Indications

    Refer to Section 4.5.3.1.4, "CRDM Subassemblies."

11. Motor Tube Design Criteria

    Refer to Section 4.5.3.1.4, "CRDM Subassemblies."

9 The Type C mechanism is presented in Figure 4-34 and is described in Reference 5.

4.5.4 INTERNALS TESTS AND INSPECTIONS

4.5.4.1 Reactor Internals

The hydraulic design of the upper and lower plena of the internals is evaluated and guided by the results from the 1/6 scale model flow test which is described in Section 4.4.4, "Thermal and Hydraulic Tests and Inspection." These test results have guided the design to obtain minimum flow maldistribution, and the test data allowed verification of vessel flow and pressure drop.

The effects of internals misalignment was evaluated on the basis of the test results from the CRDL tests
7 described in Section 4.2.4, "Fuel Assembly, Control Rod Assembly, and Control Rod Drive Mechanical
7 Tests and Inspection." These test results, correlated with the internals guide tube design, insure that the CRA can be inserted at specified rates under conditions of maximum misalignment.

Internals shop fabrication quality control tests, inspection, procedures, and methods are similar to those for the pressure vessel described in Section 5.2.3.11, "Quality Assurance." The internals surveillance specimen holder tubes and the material irradiation program are described in Section 5.2.3.13, "Reactor Vessel Material Surveillance Program."

A listing is included herewith for all internals nondestructive examinations and inspections with applicable codes or standards applicable to all core structural support material of various forms. In addition, one or more of these examinations are performed on materials or processes which are used for functions other than structural support (i.e. alignment dowels, etc.) so that virtually 100 percent of the completed internals materials and parts are included in the listing. Internals raw materials are purchased to ASME Code Section II or ASTM material specifications. Certified material test reports are obtained and retained to substantiate the material chemical and physical properties. All internals materials are purchased and obtained to a low cobalt limitation. The ASME Code Section III, as applicable for Class A vessels, is generally specified as the requirement for reference level nondestructive examination and acceptance. In isolated instances when ASME III cannot be applied, the appropriate ASTM Specifications for non-destructive testing are imposed. All welders performing weld operations on internals are qualified in accordance with ASME Code Section IX applicable Edition and Addenda. The primary purpose of the following list of non-destructive tests is to locate, define, and determine the size of material defects to allow an evaluation of defect, acceptance, rejection, or repair. Repaired defects are similarly inspected as required by applicable codes.

4.5.4.1.1 Ultrasonic Examination

1. Wrought or forged raw material forms are 100 percent inspected throughout the entire material volume to ASME III, Class A.
2. Personnel conducting these examinations are trained and qualified.

4.5.4.1.2 Radiographic Examination (includes X-ray or radioactive sources)

1. Cast raw material forms are 100 percent inspected to ASME III Class A or ASTM.
2. All circumferential full penetration structural weld joints which support the core are 100 percent inspected to ASME III Class A.
3. All radiographs are reviewed by qualified personnel who are trained in their interpretation.

4.5.4.1.3 Liquid Penetrant Examination

1. Cast form raw material surfaces are 100 percent inspected to ASME III Class A or ASTM.
2. Full penetration non-radiographic or partial penetration structural welds are inspected by examination of root, and cover passes to ASME III Class A.
3. All circumferential full penetration structural weld joints which support the core have cover passes inspected to ASME III Class A.
4. Personnel conducting these examinations are trained and qualified.

4.5.4.1.4 Visual (5X Magnification) Examination

This examination is performed in accordance with and results accepted on the basis of a B&W Quality Control Specification which complies with NAV-SHIPS 250-1500-1. Each entire weld pass and adjacent base metal are inspected prior to the next pass from the root to and including the cover passes.

1. Partial penetration non-radiographically or non-ultrasonically feasible structural weld joints are 100 percent inspected to the above specification.
2. Partial or full penetration attachment weld joints for nonstructural materials or parts are 100 percent inspected to the above specification.
3. Partial or full penetration weld joints for attachment of mechanical devices which lock and retain structural fasteners.
4. Personnel conducting these examinations are trained and qualified.

After completion of shop fabrication, the internals components are shopfitted and assembled to final design requirements. The assembled internals components undergo a final shop fitting and alignment of the internals with the "as built" dimensions of the reactor vessel. Dummy fuel and CRAs are used to insure that ample clearances exist between the fuel and internals structures guide tubes to allow free movement of the CRA throughout its full stroke length in various core locations. Fuel assembly mating fit is checked at all core locations. The dummy fuel and CRAs are identical to the production components except that they are manufactured to the most adverse tolerance space envelope, and they contain no fissionable or absorber materials.

All internal components can be removed from the reactor vessel to allow inspection of all vessel interior surfaces. Internals components surfaces can be inspected when the internals are removed to the canal underwater storage location.

4.5.4.2 Internals Vent Valves Tests and Inspection

The internals vent valves are designed to relieve the pressure generated by steaming in the core following a LOCA so that the core will remain sufficiently cooled. The valves were designed to withstand the forces resulting from rupture of either a reactor coolant inlet or outlet pipe. To verify the structural adequacy of the valves to withstand the pressure forces and perform the venting function, the following tests were performed:

1 4.5.4.2.1 Hydrostatic Testing

A full-size prototype valve assembly (valve disc retaining mechanism and valve body) was hydrostatically tested to the maximum pressure expected to result during the blowdown.

1 4.5.4.2.2 Frictional Load Tests

Sufficient tests were conducted at zero pressure to determine the frictional loads in the hinge assembly, the inertia of the valve disc, and the disc rebound resulting from impact of the disc on the seat so that the valve response to cyclic blowdown forces may be determined analytically.

1 4.5.4.2.3 Pressure Testing

A prototype valve was pressurized to determine the pressure differential required to cause the valve disc to begin to open. A determination of the pressure differential required to open the valve disc to its maximum open position was simulated by mechanical means.

1 4.5.4.2.4 Handling Test

A prototype valve assembly was successfully installed and removed remotely in a test stand to confirm the adequacy of the vent valve handling tool.

1 4.5.4.2.5 Closing Force Test

A 1/6 scale model valve disc closing force (excluding gravity) test is described in Section 4.4.4, "Thermal and Hydraulic Tests and Inspection."

1 4.5.4.2.6 Vibration Testing

The full-size prototype valve's response to vibration was determined experimentally to verify prior analytical results which indicated that the valve disc would not move relative to the body seal face as a result of vibration caused by transmission of core support shield vibrations. The prototype valve was mounted in a test fixture which duplicated the method of valve mounting in the core support shield. The test fixture with valve installed was attached to a vibration test machine and excited sinusoidally through a range of frequencies which encompassed those which may reasonably be anticipated for the core support shield during reactor operation. The relative motion between the valve disc and seat was monitored and recorded during test. The test results indicated that there was no relative motion of the valve to its seat for conditions simulating operating conditions. After no relative motion was observed or recorded during test, the valve disc was manually forced open during test to observe its response. The disc closed with impact on its seat, rebounded open and reseated without any adverse affects to valve seal surfaces, characteristics, or performance. From this oscillograph record, the natural frequency of the valve disc was conservatively calculated as approximately 1500 cps; whereas, the range of frequencies for the Oconee system (including internals components) has been established as 15 to 160 cps.

These frequencies are separated by an ample margin to conclude that no relative motion between the valve disc and its seal will occur during normal reactor operation.

9 4.5.4.2.7 Production Valve Testing

6 Each production valve will be subjected to tests described in Sections 4.5.4.2.2, "Frictional Load Tests"
6 and 4.5.4.2.3, "Pressure Testing" except that no additional analysis will be performed in conjunction with
6 the test described in Section 4.5.4.2.2, "Frictional Load Tests."

The valve disc, hinge shaft, shaft journals (bushings), disc journal receptacles, and valve body journal receptacles are designed to withstand without failure the internal and external differential pressure loadings resulting from a loss-of-coolant accident. These valve materials will be nondestructively tested and accepted in accordance with the ASME Code III requirements for Class A vessels as a reference quality level.

9 4.5.4.2.8 Subsequent Operations

During scheduled refueling outages after the reactor vessel head and the internals plenum assembly have
been removed, the vent valves are accessible for visual and mechanical inspection. A hook tool is
9 provided to engage with the valve disc exercise lug described in Item 7 of Section 4.5.1.3.2, "Core Support
9 Assembly." With the aid of this tool, the valve disc will be manually exercised to evaluate the disc
9 freedom. The hinge design incorporates special features, as described in Item 7 of Section 4.5.1.3.2, "Core
9 Support Assembly" to minimize the possibility of valve disc motion impairment during its service life.
With the aid of the hook tool, the valve disc can be raised and a remote visual inspection of the valve
body and disc sealing faces can be performed for evaluation of observed surface irregularities.

Remote installation and removal of the vent valve assemblies if required is performed with the aid of the vent valve handling tool which includes unlocking and operating features for the retaining ring jackscrews.

An inspection of hinge parts is not planned until such time as a valve assembly is removed because its free-disc motion has been impaired. In the unlikely event that a hinge part should fail during normal operation, the most significant indication of such a failure would be a change in the free-disc motion as a result of altered rotational clearances.

4.5.5 REFERENCES

1. E. O. Hooker, H. J. Fortune, "Design of Reactor Internals and Incore Instrument Nozzles for Flow Induced Vibrations," B&W, *BAW-10051*, Lynchburg, VA., September 1972.

7 2. BAW-10008, Part 1, Rev. 1, Reactor Internals Stress and Deflection due to Loss-of-Coolant Accident
7 and Maximum Hypothetical Earthquake, June 1970.

7 3. Deleted Per 1997 Update

4. "Internals Vent Valve Evaluation," B&W, BAW-10005, Revision 1, Lynchburg, VA., June 1970.

5. James T. Williams, R. E. Harris, John Ficor, "Control Rod Drive Mechanism," B&W, *BAW-10029A*, *Revision 3*, Lynchburg, VA., August 1976.

9 6. *Demonstration of the Management of Aging Effects for the Reactor Vessel Internals*, BAW-2248A, The
9 B&W Owners Group Generic License Renewal Program, March 2000.

9 7. *Application for Renewed Operating Licenses for Oconee Nuclear Station, Units 1, 2, and 3*, submitted
9 by M. S. Tuckman (Duke) letter dated July 6, 1998 to Document Control Desk (NRC), Docket Nos.
9 50-269, 50-270, and 50-287.

9 8. NUREG-1723, *Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station,*
9 *Units 1, 2, and 3*, Docket Nos. 50-269, 50-270, and 50-287.

THIS IS THE LAST PAGE OF THE CHAPTER 4 TEXT PORTION

APPENDIX 4. CHAPTER 4 TABLES AND FIGURES

9 **Table 4-1 (Page 1 of 4). Core Design, Thermal, and Hydraulic Data**

	Reactor	
	Rated Heat Output, MWt	2,568
9	Vessel Coolant Inlet Temperature, 100% power, F	557.8
9	Vessel Coolant Outlet Temperature, 100% power, F	602.4
9	Core Outlet Temperature, 100% power	606.2
9	Core Operating Pressure, psia	2200
9	Reactor Coolant flow, % design flow	107.5
9	**Note:** The following parameters specified below are based on the fuel assembly nomenclature.	
9	Core and Fuel Assemblies (Note 1)	
	Total Number of Fuel Assemblies in Core	177
	Number of Fuel Rods per Fuel Assembly	208
	Number of Control Rod Guide Tubes per Assembly	16
	Number of In-Core Instrumentation Positions per Fuel Assembly	1
	Fuel Rod Outside Diameter, in.	
9	Mk-B10	0.430
9	Mk-B11	0.416
	Clad Thickness, in.	
9	Mk-B10 to B10E	0.0265
7	Mk B-10F, MK B-10G, and MK B-10L	0.0250
9	Mk-B11	0.0240
	Fuel Rod Pitch, in.	0.568
	Fuel Assembly Pitch Spacing, in.	8.587
9	Fuel Assembly Overall Length (Typical), in.	
9	Mk-B20 to B10L and Mk-B11	165.695
	Unit Cell Metal/Water Ratio (Volume Basis)	0.82
9		
	Fuel	
	Material	UO_2
	Form	Dished-End, Cylindrical Pellets

9 **Table 4-1 (Page 2 of 4). Core Design, Thermal, and Hydraulic Data**

1	Pellet Diameter, in.	
9		
9	MK B-10 to B-10E	0.3700
7	MK B-10F, MK B-10G, and MK B-10L	0.3735
9	Mk-B11	0.3615
1	Active Length, in.	
9		
9		
9	MK B10 to B-10E	140.5 - 140.7
9		
9		
7	MK B-10F, MK B-10G, and MK B-10L	142.3
9	Mk-B11	143.05
	Density, % of Theoretical	
9	MK B-10 to B-10E	95.0
7	MK B-10F, MK B-10G, and MK B-10L	96.0
9	Mk-B11	96.0
9	Heat Transfer and Fluid Flow at Rated Power (Note 2)	
9	Total Heat Transfer Surface in Core, ft^2	
9	Mk-B10 to B-10E	48,525
9	Mk-B10F, Mk-B10G and Mk-B10L	49,147
9	Mk-B11	47,797
9	Average Heat Flux, Btu/hr-ft^2	
9	MK-B10 to B-10E	175.7×10^3
9	Mk-B10F, Mk-B10G and Mk-B10L	173.5×10^3
9	Mk-B11	178.4×10^3
9	Maximum Heat Flux, Btu/hr-ft^2	
9	Mk-B10 to B-10E	452×10^3
9	Mk-B10F, Mk-B10G and Mk-B10L	446×10^3
9	Mk-B11	458×10^3
9	Average Power Density in Core, kW/ℓ	
9	Mk-B10 to B-10E	85.46
9	Mk-B10F, Mk-B10G and Mk-B10L	84.38

9 **Table 4-1 (Page 3 of 4). Core Design, Thermal, and Hydraulic Data**

9	Mk-B11	83.94
9	Average Thermal Output, kW/ft of Fuel Rod	
9	Mk-B10, Mk-B10D, and Mk-B10E	5.8
9	Mk-B10F, Mk-B10G, Mk-B10L and Mk-B11	5.7
9	Maximum Thermal Output, kW/ft of Fuel Rod	
9	Mk-B10, Mk-B10D and Mk-B10E	14.9
9	Mk-B10F, Mk-B10G and Mk-B10L	14.7
9	Mk-B11	14.6
9	Average Core Fuel Temperature, F	
9	Mk-B10,Mk-B10D and Mk-B10E	1215
9	Mk-B10F, Mk-B10G and Mk-B10L	1162
9	Mk-B11	1175
9	Total Reactor Coolant Flow, lb/hr	141.0×10^6
9	Core Flow Area (Effective for Heat Transfer), ft^2	
9	Mk-B10 though Mk-B10L	49.645
9	Mk-B11	52.032
9	Core Coolant Average Velocity, fps	
9	Mk-B10 through Mk-B10L	15.79
9	Mk-B11	14.99
	Power Distribution	
9	Maximum/Average Power Ratio, Radial x Local ($F_{\Delta h}$ Nuclear)	1.714
9	Maximum/Average Power Ratio, Axial (F_z Nuclear)	1.5 cos
	Overall Power Ratio (F_q Nuclear)	2.57
	Power Generated in Fuel and Cladding, %	97.3
	Hot Channel Factors	
9	Power Peaking Factor (F_Q)	
9	Mk-B10, Mk-B10D and Mk-B10E	1.0107
9	Mk-B10F, Mk-B10G and Mk-B10L	1.0132
9	Mk-B11	1.0133
4		
	Hot Spot Maximum/Average Heat Flux Ratio	
9	(F_q nuc and mech)	
9	Mk-B10, Mk-B10D and Mk-B10E	2.71

9	**Table 4-1 (Page 4 of 4). Core Design, Thermal, and Hydraulic Data**	
9	Mk-B10F to B10L and Mk-B11	2.72
9 9	Flow Area Reduction Factor(F_A) for MK-B10 through Mk-B10L & B11	
0	Unit/CRGT Bundle Cells	0.98
0	IGT Bundle Cells	0.97
4		
	DNB Data	
	Design Overpower (% Rated Power)	112
9	CHF Correlation	
9	MK-B10 through Mk-B10L	BWC
9 9	Mk-B11	BWU-Z with FB11 Multiplier
9	DNB Limit - Non SCD	
9	Mk-B10 through Mk-B10L	1.18
9	Mk-B11	1.19
9	DNB Limit - SCD	
9	Mk-B10 through Mk-B10L	1.43
9	Mk-B11	1.33
0		
9		
9	Typical minimum DNBR	
9	Mk-B10 through Mk-B10L	2.47
9	Mk-B11	2.76
0		
4		

Notes:

9 9 1. Parameters are based on cold dimensions for each of the respective fuel assembly designs, as applicable.

9 9 9 2. Based on reference peaking and active fuel length for each fuel rod type specified at BOL conditions. Flow is based on 107.5 percent design flow and 7.0 and 7.5 percent bypass flow for Mk-B10F/G/L and Mk-B11 fuel assembly designs, as applicable.

9 **Table 4-2 (Page 1 of 2). Fuel Assembly Components**

	Item	Material	Dimensions (In)
7			
9	Fuel Clad (in.)		
9	MK B-10 through B10E	Zircaloy-4	0.430 OD x 0.377 ID
7	MK B-10F through MK B-10L	Zircaloy-4	0.430 OD x 0.380 ID
9	Mk-B11	M5	0.416 OD x 0.368 ID
9	Fuel Rod Length (Typical), in.		
9	Mk-B10 to B10L		154.16
9	Mk-B11		155.30
9			
9			
	Fuel Assembly:		
9			
9	Overall Length B10 & B11 (Typical), in.		165.695
9	Control Rod Guide Tube (in.)		
9	Mk-B10 to Mk-B10L	Zircaloy-4	0.530 OD x 0.016 wall
9	Mk-B11	Zircaloy-4	0.530 OD x 0.016 wall
9	Instrumentation Tube (in.)		
9	Mk-B10 to B10L	Zircaloy-4	0.493 OD x 0.441 ID
9	Mk-B11	Zircaloy-4	0.493 OD x 0.441 ID
	End Fittings		
9 9	Mk-B10 to B10L	Stainless Steel (Castings)	
9	Mk-B11	Stainless Steel	
5	End Spacer Grid		
9			
9	MK B-10 to MK B-10L	Inconel-718	0.020 thick exteriors 0.018 thick interiors
9 9	Mk-B11	Inconel-718	0.020 thick exteriors 0.018 thick interiors
9	Intermediate Spacer Grid		
9 9	MK-B10 to MK-B10L	Zircaloy-4	0.021 thick exteriors 0.018 thick interiors

9 **Table 4-2 (Page 2 of 2). Fuel Assembly Components**

	Item	**Material**	**Dimensions (In)**
9 9	Mk-B11	Zircaloy-4	0.021 thick exteriors 0.018 thick interiors
9	Spacer Sleeve		
9	Mk-B10 to B10L	Zircaloy-4	0.554 OD x 0.502 ID
9	Mk-B11	Zircaloy-4	0.554 OD x 0.502 ID
7	Fuel Assembly Design:		Fuel Assembly Burnup
9 8	MK B-10 through MK B-10L, MK-B11		Consistent with a Maximum rod burnup of 62,000 MWD/MTU (Reference 10) of Section 4.2.5
7			

Notes:

9 1. Typical geometry. Batch specific is reported in individual reload reports.

9 9 2. Mk-B9 fuel rods are used in Mk-B0 and Mk-B10D/E fuel assembly designs. Mk-B10 design fuel rods are used in Mk-B10F/G/L fuel assembly designs (See Table 4-23).

9 **Table 4-14. Deleted per 1999 Update**

7 **Table 4-15. Deleted per 1997 Update**

Table 4-16. Internals Vent Valve Materials

Valve Part Name	Material and Form	Material Specification No.
Valve Body	304 S.S. Casting [a]	ASTM A351-CF8
Valve Disc	304 S.S. Casting [a]	ASTM A351-CF8
Disc Shaft	431 S.S. Bar [b]	ASTM A276 Type 431 Cond. T
Shaft Bushings	Stellite No. 6	
Retaining Rings (Top and Bottom)	15-5 pH (H1100) S.S. forgings	AMS 5658
Ring Jack Screws	"A-286 Superalloy" S.S. [c]	AMS 5737 C
9 Jackscrew Bushings	431 S.S. Bar	ASTM A276 Type 431 Cond. A
Misc. Fasteners, covers, locking devices, etc.	304 S.S. plate bar, etc.	ASTM A240 ASTM A276

Note:

(a) Carbide solution annealed, C_{max} 0.08%, Co_{max} 0.2%

(b) Heat treated and tempered to Brinnel Hardness Number (BHN) range of 290-320.

(c) Heat treated to produce a BHN of 248 min.

The hinge assembly consists of a shaft, two valve body journal receptacles, two halve disc journal receptacles, and four flanged shaft journals (bushings). Loose clearances are used between the shaft and journal inside diameters, and between the journal outside diameters and their receptacles. The hinge assembly is shown and the clearance gaps are identified in Figure 4-30. The bushing clearances are listed in Table 4-17.

The valve disc hinge journal contains integral exercise lugs for remote operation of the disc with the valve installed in the core support shield.

Table 4-21. Control Rod Drive Mechanism Design Data

		Shim Safety	Axial Power Shaping
	Type	Roller Nut Drive	Roller Nut Drive
	Quantity	61	8
	Location	Top-mounted	Top-mounted
	Direction of Trip	Down	Does not trip
9	Velocity of Normal (Run) Withdrawal and Insertion, in./min.	30	30
9	Velocity of Jog Withdrawal and Insertion in./min.	3	3
	Maximum Travel Time for Trip		
2 2	2/3 Insertion, sec	1.40*	Drive has no trip function
2 2	3/4 Insertion, sec	1.52*	Drive has no trip function
	Length of Stroke, in.	139	139
	Design Pressure, psig	2,500	2,500
	Design Temperature, °F	650	650
	Weight of Mechanism (App.)	940 lb	940 lb

2 *These time values include rod motion only. The Technical Specification surveillance requirement for
2 maximum control rod drop time includes, in addition, 0.14 seconds from the time the control rod drive
2 breakers receive the signal to trip to the beginning of rod motion. This is appropriate since the elapsed
2 time measured in the test begins with that signal to trip the CRD Breaker.

Table 4-22. Fuel Assembly / APSR Compatibility

Plant and Unit	Drive Type	Type of APSR Coupling-Spider Assembly Required for Mk-B9 Fuel Design and Earlier	Type of APSR Coupling-Spider Assembly Required for Mk-B10 and Mk-B11 Fuel Designs
Oconee Unit 1&2	Type A APSR Drive	Mk-B Standard	Extended Coupling
Oconee Unit 1&2	Type C APSR Drive*	Mk-B Standard	Mk-B Standard OR Extended Coupling
Oconee Unit 3	Type C APSR Drive	Mk-B Standard	Mk-B Standard OR Extended Coupling

*Type C APSR Drive has R4C position indicators and hydraulic tension closures.

Note:

1. The length of the Mk-B Standard and Extended Coupling APSR Hubs is 7.0 in. (nom.) and 7.57 in. (nom.), respectively. The length equals the sum of the female coupling, spider, and lower hub B, which is the distance from the bottom seating surface to the top of the female coupling.
2. Oconee Unit 1 and Unit 2 will replace the existing Type A APSR Drives with Type C APSR Drives*. As shown above, Oconee Unit 2 will operate with both drive types for a temporary period.

9 **Table 4-23. Fuel Assembly Design Descriptions**

	Assembly Designation	Cage Design	Rod Design	Clad Material	Axial Blanket	Zoned Enrichment	HDS [2] Design	UEF [3] Attachment	Debris Filter
9	Mk-B10	B10	B9	Zirc-4	No	No	Cruciform	Lock Nut	Plug/Grid
9	Mk-B10D	B10	B9	Zirc-4 [1]	No	No	Cruciform	Lock Nut	Plug/Grid
9	Mk-B10E	B10	B9	Zirc-4	Yes	No	Cruciform	Lock Nut	Plug/Grid
9	Mk-B10F	B10	B10	Zirc-4	Yes	No	Cruciform	Lock Nut	Plug/Grid
9	Mk-B10G	B10	B10	Zirc-4	Yes	No	Cruciform	Quick Disconnect	Plug/Grid
9	Mk-B10L	B10	B10	Zirc-4	Yes	Yes	Cruciform	Quick Disconnect	Plug/Grid
9	Mk-B11	B11	B11	M5	Yes	Yes	Cruciform	Quick Disconnect	Plug/Grid

9 **Note:**

9 1. Consumer's or Smud Cladding

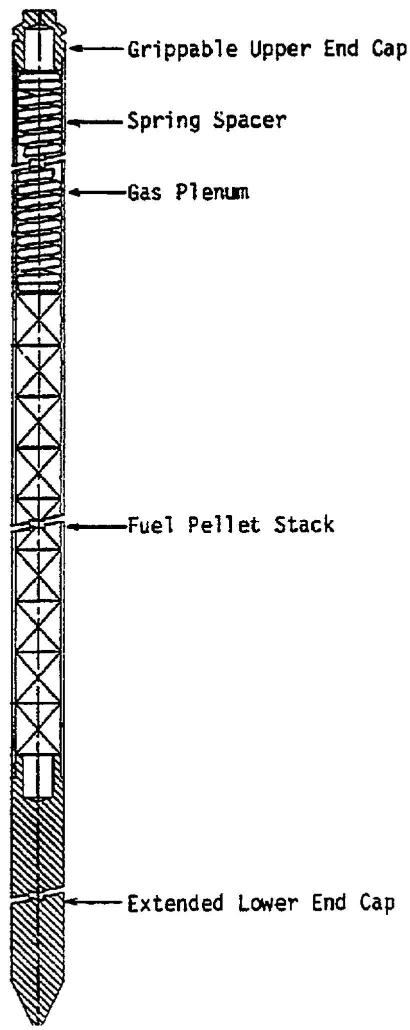
9 2. HDS = Hold Down Spring

9 3. UEF = Upper End Fitting

9

9 Figure 4-2.
9 Deleted per 1999 Update

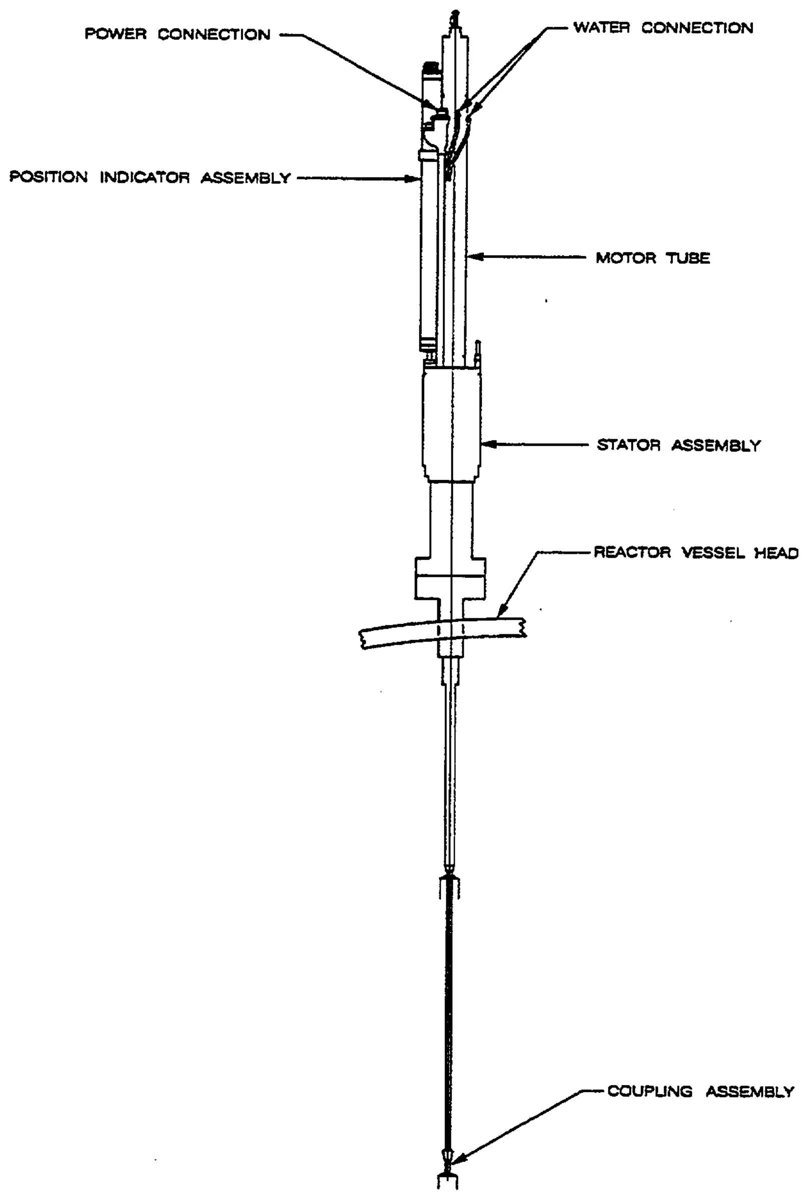
9 Figure 4-3.
9 Deleted per 1999 Update



9 **Figure 4-4.**

9 **Typical Pressurized Fuel Rod**

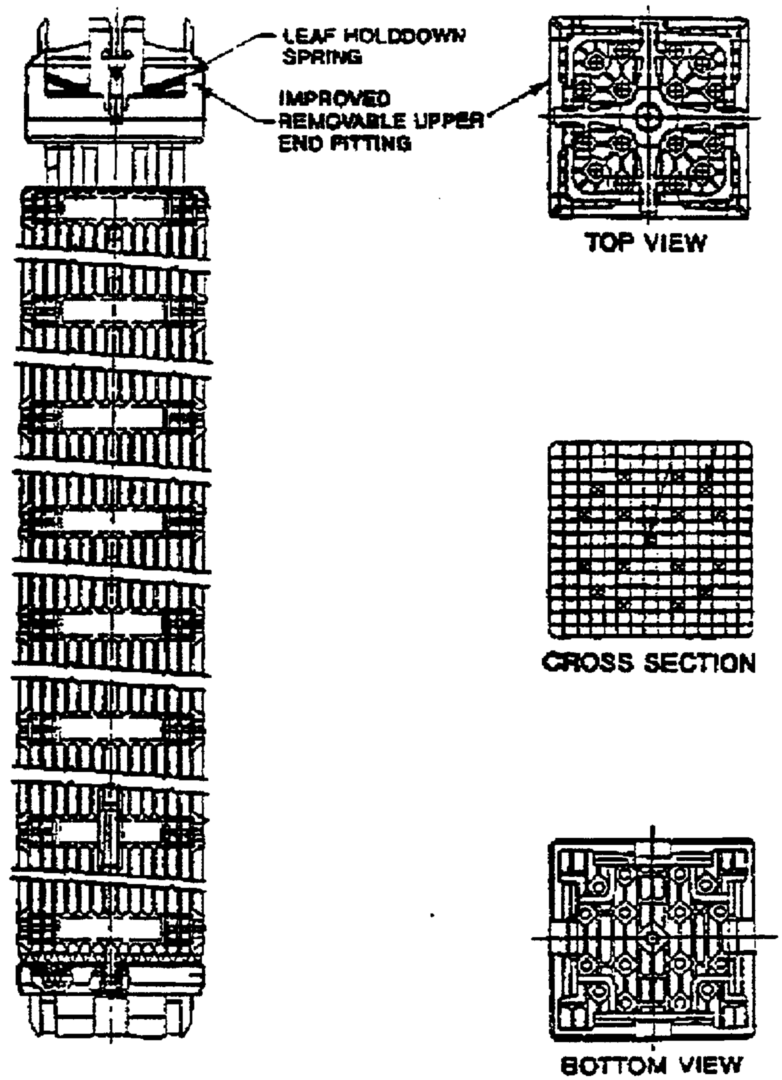
9 Figure 4-33.
9 Deleted Per 1999 Update



9 Figure 4-34.
9 Control Rod Drive - General Arrangement

9 Figure 4-35.
9 Deleted Per 1999 Update

9 Figure 4-36.
9 Deleted per 1999 Update



9 Figure 4-37.
9 Typical Fuel Assembly

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

CHAPTER 5. REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 SUMMARY DESCRIPTION

5.1.1 GENERAL

5.1.1.1 System

The Reactor Coolant System consists of the reactor vessel, two vertical once-through steam generators, four shaft-sealed reactor coolant pumps, an electrically heated pressurizer and interconnecting piping. The system is arranged in two heat transport loops, each with two reactor coolant pumps and one steam generator. The reactor coolant is transported through piping connecting the reactor vessel to the steam generators and flows downward through the steam generator tubes transferring heat to the steam and water on the shell side of the steam generator. In each loop, the reactor coolant is returned to the reactor through two lines, each containing a reactor coolant pump, to the reactor vessel. In addition to serving as a heat transport medium, the coolant also serves as a neutron moderator and reflector, and a solvent for the soluble poison (boron in the form of boric acid). The system pressure settings are listed in Table 5-1; the integrity of the reactor coolant pressure boundary is described in Section 5.2, "Integrity of Reactor Coolant Pressure Boundary"; the reactor vessel design is described in Section 5.3, "Reactor Vessel"; and other major components and subsystems in the reactor coolant pressure boundary (RCPB) are described
9 in Section 5.4, "Component and Subsystem Design." The maximum reactor coolant system volume is
9 12,200 ft^3.

The Reactor Coolant System piping diagrams are Figure 5-1 (Oconee 1) and Figure 5-2 (Oconee 2 & 3).

In 1970, the Oconee 1 reactor coolant pumps were replaced with Westinghouse Model 93A pumps. The reactor coolant piping was modified slightly to accommodate the replacement pumps. Both the original pumps and the replacement pumps were bottom suction and side discharge allowing installation of the replacement pumps on the same centerlines as the original pumps. The original motors were utilized with the replacement pumps.

Figure 5-3 and Figure 5-4 show the revised arrangement of the reactor coolant piping for Oconee 1.

5.1.1.2 System Protection

Engineered safety features and associated systems are protected from missiles which might result from a loss of coolant accident. Protection is provided by concrete shielding and/or segregation of redundant components.

The reactor vessel is surrounded by a concrete primary shield wall and the heat transport loops are surrounded by a concrete secondary shield wall. These shielding walls provide missile protection for the Reactor Building liner plate and equipment located outside the secondary shielding.

Removable concrete slabs over the reactor vessel area and the concrete deck over the area outside of the secondary shield wall also provide shielding and missile protection.

The Reactor Coolant System is analyzed for maximum hypothetical earthquake to determine that resultant stresses do not jeopardize the safe shutdown of the Reactor Coolant System and removal of decay heat.

5.1.1.3 System Arrangement

The system arrangement in relation to shielding walls, the Reactor Building and other equipment in the building are described in Chapter 1, "Introduction and General Description of Plant." Plan and elevation drawings showing principal dimensions of the Reactor Coolant System in relation to the supporting or surrounding concrete structures are provided in Figure 5-3, Figure 5-4 (Oconee 1), Figure 5-5, Figure 5-6 (Oconee 2) and Figure 5-7, Figure 5-8 (Oconee 3).

5.1.1.4 System Parameters

5.1.1.4.1 Flow

The Reactor Coolant System is designed on the basis of 176,000 gpm flow rate in each heat transport loop.

5.1.1.4.2 Temperatures

Reactor Coolant System temperatures as a function of power are shown in Figure 5-9. The system is controlled to a constant average temperature throughout the power range from 15 percent to 100 percent full power. The average system temperature is decreased between 15 percent and 0 percent of full power to the saturation temperature at 900 psia.

5.1.1.4.3 Heatup

All Reactor Coolant System components are designed for a continuous heatup rate of 100°F/hr.

5.1.1.4.4 Cooldown

All Reactor Coolant System components are structurally designed for a continuous cooldown rate of 100°F/hr. System cooldown to 250°F is accomplished by use of the steam generators and by bypassing steam to the condenser with the Turbine Bypass System. The Low Pressure Injection System provides the heat removal for system cooldown below 250°F.

5.1.1.4.5 Volume Control

The only coolant removed from the Reactor Coolant System is that which is letdown to the High Pressure Injection System. The letdown flow rate is set at the desired rate by the operator positioning the letdown control valve and/or opening the stop valve for the letdown orifice.

To maintain a constant pressurizer water level, total makeup to the Reactor Coolant System must equal that which is letdown from the system. Total makeup consists of the seal injection water through the reactor coolant pump shaft seals and makeup returned to the system through the reactor coolant volume control valve (High Pressure Injection System). The pressurizer level controller provides automatic control of the valve to maintain the desired pressurizer water level. Reactor coolant volume changes during plant load changes exceed the capability of the reactor coolant volume control valve, and thus result in variations in pressurizer level. The level is returned to normal as the system returns to steady state conditions.

5.1.1.4.6 Chemical Control

Control of the Reactor Coolant Chemistry is a function of the Chemical Addition and Sampling System. Sampling lines from the letdown line of the High Pressure Injection System provide samples of the reactor coolant for chemical analysis. All chemical addition is made from the Chemical Addition and Sampling

System to the High Pressure Injection System. See Chapter 9, "Auxiliary Systems" for detailed information concerning the Chemical Addition and Sampling System and Chapter 6, "Engineered Safeguards" for the High Pressure Injection System.

5.1.1.4.7 Boron

Boron in the form of boric acid is used as a soluble poison in the reactor coolant. Concentrated boric acid is stored in the Chemical Addition and Sampling System and is transported to the Reactor Coolant
2 System in the same manner as described above for chemical addition. The concentrated boric acid may
2 be stored in the concentrated boric acid storage tank (CBAST) or directly in the boric acid mix tank. The
2 CBAST receives concentrated boric acid from the boric acid mix tank. The CBAST is required to
2 contain a specified concentration of boric acid based on the volume in the tank in order to supply a
2 source of concentrated soluble boric acid to the Reactor Coolant System in addition to the borated water
2 storage tank. The concentrated boric acid is pumped to the High Pressure Injection System which
2 transports it to the Reactor Coolant System. Boron concentrations are reduced by running letdown flow
2 through the deborating demineralizers and/or diluting the reactor coolant with demineralized water. All bleed and feed operations for changing the boric acid concentrations of the reactor coolant are made between the High Pressure Injection System and the Coolant Storage System.

5.1.1.4.8 pH

The pH of the reactor coolant is controlled to minimize corrosion of the Reactor Coolant System surfaces which minimizes coolant activity and radiation levels of the components.

5.1.1.4.9 Water Quality

The reactor coolant water chemistry specifications have been selected to provide the necessary boron content for reactivity control and to minimize corrosion of Reactor Coolant System surfaces. The solids content of the reactor coolant is maintained below the design level by minimizing corrosion through chemistry control and by continuous purification by the demineralizer of the High Pressure Injection System. Excess hydrogen is maintained in the reactor coolant to chemically combine with the oxygen produced by radiolysis of the water.

5.1.1.4.10 Vents and Drains

Vent and drain lines are located at the high and low points of the system and provide the means for draining, filling, and venting the heat transport loops and pressurizer. The reactor vessel cannot be drained below the top of the reactor outlet nozzle using these drain lines. Each vent and drain line contains two manual valves in series. Vent lines are routed to a header connected to the quench tank gas space and drain lines are routed to a header connected to the suction of the component drain pump.

5.1.2 PERFORMANCE OBJECTIVES

5.1.2.1 Steam Output

The Reactor Coolant System is designed to operate at a core power level of 2,568 MWt and transfer a total of 2,584 MWt (including 16 MWt input from reactor coolant pumps) to the steam generators. The system will produce a total steam flow of 11.2 million lb/hr.

5.1.2.2 Transient Performance

The Reactor Coolant System will follow step or ramp load changes under automatic control without relief valve or turbine bypass valve action as follows:

5.1.2.2.1 Step Load Changes

Increasing or decreasing load steps of 10 percent of full power in the range between 20 percent and 90 percent full power.

5.1.2.2.2 Ramp Load Changes

7 Increasing load ramps of 1 percent per minute between 2 percent and 15 percent, 5 percent per minute
7 between 15 percent and 20 percent, 9.9 percent between 20 percent and 95 percent and 5 percent per
7 minute between 95 percent and 100 percent full power are acceptable. Decreasing load ramps of 9.9
7 percent between 100 percent and 20 percent, 5 percent per minute between 20 percent and 15 percent and
7 1 percent per minute between 15 percent and 2 percent full power are acceptable.

The combined actions of the Control System and the Turbine Bypass System permit a 40 percent load rejection or a turbine trip from 40 percent full power without safety valve action. The combined actions of the Control System, the turbine bypass valves, and the main steam safety valves are designed to accept separation of the generator from the Transmission System without reactor trip.

5.1.2.3 Partial Loop Operation

The Reactor Coolant System will permit operation with less than four reactor coolant pumps in operation. The nominal steady-state operating power levels for combinations of reactor coolant pumps operating are as follows:

Reactor Coolant Pumps Operating	Rated Power, %
4	100
3	75

2

5.1.2.4 Natural Circulation

Natural circulation provides an acceptable method of energy removal from the core with transfer of energy to the Secondary System through the steam generators. The controlling parameters which determine the magnitude of the natural circulation flow rates, i.e., steam generator liquid level and source of feedwater (emergency or main), produce more than adequate circulation rates under steady conditions. The margins to the limits for acceptable operation are more than adequate for steady-state and expected transients.

Natural circulation cooldown mode of operation is not expected to be undertaken at Oconee Nuclear Station except for SBLOCA events which do not allow continued operation of or restart of reactor
8 coolant pumps. In all other situations, procedures recommend that MODE 3 with average Reactor
8 Coolant temperature ≥525°F be maintained until those systems required for forced circulation are put back into service.

In response to Generic Letter 81-21, Duke has developed a procedure to continuously vent the reactor vessel head to containment during a natural circulation cooldown to Decay Heat Removal System conditions, as well as prevent upper head voiding. NRC Safety Evaluation Report (Reference 1) concurs with Duke that natural circulation cooldown is not a safety concern due to operator training and procedures.

5.1.3 REFERENCES

1. Letter from J. F. Stolz (NRC) to H. B. Tucker (Duke) dated June 5, 1985. Subject: NRC Safety Evaluation Report on Duke Response to Generic Letter 81-21 Natural Circulation Cooldown.

5.2 INTEGRITY OF REACTOR COOLANT PRESSURE BOUNDARY

5.2.1 DESIGN CONDITIONS

5.2.1.1 Pressure

The Reactor Coolant System components are designed structurally for an internal pressure of 2,500 psig.

5.2.1.2 Temperature

7 With the exception of the components associated with the pressurizer, the Reactor Coolant System
7 pressure boundary components are designed for a temperature of 650°F. The pressurizer and associated
7 code safety valves, power operated relief valve and piping, surge line, sample and drain lines and associated
7 valves, and a portion of the spray line piping are designed for 670°F.

5.2.1.3 Reactor Loads

Reactor Coolant System components are supported and interconnected so that stresses resulting from combined mechanical and thermal forces are within established code limits. Equipment supports are designed to transmit piping rupture reaction loads to the foundation structures.

The Reactor Coolant System supports are on an eight foot six inch thick, heavily reinforced concrete slab which rests on a solid rock subgrade. The minimum ultimate crushing strength of rock cores tested was 720 kips per square foot and the maximum applied dynamic gross load is 30 kips per square foot. Based on the subgrade, the ratio of applied load to bearing capacity of the subgrade, and the monolithic nature of the base slab, differential settlement of the foundation is not anticipated.

5.2.1.4 Cyclic Loads

All Reactor Coolant System components are designed to withstand the effects of cyclic loads due to
1 system temperature and pressure changes. Design transient cycles are shown in Table 5-2 and
1 Table 5-23.

Flow-induced vibration analyses have been performed for the fuel assembly, including fuel rods, and for the reactor internals components. The analyses and design criteria for the thermal shield, flow distributor assembly, surveillance holder tubes and shroud tubes, and the "U" baffles are given in B&W Topical Report BAW-10051, Reference 1.

Components subjected to cross flow are checked for response during design, so that the fundamental frequencies associated with cross flow are above the vortex shedding frequencies. It has also been conservatively determined that the flow induced pressure fluctuations acting on the disc of the vent valve are such that for normal operation there is always a positive net closing force acting on the disc.
7 Emergency operational modes are covered in B&W Topical Report BAW-10008, Part 1, Reference 2.

9 Oconee Technical Specification 5.5.6 establishes the requirement to provide controls to track the number
9 of UFSAR Section 5.2.1.4, "Cyclic Loads" cyclic and transient occurrences to assure that components are
9 maintained within design limits. This requirement is managed by the Oconee Thermal Fatigue
9 Management Program.

9 For license renewal, continuation of the Oconee Thermal Fatigue Management Program into the period
9 of extended operation will provide reasonable assurance that the thermal fatigue analyses, including
9 applicable flaw growth calculations, will remain valid or that appropriate action is taken in a timely
9 manner to assure continued validity of the design.

9 References for this Section: Application [Reference 38] and Final SER [Reference 39]

5.2.1.5 Seismic Loads and Loss-of-Coolant Loads

Reactor Coolant System components are designated as Class I equipment and are designed to maintain
9 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. A discussion of each of the cases of
loading combinations follows:

5.2.1.5.1 Seismic Loads

Case I - Design Loads Plus Design Basis Earthquake (DBE) Loads - For this combination, the reactor must be capable of continued operation; therefore, all components excluding piping are designed to Section III of the ASME Code for Reactor Vessels. The primary piping is designed according to the requirements of USAS B31.1 and B31.7. The S_m values for all components, excluding bolting, are those specified in Table N-421 of the ASME code. The S_m value for bolts are those specified in Table N-422 of the ASME Code.

CASE II - Design Loads Plus Maximum Hypothetical Earthquake (MHE) - In establishing stress levels
for this case, a "no-loss-of-function" criterion applies, and higher stress values than in Case I can be
allowed. The multiplying factor of 1.2 has been selected in order to increase the code-based stress limits
9 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 BAW-10008, Part 1, "Reactor Internals Stress & Deflection Due to
LOCA and Maximum Hypothetical Earthquake." The S_m value for all components are those specified
in Table N-421 of the ASME Code.

5.2.1.5.2 Loss-of-Coolant Loads

A loss-of-coolant accident coincident with a seismic disturbance has been analyzed to assure that no loss of function occurs. In this case, primary attention is focused on the ability to initiate and maintain reactor shutdown and emergency core cooling. Two additional cases are considered as follows:

Case III - Design Loads Plus Pipe Rupture Loads - For this combination of loads, the stress limits for Case II are imposed for those components, systems, and equipment necessary for reactor shutdown and emergency core cooling.

Case IV - Design loads plus Maximum Hypothetical Earthquake (MHE) Loads Plus Pipe Rupture Loads - Two thirds of the ultimate strength has been selected as the stress limit for the simultaneous occurrence of MHE and reactor coolant pipe rupture. As in Case III, the primary concern is to maintain the ability to shut the reactor down and to cool the reactor core. This limit assures that a materials strength margin of safety of 50 percent will always exist.

The design allowable stress of Case IV loads is given in BAW-10008 for 304 stainless steel. This curve is used for all reactor vessel internals including bolts. It is based on adjusting the ultimate strength curves published by U.S. Steel to minimum ultimate strength values by using the ratio of ultimate strength given by Table N-421 of Section III of the ASME code at room temperature to the room temperature strength given by U.S. Steel.

In Cases II, III, and IV, secondary stresses were neglected, since they are self-limiting. Design stress limits in most cases are in the plastic region, and local yielding would occur. Thus, the conditions that caused the stresses are assumed to have been satisfied. BAW-10008, Part 1, contains a more extensive discussion of the margin of safety, the effects of using elastic equations, and the use of limit design curves for reactor internals. Table 5-3 provides the stress limits for seismic, pipe rupture, and combined loads.

5.2.1.6 Service Lifetime

The design service lifetime for the major Reactor Coolant System components is 40 years. The number
1 of cyclic system temperature and pressure changes (Table 5-2 and Table 5-23), is based on operation for
this design lifetime.

5.2.1.7 Water Chemistry

The water chemistry is selected to provide the necessary boron content for reactivity control and to minimize corrosion of the Reactor Coolant System surfaces. To ensure the best protection is provided, reactor coolant water quality specifications are based upon the most current revision of the EPRI PWR Primary Water Chemistry Guidelines and vendor recommendations. These are addressed in the Chemistry Section Manual.

5.2.1.8 Vessel Radiation Exposure

The reactor vessel is the only Reactor Coolant System component exposed to a significant level of
1 neutron irradiation and is therefore the only component subject to material radiation damage. The
8 maximum predicted exposure from fast neutrons (E > 1.0 MeV) at the inside vessel surface over a
1 40-year life with an 80 percent load factor has been computed to be as follows (per References 27, 28, and
8 29):

8	Oconee Unit 1	9.32×10^{18} neutrons/cm^2
8	Oconee Unit 2	9.02×10^{18} neutrons/cm^2
8	Oconee Unit 3	8.90×10^{18} neutrons/cm^2

9 5.2.1.9 Leak Before Break

9 Leak-before-break is used at Oconee to establish Mark - B fuel assembly spacer grid impact loads and
9 displacement time histories.

9 The successful application of Leak-Before-Break (LBB) to the Oconee Reactor Coolant System main
9 coolant piping is described in B&WOG topical report entitled, "The B&W Owners Group
9 Leak-Before-Break Evaluation of Margins Against Full Break for RCS Primary Piping of B&W Designed
9 NSSS," BAW-1847, Revision 1, September 1985. This report provides the technical basis for evaluating
9 postulated flaw growth in the main Reactor Coolant System piping under normal plus faulted loading
9 conditions and was approved by the NRC for the current term of operation. The time-limited aging
9 analyses in BAW-1847, Revision 1, include fatigue flaw growth and the qualitative assessment of thermal
9 aging of cast austenitic stainless steel reactor coolant pump inlet and exit nozzles.

9 Fatigue flaw growth evaluations are based on transient definitions defined by the Reactor Coolant System
9 design specification. The original transient cycles that were defined for 40 years of operation are being
9 monitored by the Oconee Thermal Fatigue Management Program. If a transient cycle count approaches
9 or exceeds the allowable design limit, corrective actions are taken. The cast austenitic stainless steel

9 reactor coolant pump inlet and outlet nozzles are susceptible to thermal aging. Thermal aging of cast
9 austenitic stainless steel causes a reduction of fracture toughness. Reduction of fracture toughness of the
9 reactor coolant pump nozzles has been determined to be acceptable for the period of extended operation
9 through a flaw stability analysis.

9 References for this Section: [Reference 40] and [Reference 41]

5.2.2 CODES AND CLASSIFICATIONS

The codes listed in this section and Table 5-4 include the code addenda and case interpretations issued through Summer 1967 unless noted otherwise. Quality control and quality assurance programs relating to the fabrication and erection of system components are summarized in Section 5.2.3.11, "Quality Assurance."

5 For inservice inspection of all three units, the applicable ASME Boiler and Pressure Vessel Code is: 1989
5 Edition, no Addenda.

5.2.2.1 Vessels

The design, fabrication, inspection and testing of the reactor vessel and closure head, steam generator (both reactor coolant side and secondary side), pressurizer and attachment nozzles on the vessels is in accordance with the ASME Boiler and Pressure Vessel Code, Section III, for Class A vessels.

5.2.2.2 Piping

The design, fabrication, inspection and testing of the reactor coolant piping excluding the pressurizer surge line and the spray line is in accordance with USAS B31.7, Code for Pressure Piping, Nuclear Power Piping, dated February, 1968, and as corrected for Errata under date of June, 1968. The pressurizer surge and spray lines were fabricated and initially inspected in accordance with USAS B31.7, February 1968
8 with June, 1968, Errata. However, the surge line, which was analyzed in accordance with the ASME
8 Code, 1977 edition, Summer 1979 Addenda, has been reanalyzed to the 1986 ASME Code in response to
9 NRC Bulletin 88-11 concerns. The spray line has been reanalyzed to the 1983 Edition of the ASME
9 Code. The following Reactor Coolant Branch lines were analyzed, up to the first isolation valve from the
9 Reactor Coolant Loop, to Class 1 rules of the 1983 Edition, no addenda, of the ASME code:

9 • High Pressure Injection (Emergency Injection)
9 • High Pressure Injection (Normal Injection)
9 • High Pressure Injection (Letdown)
9 • Low Pressure Injection (Decay Heat Removal Drop-line)
9 • Low Pressure Injection (Core Flood)
9 • Reactor Coolant Drains
9 • Pressurizer Relief Valve Nozzles

7 The feedwater header and the emergency feedwater header for the steam generator meet the requirements
of the Code for Pressure Piping, Power Piping USAS B31.1.0 - 1967.

5.2.2.3 Reactor Coolant Pumps

The reactor coolant pump casings are designed, fabricated, inspected and tested to meet the intent of the ASME Boiler and Pressure Vessel Code, Section III, for Class A vessels, but are not code stamped.

5.2.2.4 Relief Valves

The pressurizer code safety valves and the electromatic relief valve comply with Article 9, Section III, of the ASME Boiler and the Pressure Vessel Code.

5.2.2.5 Welding

Welding qualifications are in accordance with the ASME Boiler and Pressure Vessel Code, Section III and Section IX and Section XI, as applicable.

5.2.3 SYSTEM DESIGN EVALUATION

5.2.3.1 Design Margin

The Reactor Coolant System is designed structurally for 2,500 psig and 650°F. The system will normally operate at 2,155 psig and 604°F.

In the event of a complete loss of power to all reactor coolant pumps, reactor coolant flow, coastdown, and subsequent natural circulation flow is more than adequate for core cooling and decay heat removal as shown by the analysis in Chapter 15, "Accident Analyses."

1 The number of transient cycles specified in Table 5-2 and Table 5-23 for the fatigue analysis is conservative.

5.2.3.2 Material Selection

Each of the materials used in the Reactor Coolant System has been selected for the expected environment and service conditions. The major component materials are listed in Table 5-5. All Reactor Coolant System materials normally exposed to the coolant are corrosion-resistant materials consisting of 304 or 316 stainless steel, Inconel, 17-4PH (H1100), Zircaloy, or weld deposits with corrosion-resistant properties equivalent to or better than those of 304 SS. These materials were chosen for specific uses at various locations within the system because of their compatibility with the reactor coolant. There are no novel material applications in the Reactor Coolant System.

To assure long steam generator tube lifetime, feedwater quality entering the steam generator is maintained as high as practical. The current revision of the SGOG EPRI PWR Secondary Chemistry Guidelines and vendor recommendations are used to prepare operating specifications which are addressed in the Chemistry Section Manual.

The selection of materials and the manufacturing sequence for the Reactor Coolant System components, is arranged to insure that no pressure boundary material is furnace-sensitized stainless steel. Safe ends are provided on those carbon steel nozzles of the system vessels which connect to stainless steel piping. All dissimilar metal welds, with the exception of Inconel to Stainless Steel pipe welds, will be made in the manufacturer's shops.

Piping systems designed to resist seismic forces have been restrained by steel supports capable of
withstanding these seismic forces. The restraints also act as pipe stops restraining the lines against
9 whipping. In systems, where it was necessary to use hydraulic or mechanical snubbers to resist seismic
forces, the mechanical action associated with the snubbers makes it possible to consider them as restraints
9 against pipe whipping. A more detailed discussion of the types of snubbers in use at Oconee is provided
9 in Section 3.9.3.4.2.2, "Snubbers."

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

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

The pressure boundary of the RCS is fabricated primarily from ferritic materials, while that of the attached systems is fabricated primarily from austenitic material.

Consequently, the RCS components are the only ones that require special protection against nonductile failure and that must comply with the fracture toughness requirements of Appendix G to 10 CFR 50. This protection is ensured by establishing pressure-temperature limitations on the RCS. The margin of safety is controlled by not exceeding the calculated allowable pressure at any given temperature. The following loading conditions require pressure-temperature limits:

1. Normal operations including bolt preloading, heatup, and cooldown.
2. Preservice system hydrostatic test.
3. Inservice system leak and hydrostatic tests.
4. Reactor core operation.

For a better understanding of the required protection against non-ductile failure, typical operational parameters of the RCS are described in the following sections for each of the loading conditions.

5.2.3.2.1 Normal Operation

During bolt preload, the reactor vessel closure studs are tensioned to the specified load. Bolt preloading is not allowed until the reactor coolant temperature or the volumetric average temperature of the closure head region (including the studs) is higher than the specified minimum preload temperature. After the studs are tensioned, system pressure can be increased by the pressurizer until it is above the net positive suction head (NPSH) required for reactor coolant pump (RCP) operation. The heatup transient begins when the RCP is started.

8 During heatup, the RCS is brought from MODE 5 to MODE 3 with average Reactor Coolant
8 temperature ≥525°F. The heat sources used to increase the temperature of the system are the RCP and any residual (decay) heat from the core. Normally, when the pumps are started, the temperature of the water in the pressurizer is about 400°F; this corresponds to the pressure in the RCS, which is about 300 psig. The coolant temperature is at or above the minimum specified bolt preload temperature.

Initially, the reactor coolant temperature may be as low as room temperature for initial core loading or as high as 130°F for subsequent refueling. The system pressure is maintained below the maximum allowable pressure of approximately 625 psig (20 percent of preoperational system hydrostatic test pressure) until the reactor coolant temperature is approximately 270°F.

At any given time throughout the heatup transient, the temperature of the reactor coolant is essentially the same throughout the system except, of course, in the pressurizer. The system pressure, as controlled by the pressurizer heaters, is maintained between the minimum required for RCP NPSH and the maximum established to meet the fracture toughness requirements. The heatup rate is maintained below the maximum rate used to establish the maximum allowable pressure-temperature limit curve.

8 RCS cooldown brings the system from MODE 3 to MODE 5. The cooldown is normally accomplished in two phases: The first phase reduces the fluid temperature from approximately 550°F to below the design temperature of the decay heat removal system (approximately 300°F). This temperature reduction is accomplished using the steam generators but bypassing the turbine and dumping the steam directly to the condenser. Once below its design temperature (and pressure), the Decay Heat Removal System (DHRS) is activated in the second phase to further reduce the reactor coolant temperature to that desired.

Before cooldown, the RCS temperature is maintained constant by balancing the heat removal rate from the steam dump with the heat contributed by the RCP and core decay heat. The system pressure is
9 maintained by the pressurizer. The cooldown is normally initiated by stopping two RCPs in one loop. The two remaining pumps provide coolant circulation through both steam generators, and the turbine steam bypass flow controls the cooldown rate. The primary pressure during cooldown is controlled with the pressurizer heaters and spray. After cooling down below the DHRS design temperature and pressure, the cooling mode is changed from the steam generators to the DHRS. Before the switch, the RCS pressure is below 625 psig (20 percent of preoperational system hydrostatic test pressure) and below the DHRS pressure but above the pressure required for the RCP to operate.

To minimize the thermal shock on the RCPB, the two RCP remain in operation as the water flow of the DHRS is initiated. The DHRS flow rapidly mixes with the reactor coolant; but during this period, the indicated RCS temperature may fluctuate until mixing is complete. After the switch is completed, the RCP are stopped. During this phase, the cooldown rate is controlled by the temperature and flow of the DHRS.

5.2.3.2.2 Preservice System Hydrostatic Test

Prior to initial operation, the RCS is hydrostatically tested in accordance with ASME Code requirements. During this test, the system is brought up to an internal pressure not less than 1.25 times the system design pressure. This minimum test pressure is in accordance with Article NB-6000 of ASME Section III. Since the system design pressure is 2500 psig, the preservice system hydrostatic test pressure is 3125 psig. Initially, the RCS is heated to a temperature above the calculated minimum test temperature required for adequate fracture toughness. This heatup is accomplished by running the RCP. The pressurizer heaters are used to heat the pressurizer to the required temperature. Before the test temperature is reached, the pressure is maintained above NPSH required for the RCP but below the maximum allowable pressure for adequate fracture toughness. When the test temperature is reached, the RCP are stopped and RCS makeup water is added to fill the pressurizer. The test pressure is then reached using either the pressurizer heaters or the hydrostatic pumps connected to the RCS. The test pressure is held for the minimum specified time, and the examination for leakage follows in accordance with the ASME Code.

5.2.3.2.3 Inservice System Leak and Hydrostatic Tests

When the inservice system leak and hydrostatic tests are required, the system is brought from a cold to a hot shutdown condition. The means of heating the system and increasing the pressure are the same as those used during normal heatup. If it is necessary to cool the system down after either test, normal cooldown procedures are used. These two tests are conducted in accordance with the requirements of ASME Section XI, Article IWB-5000. The test pressure for the inservice leak tests is the pressure that, for the component located at the highest elevation in the system, is no less than the system nominal operating pressure at 100 percent rated reactor power. For the inservice hydrostatic test, ASME Section XI gives a table of the minimum test pressure versus the test temperature at which the system must be tested. However, the test temperature for both the inservice leak and hydrostatic tests is determined by the requirements for fracture toughness.

5.2.3.2.4 Reactor Core Operation

The reactor core is not allowed to become critical until the RCS fluid temperature is above 525°F except for brief periods of low-power physics testing. This temperature is much higher than the minimum permissible temperature for the inservice system hydrostatic pressure test, and it is also at least 40°F above the calculated minimum temperature required at normal pressure for operation throughout the service life of the plant.

5.2.3.3 Reactor Vessel

The ability of the reactor pressure vessel to resist fracture is the primary factor in ensuring the safety of the
primary system in light water cooled reactors. The beltline region of the reactor vessel is the most critical
region of the vessel because it is exposed to neutron irradiation. The general effects of fast neutron
irradiation on the mechanical properties of such low-alloy ferritic steels as SA302B, Code Case 1339, used
1 in the fabrication of the Oconee 1 reactor vessel, and SA508, Class 2, used in the fabrication of Oconee 2
1 and 3 reactor vessels, are well characterized and documented in the literature. The low-alloy ferritic steels
used in the beltline region of reactor vessels exhibit an increase in ultimate and yield strength properties
with a corresponding decrease in ductility after irradiation. In reactor pressure vessel steels, the most
serious mechanical property change is the increase in temperature for the transition from brittle to ductile
1 fracture accompanied by a reduction in the Charpy upper-shelf impact strength.

1 10 CFR 50, Appendix G, "Fracture Toughness Requirements," specifies minimum fracture toughness
requirements for the ferritic materials of the pressure-retaining components of the reactor coolant pressure
boundary (RCPB) of water-cooled power reactors and provides specific guidelines for determining the
pressure-temperature limitations on operation of the RCPB. The toughness and operational requirements
are specified to provide adequate safety margins during any condition of normal operation, including
anticipated operational occurrences and system hydrostatic tests, to which the pressure boundary may be
1 subjected over its service lifetime. Although the requirements of 10 CFR 50, Appendix G, became
effective on August 13, 1973, the requirements are applicable to all boiling and pressurized water-cooled
nuclear power reactors, including those under construction or in operation on the effective date.

1 10 CFR 50, Appendix H, "Reactor Vessel Materials Surveillance Program Requirements," defines the
material surveillance program required to monitor changes in the fracture toughness properties of ferritic
materials in the reactor vessel beltline region of water-cooled reactors resulting from exposure to neutron
irradiation and the thermal environment. Fracture toughness test data are obtained from material
specimens withdrawn periodically from the reactor vessel. These data will permit determination of the
condition under which the vessel can be operated with adequate safety margins against fracture
throughout its service life.

1 A method for guarding against brittle fracture in reactor pressure vessels is described in the ASME Boiler
1 and Pressure Vessel Code, Section III, Appendix G. This method utilizes fracture mechanics concepts
1 and the reference nil-ductility temperature, RT_{NDT}, which is defined in ASME Section III, Paragraph NB
1 2331. The RT_{NDT} of a given material is used to index that material to a reference stress intensity factor
curve (K_{IR} curve), which appears in Appendix G of ASME Section III. The K_{IR} curve is a lower bound
of dynamic, static, and crack arrest fracture toughness results obtained from several heats of pressure vessel
steel. When a given material is indexed to the K_{IR} curve, allowable stress intensity factors can be obtained
1 for the material as a function of temperature. Allowable operating limits can then be determined using
1 the allowable stress intensity factors.

The RT_{NDT} and, in turn the operating limits of a nuclear power plant, can be adjusted to account for the
effects of radiation on the properties of the reactor vessel materials. The radiation embrittlement and the
1 resultant changes in mechanical properties of a given pressure vessel steel can be monitored by a
surveillance program in which a surveillance capsule containing prepared specimens of the reactor vessel
materials is periodically removed from the operating nuclear reactor and the specimens tested. The
7 increase in the Charpy V-notch 30-ft-lb temperature, is added to the original RT_{NDT} along with a margin
1 value to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} is used to index the
material to the K_{IR} curve, which, in turn is used to set operating limits for the nuclear power plant.
These new limits take into account the effects of irradiation on the reactor vessel materials.

5.2.3.3.1 Stress Analysis

A stress evaluation of the reactor vessel was initially performed in accordance with Section III of the ASME Code. The evaluation showed that stress levels are within the Code limits.

7 Table 5-6 lists the reactor vessel steady-state stresses from the initial stress evaluation at various load
7 points. The results of the initial transient analysis and the determination of the initial fatigue usage factor
7 at the same load points are listed in Table 5-7. Calculation OSC-1815 provides the current stress values
7 and fatigue usage factors for the reactor vessel. As specified in the ASME Code, Section III, Paragraph
415.2(d)(6), the cumulative fatigue usage factor is less than 1.0 for the design cycles listed in Table 5-2.
Figure 5-10 illustrates the points of stress analysis for the stresses listed in Table 5-6 and the fatigue usage
factors listed in Table 5-7.

7 The initial stress summaries provided in UFSAR Table 5-6 and Table 5-7 demonstrated that all of the
requirements for stress limits and fatigue required by ASME Section III for all of the operational
7 requirements imposed by the design specifications were met (the current stress analysis is presented in
7 calculation OSC-1815). The values tabulated in these summaries were the maximum value obtained in
each region. The imposed transients are based on description of the realistic behavior that might be
expected for this plant. Transients such as loss of flow and load that cause temperature and pressure
variations are included in the reactor vessel specification and Table 5-2. Their effect on accumulated
7 usage factor were included in the initial stress analysis as summarized in Table 5-7. These transients were
7 not the major contributors to the largest usage factor of 0.38 for the stud bolts from the initial fatigue
7 evaluation as given in Table 5-7. The current reactor vessel fatigue evaluation provided in OSC-1815
7 shows that the largest usage factor for the stud bolts remains less than 1.0."

5.2.3.3.2 Reference Nil-Ductility Temperature (RT_{NDT})

Throughout the lifetime of a reactor vessel, the impact and tensile properties of the ferritic beltline region materials will change because of neutron irradiation. These changes require periodic adjustment of pressure-temperature relationships for heatup and cooldown during normal, upset, and testing conditions.

To determine the pressure-temperature operating limitations for the RCPB the reference nil-ductility
temperature (RT_{NDT}) of the ferritic materials must be established. The RT_{NDT} is needed to calculate the
1 critical stress intensity factor (K_{IR}). In ASME Section III, Appendix G, K_{IR} is related to temperature, T,
and to RT_{NDT} by the following equation:

$$K_{IR} = 26.777 + 1.223 \exp[0.0145(T - RT_{NDT} + 160)] \text{ksi}\sqrt{\text{in.}}$$

This relationship is applicable only to ferritic materials that have a specified minimum yield strength of 50,000 psi or less at room temperature.

Since the impact properties of the beltline region materials of a reactor vessel will change throughout its lifetime, periodic adjustments are required on the pressure-temperature limit curves of the RCPB. The magnitude of these adjustments is proportional to the shift in RT_{NDT} caused by neutron fluence. Therefore, it is essential to determine the radiation-induced ΔRT_{NDT} of the beltline region materials.

1

The RT_{NDT} of the ferritic materials, which were specified and tested in accordance with the fracture toughness requirements of the ASME Section III Summer 1972 Addenda (to 1971 Edition) or subsequent addenda, are determined as required by that Code. When enough material is available, the RT_{NDT} of those beltline region materials, which were specified and tested in accordance with an edition or addenda of ASME Section III prior to the Summer 1972 Addenda, are obtained by testing specimens oriented normal to the principal working direction. The test procedure is in accordance with ASME Section III, paragraph NB 2300 (Summer 1972 Addenda).

The Oconee pressure boundaries were designed and constructed in accordance with the requirements of an edition or addenda of ASME Section III issued before the Summer 1972 Addenda. Except for the beltline region materials for which sufficient test material is available, the RT_{NDT} of the ferritic materials must be estimated. This is necessary because the test data required for the exact determination of RT_{NDT} were not required by the applicable ASME Code.

Generally, drop weight tests were not performed, and the Charpy V-notch tests were limited to "fixed" energy level requirements for specimens oriented in the longitudinal (principal working) direction at a temperature of 40°F or lower.

To obtain an RT_{NDT} estimate that is appropriately conservative, B&W has collected and evaluated the data from tests conducted on pressure-retaining ferritic materials to which the new fracture toughness requirements were applied. Based on these evaluations, techniques were developed to estimate RT_{NDT}. These techniques as well as the results are described in B&W Topical Report BAW-10046P, Reference 4.

1 10 CFR 50, Appendix G, requires complete characterization of the unirradiated impact properties of all the beltline region materials of the reactor vessel. The complete characterization includes the determination of RT_{NDT} and Charpy (C_v) test curves for the directions normal to and parallel to the principal working direction (other than the thickness direction). Appendix G also requires a minimum C_v
1 USE of 75 ft-lb for all beltline region materials unless it is demonstrated that lower values of upper-shelf fracture energy provide an adequate margin for deterioration from irradiation.

For the beltline region materials of reactor vessels that were specified in accordance with the requirements of an edition or addenda of ASME Section III issued before the Summer 1972 Addenda, the complete C_v test curves, including C_v USE, is determined when the material forms part of the reactor vessel surveillance program. For the beltline region materials that do not form part of the surveillance program, and when enough material is available, the C_v test curve and USE are determined only in the direction normal to the principal working direction. No minimum Charpy V-notch USE are required, other than the 50 ft-lbs/35 mils of lateral expansion for the beltline region materials of these reactor vessels. When the unirradiated USE of these materials is below 75 ft-lb/, the procedures described in BAW-10046P are applied to predict the end-of-service USE.

The C_v USE must be estimated for reactor vessel beltline region materials that were specified in accordance with the requirements of an edition or addenda of ASME Section III issued before the Summer 1972 Addenda and for which insufficient material is available for testing. All available data from tests conducted on reactor vessel beltline region materials were collected and evaluated in order to obtain an appropriately conservative estimate. Not all the data were obtained in accordance with the methods specified in ASME Section III, Appendix G, since in some cases the absorbed energy was obtained only at one temperature. Based on these evaluations, estimates of C_v USE were developed. The techniques and results are described in BAW-10046P.

5.2.3.3.3 Neutron Flux at Reactor Vessel Wall

The design value for the fast neutron flux greater than 1.0 MeV at the inner surface of the reactor vessel is 3.0×10^{10} n/cm^2-sec at a rated power of 2,568 MWt. The most recent corresponding calculated maximum fast neutron flux at the vessel wall is approximately a factor of 3 lower. For 40 years at 80 percent load this corresponds to a fluence of approximately 1×10^{19} n/cm^2 for the vessel wall.

A semiempirical method is used to calculate the surveillance capsule and reactor vessel flux. The method employs explicit modeling of the surveillance capsule, reactor vessel, and internals and uses a time-weighted average pin-by-pin core power distribution in the two-dimensional DOT IV, version 4.3, computer code. DOT IV is a two-dimensional code which is used to calculate the energy- and space-dependent neutron flux at all points of interest in the specific reactor system configuration. DOT IV employs the discrete ordinates method of solution of the Boltzmann transport equation and has multigroup and asymmetric scattering capability.

The calculational model is an R-theta geometric representation of a plan view through the reactor core midplane using one-eighth core symmetry. The model includes the core with a time-averaged radial power distribution core liner, coolant regions, core barrel, thermal shield, pressure vessel, and concrete. The DOT calculation is carried out with an S_8 order of angular quadrature, a P_3 expansion of the scattering matrix, and the CASK23E cross-section set. The P_3 order of scattering indicates a third order LeGendre polynomial scattering approximation which adequately describes the predominately forward scattering of neutrons observed in the deep penetration of steel and water media. This calculation provides the neutron flux as a function of energy at the detector position and, in addition to the flux, the DOT IV code calculates the saturated specific activity of the various neutron dosimeters located in the surveillance capsule using the ENDF/B5 dosimeter reaction cross-sections. The saturated activity of each dosimeter is then adjusted by a factor which corrects for fraction of saturation attained during the dosimeter's actual detailed irradiation history. Additional corrections are normally made to account for the effects of the following:

1. photon-induced fissions in the U and Np dosimeters,
2. short half-life of isotopes produced in Fe and Ni dosimeters, and
3. Pu-239 generated in the U-238 dosimeter.

These calculated activities are used for comparison with the measured dosimeter activity values. The basic equation for the calculated activity (μCi/g) is

$$D_i = \frac{N}{A_n\, 3.7\text{x}10^4}\, f_i \sum_E \sigma_n(E)\, \phi(E) \sum_{j=1} F_j (1 - e^{-\lambda_i t_j}) e^{-\lambda_i (T - \tau_j)}$$

where:

1

N = Avagadro's number,

A_n = atomic weight of target material n,

f_i = either weight fraction of target isotope in nth material or fission yield of desired isotope,

$\sigma_n(E)$ = group-averaged cross sections for material n

$\phi(E)$ = group-averaged fluxes calculated by DOT analysis,

F_j = fraction of full power during jth time interval t_j,

λ_i = decay constant of ith material,

1 t_j = length of the jth time period,

T = sum of total irradiation time, i.e., residual time in reactor and wait time between reactor shutdown and counting,

τ_j = cumulative time from reactor startup to end of jth time period, i.e.,

$$\tau_j = \sum_{k=1}^{j} t_k.$$

1 The flux normalization factor C_i is then obtained by the following equation:

$$C_i = \frac{D_i\,(\text{measured})}{D_i\,(\text{calculated})}$$

1 With C specified, the neutron fluence greater than 1 MeV can be calculated from

$$\phi t(E > 1.0\,\text{MeV}) = C \sum_{E=1}^{E=15\text{MeV}} \phi(E) \sum_{j=1}^{j=M} F_j t_j$$

where M is the number of irradiation time intervals; the other values are defined above.

The specific results of these calculations are included in the specific capsule evaluation reports prepared as
8 part of the Reactor Vessel Materials Surveillance Program (FSAR Section5.2.3.12, "Tests and
8 Inspections").

5.2.3.3.4 Radiation Effects

1 The adjusted reference temperatures are calculated by adding the predicted radiation-induced ΔRT_{NDT}, the
1 unirradiated RT_{NDT}, and a margin value. The predicted ΔRT_{NDT} is calculated using the respective
1 neutron fluence and copper and nickel contents. The design curves of Regulatory Guide 1.99 were used to predict the radiation-induced ΔRT_{NDT} values as a function of the material's copper and phosphorous
1 content and neutron fluence. With the issuance of Rev. 2 of Regulatory Guide 1.99 in May, 1988,
1 ΔRT_{NDT} values are obtained on the basis of copper and nickel contents.

The effects of radiation on the Charpy USE level of the beltline region material is estimated using the
1 curves shown in Regulatory Guide 1.99, Rev. 2, Figure 2.

1

Several operating plant reactor vessels were manufactured with "high-copper MnMoNi/Linde 80"
1 submerged-arc weld metal. This class of weld metal is susceptible to relatively large changes in impact
1 properties when exposed to fast neutron irradiation. The Charpy V-notch upper-shelf energy (C_vUSE) of
1 some of these welds may drop below the 50 ft-lb threshold required by federal regulatory requirements (10
1 CFR 50, Appendix G) during the 40-year reactor design life. Should the C_vUSE drop below 50 ft-lb, certain corrective actions would be required that could severely impact plant availability.

One of the major goals of the B&W Owners Group Program has been to determine the period of time each 177-fuel assembly (FA) reactor vessel can operate without violating the 50 ft-lb C_vUSE threshold. The work that has been completed in this program includes reports entitled "Prediction of Charpy Upper Shelf Energy Drop in Irradiated Weld Metals," "Pressure Vessel Fluence Analysis for 177-FA Reactors," "Chemistry of B&W 177-FA Owners Group Reactor Vessel Beltline Welds."

1 BAW-1803, Rev. 1, Reference 6, describes the implementation of predictive methodology developed in this program to determine the service life to reach the 50 ft-lb C_vUSE threshold for each of the Owners Group reactor vessels. It was also necessary to establish a means of predicting the pre-service C_vUSE of each of the beltline region reactor vessel welds. The available C_vUSE data obtained from B&W manufactured, early vintage welds (high-Cu MnMoNi/Linde 80 submerged-arc) were analyzed collectively for this purpose.

1 Based on the developed methods, the limiting Oconee reactor welds are predicted to exhibit a C_vUSE of
8 more than 50 ft-lb for >32 EFPY plant operation for the 40-year design life (BAW-2192PA, Reference 30
8 and BAW-2178PA, Reference 31).

5.2.3.3.5 Fracture Mode Evaluation

An analysis has been made to demonstrate that the reactor vessel can accommodate without failure the rapid temperature change associated with the postulated operation of the Emergency Core Cooling System (ECCS) at end of vessel design life. A summary of the evaluation follows:

The state of stress in the reactor vessel during the loss-of-coolant accident was evaluated for an initial vessel temperature of 603°F. The inside of the vessel wall is rapidly subjected to 90°F injection water of the maximum flow rate obtainable. The results of this analysis show that the integrity of the vessel is not violated.

The assumed modes of failure are ductile yielding and brittle fracture, which includes the nil-ductility approach and the fracture mechanics approach. The modes of failure are considered separately in the following paragraphs.

Ductile Yielding

The criterion for this mode of failure is that there shall be no gross yielding across the vessel wall using the minimum specified yield strength in the ASME Code, Section III. The analysis considered the maximum combined thermal and pressure stresses through the vessel wall thickness as a function of time during the safety injection. Comparison of calculated stresses to the material yield stress indicated that local yielding may occur in the inner 8.0 percent of the vessel wall thickness.

Brittle Fracture

Because the reactor vessel wall in the core region is subjected to neutron flux resulting in embrittlement of the steel, this area was analyzed from both a nil-ductility approach and a fracture mechanics approach.

The results of the two methods of analysis compare favorably and show that pressure vessel integrity is maintained.

The criterion used in the nil-ductility approach is that a crack cannot propagate beyond any point where the applied stress is below the threshold stress for crack initiation (5-8 ksi), or when the stress is compressive. This approach involves making the very conservative assumption that all of the vessel
1 material could propagate a crack by a low-energy absorption or cleavage mode. End-of-life vessel conditions were assumed. The crack arrest temperature through the thickness of the wall was developed on a stress-temperature coordinate system. The actual quench-induced, stress-temperature condition through the thickness of the wall at several times during the quench was developed and plotted. The maximum depth at which the material in the vessel wall would be in tension or at which the stress in the
1 material would be in excess of the threshold stress for crack initiation (5-8 ksi) was determined by comparison of the plots. The comparison showed that a crack could propagate only through the inner 35 percent of the wall thickness if a crack initiation threshold of 5-8 ksi is applicable.

The foregoing method of analysis is essentially a stress analysis approach which assumes the worst conceivable material properties and a flaw size large enough to initiate a crack. Actually, the outer 83 percent of the vessel wall is at a temperature above the Ductility Transition Temperature (DTT) (NDTT + 60°F) when credit is taken for the neutron shielding, and for the original DTT profile through the wall thickness. The analysis is conservative in that it does not deny that cracks can be initiated, and in that it assumed a crack from 1 to 2 ft long to exist in the vessel wall at the time of the accident. Therefore, it can be concluded that, if a crack were present in the worst location and orientation (such as a circumferentially oriented crack on the inside of the vessel wall), it could not propagate through the vessel wall.

A fracture mechanics analysis was conducted which assumed a continuous surface flaw to exist on the inside surface of the vessel wall. The criterion used for the analysis is that a crack cannot propagate when the stress intensity at the tip of the crack is below the critical crack stress intensity factor (K_{IC}). Topical Report BAW-10018, Reference 7, provides the details of the analysis. This report includes an evaluation considering the Irwin fracture mechanics method and performs a sensitivity analysis of the effect of varying the conservatism of several major parameters on the result.

5.2.3.3.6 Pressurized Thermal Shock

In response to the TMI Action Plan (Item II.K.2.13 "Thermal-Mechanical Report") the effect of cold high pressure injection water entering the reactor vessel during a small break loss of coolant accident or an overcooling transient was considered. The concern was that the cold injection water could rapidly cool the reactor vessel welds and that the resulting thermal stresses, coupled with the relatively high pressure stress on the vessel, would lead to a loss of vessel integrity. This type of event is a particular concern later in life as the vessel neutron fluence increases and the metal becomes more brittle. Various vendor, utility, and EPRI research performed in response to this action item showed that good mixing of the injection water with the warmer Reactor Coolant System fluid would occur, even under near zero loop flow conditions. In particular, the vent valves in the Oconee plant would provide a source of heated water flowing directly from the vessel upper plenum to the downcomer, thus mitigating the cooling effect of the injection flow. The NRC Staff concluded that there is reasonable assurance that vessel integrity would be maintained during a II.K.2.13 event (Reference 12).

1 The NRC amended its regulations for light water nuclear power plants, effective July 23, 1985, to
1 establish a screening criterion related to the fracture resistance of PWR vessels during PTS events. Only
1 those plants that exceed the screening criterion are required to perform further analysis using Regulatory
8 Guide 1.154. All Oconee units passed the screening criterion (References 32, 35, and 36) and, therefore,
1 met the regulations regarding the PTS concern. This rule was further amended on June 14, 1989, to make
1 the definition of RT_{PTS} equal to RT_{NDT} in Regulatory Guide 1.99, Rev. 2. Assessment in accordance

8 with the amended rule is complete (BAW-2143, Reference 23). All Oconee units satisfy this revised
1 screening criterion.

9 Section 50.61(b)(1) provides rules for protection against pressurized thermal shock events for pressurized
9 water reactors. Licensees are required to perform an assessment of the projected values of reference
9 temperature whenever there is a significant change in projected values of RT_{PTS}, or upon request for a
9 change in the expiration date for the operation of the facility. For license renewal, RT_{PTS} values are
9 calculated for 48 EFPY for Oconee Units 1, 2, and 3.

9 Section 50.61(c) provides two methods for determining RT_{PTS}: (Position 1) for material that does not
9 have credible surveillance data available, and (Position 2) for material that does have credible surveillance
9 data. Availability of surveillance data is not the only measure of whether Position 2[1] may be used; the
9 data must also meet tests of sufficiency and credibility.

9 $\mathbf{RT_{PTS}}$ is the sum of the initial reference temperature (IRT_{NDT}), the shift in reference temperature caused
9 by neutron irradiation (ΔRT_{NDT}), and a margin term (M) to account for uncertainties.

9 $\mathbf{IRT_{NDT}}$ is determined using the method of Section III of the ASME Boiler & Pressure Vessel Code.
9 That is, IRT_{NDT} is the greater of the drop weight nil-ductility transition temperature or the temperature
9 that is 60°F below that at which the material exhibits Charpy test values of 50 ft-lbs and 35 mils lateral
9 expansion. For a material for which test data is unavailable, generic values may be used if there are
9 sufficient test results for that class of material. For Linde 80 weld material with the exception of WF-70,
9 the IRT_{NDT} is taken to be the currently NRC accepted values of -7°F or -5°F. For WF-70, the IRT_{NDT}
9 is similarly taken to be a measured value, -26.5°F, in accordance with the discussion and results presented
9 in BAW-2202[2] [Reference 42]. For forgings and plate material, measured values are used where
9 appropriate data is available. Where not available, the generic value of +3°F is used for forgings and
9 +1°F is used for plate material [Reference 43].

9 For Position 1 material (surveillance data not available), $\mathbf{\Delta RT_{NDT}}$ is defined as the product of the
9 chemistry factor (**CF**) and the fluence factor (**ff**). **CF** is a function of the material's copper and nickel
9 content expressed as weight percent. "Best estimate" copper and nickel contents are used which is the
9 mean of measured values for the material. For Oconee, best estimate values were obtained from the
9 following FTI reports: BAW-1820, BAW-2121P, BAW-2166, and BAW-2222[3] [References 44, 45, 46, and
9 47]. The value of **CF** is directly obtained from tables in Section 50.61. **ff** is a calculated value[4] using
9 end-of-license (EOL) peak fluence at the inner surface at the material's location. Fluence values were
9 obtained by extrapolation to 48 EFPY of the current 32 EFPY values for each Oconee unit.

9 For beltline welds and plate materials for which surveillance data is available, evaluations were performed
9 in accordance with Regulatory Guide 1.99, Revision 2, Position 2. The applicable chemistry factors,
9 margin, and RT_{PTS} at 48 EFPY are summarized in Table 5-24, Table 5-25, and Table 5-26.

---

9 [1] The term "Position" is taken from Regulatory Guide 1.99, the methodology of which was incorporated into 10
9 CFR 50.61.

9 [2] BAW-2202 is an FTI topical report submitted to the NRC for their acceptance on September 29, 1993. The
9 NRC's acceptance for use at the Zion plants was published in the Federal Register, Vol. 59, No. 40 Pages 9782 -
9 9785, March 1, 1994.

9 [3] BAW-1820 and BAW-2121P were provided to the NRC for their information. BAW-2166 and BAW-2222 were
9 provided to the NRC as part of the Generic Letter 92-01 program.

9 [4] $ff = f^{(0.28-0.1 * \log f)}$, where $f = fluence * 10^{-19}$ (n/cm^2, $E > 1MeV$).

9 For Position 2 material (surveillance data available), the discussion above for Position 1 applies except for
9 determination of CF, which in this instance is a material-specific value calculated as follows:

9 1. Multiply each ΔRT_{NDT} value by its corresponding **ff**.

9 2. Sum these products.

9 3. Divide this sum by the sum of the squares of the **ffs**.

9 The margin term (M) is generally determined as follows:

$$M = 2(\sigma_I^2 + \sigma_\Delta^2)^{0.5}$$

9 where σ_I is the standard deviation for IRT_{NDT}

9 and σ_Δ is the standard deviation for ΔRT_{NDT}.

9 For Position 1, $\sigma_I = 0$ if measured values are used. If generic values are used, σ_I is the standard deviation
9 of the set of values used to obtain the mean value. For ΔRT_{NDT}, σ_Δ = 28°F for welds and 17°F for base
9 metal (plate and forgings), except that σ_Δ need not exceed one-half of the mean value of ΔRT_{NDT}. For
9 Position 2, the same method for determining the σ values are used except that the σ_Δ values are halved
9 (14°F for welds and 8.5°F for base metal).

9 Section 50.61(b)(2) establishes screening criteria for RT_{PTS} 270°F for plates, forgings, and axial welds and
9 300°F for circumferential welds. The values for RT_{PTS} at 48 EFPY are provided in Table 5-24,
9 Table 5-25, and Table 5-26 for Units 1, 2, and 3, respectively. The RT_{PTS} values reported herein are
9 based on updated 48 EFPY fluence projections using the evaluation based methodology described in
9 BAW-2251 [Reference 48, Appendix D] and BAW-2241P [Reference 49]. The chemistry and
9 surveillance data for the beltline materials are reported in BAW-2325 [Reference 50].

9 The projected RT_{PTS} values for Units 1, 2 and 3 are within the established screening criteria for 48 EFPY.
9 For Unit 1, the limiting weld is SA-1073 with a projected value of RT_{PTS} at 48 EFPY of 230.3°F
9 (screening limit of 270°F). For Unit 2, the limiting weld is WF-25, with a projected value of RT_{PTS} at 48
9 EFPY of 296.8°F (screening limit of 300°F). For Unit 3, the limiting weld is WF-67 with a projected
9 value of RT_{PTS} at 48 EFPY of 253.5°F (screening limit of 300°F). [Reference 51]

9 Reference for this section: Final SER [Reference 39].

5.2.3.3.7 Closure

The reactor closure is bolted to a ring flange on the reactor vessel. The vessel closure seal is formed by two concentric metal O-ring seals with provisions for leak-off between the O-rings. Reactor closure head leakage will be negligible from the annulus between the metallic O-ring seals during vessel steady-state and virtually all transient operating conditions. Only in the event of a rapid transient operation, such as an emergency cooldown, would there be some leakage past the inner-most O-ring seal. A stress analysis on a similar vessel design indicates this leak rate would be approximately 10 cc/min and no leakage would occur past the outer O-ring seal.

The reactor closure head is attached to the reactor vessel with sixty 6-1/2 in. diameter studs. The studs have a minimum yield strength of 130,000 psi. The studs, when tightened for operating conditions, will have a tensile stress of approximately 30,000 psi. An evaluation of stud failures shows that:

1. 10 adjacent studs can fail before leak occurs.
2. 25 adjacent studs can fail before the remaining studs reach yield strength.

3. 26 adjacent studs can fail before the remaining studs reach the ultimate tensile strength.
4. 43 symmetrically located studs can fail before the remaining studs reach yield strength.

The fatigue evaluation results of the studs is included in Table 5-7.

5.2.3.3.8 Control Rod Drive Service Structure

The control rod drive service structure is designed to support the control rod drives to assure no loss of function in the event of a combined loss of coolant accident and maximum hypothetical earthquake. Requirements for rigidity, imposed on the structure to avoid adversely affecting the natural frequency of vibration of the vessel and internals, as well as space requirements for service routing, result in stress levels considerably lower than design limits. The structure is more than adequate to perform its required function.

6 5.2.3.3.9 Control Rod Drive Mechanism

Appendix G to 10 CFR 50 requires that the adequacy of the fracture toughness properties of ferritic materials such as type 403 modified stainless steel be demonstrated to the Commission on a case-by-case basis. The type 403 modified steel is used as an RCPB material in the motor tube of the control rod drive mechanism. This section demonstrates that, for this application, the material has adequate fracture toughness for protection against non-ductile failure.

The nominal wall thickness of the motor tube section of interest is more than 1/2 inch and less than 5/8 inch. In the early editions of ASME Section III up to the Winter 1971 Addenda to the 1971 Edition, materials with a nominal section thickness of 1/2 inch or less did not require impact testing. Starting with the Summer 1972 Addenda, the nominal section thickness increased to 5/8 inch or less. Thus, in the early editions of ASME Section III, the Type 403 modified steel required impact testing, but in the new editions it does not. However, since this material was selected for use, B&W has ordered it to meet the impact toughness requirements for ASME Section III, Summer 1972 and later Addenda, the imposed acceptance standard for nominal wall thicknesses from 5/8 to 3/4 inch, inclusive is presented in paragraph NB-2332. The material has also been specified to meet the requirements of SA 182 grade F6 (forgings) or ASTM A276 (bars) as modified by ASME Code Case 1337.

When ordered according to the early revisions of Code Case 1337 (including Revision 6) and to the early editions of ASME Section III, the type 403 modified forgings or bars were required to be impact-tested at 20°F. The minimum average energy of a set of three Charpy V-notch specimens was 35 ft-lb, with one specimen allowed to be less than 35 but not less than 30 ft-lb. For both forgings and bars, the Charpy specimens were oriented in the axial (longitudinal) direction.

In the Summer 1972 Addenda to the 1971 Edition of ASME Section III, the fracture toughness requirements of all pressure boundary ferritic materials changed; however, no acceptance criterion was given for the martensitic high-alloy chromium steels, such as type 403 modified steel. A year later, the Summer 1973 Addenda re-established the acceptance criteria for the type 4XX steels. Beginning with this addenda, the fracture toughness requirements and acceptance criteria for the type 4XX steels are described in paragraph NB-2332 of ASME Section III. This paragraph requires that three Charpy V-notch specimens be tested at temperatures lower than or equal to the lowest service temperature. The lateral expansion of each specimen must be equal to or greater than 20 mils. The test temperature has been specified as equal to or less than 40°F. The orientations of the specimens are transverse (normal to principal working direction) for the forgings and axial for the steel bars.

The fracture toughness requirements of Code Case 1337, starting with Revision 7, are the same as those of ASME Section III, Summer 1973 Addenda to the 1971 Edition.

It is considered that the fracture toughness requirements of the new edition. of ASME Section III provide adequate protection against nonductile failure. The proof of adequate toughness is based on demonstrating that the type 403 modified steels used in the construction of components designed to an edition or addenda of ASME Section III prior to the Summer 1973 Addenda meet or exceed the toughness requirements of that addenda.

Based on actual test data, the lowest service temperature of the control rod drive mechanism can be as low as 40°F; however, for additional protection against non-ductile failure, B&W has defined the component's lowest service temperature at 100°F. This specified lowest service temperature is 60°F above the temperature at which the fracture toughness requirements are specified and met. The additional 60°F provides margins of safety beyond that required by the ASME code and by Appendix G to 10 CFR 50.

9 5.2.3.3.10 Charpy Upper-Shelf Energy

9 Appendix G of 10 CFR 50 requires that reactor vessel beltline materials "have Charpy upper-shelf energy
9 ... of no less than 75 ft-lb initially and must maintain Charpy upper-shelf energy throughout the life of
9 the vessel of no less than 50 ft-lb ... ." The B&WOG positions on upper shelf energy for 32 EFPY are
9 documented in the responses to Generic Letter 92-01, as reported in BAW-2166 and BAW-2222 and, the
9 low upper shelf toughness analyses documented in BAW-2275 [Reference 52], which is included in
9 BAW-2251 as Appendix B.

9 Regulatory Guide 1.99, Revision 2 provides two methods for determining Charpy upper-shelf energy
9 (C_VUSE): Position 1 for material that does not have credible surveillance data available and Position 2
9 for material that does have credible surveillance data. For Position 1, the percent drop in C_VUSE, for a
9 stated copper content and neutron fluence, is determined by reference to Figure 2 of Regulatory Guide
9 1.99, Revision 2. This percentage drop is applied to the initial C_VUSE to obtain the adjusted C_VUSE.
9 For Position 2, the percent drop in C_VUSE is determined by plotting the available data on Figure 2 and
9 fitting the data with a line drawn parallel to the existing lines that upper bounds all the plotted points.

9 The 48 EFPY C_VUSE values were determined for the reactor vessel beltline materials for each Oconee
9 Unit and are reported in Table 5-27, Table 5-28, and Table 5-29. The T/4 fluence values reported in
9 these tables were calculated in accordance with the ratio of inner surface to T/4 values (i.e. neutron
9 fluence lead factors at T/4) determined in the latest Reactor Vessel Surveillance Program report. As
9 shown in these tables, the C_VUSE is maintained above 50 ft-lb for base metal (plates and forgings),
9 however, for Oconee the C_VUSE for weld metal drops below the required 50 ft-lb level at 48 EFPY.
9 Appendix G of 10 CFR 50 provides for this by allowing operation with lower values of C_VUSE if "it is
9 demonstrated ... that the lower values of Charpy upper-shelf energy will provide margins of safety against
9 fracture equivalent to those required by Appendix G of Section XI of the ASME Code."

9 This equivalent margin analysis was performed for 48 EFPY and is reported in BAW-2275 for service
9 levels A, B, C, and D. The analysis used very conservative material models and load combinations, i. e.,
9 treating thermal gradient stress as a primary stress. For service levels A and B, the analytical results
9 demonstrate that there is sufficient margin beyond that required by the acceptance criteria of Appendix K
9 of the ASME Code (1995 Edition). For service levels C and D, the most limiting transient was evaluated,
9 and again the analytical results demonstrate that there is sufficient margin beyond that required by the
9 acceptance criteria of Appendix K of the ASME Code. The evaluations for all service levels conclusively
9 demonstrate the adequacy of margin of safety against fracture for the reactor vessels within the scope of
9 this report for 48 EFPY.

9 5.2.3.3.11 Intergranular Separation in HAZ of Low Alloy Steel under Austenitic SS Weld Cladding

9 Intergranular separations in low alloy steel heat-affected zones under austenitic stainless steel weld
9 claddings were detected in SA-508, Class 2 reactor vessel forgings manufactured to a coarse grain practice,

9 and clad by high-heat-input submerged arc processes. BAW-10013 contains a fracture mechanics analysis
9 that demonstrates the critical crack size required to initiate fast fracture is several orders of magnitude
9 greater than the assumed maximum flaw size plus predicted flaw growth due to design fatigue cycles. The
9 flaw growth analysis was performed for a 40-year cyclic loading, and an end-of-life assessment of radiation
9 embrittlement (i.e., fluence at 32 EFPY) was used to determine fracture toughness properties. The report
9 concluded that the intergranular separations found in B&W vessels would not lead to vessel failure. This
9 conclusion was accepted by the Atomic Energy Commission[5] To cover the period of extended operation,
9 an analysis was performed using current ASME Code requirements; this analysis is fully described in
9 BAW-2274 [Reference 53] which is contained in BAW-2251 as Appendix C.

9 In May 1973, the Atomic Energy Commission issued Regulatory Guide 1.43, "Control of Stainless Steel
9 Weld Cladding of Low-Alloy Steel Components," [Reference 54]. The guide states that underclad
9 cracking "has been reported only in forgings and plate material of SA-508 Class 2 composition made to
9 coarse grain practice when clad using high-deposition-rate welding processes identified as 'high-heat-input'
9 processes such as the submerged-arc wide-strip and the submerged-arc 6-wire processes. Cracking was not
9 observed in clad SA-508 Class 2 materials clad by 'low-heat-input' processes controlled to minimize
9 heating of the base metal. Further, cracking was not observed in clad SA-533 Grade B Class 1 plate
9 material, which is produced to fine grain practice. Characteristically, the cracking occurs only in the
9 grain-coarsened region of the base-metal heat-affected zone at the weld bead overlap." The guide also
9 notes that the maximum observed dimensions of these subsurface cracks is 0.165-inch deep by 0.5-inch
9 long.

9 The BAW-10013 fracture mechanics analysis is a flaw evaluation performed before the ASME Code
9 requirements for flaw evaluation, the K_{Ia} curve for ferritic steels as indexed against RT_{NDT}, and the
9 ASME Code fatigue crack growth curves for carbon and low alloy ferritic steels were available. The
9 revised analysis uses current fracture toughness information, applied stress intensity factor solutions, and
9 fatigue crack growth correlations for SA-508 Class 2 material. The objective of the analysis is to
9 determine the acceptability of the postulated flaws for 48 EFPY using ASME Code, Section XI, (1995
9 Edition), IWB-3612 acceptance criteria.

9 The revised analysis was applied to three relevant regions of the reactor vessel: the beltline, the nozzle belt,
9 and the closure head/head flange. The analysis conservatively considered 360 cycles of 100°F/hr normal
9 heatup and cooldown transients. For the power maneuvering transients, the range in applied stress
9 intensity factors for the closure head region were assumed to be the same as that determined for the
9 beltline region. This assumption is considered conservative since the closure head region is subject to a
9 low flow condition while the beltline region is subject to a forced flow condition.

9 An initial flaw size of 0.353-inch deep by 2.12-inch long (6:1 aspect ratio) was conservatively assumed for
9 each of the three regions. The flaw was further assumed to be an axially oriented, semi-elliptical surface
9 flaw in contrast to the observed flaws which are subsurface with a maximum size of 0.165-inch deep by
9 0.5-inch long.

9 The maximum crack growth and applied stress intensity factor for the normal and upset conditions were
9 found to occur in the nozzle belt region. The maximum crack growth, considering all the normal and
9 upset condition transients for 48 EFPY, was determined to be 0.180-inch, which results in a final flaw
9 depth of 0.533-inch. The maximum applied stress intensity factor for the normal and upset condition
9 results in a fracture toughness margin of 3.6 which is greater than the IWB-3612 acceptance criterion of
9 3.16.

9 [5] R. C. DeYoung (USAEC) to J. F. Mallay (B&W), letter transmitting topical report evaluation, October 11, 1972.

9 The maximum applied stress intensity factor for the emergency and faulted conditions occurs in the
9 closure head to head flange region and the fracture toughness margin was determined to be 2.24, which is
9 greater than the IWB-3612 acceptance criterion of 1.41. It is therefore concluded that the postulated
9 intergranular separations in the Oconee Unit 1, 2, and 3 reactor vessel 508 Class 2 forgings are acceptable
9 for continued safe operation through the period of extended operation.

5.2.3.4 Steam Generators

Research and Development

In August of 1964, B&W began design and construction of facilities to test full scale sections of the Once Through Steam Generator. Since that time, three different test models of the Once Through Steam Generator have been tested. The design criteria for the test steam generators were as follows:

1. To provide a test steam generator for investigation of the operational characteristics of the steam generator such as heat transfer, pressure drop, control characteristics (including measurements necessary for control), and stability.
2. To provide a test steam generator for investigation of manufacturing procedures, fouling characteristics, and cleaning procedures.
3. To provide a test steam generator which could be non-destructively examined and analyzed with respect to vibration, corrosion, and unit integrity.

The design bases for the test steam generators were:

1. To duplicate tube length, tube thickness, and tube diameters of the full size steam generator.
2. To duplicate important dynamic characteristics such as secondary flow area per tube, downcomer annulus area, and feedwater spray velocity.
3. To operate the test units under temperature, pressure, and control conditions of the full size units.

The general objectives of the model tests include:

1. Heat transfer tests
2. Pressure drop tests
3. Stability tests
4. Fouling and cleaning tests
5. Mechanical design tests including vibration, and structural tests.

In April, 1971, B&W submitted a topical report, BAW-10027, Reference 8 General results and evaluation of the model tests including the following were reported in BAW-10027:

1. The steady state and transient operation tests have confirmed the analytically predicted performance characteristics of the steam generator, and have provided the data for the control system.
2. Feedwater spray nozzle tests have demonstrated that the design will satisfactorily heat the feedwater.
3. Tube leak simulation tests have demonstrated that a leak in one tube will not propagate by causing a failure in adjacent tubes.
4. Mechanical tests have demonstrated that the tubes can withstand, without failure, the mechanical loads they may experience either during normal operation or accident conditions.
5. Vibration testing demonstrated that the unit contained no undesirable resonance characteristics.

6. Tests to simulate a steam line failure or reactor coolant system failure have demonstrated the integrity of the steam generator under conditions of rapid depressurization and large temperature differentials between the tubes and the shell of the unit.

7. Secondary side fouling tests demonstrated that fouling will be detected by increased pressure drop in the downcomer. Feedwater nozzle flooding causes the downcomer water temperature to fall below saturation temperature. Feedwater nozzle flooding is prevented in high downcomer level limits which restrict and/or limit feedwater flow. Cleaning of the secondary side of the steam generator is required when the high downcomer level limit is activated at full power. If the operator chooses, cleaning may be postponed indefinitely by reducing the power level to the point at which the high downcomer level limit is not actuated.

8. Additional information concerning steam generator research and development, design programs, and evaluations are contained in BAW-10027 as follows:

   a. Objectives and evaluations of all model steam generator tests.

   b. Extrapolation of model tests to full size performance.

   c. Verification test program to be conducted at Oconee 1.

   d. Cleaning processes to be used.

   e. Computer programs used in the design of the steam generator and transient analysis.

Because the steam generator is of a, straight tube-straight shell design and because of a minor difference in the coefficient of thermal expansion between Inconel and carbon steel, there exists structural limitation on the mean temperature difference between the tubes and the shell. During normal operation of the steam generator, the tube mean temperature should not be more than 32°F higher than the shell mean temperature. The maximum calculated mean tube to shell ΔT at normal operating conditions poses no problems to the structural integrity of the reactor coolant boundary. The effect of loss of reactor coolant would impose tensile stresses on the tubes and cause slight yielding across the tubes. Such a condition would introduce a small permanent deformation in the tubes but would in no way violate the boundary integrity. The rupture of a secondary pipe would cause the tubes to become warmer than the shell and may cause tube deformation. Blowdown tests simulating secondary side blowdown on a 37-tube model boiler, show that although a slight buckling in the tubes occurred, there was no loss of reactor coolant.

Calculations confirm that the steam generator tube sheet will withstand the loading resulting from a loss-of-coolant accident. The basis for this analysis is a hypothetical rupture of a reactor coolant pipe resulting in a maximum design pressure differential from the secondary side of 1050 psi. Under these conditions there is no rupture of the primary to secondary boundary (tubes and tube sheet).

The maximum primary membrane plus primary bending stress in the tube sheet under these conditions is 15,900 psi across the center ligaments which is well below the ASME Section III allowable limit of 40,000 psi at 650°F. Under the condition postulated, the stresses in the primary head show only the effect of its role as a structural restraint on the tube sheet. The stress intensity at the juncture of the spherical head with the tube sheet is 14,970 psi which is well below the allowable stress limit. It can therefore be concluded that no damage will occur to the tube sheet or the primary head as a result of this postulated accident.

In regard to tube integrity under loss of reactor coolant, actual pressure tests of 5/8 in. o.d./0.034 inch wall Inconel Tubing show collapse under an external pressure of 4,950 psig. This is a factor of safety of 4.7 against collapse under the 1,050 psig accidental application of external pressure to the tubes.

The rupture of a secondary pipe has been assumed to impose a maximum design pressure differential of 2,500 psi across the tubes and tube sheet from the primary side. The criterion for this accident permits no violation of the reactor coolant boundary (primary head, tube sheet, and tubes).

To meet this criterion, the stress limits delineated in the ASME Pressure Vessel Code, Section III, Paragraph N-714.2 for hydrotest limitations are applicable for the aforementioned abnormal operating circumstance. The referenced section states that the primary membrane stresses in the tube sheet ligaments, averaged across the ligament and through the tube sheet thickness, do not exceed 90 percent of the material yield stress at the operating temperature; in addition, the primary membrane plus primary bending stress in the tube sheet ligaments, averaged across the ligament width at the tube sheet surface location giving a maximum stress, does not exceed 135 percent of the material yield stress at the operating temperature.

An examination of stresses under these conditions show that for the case of a 2,500 psi design pressure differential, the stresses are within acceptable limits. These stresses together with the corresponding stress limits are given in Table 5-8.

The basic design criterion for the tubes assumes a pressure differential of 2,500 psi in accordance with Section III. Therefore, the secondary pressure loss accident condition imposes no extraordinary stress on the tubes beyond that normally expected and considered in Section III requirements.

The superimposed effect of secondary side pressure loss and maximum hypothetical earthquake has been considered. For this condition, the criterion is that there be no violation of the primary to secondary boundary (tube and tube sheet). For the case of the tube sheet, the maximum hypothetical earthquake loading will contribute an equivalent static pressure loading over the tube sheet of less than 5 psi (for vertical shock).

The effect of fluid dynamic forces on the steam generator internals under secondary steam break accident conditions has been simulated in a 37-tube laboratory boiler. Results of the tests show that reactor coolant boundary integrity is maintained under the most severe mode of secondary blowdown.

The ratio of allowable stresses (based on an allowable membrane stress of 0.9 of the nominal yield stress of the material) to the computed stresses for a design pressure differential of 2,500 psi are summarized in Table 5-9.

9 An assessment of SG tube loads following a large steam line break has been performed. Based upon the
9 results of testing and analysis done by B&W, steam generator tubes will not rupture concurrently with
9 rupture of the main steam line as a result of blowdown loadings. Data taken from a steam line rupture
9 test on a 37-tube model OTSG show that during the transient, the tubes become 100°F cooler than the
9 shell. The steam line rupture test has no feedwater flow after approximately 80 sec. even though the
9 worst case accident assumed continuous feedwater flow controlling on minimum level. The lack of
9 feedwater flow causes the steam pressure to drop rapidly with an associated drop in the saturation
9 temperature. This rapid drop in steam temperature causes the 100°F tube to shell temperature difference.

9 The calculations used to arrive at the tube stress intensity for an assumed worst case degraded tube are
9 based on pressure and temperature conditions as follows:

9	Primary Pressure	2185 psi
9	Secondary Pressure	0 psi
9	Mean Shell Temperature	553°F
9	Mean Tube Temperature	453°F

9 Power Level - 100 percent prior to rupture.

9 The shell temperature is not assumed to change as rapidly as tube temperature during the steam line
9 rupture transient.

9 The pressure and temperature conditions produce axial tensile tube stresses as follows:

9	1. Design size tube (no degradation)	tube bundle @ edge	15.2 ksi
9		tube bundle @ center	7.6 ksi
9			
9	2. Degraded tube with assumed 1/2 original wall thickness		
9		tube bundle @ edge	31.5 ksi
9		tube bundle @ center	15.3 ksi

9 For the assumed worst case (2) the maximum stress intensity is calculated as follows:

9 <u>Stress Intensities</u>

9 For Edge Tube (Worst Case)

$\sigma_L = 31.5$ ksi

$$\sigma_c = \frac{2185(.574)}{0.017(2)} = 37.0 \text{ ksi}$$

$\sigma_R = -2.2$ ksi

Mean Diam $= 0.557 + 0.017 = .574$

$\sigma_C = 37.0$ ksi

$$\sigma_C = \frac{P\, r_M}{t}$$

$$r_M = \frac{.574}{2}$$

$\sigma_{C-R} = 39.2$ ksi

$\sigma_{L-R} = 33.7$ ksi

$\sigma_{L-C} = 5.5$ ksi

9 As shown above, the circumferential-radial stress combination yield the maximum stress intensity which is
9 with the allowable stress limits, as calculated below:

9 σ circumferential - σ radial $= 39.2 \text{ ksi} < 1.5\,(1.2)\, S_m$

9 $39.2 \text{ ksi} < 42.0 \text{ ksi}$

9 The degraded tube primary plus secondary stress intensity is less than the Oconee Nuclear Station FSAR
9 Case III stress limit for primary stresses resulting from design loads plus pipe rupture loads.

9 The axial stress in the presumed degraded tube is at the yield point. However, the nature of thermal
9 restraint stresses limits the amount of tube deformation during the steam line rupture transient.

9 The assumption that the tube wall thickness is reduced to one-half the original value is used to
9 demonstrate that the chance of rupture is slight even when extreme effects of erosion, corrosion, vibration
9 or leakage are considered. Results of tests conducted by B&W show that the tubes are not degraded to
9 such an extent by those effects.

9 The steam line blowdown loads have been analyzed empirically and by simulation, and the results indicate
9 these loads will not cause tube failure.

Additional information discussed in BAW-10027 includes:

1. Discussion of thermal fatigue due to fluctuation and shifting of the liquid-vapor interface on the tubes,
2. Stress distributions and effective elastic constants obtained under thermal inplane and transverse loadings, and analysis of tube to tube sheet complex,
3. Vibration Analysis.

Electroslag welding is utilized on longitudinal seams of the 7-inch shell courses of the steam generator as shown in Figure 5-11. The techniques used in the electroslag welding for the Oconee steam generators are identical to those used in the electroslag welding program reported as Appendix F of Dockets No. 50-237 and 50-249 (Dresden Units 2 and 3). The procedures used were appropriately modified to reflect the difference in materials of the components being welded.

Each weld is subjected to radiographic inspection, ultrasonic inspection, and the finished surfaces of the weld are magnafluxed. In addition, each plate is ordered with excess width so that test specimens may be removed after heat treatment. Physical property test specimens including tensile and impact specimens of the base material heat affected zone and weld metal is obtained from this excess material in accordance with Section III of the ASME Code. Radiographic, ultrasonic, and magnetic particle inspection is performed in accordance with Section III of the ASME Code and as required by Code Case 1355 which permits such welds for Class A vessels.

Physical tests are performed per Section N-511 of Section III of the ASME Code. For example:

1. All weld metal tensile specimens from each heat of weld wire, batch of flux, and for each combination of heat of wire and batch of flux used is obtained and tested after heat treatment.
2. Charpy impact test specimens representing weld metal and heat affected base material for every heat of wire, batch of flux, and combination of heat of wire and batch of flux used is tested.
3. Charpy V-notch impact specimens and tensile specimens are tested for 15 percent of all production welds. Included in this 15 percent are the tests required by 1 and 2 above.

All electroslag welds are made in the vertical position. Two men, one on the inside and one on the outside of the vessel, are used to check the progress of the weld, and to insure that the prescribed welding procedure is being followed. The weld is started in a U-shaped starting fixture about six inches deep attached to the bottom of the joint. The weld stabilizes in this starting tab which is later cut off and discarded. The weld once started is not stopped until the total seam is completed.

The weld receives a heat treatment which consists of a water quench from 1625°F, and a temper of 1150°F, followed by an air cool. This post-weld heat treatment refines the grain of the weld and the base material heat affected zone such that it is virtually indistinguishable from the unaffected base material. The microstructure is the same through the weld.

In 1972, an audit revealed that documentation was incomplete for certain lots of weld filler metal used in the fabrication of the Oconee steam generators. Consequently, an investigation was conducted, the results of which are documented in B&W Topical Report, BAW-1402, Reference 9, which showed that all weld filler metal having incomplete documentation is satisfactory and acceptable.

5.2.3.5 Reliance on Interconnected Systems

The principal heat removal system interconnected with the Reactor Coolant System is the Steam and Power Conversion System. This system provides capability to remove reactor decay heat for the hypothetical case where all station power is lost. Under these conditions decay heat removal from the reactor core is provided by the natural circulation characteristics of the Reactor Coolant System. The turbine driven emergency feedwater pump supplies feedwater to the steam generators. Cooling water flow to the condenser is provided by the emergency discharge line which discharges to the tailrace of the
7 Keowee Dam. Should the condenser not be available to receive the steam generated by decay heat, which is unlikely in view of emergency discharge line flow, the water stored in the feedwater system can be pumped to the steam generators and the resultant steam vented to atmosphere to provide required
7 cooling. The analysis of the plant component functions credited for coping with the unlikely condition of
7 total loss of station power is presented in Section 8.3.2.2.4, "Station Blackout Analysis."

5.2.3.6 System Integrity

7 The Reactor Protective System (Chapter 7, "Instrumentation and Control") monitors parameters related to safe operation and trips the reactor to protect against Reactor Coolant System damage caused by high system pressure. The pressurizer code safety valves prevent Reactor Coolant System overpressure after a reactor trip as a result of reactor decay heat and/or any power mismatch between the Reactor Coolant System and the Secondary System.

As a pump-motor shaft is designed to have a natural frequency at least 20 percent above the critical speed, the shaft is too stiff to respond to any of the lower seismic frequencies. The pump and motor bearings are designed to be capable of meeting the seismic design criteria.

The design specification for the control rod drives requires that the drives be capable of withstanding the seismic loadings within the stress limits for Class I equipment.

The purchase specifications for the Emergency Core Cooling System (ECCS) pumps and valves require that the units be capable of operating under the seismic loads predicted to exist at the building elevations where the units will be located. The equipment supplier has certified that the units, based on tests which exceeded the specification requirements on similar units, do adequately meet the purchase specification requirements for operation under seismic loads. The instrumentation transmitters are tested to demonstrate their suitability for the specified seismic conditions.

The center of gravity for this type of equipment is low and both the pump and the driver are rigidly connected to a structural baseplate which in turn is bolted to the building. This type of equipment is structurally quite rigid and in most instances will accommodate very high "g" loadings.

5.2.3.7 Overpressure Protection

The Reactor Coolant System is protected against overpressure by the pressurizer code safety valves mounted on top of the pressurizer. The capacity of these valves is determined from considerations of: (1) the Reactor Protective System; (2) pressure drop (static and dynamic) between the points of highest pressure in the Reactor Coolant System and the pressurizer; and (3) accident or transient overpressure conditions.

The combined capacity of the pressurizer code safety valves is based on the hypothetical case of withdrawal of a regulating control rod assembly bank from a relatively low initial power. The accident is terminated by high pressure reactor trip with resulting turbine trip. This accident condition produces a power mismatch between the Reactor Coolant System and Secondary System larger than that caused by a turbine trip without immediate reactor trip, or by a partial load rejection from full load.

The Low Temperature Overpressure Protection (LTOP) System protects the reactor vessel from excessive pressures at low temperature conditions. As a result of Generic Letter 88-11 and a review of operating practices at Oconee, the supporting analyses for the LTOP System have been revised.

The following low temperature overpressure events have been evaluated:

1. Erroneous actuation of the High Pressure Injection System.
2. Erroneous opening of the core flood tank discharge valve.
3. Erroneous addition of nitrogen to the pressurizer.
4. Makeup control valve (makeup to the RCS) fails full open.
5. All pressurizer heaters erroneously energized.
6. Temporary loss of the Decay Heat Removal System's capability to remove decay heat from the RCS.
7. Thermal expansion of the RCS after starting a reactor coolant pump, as a result of the stored energy in the steam generators.

The reactor vessel is protected from damage during these events by the LTOP System. The LTOP System consists of two diverse trains. One train consists of the pressurizer power operated relief valve (PORV) with a lift setpoint based on the low temperature pressure limits. The pressure limits for low
9 temperature operation are 100% of the steady-state Appendix G curve. The second train consists of operator action, assisted by administrative controls, alarms, and an operating philosophy that maintains a steam or gas bubble in the pressurizer during all modes of operation (except for inservice hydrostatic testing).

The pressurizer PORV has a dual setpoint. During normal operation, the lift setpoint is 2450 psig. A lower PORV lift setpoint is used during startup and shutdown conditions. The lower setpoint is enabled by actuation of a switch in the control room whenever the RCS temperature is below 325°F. In order to
2 prevent the LTOP pressure limits from being exceeded, a low pressure setpoint is specified within
2 Technical Specifications.

The second LTOP train relies on operator action to mitigate a low temperature overpressure event. In order to assure that adequate time is available for operator action, administrative controls exist for:

1. RCS pressure;
2. Pressurizer level;
3. 2 Nitrogen addition system;
4. Number of operating reactor coolant pumps;
5. Deactivation of the A and B injection trains of the HPI System;
6. Deactivation of both core flood tanks.
7. A dedicated operator provided with approved procedures monitors RCS pressure and pressurizer level during operations at RCS temperatures below 325°F. The sole duty of the operator is to detect and mitigate LTOP transients before the RCS pressure exceeds the low temperature pressure limits.
8. 8 In addition, alarms are provided to alert the operator that an overpressure event is occurring. The
8 LTOP analysis credits either a RCS pressure or pressurizer level alarm to alert the operator. These

8 alarms ensure that a time is available for the operator to mitigate an overpressure event prior to
8 exceeding the low temperature pressure limits.

8 9. Deactivation of one bank of pressurizer heaters

The low temperature overpressure scenarios have been analyzed using conservative assumptions
8 (Reference 37). Assuming a single failure of either of the two diverse methods of overpressure protection,
the analyses demonstrate that the reactor vessel is protected from damage during events which cause
increasing pressure.

7 The two trains (active and passive) of the LTOP System taken together are single failure proof. The
7 individual trains are not single failure proof.

7 LTOP System seismic, loss of air, loss of offsite power, and IEEE-279 design requirements are as follows:

7 1. The active (PORV) and passive (Operator action) LTOP mitigation trains do not have to be
7 seismically designed,
7 2. A loss of instrumentation air event does not affect the LTOP mitigation trains' ability to mitigate an
7 LTOP event,
7 3. A loss of offsite power event does not affect the LTOP mitigation trains' ability to mitigate an LTOP
7 event,
7 4. The LTOP System does not meet IEEE-279 design requirements,

7 <u>Because:</u>

7 1. A pressurizer nitrogen or steam bubble is maintained in the RCS at all times (except for hydrostatic
7 testing).
7 2. It can be shown that a seismic event, a loss of air event, and a loss of offsite power do not cause an
7 LTOP event.
7 3. Sufficient administrative controls are in place, per Technical Specifications, to further minimize the
7 probability of an LTOP event.

7 The above criteria are based on the premise that neither a seismic event nor loss of instrumentation air
7 event nor a loss of offsite power event randomly occur at the same time as an LTOP event at Oconee
7 Nuclear Station.

5.2.3.8 System Incident Potential

Potential accidents and their effects and consequences as a result of component or control failures are analyzed and discussed in Chapter 15, "Accident Analyses."

The pressurizer spray line contains an electric motor-operated backup valve which can be closed should the pressurizer spray valve malfunction and fail to close; this would prevent depressurization of the system to the saturation pressure of the reactor coolant. An electric motor-operated valve located between the pressurizer and the pressurizer electromatic relief valve can be closed to prevent pressurizer steam blowdown in the unlikely event the electromatic relief valve fails to reclose after being actuated. Because of the other protective features in the plant, it is unlikely that the code valves will ever lift during operation. In addition, it is extremely unlikely these valves would stick open, since there is adequate experience to indicate the reliability of code safety valves. The analysis in Chapter 15, "Accident
9 Analyses" indicate that the High Pressure Injection System will protect the core for a
n opening in the system considerably larger than one pressurizer code safety valve in the open position.

The consequences of crud filling one of the two instrument lines from the flow annulus to the flow transmitters has been evaluated.

No mechanism can be postulated which would completely block one of these lines. The Reactor Coolant System is a very clean system and is continuously filtered to assure that no significant particulate matter is circulated. The boric acid in the coolant is in concentrations about a factor of two below its solubility limit at 70°F and no precipitation would occur. The entire flow monitoring system is essentially stagnant because it is a pressure-sensitive device. There is no flow in the sensing lines to induce material into these lines. Any matter of sufficient size to block the instrument lines would have to penetrate the annulus which is of a smaller size than the instrument lines. Blockage of less than four entry ports to the annulus does not significantly impare the flow reading.

If the assumption is made that the line did become blocked, however, two possible situations would arise.
The blockage of the high-pressure line would cause the average flow to appear high as flow decreases.
Similarly, if the low pressure line is blocked, the average flow will appear higher than normal as flow is
9 decreased. In both cases, the loss of one pump will not cause trip based on flux-flow if the power is
9 constant at rated power. The results of a single pump coastdown from rated power was analyzed without
trip or power runback. The minimum Departure from Nucleate Boiling Ratio (DNBR) reached when
the flow has settled to the three-pump steady state values is 1.34.

If power runback from the Integrated Control System (ICS) is assumed, the reactivity added by control
rod insertion is sufficient to reduce the power to 89 percent by the time the flow has reached its new
value. Therefore, the hypothetical blocking of the instrument line would not cause the core thermal
9 design limit to be exceeded as a result of the loss of one pump from rated power. These analyses of crud
9 filling one of the two instrument lines from the flow annulus to the flow transmitters are not reflective of
9 the current methods described in Section 15.6, "Loss of Coolant Flow Accidents." These analyses are
9 being retained for historical purposes only.

5.2.3.9 Redundancy

Each heat transport loop of the Reactor Coolant System contains one steam generator and two reactor coolant pumps. Operation at reduced reactor power is possible with one or more pumps out of service. For added reliability, power to each pump is normally supplied by one of two electrically separated buses. Each of the two pumps per loop is fed from separate buses.

Two core flooding nozzles are located on opposite sides of the reactor vessel to ensure core reflooding water in the event of a single nozzle failure. Reflooding water is available from either the core flooding tanks or the low pressure injection pumps. The high pressure injection lines are connected to the Reactor Coolant System on each of the four reactor coolant inlet pipes.

5.2.3.10 Safety Limits and Conditions

5.2.3.10.1 Maximum Pressure

The Reactor Coolant System serves as a barrier which prevents release of radionuclides contained in the reactor coolant to the Reactor Building atmosphere. In the event of a fuel cladding failure, the Reactor Coolant System is the primary barrier against the release of fission products to the Reactor Building. The safety limit of 2,750 psig (110 percent of design pressure) has been established. This represents the maximum transient pressure allowable in the Reactor Coolant System under the ASME Code, Section III.

5.2.3.10.2 Maximum Reactor Coolant Activity

Release of activity into the reactor coolant in itself does not constitute a hazard. Activity in the reactor coolant constitutes a hazard only if the amount of activity is excessive and it is released to the environment. The plant systems are designed for operation with activity in the Reactor Coolant Systems

resulting from 1 percent defective fuel. Activity would be released to the environment if the reactor
coolant containing gaseous activity were to leak to the steam side of the steam generator. Gaseous
activity could then be released to the environment by the steam jet air ejector on the main condenser. In
6 10 CFR 20, maximum effluent concentrations (EC) for continuous exposure to gaseous activity have been
6 established. These ECs will be used as the basis for maximum release of activity to the environment
which has unrestricted access.

5.2.3.10.3 Leakage

Reactor Coolant System leakage rate is determined by comparing instrument indications of reactor coolant average temperature, pressurizer water level and letdown storage tank water level over a time interval. All of these indications are recorded. The letdown storage tank capacity is 31 gallons per inch of height, and each graduation on the level recorded represents two inches of tank height.

Reactor Coolant System leak detection is also provided by monitoring the Reactor Building sump level
7 and the letdown storage tank level. The Reactor Building sump capacity is 15 gallons per inch of height.
Since the pressurizer level controller maintains a constant pressurizer level, any Reactor Coolant System
volume change due to a leakage would manifest itself as a Reactor Building sump level change and/or a
7 corresponding letdown storage tank level change. Alarm indication in the control room for the Reactor
7 Building sump is provided at a low level of 6 inches of water and a high level of 15 inches of water. For
9 the Letdown Storage Tank, alarm (statalarm) indication is provided at a low level of 60 inches of water
9 and a high level of 90 inches of water. Considering the most adverse initial conditions of a low level in
the Reactor Building sump and a high level in the letdown storage tank, a 1 gpm leak from the Reactor
Coolant System would initiate a Reactor Building sump high level alarm indication in the control room
9 within 3 hours and a letdown storage tank low level alarm indication in the control room within 17 hours.
A three gpm leak would be detected in 1/3 the time given above for detection of a one gpm leak.
Normally, with the Reactor Building sump level and the letdown storage tank level between their high
alarm and the low alarm respectively, these detection times would be reduced.

If the leak allows primary coolant into the containment atmosphere, additional leak detection is provided
8 by the Reactor Building Process Monitoring System and the Reactor Building Area Monitoring System.
The sensitivity and time for detection of a Reactor Coolant System leak by any of the radioactivity
monitoring systems depends upon reactor coolant activity and the location of the leak. Alarm indication
for each sample point in these systems is in the control room.

9 If the leak is in a steam generator, the leak can be detected by a decrease in the level of the letdown
9 storage tank as described above, Secondary Tritium Analysis, Xenon Analysis, and also by main steam
line and condenser air ejector off gas radiation monitors. The sensitivity of the radiation monitors for leak
detection depends upon the activity of the Reactor Coolant System.

Class I fluid systems other than the Reactor Coolant System pressure boundary will be monitored for leakage by monitoring the various storage and/or surge tanks for the applicable systems. The Radiation Monitoring System for the station will aid in leak detection of systems containing radioactive fluids. In addition to the above, routine Operator and/or Health Physics radiation surveillance will detect leakage in both radioactive and non-radioactive systems.

Natural circulation can be maintained in the Reactor Coolant System for decay heat removal following a complete loss of station power even if the system has been operating with an equipment leak. The natural circulation path will be maintained solid with water until the pressurizer has emptied, which is 6,000 gallons of coolant. A 30 gpm leakage rate in conjunction with a complete loss of station power and subsequent cooldown of the Reactor Coolant System by the Turbine Bypass System (set at 1,040 psia) and steam driven emergency feedwater pump would require a minimum of 60 minutes to empty the pressurizer from the combined effect of system leakage and contraction. Sixty minutes is ample time to restore electrical power to the plant and makeup flow to the Reactor Coolant System.

5.2.3.10.4 System Minimum Operational Components

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources, i.e., pump energy, pressurizer heaters, and reactor decay heat. Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. One steam generator is required to be operable prior to criticality as the steam generator is the means for normal decay heat removal at temperatures above 250°F.

A reactor coolant pump or low pressure injection pump is required to be in operation prior to reducing boron concentration by dilution with make-up water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor.

5.2.3.10.5 Leak Detection

The entire Reactor Coolant System is located within the secondary shielding and is inaccessible during reactor operation. Any leakage drains to the Reactor Building normal sump. Any coolant leakage to the atmosphere will be in the form of fluid and vapor. The fluid will drain to the sump and the vapor will be condensed in the Reactor Building coolers and also reach the sump via a drain line from the cooler.

7 For the reactor coolant pump, any leakage past the mechanical seals is routed as follows: For the Unit 1
7 RCPs, the leakage past the innermost seal is routed to the Letdown Storage Tank; leakage past the middle
7 and outermost seals is routed either to the quench tank or reactor building sump. For the Unit 2 and 3
7 RCPs, the leakage past the middle seal is routed to the Letdown Storage Tank; leakage past the outermost
7 seal is routed to the quench tank or reactor building sump.

Locating the actual point of Reactor Coolant System leakage can most readily be accomplished when the reactor is shutdown, thereby allowing personnel access inside the secondary shielding. Location of leaks can then be accomplished by visual observation of escaping steam or water, or of the presence of boric acid crystals which would be deposited near the leak by evaporation of the leaking coolant.

Leakage of reactor coolant into the Reactor Building during reactor operation will be detected by sump/tank levels radioactivity, or both.

All leakage, both reactor coolant and cooling water is collected in the Reactor Building Sump. The sump water level is indicated and annunciated at high level in the control room. Changes in sump water level are an indication of total leakage. Pursuant to the NUREG 0737, Item II.F.1.5 safety grade redundant level transmitter to the normal and emergency containment sumps have been installed. Both sump levels are indicated and recorded in the control room. Measurement of the letdown storage tank coolant level provides a direct indication of reactor coolant leakage. Since the pressurizer level is maintained constant by the pressurizer level controller, any coolant leakage is replaced by coolant from the letdown storage tank resulting in a decrease in tank level. Both the pressurizer and letdown storage tank coolant levels are recorded in the control room. A comparison of these two recordings over a time period yields the total reactor coolant leakage rate.

Changes in the reactor coolant leakage rate in the Reactor Building may cause changes in the control room indication of the Reactor Building atmosphere particulate and gas radioactivities.

5.2.3.11 Quality Assurance

Assurance that the Reactor Coolant System will meet its design bases insofar as the integrity of the pressure boundary is concerned, is obtained by analysis, inspection, and testing.

5.2.3.11.1 Stress Analyses

Detailed stress analyses of the individual Reactor Coolant System components including the vessel, piping, pumps, steam generators, and pressurizer have been performed for the Design Bases.

Dynamic analyses have been performed on the complete system treating each steam generator and associated coolant piping as an independent system to include the effect of the design bases earthquake or the maximum hypothetical earthquake in the piping stresses and nozzle stresses.

Independent thermal and dynamic analyses have been performed to insure that piping connecting to the Reactor Coolant System is of the proper schedule and that it does not impose forces on the nozzles greater than allowable. Small nozzles are conservatively designed and utilize ASA schedule 160. The reactor coolant pump casing has been completely analyzed including a dynamic analysis separately from the loop to insure that the stresses throughout the casing are below the allowable for all design conditions.

Stress analysis reports required by codes for the several components have been prepared by the manufacturer and reviewed for adequacy by a separate organization.

5.2.3.11.2 Shop Inspection

Inspection and non-destructive testing of materials prior to and during manufacturing in accordance with applicable codes and additional requirements imposed by the manufacturer have been carried out for all of the Reactor Coolant System components and piping. The extent of these inspections and testing is listed in Table 5-10 for each of the components in the system. Shop testing culminates with a hydrostatic test of each component followed by magnetic particle inspection of the component external surface. Piping will be hydrostatically tested in the field and will undergo a final field inspection.

Preoperational mapping of the reactor vessel by ultrasonic examination was accomplished to establish acceptability of the vessel for service. To meet the requirements of IS-232 of Section XI of the ASME Code, the acceptance standards contained in N625.4 of the 1965 edition of Section III of the ASME Code with Addenda through Summer 1967 were used.

7 Components were cleaned, packaged to prevent contamination, and shipped over a pre-selected route to
the site. For materials purchased or manufactured outside of B&W, the results of the material inspection
and testing program have been observed or audited by B&W, and audited by the applicant. In addition
7 there was an independent audit by B&W's Nuclear Power Generation Department Quality Assurance
Section.

5.2.3.11.3 Field Inspection

Field welding of reactor coolant piping and piping connecting to nozzles is performed using procedures which will result in weld quality equal to that obtained in shop welding. Non-destructive testing of the welds is identical to that performed on similar welds in the shop and is shown in Table 5-10. Accessible shop and field welds and weld repairs in the reactor coolant piping are inspected by magnetic particle or liquid penetrant tests following the system hydrostatic test.

5.2.3.11.4 Testing

The Reactor Coolant System including the reactor coolant pump internals, reactor closure head, control rod drives, and associated piping out to the first stop valve undergoes a hydrostatic test following completion of assembly. The hydrostatic test is conducted at a temperature 60°F greater than the highest nil-ductility temperature. During the hydrostatic test, a careful examination is made of all pressure boundary surfaces including gasketed joints.

5.2.3.12 Tests and Inspections

This section discusses tests and inspections performed during and after the assembly of the individual components into a completed Reactor Coolant System. These tests and inspections are performed to demonstrate the functional capabilities of the components after assembly into a completed system, to inspect the quality of the system closure weldments, and to monitor system integrity during service.

5.2.3.12.1 Construction Inspection

The coolant piping for each loop is shipped to the field in six subassemblies. The loops are then assembled in the field. In order to accommodate the small fabricating and field installation tolerances, a number of the subassemblies are fabricated with excess length. Thus, the final fitting of the coolant piping is accomplished in the field. The ends with excess length are field machined. All carbon steel-to-carbon steel field welds are back-clad with stainless steel following removal of the backing rings. Consumable inserts are used in stainless-to-stainless welds, such as surge line and some coolant pump welds. All welding is inspected in accordance with requirements of the applicable codes or better.

Welding of the auxiliary piping to Reactor Coolant System nozzles is done to the same standards as the main coolant piping. Consumable inserts are used in all cases.

Cleaning of reactor coolant piping and equipment is accomplished both before and after erection of various equipment. Piping and equipment nozzles will require cleaning in the area of the connecting weldments. Most of the piping and equipment are large enough for personnel entry and are cleaned by locally applying solvents and demineralized water and by wire brush to remove trapped foreign particles. Where surfaces and equipment cannot be reached by personnel entry and have been cleaned in vendor shops to the required cleanliness for operation and appropriately protected to maintain cleanliness during handling, shipping, storage, and installation, further cleaning will not be performed. Appropriate checks to verify maintenance of required cleanliness will be performed prior to operation.

5.2.3.12.2 Installation Testing

The Reactor Coolant System will be hydrostatically tested in accordance with USAS B31.7, Nuclear Power Piping Code. The test pressure will affect all parts of the Reactor Coolant System up to and including means of isolation from auxiliary systems, such as valves and blank flanges. The hydrostatic test will be performed at temperature above Design Transition Temperature.

The Reactor Coolant System relief valves will be inspected and shop-tested in accordance with Section III of the ASME code for Nuclear Vessels. The relief pressure setting will be made during the shop test.

5.2.3.12.3 Functional Testing

Prior to initial fuel loading, the functional capabilities of the Reactor Coolant System components will be demonstrated at operating pressures and temperatures. Measurement of pressures, flows, and temperatures will be recorded for various system conditions. Operation of reactor coolant pumps, pressurizer heaters, Pressurizer Spray System, control rod drive mechanism, and other Reactor Coolant

System equipment will be demonstrated. For descriptions of the various functional tests performed, refer to Chapter 14, "Initial Tests and Operation."

5.2.3.12.4 Inservice Inspection

Inservice examination of ASME Code Class 1, 2 and 3 components are performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR
0 50, Section 50.55a(g)(4), to the extent practical within the limitations of design, geometry and materials of
8 construction of the components, except where specific written relief has been granted by the Commission.

Vessels, heat exchangers, pumps, valves, and piping, are classified in accordance with 1OCFR50.55a and NRC Regulatory Guide 1.26. For each ASME code class, systems have been identified which will be examined. Appropriate Duke drawings and documents provide the exact boundaries for each system to be examined.

6 The examination categories to be used are those listed in Tables IWB, IWC, IWD, and IWF-2500-1 of
6 ASME Section XI. Specific examinations will be identified by an Item Number specified in the Oconee
6 Inservice Inspection Plan. Appropriate tests are contained in Subsections IWP and IWV.

6 The examination techniques to be used for inservice inspection include radiographic, ultrasonic, magnetic
6 particle, liquid penetrant, eddy current and visual examination methods.

7 Repair procedures are prepared as necessary by Duke Power Company Nuclear Generation Department.
1 These procedures are reviewed for compliance with Section XI. Reexamination to Section XI is included in the repair process.

5.2.3.13 Reactor Vessel Material Surveillance Program

The original Oconee design included three reactor vessel surveillance specimen holder tubes (SSHT) located near the reactor inside vessel wall. Each of these SSHT housed two capsules containing reactor vessel surveillance specimens. When failures of the SSHT occurred at other Babcock & Wilcox (B&W) designed plants, the three Oconee units were shut down in succession, starting in March 1976 to inspect the SSHT. The inspection revealed that all of the SSHTs had suffered some damage. To prevent further damage all surveillance capsules and all parts of the SSHT that had failed or were deemed likely to fail during the remainder of that operating cycle were removed from the vessels.

Since the discovery of the damage to the SSHT, B&W has undertaken the design, manufacture and testing of an improved SSHT. SSHT of this improved design were installed in Davis-Besse 1, Crystal River 3
1 and Three Mile Island 2. (Three Mile Island 2 no longer operating but capsules were salvaged for
1 irradiations at other host plants.) All of these plants have the same basic B&W 177 fuel assembly reactor design as Oconee 1, 2, and 3. The acceptability of the redesigned SSHT has been demonstrated by a test program reviewed and approved by the NRC staff and conducted in conjunction with the hot functional test performed at Davis-Besse 1.

Installation of the redesigned SSHT in the Davis Besse l, Crystal River 3 and Three Mile Island 2 reactor vessels did not present any unusual radiological difficulties because installation was prior to neutron
1 activation of the reactor internals. Studies of methods of installing the redesigned SSHT in the irradiated B&W reactors indicate that substantial installation difficulties will be experienced primarily because precision machining, alignment and inspection must be performed remotely and under water. Although such problems do not in themselves justify relief from a requirement to reinstall the SSHT in Oconee 1, 2, and 3, they would be likely to cause significant radiation to personnel. Based on its experience in removing the SSHT at Three Mile Island 1 and Rancho Seco 1, B&W estimated that installing SSHT in irradiated reactors could result in personnel exposures totaling about 100 man-rem per reactor. In the

interest of maintaining the radiation exposure of plant personnel as low as reasonably achievable, the licensee, in cooperation with B&W and the owners of other B&W 177 fuel assembly plants, has proposed an alternative program that does not require reinstalling the SSHT in Oconee 1, 2, and 3 and the other irradiated B&W plants.

1 The capsules removed from the Oconee vessels which had damaged SSHT were placed in a host reactor, Crystal River 3, as part of the integrated surveillance program discussed herein. These capsules contain samples of plate or forging material and heat-affected zone material from the vessel beltline as well as weld metal. The weld metal is expected to be controlling because it is more radiation sensitive. However, capsules containing other than weld metal will be irradiated also, since the purpose of the surveillance program is to obtain data on materials which would prove to be important later on.

1 This program includes provisions to provide additional information, if required under 10 CFR 50,
1 Appendix G, Paragraph V.C., in addition to the normal requirements of Appendix H.

The plan involves integrating the interrupted surveillance program at Oconee and other plants with the
1 programs for new plants in a manner generally similar to that covered in 10 CFR 50, Appendix H,
Paragraph II.C.4, except that the plants are at different sites. There are three distinct features of this plan.

1. The original surveillance materials from one or more reactors that have been in service will now be irradiated in a new host reactor, that can be fitted with the newly-designed capsule holders on the thermal shield in less time and without significant radiation exposure of the workmen, and
2. There will be more weld metal specimens and some larger fracture mechanics (compact tension or CT) specimens placed in the capsules, and
3. A data-sharing feature in which all available irradiation data for the beltline welds of a given reactor some of which will come from other surveillance programs, will be considered in predicting its adjusted reference temperature and in making any fracture analyses for that reactor. Typically, several of the welds in any one vessel were made with the same weld wire and flux as those used on some other reactors. The data sharing feature is required because the welds in these reactors have high radiation sensitivity due to high copper content, large and random variation of copper from point to point in the weld, and low initial upper shelf energy.

The specific program for Oconee 1, 2, and 3 involves installing the Oconee surveillance capsules in extra locations provided in the Crystal River 3 vessel. This plan will accomplish the original purpose of obtaining information on the effect of radiation on material that is representative of the material in the Oconee reactor vessels on a schedule that provides an appropriate lead time over the vessel irradiation rate. The overall integrated program also will provide information relevant to Oconee 1, 2, and 3 from surveillance programs in Crystal River 3, and Davis Besse 1 on material considered to be essentially identical to the actual welds in the Oconee vessels. It is also important to note that still more information relevant to the Oconee vessel materials will be obtained from the NRC funded High Strength Steel Test (HSST) irradiation programs underway. Details are provided below.

5.2.3.13.1 Oconee 1

1 The limiting weld materials for the Oconee 1 vessel are Procedure Qualification (P.Q.) numbers SA-1426,
8 SA-1430, SA-1229, and SA-1585, except for pressurized thermal shock (PTS) for which the limiting
8 material is SA-1073 (Reference 32)* (BAW-2192, BAW-2178, References 30 and 31). The first two are
8 longitudinal welds in the lower shell course, the second two are beltline circumferential welds, and the last

* Weld materials are specifically identified by the ASME Code by the procedure Qualification Test number. A procedure qualification test is required on each combination of heat of weld wire and batch of flux.

8 material is a longitudinal weld in the intermediate shell. The end of life (EOL) fluences for these welds
8 are estimated to be 7.67 x 10^{18}, 7.67 x 10^{18}, 8.44 x 10^{18}, 8.99 x 10^{18}, and 6.55 x 10^{18} (Reference 27) nvt, (E
8 > 1 MeV) at the inner surface, respectively. All of the welds have compositions that are expected to make
1 them relatively sensitive to radiation damage.

The original surveillance material, WF-112, was made using the same heat of filler wire but a different
1 batch of flux as WF-154, one of the radiation sensitive welds in Oconee 2. Metallurgical considerations
suggests that the radiation behavior is affected more by the wire than the flux, thus WF-112 is expected to
respond to radiation much like WF-154. This data will be a useful part of the data base for B&W vessels.

1 Reference 14 documents where samples of the pertinent weld materials will be irradiated in the proposed
integrated program, what kinds of specimens will be used, and when information will be available. The
irradiation schedule and withdrawal dates will be modified to optimize the information obtained as
1 indicated to be appropriate as test results are obtained and evaluated. Reference 14 is updated periodically
to reflect the most recent capsule reports.

5.2.3.13.2 Oconee 2

1 The limiting weld material for the Oconee 2 vessel is P.Q. number WF-25 which is used in the center
8 circumferential weld. (BAW-2192 and BAW-2178, References 30 and 31). The end of life (EOL) fluence
8 for this weld is estimated to be 8.70 x 10^{18} nvt (E > 1 MeV) (Reference 28) at the inner surface; this weld
1 has a composition that is expected to make it relatively sensitive to radiation damage.

The original surveillance material, WF-209-1, while not identical to any of the beltline welds in B&W
1 reactors, is of the same weld wire heat as WF-70 (but different flux lot) and is predicted to be radiation
1 sensitive, based on its copper and nickel contents. Data from WF-209-1 will be a useful addition to the
data base for these reactors.

1 Reference 14 documents where samples of the pertinent weld materials will be irradiated in the proposed
integrated program, what kinds of specimens will be used, and when information will be available.
1 Reference 14 is updated periodically to reflect the most recent capsule reports.

5.2.3.13.3 Oconee 3

8 The limiting weld material for the Oconee 3 vessel is P.Q. Number WF-67 (BAW-2192 and BAW-2178,
8 References 30 and 31). WF-67 is used for the center circumferential weld (inner 75%). The end of life
8 (EOL) fluence for WF-67 is estimated to be 8.59 x 10^{18} nvt, (E > 1 MeV) (Reference 29) at the inner
8 surface; this weld has a composition that is expected to make it relatively sensitive to radiation damage.

1

1 The original surveillance material, WF 209-1, is the same as that used in Oconee 2. This discussion of
1 WF-209-1 in 5.2.3.13.2, "Oconee 2" applies here.

1 Reference 14 documents where samples of the pertinent weld materials will be irradiated in the proposed
integrated program, what kinds of specimens will be used, and when information will be available.
1 Reference 14 is updated periodically to reflect the most recent capsule reports.

1 5.2.3.13.4 Integrated Surveillance Program

8 BAW-1543A, Rev. 4, Master Integrated Reactor Vessel Material Surveillance Program, June 1996,
specifies the Oconee specimens capsules that are to be irradiated in Crystal River 3. These capsules

include the weld material and other materials such as plate or forging material samples and weld heat
1 affected zone material samples from the Oconee vessels.

For those welds where no surveillance specimens exist, guidance for predictions will be based on the
1 known chemical composition of those welds. Predictions are based on statistical analysis of the family of
1 "Linde 80" weld metals.

8 BAW-1543, Rev. 4, presents a "Master Integrated Reactor Vessel Surveillance Program" that provides for
1 additional surveillance capsules which contain tension test, Charpy V-notch, and larger-sized compact
1 fracture specimens of 12 different "Linde 80" weld metals (different wire/flux combinations). These
1 specimens will provide direct data for those materials represented and will provide a statistical base for
1 those other materials for which archive material was not available. In particular, for Oconee-1, SA-1426
1 and SA-1430 are both from the same weld wire heat for which no archive material was available. For
1 Oconee-2, WF-25 is irradiated in 7 supplementary capsules. For Oconee-3, WF-67 is irradiated in 6
1 supplementary capsules, and WF-200 is from the same weld wire heat as WF-182-1. WF-182-1 is
1 irradiated in one supplementary capsule and is the weld material in the Davis-Besse RVSP.

1 All Oconee RVSP capsules, except for standby capsules, have been tested, essentially completing the
1 requirement for reactor vessel surveillance irradiations. However, the supplementary capsules will provide
1 additional fracture toughness data.

1

1 Research programs being funded by the NRC have provided information on the effect of radiation on
these specific weld materials and on several additional B&W weld materials expected to respond to
1 radiation in a similar manner. These programs, HSST-2 and HSST-3, consist of many tension test, C_v
and CT specimens irradiated in a test reactor. Although information on shift in RT_{NDT} will be obtained,
1 the main emphasis of the HSST programs was to develop methods that can be used to better evaluate
1 upper-shelf toughness as compared to using the rather small specimens used in the power reactor
programs.

The information to be developed from this program that is directly and indirectly relevant to the Oconee
reactor vessels will be sufficient to provide assurance of safety margins against vessel failure that comply
1 with 10 CFR 50, Appendix G.

1

There are uncertainties involved in applying radiation effects information obtained in other reactors to the Oconee vessels. The major uncertainties involved include:

1. Accuracy of neutron fluence calculations,
2. Magnitude and effect of variation in neutron spectra between reactors,
3. Magnitude and effect of variations in irradiation temperature between reactors,
4. Magnitude and effect of variations in rate of irradiation on material properties.

1 The effects of these variables have been studied for many years and are discussed below.

1 1. Neutron flux calculations for the reactor vessel wall and irradiation capsule locations have been developed over many years. The dosimetry used in irradiation capsules has furnished information that was used to check out and refine the calculational methods. It is generally believed that the fast neutron flux and fluence in these locations can be calculated to an accuracy of ±20 percent,

particularly if some dosimetry checks are available. Dosimeters from the original Oconee surveillance program were removed and tested for verification of vessel fluence calculations.

It should be emphasized that the effect of neutron radiation on reactor vessel steel varies as the square root of the fluence, so uncertainties of 20 to 50 percent influence are not highly significant.

The design of the Oconee vessels, internals and cores is almost identical to that of the other reactors that will be used to obtain radiation effects information.

These considerations are the basis for the conclusion that uncertainties in the calculation of neutron fluence will be small, and the effect of such uncertainties on the assessment of the radiation effects on the vessel material will also be small.

2. Although differences in neutron energy spectra can cause uncertainties in the effects of radiation on material when evaluated without considering spectrum effects, only very large differences in spectra are significant. The variations from one B&W 177 fuel assembly reactor to another are relatively minor, because they have almost identical geometry.

   The possible differences in neutron spectra that could occur between the B&W power reactors to be involved in the integrated program has been considered. Such effects can be dealt with, if necessary, through the use of neutron damage functions that are being developed for that purpose. However, the worst expected differences are judged inconsequential based on present knowledge of irradiation effects.

3. The effect of the temperature of irradiation has also been the subject of considerable research. It is well known that radiation damage is less severe at 600°F than at 500°F (the temperature range of concern). The differences in effect on the steel appear to be noticeable and should be taken into account if the irradiation temperature difference is over about 25°F. Enough information is known to permit conservative evaluations of the effect of temperature differences of at least 50°F, and probably even 100°F or more. The differences in the temperature of the surveillance capsules and vessel walls between the B&W power reactors involved in the proposed integrated program are expected to be less than 25°F, and can be conservatively evaluated.

4. The effect of irradiation has also been evaluated by research programs at NRL and other laboratories. The general consensus of experts on this subject is that there will be no major differences in material property changes by irradiation rates varying over 2 to 3 orders of magnitude. However, the differences in the rates of irradiation of specimens in the integrated program and the limiting material in the walls of the affected vessels will be less than one order of magnitude, therefore, it is concluded that there will be no significant uncertainties in this program associated with differences in rate of irradiation.

The integrated, augmented reactor vessel material irradiation program for Oconee 1, 2, and 3, as an alternative to the original program that was interrupted by failure of the associated hardware, will provide the information required to comply with 10 CFR 50, Appendix G, and that the uncertainties involved in using data obtained from surveillance specimens irradiated in various other B&W power reactors to establish Oconee 1, 2, and 3 vessel operating limitations are small and can be accounted for by imposition of appropriate margins.

Additionally, the integrated, augmented program should provide more useful information than could have been extracted from the original surveillance program. The proposed program will also give results of the kind required to meet 10 CFR 50, Appendix G, Paragraph V.C.

An extension of the exemption for Oconee Units 1, 2 and 3 from the requirements for an in-vessel material surveillance program as set forth in 10 CFR 50, Appendix H, was requested by the Duke Power Company in January 1982 (Reference 10). In its submittal to the NRC, Duke Power Company stated that at present there were no plans to modify the Surveillance Specimen Holder Tubes (SSHT's) or the

Core Support Assembly on any Operating B&W plant which would change the geometrical similarity of
the reactors or preclude the continued irradiation of the surveillance capsules in the host plants. Thus,
adequate surveillance information will continue to be obtained for the Oconee units. An evaluation of the
Surveillance Capsules removed from operating B&W plants and an evaluation of the reactor vessel fluence
7 were included in the Duke Submittal to demonstrate the adequacy of the Surveillance Program. Duke
Power Company submittal concluded that:

1. Based on the Surveillance capsule data obtained on all the B&W-177FA plants to date, it has been
demonstrated that the prediction techniques used in establishing the vessel operation limits (i.e., Reg.
1 Guide 1.99, Rev. 2) are conservative.

2. A high degree of accuracy has been demonstrated by B&W in estimation of the reactor vessel fluence using the power histories of the reactors and the dosimetry measurements from the host plants with SSHT's.

3. The Specimen Capsules being irradiated at Crystal River-3 have received neutron fluence greater than the fluence received by the Oconee Reactor Vessels by 7 to 10 EFPYs. The Specimen Capsules are expected to continue to lead the respective reactor vessels accumulated peak fluence for the life of the plant.

NRC granted an extension to the exemption for the Oconee Nuclear Station, Units 1, 2 and 3 from the
1 requirement for an in-vessel Material Surveillance program as set forth in 10 CFR 50, Appendix H, for a
period of five years in June 1982 (Reference 11). The Commission stated in its safety evaluation that the
information derived from the surveillance specimens in the host vessel, relevant to Oconee Nuclear Station
Units 1, 2 and 3 reactor vessels would be sufficient to provide assurance of safety margins and comply with
8 10 CFR 50, Appendix G. In addition, the NRC concurred with the Duke position that the dosimetry
results have shown that the fluences can be estimated from the power histories with reasonable accuracy
1 and accepted the methodology contained in BAW 1485, June 1978. In June, 1991, the NRC accepted
1 BAW-1543, Rev. 3, and found the program capable of monitoring the effect of neutron irradiation and
1 the thermal environment on the fracture toughness of ferritic reactor vessel beltline materials in the plants
1 that are participating in the material surveillance program. This includes Oconee-1, Oconee-2, and
1 Oconee-3.

5.2.4 REFERENCES

1. BAW-10051, Design of Reactor Internals and Incore Instrument Nozzles for Flow Induced Vibrations.

7 2. BAW-10008, Part 1, Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Accident.

7 3. Deleted per 1997 Update.

4. BAW-10046P, Methods of Compliance with Fracture Toughness and Operational Requirements of 10 CFR50, Appendix G.

5. BAW-10100A, Reactor Vessel Material Surveillance Program.

1 6. BAW-1803, Rev. 1, Correlations for Predicting the Effects of Neutron Radiation on Linde 80
1 Submerged-Arc Welds.

7. BAW-10018, Analysis of the Structural Integrity of a Reactor Vessel Subjected to Thermal Shock.

8. BAW-10027, Once-Through Steam Generator Research and Development (Nonproprietary version of BAW-10002, and BAW-10002, Sup. 1).

9. BAW-1402, Steam Generator Weld Records

10. Letter from W. O. Parker, Jr. to H. R. Denton (NRC) dated January 14,1982 Subject: Exemption from 10CFR 50 Appendix H requirements for 5 years.

11. Letter from D. G. Eisenhut (NRC) to W. 0. Parker, Jr. dated June 16, 1982.

12. Safety Evaluation by the Office of Nuclear Reactor Regulation Concerning NUREG 0737 Item II.K.2.13, United States Nuclear Regulatory Commission, June 6, 1984.

13. Letter from H. B. Tucker (Duke) to NRC, Oconee Nuclear Station P/T Limit and LTOP Proposed Technical Specifications, November 15, 1989.

8 14. BAW-1543, Rev. 4, Supplement 2, Master Integrated Reactor Vessel Surveillance Program, June
8 1996.

1 15. Letter from J. N. Hannon, Office of Nuclear Reactor Regulation, NRC, to J. H. Taylor, B&W
1 Owners Group, June 11, 1991.

2 16. BAW-2108, Rev. 1 Fluence Tracking System.

1 17. BAW-2050, Analysis of Capsule OC1-C, Duke Power Company, Oconee Nuclear Station Unit-1,
1 Reactor Vessel Material Surveillance Program.

1 18. BAW-2051, Analysis of Capsule OCII-E, Duke Power Company, Oconee Nuclear Station Unit-2,
1 Reactor Vessel Material Surveillance Program.

1 19. BAW-2128, Rev. 1, Analysis of Capsule OCIII-D, Duke Power Company, Oconee Nuclear Station
1 Unit-3, Reactor Vessel Material Surveillance Program.

1 20. BAW-1895, Pressurized Thermal Shock Evaluations in Accordance with 10 CFR 50.61 for Babcock
1 & Wilcox Owners Group Reactor Pressure Vessels.

8 21. Deleted per 1998 Update.

1 22. BWNS Document 18-1202139-00, "Functional Specification - Pressurizer Surge Line for B&W
1 Lowered-Loop 177 FA Plant".

23. BAW-2143P, Evaluation of Reactor Vessel Material Reference Temperatures and Charpy Upper-Shelf
8 Energies, August 1992.

7 24. W. O. Parker, Jr., Duke Power Company letter to the USNRC, A. Schwencer, Jr., dated October 14,
7 1976, Response to NRC information request on LTOP Systems.

7 25. W. O. Parker, Jr., Duke Power Company letter to the USNRC, B. C. Rusche, dated April 1, 1977,
7 Response to the RAI #1 on LTOP System.

7 26. J. F. Stolz, USNRC letter to Duke Power Company, H. B. Tucker, dated August 8,1983, SER for
7 Oconee LTOP Systems.

8 27. FTI document 86-1266231-01, ONS-1 PT Fluence Analysis Report - Cycles 11-16, February 1998.

8 28. FTI document 86-1258198-01, ONS-2 PT Fluence Analysis Report - Cycles 9-14, December 1997.

8 29. FTI document 86-1266235-01, ONS-3 PT Fluence Analysis Report - Cycles 12-15, February 1998.

8 30. BAW-2192PA, Low Upper Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of
8 BWOG RVWG for Level A & B Service Loads, April 1994.

8 31. BAW-2178PA, Low Upper Shelf Toughness Fracture Mechanics Analysis of Reactor Vessels of
8 BWOG RVWG for Level C & D Service Loads, April 1994.

8 32. FTI document 32-5000884-00, Pressurized Thermal Shock Reference Temperatures for ONS-1,
8 February 1998.

8 33. BAW-1820, B&WOG 177 Fuel Assembly Reactor Vessel and Surveillance Program Materials
8 Information, December 1984.

8 34. BAW-2313P, Reactor Vessel Materials and Surveillance Data Information, November 1997.

8 35. FTI document 32-5000705-01, Pressurized Thermal Shock Reference Temperatures for ONS-2,
8 February 1998.

8 36. FTI document 32-5000885-00, Pressurized Thermal Shock Reference Temperatures for ONS-3,
8 February 1998.

8 37. W. R. McCollum, Duke Power Company letter to the USNRC, Document Control Desk, dated
8 October 15, 1998, Proposed Revision to Technical Specifications Pressure-Temperature Operating
8 Curves

9 38. *Application for Renewed Operating Licenses for Oconee Nuclear Station, Units 1, 2, and 3*, submitted
9 by M. S. Tuckman (Duke) letter dated July 6, 1998 to Document Control Desk (NRC), Docket
9 Nos. 50-269, - 270, and -287.

9 39. NUREG-1723, *Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station,*
9 *Units 1, 2, and 3*, Docket Nos. 50-269, 50-270, and 50-287.

9 40. W. R. McCollum, Jr. (Duke) letter dated February 17, 1999, Response to Request For Additional
9 Information, Attachment 1, Response to RAI 5.4.1-1, Oconee Nuclear Station, Units 1, 2, and 3,
9 Docket Nos. 50-269, -270, and -287.

9 41. M. S. Tuckman (Duke) letter dated December 17, 1999, Response to NRC letter dated November 17,
9 1999, Attachment 1, page 26, Oconee Nuclear Station, Units 1, 2, and 3, Docket Nos. 50-269, -270,
9 and -287.

9 42. BAW-2202, *Fracture Toughness Characterization of WF-70 Weld Metal*, B&W Nuclear Service
9 Company, Lynchburg, VA, September 1993.

9 43. BAW-10046A, Revision 2, *Methods of Compliance With Fracture Toughness and Operational*
9 *Requirements of 10 CFR 50, Appendix G*, B&W Nuclear Power Division/Alliance Research Center,
9 June 1986.

9 44. BAW-1820, *177-Fuel Assembly Reactor Vessel and Surveillance Program Materials Information*, B&W
9 Nuclear Power Division, Lynchburg, VA, December 1984.

9 45. BAW-2121P, *Chemical Composition of B&W Fabricated Reactor Vessel Beltline Welds*, B&W Nuclear
9 Technologies, Inc., Lynchburg, VA, April 1991.

9 46. BAW-2166, *Response to Generic Letter 92-01*, B&W Nuclear Service Company, Lynchburg, VA, June
9 1992.

9 47. BAW-2222, *Response to Closure Letters to Generic Letter 92-01, Revision 1*, B&W Nuclear
9 Technologies, Lynchburg, VA, June 1994.

9 48. BAW-2251, *Demonstration of the Management of Aging Effects for the Reactor Vessel*, The B&W
9 Owners Group Generic License Renewal Program, June 1996.

9 49. BAW-2241P, *Fluence and Uncertainty Methodologies*, April 1997.

9 50. BAW-2325, *Response to Request for Additional Information (RAI) Regarding Reactor Pressure Vessel
9 Integrity*, Revision 1, January 1999.

9 51. W. R. McCollum, Jr. (Duke) letter dated February 17, 1999, Response to Request For Additional
9 Information, Attachment 1, Response to RAI 5.4.2-1, pages 78, 79, and 80; Oconee Nuclear Station,
9 Units 1, 2, and 3, Docket Nos. 50-269, -270, and -287.

9 52. BAW-2275, T. Wiger and D. Killian, *Low Upper-Shelf Toughness Fracture Mechanics Analysis of
9 B&W Designed Reactor Vessels for 48 EFPY*, Framatome Technologies, Inc. Lynchburg, VA.

9 53. BAW-2274, A. Nana, *Fracture Mechanics Analysis of Postulated Underclad Cracks in B&W Designed
9 Reactor Vessels for 48 EFPY*, Framatome Technologies, Inc. Lynchburg, VA.

9 54. U.S. Atomic Energy commission, *Control of Stainless Steel Weld Cladding of Low-Alloy Steel
9 Components*, Regulatory Guide 1.43, May 1973.

5.3 REACTOR VESSEL

5.3.1 DESCRIPTION

The reactor vessel consists of a cylindrical shell, a spherically dished bottom head, and a ring flange to which a removable reactor closure head is bolted. The reactor vessel is supported by a cylindrical support skirt.

The reactor vessel closure head is a spherically dished head welded to a ring flange which mates with and is bolted to the vessel with large-diameter studs. All internal surfaces of the vessel and closure head are clad with stainless steel or nickel-chromium-iron (Ni-Cr-Fe) weld deposit.

The reactor vessel outlines are shown in Figure 5-14 (Oconee 1), Figure 5-15 (Oconee 2), and Figure 5-16 (Oconee 3). The general arrangement of the reactor vessel with internals is shown in Figure 4-26 and Figure 4-27. Reactor vessel design data is listed in Table 5-11.

All major reactor vessel nozzles are installed with full penetration welds. All control rod drive and incore instrument nozzles are installed with partial penetration welds. The gasket leakage tap is installed in each reactor vessel flange with a partial penetration weld. In addition, the Oconee 1 closure head contains eight nozzles initially for instrumentation but now blanked off.

The reactor closure head flange and the reactor vessel flange are joined by sixty 6-1/2 in. diameter studs. Two metallic O-rings seal the reactor vessel when the reactor closure head is bolted in place. Test taps are provided in the annulus between the two 0-rings to afford a means to leak test the vessel closure seal after refueling. To insure uniform loading of the closure seal, the studs are hydraulically tensioned.

The reactor vessels and closure heads are constructed of a combination of formed plates and forgings. The ring forgings in the reactor vessel shells, other than closure flanges, for Oconee 1, 2, and 3 are identified in Figure 5-14, Figure 5-15, and Figure 5-16.

The core support assembly is supported by a ledge on the inside of the vessel flanges, and its location is maintained on this elevation by the closure head flange. The core support assembly directs coolant flow through the reactor vessel and core, supports the core, and guides the control rods in the withdrawn position.

The coolant enters the reactor through the inlet nozzles, passes down through the annulus between the thermal shield and vessel inside wall, reverses at the bottom head, passes up through the core, turns around through the plenum assembly, and leaves the reactor vessel through the outlet nozzles.

The vessel has two outlet nozzles through which the reactor coolant is transported to the steam generators and four inlet nozzles through which reactor coolant reenters the reactor vessel. Two smaller nozzles located between the reactor coolant inlet nozzles serve as inlets for decay heat cooling and emergency cooling water injection (core flooding and low-pressure injection engineered safety features functions). The reactor coolant and the control rod drive penetrations are located above the top of the core to maintain a flooded core in the event of a rupture in a reactor coolant pipe or a control rod drive pressure housing. The reactor vessel is vented through the control rod drives.

The bottom head of the vessel is penetrated by instrumentation nozzles. The closure head is penetrated by flanged nozzles which provide for attaching the control rod drive mechanisms and for control rod extension shaft movement.

Guide lugs welded inside the reactor vessel's lower head limit a vertical drop of the reactor internals and core to 1/2 inch or less and prevent rotation about the vertical axis in the unlikely event of a major internals component failure.

The reactor vessel shell material is protected from fast neutron flux and gamma heating effects by a series of water annuli and stainless steel barriers located between the core and the vessel's wall.

5.3.2 VESSEL MATERIALS

5.3.2.1 Materials Specifications

The materials used in the reactor vessel are discussed in Section 5.2.3.2, "Material Selection" and listed in Table 5-5. The original reactor vessel material properties, as used in licensing Oconee, are presented in Table 5-12 and Table 5-13. Additional material physical properties are presented in Table 5-14. These properties have been updated as new data became available as explained in Section 5.2.3.3, "Reactor Vessel."

5.3.2.2 Special Processes for Manufacturing and Fabrication

The reactor vessel and appurtenances are constructed in accordance with the ASME Code, Section III
edition and addenda listed in Table 5-4. Processes and materials, including product form used in
fabrication of the reactor vessel, are discussed in Section 5.2.3, "System Design Evaluation," and were
selected to ensure reactor vessel integrity, and to meet regulatory requirements and recommendations.
7 Special or unusual processes not meeting the above requirements were not used in construction of the
reactor vessel.

5.3.2.3 Special Methods for Nondestructive Examination

The required nondestructive examinations carried out during fabrication are presented in Table 5-10.
7 These inspections were performed in accordance with procedures meeting the requirements of the edition
and addenda of the ASME Code, Section III listed in Table 5-4. Nondestructive examination techniques
7 used were selected to provide adequate sensitivity, reliability, and reproducibility to inspect surfaces and
7 detect internal discontinuities. Acceptance standards were in accordance with the requirements of the
ASME Code, Section III for the given product and/or fabrication process.

5.3.3 DESIGN EVALUATION

The summary description of the reactor vessel, including major considerations in achieving reactor vessel safety and vessels contributing to the vessel's integrity, is contained in Section 5.2, "Integrity of Reactor Coolant Pressure Boundary." B&W is the reactor vessel designer and fabricator.

5.3.3.1 Design

The ASME Code, Section III, is the Primary design criteria for the reactor vessel. Chapter 5, "Reactor Coolant System and Connected Systems" describes the reactor vessel design, including construction features and arrangement drawing. Materials of construction are listed in Table 5-5. The design code is given in Table 5-4. Table 5-11 gives the design basis values used in the design.

5.3.3.2 Materials of Construction

The materials of construction for the reactor vessel are listed in Table 5-5. Special requirements, reason for selection, and suitability of the materials used are included in Section 5.2.3, "System Design Evaluation." The materials selected have been used extensively in nuclear vessel construction and exhibit well defined properties and serviceability.

5.3.3.3 Fabrication Methods

Fabrication methods used in constructing the reactor vessel are described in Section 5.2.2, "Codes and Classifications." The suitability of the fabrication methods is demonstrated by the excellent service history of vessels constructed using these methods.

5.3.3.4 Inspection Requirements

Fabrication inspection requirements imposed on the reactor vessel are summarized in Section 5.2.3.11, "Quality Assurance" and Table 5-10. Preservice and inservice inspection requirements are summarized in Section 5.2.3.12, "Tests and Inspections."

5.3.3.5 Shipment and Installation

7 B&W specified cleanliness requirements during shipment of the reactor vessel to ensure its arrival at the
7 site in satisfactory condition. B&W also provided appropriate instructions and consultation to the owner
7 for onsite cleaning and vessel protection. Temporary protective coatings and/or covers were applied to
the vessel during shipment and storage as appropriate for expected environmental conditions. Water
7 chemistry was controlled during initial fill, testing, and operation of the vessel to prevent an environment
that may be conducive to material failure.

5.3.3.6 Operating Conditions

The operational limits specified to ensure reactor vessel safety are described in Section 5.2.1, "Design Conditions." These are compared with normal intended and upset operating conditions in Section 5.2.1, "Design Conditions." The design transients for the reactor vessel are specified in Section 5.2.1, "Design Conditions."

5.3.3.7 Inservice Surveillance

7 A discussion of the reactor vessel material surveillance program is given in Section 5.2.3.13, "Reactor
7 Vessel Material Surveillance Program."

5.3.4 PRESSURE - TEMPERATURE LIMITS

5.3.4.1 Design Bases

B&W Topical Report BAW-10046A, Reference 1, provides the bases for setting operational limits on pressure and temperature. This topical report provides detailed assurance that, throughout the life of the
1 plant, operations will comply to requirements of 10 CFR 50, Appendix G. Regulatory Guide 1.99 is used to predict the effects of neutron irradiation on the beltline region materials. For assurance of compliance
1 with 10 CFR 50, Appendix H, through out the life of the plant, see Section 5.2.3.12, "Tests and Inspections."

5.3.4.2 Limit Curves

Topical Report BAW-10046A provides the following information:

1. Procedures and criteria used
2. Safety margins
3. Bases used to determine the limits
4. Procedures that will be used to revise the limits

The limits of pressure and temperature for the following conditions are provided in Technical
8 Specification 3.4.3.

1. Inservice leak and hydrostatic tests
2. Normal operation, including heatup and cooldown
3. Reactor core operation

8

5.3.5 REFERENCES

1 1. BAW-10046A, Rev. 2, Methods of Compliance with Fracture Toughness and Operational Requirements of 1OCFR50, Appendix G.

5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 REACTOR COOLANT PUMPS

The reactor coolant pumps installed on Oconee 1 are Westinghouse Model 93A, while those installed on Oconee 2 and 3 are Bingham. The following briefly describes the significant changes for Oconee 1. Except where noted, the Oconee 1 design is the same as that of Oconee 2 and 3. The reactor coolant flow distribution with less than four pumps operating is presented in Table 5-15.

5.4.1.1 Reactor Coolant Pumps (Oconee 1 Only)

Each reactor coolant loop contains two vertical single stage centrifugal-type pumps which employ a controlled leakage seal assembly. A cutaway view of the pump is shown in Figure 5-17 and the principal design parameters for the pumps are listed in Table 5-16. The estimated reactor coolant pump performance characteristic is shown in Figure 5-18. Connections to the pumps are shown on Figure 5-1.

Reactor coolant is pumped by the impeller attached to the bottom of the rotor shaft. The coolant is drawn up through the bottom of the impeller, discharged through passages in the guide vanes and out through a discharge in the side of the casing. The motor-impeller can be removed from the casing for maintenance or inspection without removing the casing from the piping. All parts of the pumps in contact with the reactor coolant are constructed of austenitic stainless steel or equivalent corrosion resistant materials. A list of pressure containing materials is given in Table 5-5.

The pump employs a controlled leakage seal assembly to restrict leakage along the pump shaft, as well as a secondary seal which directs the controlled leakage out of the pump, and a vapor seal which minimizes the leakage of vapor from the pump into the containment atmosphere.

A portion of the high pressure water flow from the high pressure injection pumps is injected into the reactor coolant pump between the impeller and the controlled leakage seal. Part of the flow enters the Reactor Coolant System through a labyrinth seal in the lower pump shaft to serve as a buffer to keep reactor coolant from entering the upper portion of the pump. The remainder of the injection water flows along the drive shaft, through the controlled leakage seal, and finally out of the pump. A small amount which leaks through the secondary seal is also collected and removed from the pump.

Component cooling water is supplied to the thermal barrier cooling coil.

5.4.1.2 Reactor Coolant Pumps (Oconee 2 & 3)

The reactor coolant pumps are single suction, single stage, vertical, radially balanced, constant speed centrifugal pumps. This type of pump employs mechanical seals to prevent reactor coolant fluid leakage to the atmosphere. A view of the pump is shown in Figure 5-19 and the principal design parameters are
7 listed in Table 5-17. The estimated reactor coolant pump performance characteristics are shown in Figure 5-20. Connections to the pumps are shown on Figure 5-21 (Oconee 2) and Figure 5-22 (Oconee 3).

The pump casing design utilizes a quad-volute inner case permanently welded to a pressure containing outer case. The configuration of the pressure containing outer case is kept simple so that the casing quality will meet the required radiographic level and the stresses can be analyzed to meet the requirements of the design specification. The quad-volute inner casing consists of four volute passages spaced 90° apart which receive the discharge from the pump impeller and guide it efficiently into the outer casing where it flows to the discharge nozzle through a passage having a constantly increasing cross-sectional area. The pump casing is welded into the piping system and the pump internals can be removed for inspection or maintenance without removing the casing from the piping.

The pump cover and stuffing box is a unit containing a thermal barrier, recirculation impellers, shaft, journal bearing, and mechanical face-type seals. The pump shaft is coupled to the motor with a spacer coupling which will permit removal and replacement of the seals without removing the motor. The pump cover has a cooling jacket to remove the heat which passes through the thermal barrier. This jacket has a capacity large enough to remove all heat which is transmitted to the cover. However, additional cooling capacity is provided, in case injection cooling water is lost. A recirculation impeller on the shaft immediately above the journal bearing circulates water in the bearing chamber to a heat exchanger and returns it to the chamber. The pump may be operated with loss of either injection water or cooling water.

The Shaft Seal System consists of face-type mechanical seals operating in tandem. Injection water, at a pressure above the pump suction pressure, is injected into the pump bearing chamber. A small portion of the injection water flows into the pump through a restriction bushing. The major portion flows through
8 cooling slots in the o.d. of the bearing steel. The shaft seal system is made up of three mechanical seals
8 operating in tandem, wherein about one-third of the system pressure is expanded in each seal. Each seal is capable of operation at the full system pressure. The fluid which leaks past the face-type mechanical seal passes into a seal leakage chamber and then out to the quench tank. A low pressure mechanical seal at the top of the seal leakage chamber prevents the escape of fluid to the atmosphere.

Electroslag welding is used to make the seven-inch thick circumferential butt weld which welds together the upper and lower halves of the pump casing. This weld is performed in accordance with ASME Code Case 1355-2 which permits electroslag welding of Class A pressure vessels. The casings are cast and welded by ESCO, who is the leading supplier of RCP casings for the industry.

Electroslag welding is a welding process wherein coalescence is produced by heat generated in a conductive molten slag which melts the filler metal and the surfaces of the work to be welded. The weld pool is shielded by this slag and moves along the full cross section of the joint as the welding progresses. The conductive slag is maintained molten by its resistance to the flow of electric current passing between the electrode and the work. Water cooled, non-fusing metal shoes are used to contain the molten metal on both sides of the weld. The welding is performed in a vertical position with the start and finished performed on run-out tabs affixed to the casting. These run-out tabs are later cut off and discarded. The only variables contained in the method of welding are the wide range of amperage (480-units 720H) and voltage (44-52V) needed to control the molten pool of metal.

The weld is examined 100 percent using liquid penetrant and radiographic examination methods in accordance with Section III of the ASME Code. Ultrasonic inspection is not performed because the pump casing material, austenitic stainless steel, precludes achieving meaningful inspection results.

The pump casing receives two heat treatment cycles. The first is a solution annealing treatment where the pump casing halves are furnace heated to 1900°F, held for a specified time, and water quenched. The second heat treatment is a stabilizing treatment in which the welded pump casing is heated to 725°F and air cooled.

8 Three types of analyses are performed on the pump to verify compliance with ASME Section III: thermal,
8 stress and closure. The first two types are performed using mathematical models of the structure which
8 are analyzed with computer techniques, the third using a merging of preceeding math model results using

8 an assumption of displacement compatibility at contiguous boundaries. The approaches and computer
8 programs are examined in greater length in calculation OSC-1812, Section 3.0 and Section 4.0.

8 In the analysis, to determine temperatures throughout the pump, the pump is broken into two
8 mathematically modeled sections which are analyzed using the THAN thermal analysis program. The
8 first model is of typical pump casing wall section, transient analysis are performed on this section. The
8 second model is of the cover. A steady state thermal analysis is made of this region for both wet and
8 drained cooling jacket conditions.

8 The stress calculations utilize the STARDYNE I, Wilson Jones and NAOS computer programs. Stresses
8 are below their nominal allowables stated in the ASME Boiler and Pressure Vessel Code, Section III,
8 1968 Edition with addenda through summer, 1970.

A summary of the code allowables is listed in Table 5-18 and shown pictorially on Figure 5-23 and Figure 5-24. The reinforcement area is as defined in paragraph N-454 of the ASME Code Section III. The stress analysis performed on the bowl and the attached nozzles showed that the stresses are within the allowable limits. Note that a factor of two was applied to the nozzle loading due to seismic reactions and when these were combined with the dead weight and thermal expansion reactions, the stress levels were within the realistic allowable stress intensities shown in Table 5-18. A summary of maximum calculated
7 stresses is given in Table 5-19.

The casing cover analysis indicates that the thermal stresses and pressure stresses on the cover are within the Section III code allowables.

There are no deviations from the applicable ASME Code requirements in the design and fabrication of the pump casings other than code stamping.

5.4.2 STEAM GENERATOR

The steam generator general arrangement is shown in Figure 5-25 (Oconee 1 and 2) and Figure 5-26 (Oconee 3). Principal design data are tabulated in Table 5-20.

The once-through steam generator supplies superheated steam and provides a barrier to prevent fission products and activated corrosion products from entering the Steam System.

The steam generator is a vertical, straight tube, tube and shell heat exchanger which produces superheated steam at constant pressure over the power range. Reactor coolant flows downward through the tubes and transfers heat to generate steam on the shell side. The high pressure (reactor coolant pressure) parts of the unit are the hemispherical heads, the tube sheets and the tubes between the tube sheets. Tube support plates maintain the tubes in a uniform pattern along their length. The unit is supported by a skirt attached to the bottom head.

The shell, the outside of the tubes, and the tube sheets form the boundaries of the steam producing section of the vessel. Within the shell, the tube bundle is surrounded by a cylindrical baffle. There are openings in the baffle at the feedwater inlet nozzle elevation to provide a path for steam to afford contact feedwater heating. The upper part of the annulus formed by the baffle plate and the shell is the superheat steam outlet, while the lower part is the feedwater inlet heating zone.

Vent, drain, and instrumentation nozzles, and inspection handholes are provided on the shell side of the unit. The reactor coolant side has manway openings in both the top and bottom heads, and a drain nozzle on the bottom head. Venting of the reactor coolant side of the unit is accomplished by a vent connection on the reactor coolant inlet pipe to each unit.

7 Feedwater or Emergency Feedwater is supplied to the steam generator through an emergency feedwater ring located at the top of the steam generator to assure natural circulation of the reactor coolant following the unlikely event of the loss of all reactor coolant pumps.

Four heat transfer regions exist in the steam generator as feedwater is converted to superheated steam. Starting with the feedwater inlet these are:

5.4.2.1 Feedwater Heating Region

Feedwater is heated to saturation temperature by direct contact heat exchange. The feedwater entering the unit is sprayed into the downcomer annulus formed by the shell and the cylindrical baffle around the tube bundle. Steam is drawn by aspiration into the downcomer and heats the feedwater to saturation temperature.

The saturated water level in the downcomer provides a static head to balance the static head in the nucleate boiling section, and the required head to overcome pressure drop in the circuit formed by the downcomer, the boiling sections, and the bypass steam flow to the feedwater heating region. The downcomer water level varies with steam flow from 15 - 100 percent load. A constant minimum level is held below 15 percent load.

5.4.2.2 Nucleate Boiling Region

The saturated water enters the tube bundle just above the lower tube sheet and the steam-water mixture flows upward on the outside of the tubes counter current to the reactor coolant flow. The vapor content of the mixture increases almost uniformly until DNB is reached, and then film boiling and super heating occurs.

5.4.2.3 Film Boiling Region

Dry saturated steam is produced in the film boiling region of the tube bundle.

5.4.2.4 Superheated Steam Region

Saturated steam is raised to final temperature in the superheater region. The amount of surface available for superheat varies inversely with load. As load decreases the superheat section gains surface from the nucleate and film boiling regions. Mass inventory in the steam generator increases with load as the length of the heat transfer regions vary. Changes in temperature, pressure, and load conditions cause an adjustment in the length of the individual heat transfer regions and result in a change in the inventory requirements. If the inventory is greater than that required, the pressure increases. Inventory is controlled automatically as a function of load by the feedwater controls in the Integrated Control System.

7 Steam Generator Feedwater quality is addressed in the Chemistry Section Manual.

5.4.3 REACTOR COOLANT PIPING

The general arrangement of the reactor coolant piping is shown in Figure 5-3, Figure 5-4, Figure 5-5, Figure 5-6, Figure 5-7, and Figure 5-8. Principal design data are tabulated in Table 5-21.

The major piping components in this system are the 28-inch i.d. cold leg piping from the steam generator to the reactor vessel and the 36-inch i.d. hot leg piping from the reactor vessel to the steam generator. Also included in this system are the 10-inch surge line and the 2-1/2-inch spray line to the pressurizer. The system piping also incorporates the auxiliary system connections necessary for operation. In addition to drains, vents, pressure taps, injection, and temperature element connections, there is a flow meter section in each 36-inch line to the steam generators to provide a means of determining the flow in each loop.

The 28-inch and 36-inch piping is carbon steel clad with austenitic stainless steel. Short sections of 28-inch stainless steel transition piping are provided between the pump casing and the 28-inch carbon steel lines.

For Oconee 1 only a 28 in. i.d. x 31 in. i.d. stainless steel transition section is installed between the existing 28 in. i.d. coolant piping and the 31 in. i.d. pump suction.

Also a 28 in. i.d. small angle elbow section between the pump discharge nozzle and the reactor inlet pipe is installed to account for the radial discharge of the replacement pump. The original pump had a tangential discharge nozzle. The elbow section is carbon steel with a section of stainless for welding to the pump casing nozzle.

Stainless steel or Inconel safe-ends are provided for field welding the nozzle connections to smaller piping. The piping safe-ends are designed so that there will not be any furnace sensitized stainless steel in the pressure boundary material. This is accomplished either by installing stainless steel safe-ends after stress relief or using Inconel. Smaller piping, including the pressurizer surge and spray lines, is austenitic
7 stainless steel. All piping connections in the Reactor Coolant System, larger than 2 inch diameter, are
7 butt-welded except for the flanged connections on the pressurizer relief valves.

Thermal sleeves are installed where required to limit the thermal stresses developed because of rapid changes in fluid temperatures. They are provided in the following nozzles: the four high pressure injection nozzles on the reactor inlet pipes; the two core flooding low pressure injection nozzles on the reactor vessel; and the surge line nozzle and spray line nozzle on the pressurizer.

5.4.4 REACTOR COOLANT PUMP MOTORS

The reactor coolant pump motors are large, vertical, squirrel cage, induction machines. The motors have flywheels to increase the rotational-inertia, thus prolonging pump coastdown and assuring a more gradual loss of main coolant flow to the core in the event pump power is lost. The flywheel is mounted on the upper end of the rotor, below the upper radial bearing and inside the motor frame. An anti-reverse device is included in the flywheel assembly to eliminate reverse rotation when there is back flow. Prevention of back rotation also reduces motor starting time.

The motors are enclosed with water-to-air heat exchangers so as to provide a closed circuit air flow through the motor. Radial bearings are floating pad type, and the thrust bearing is a double-acting Kingsbury type designed to carry the full thrust of the pump. A High Pressure Oil System with separate pumps is provided with each motor to jack and float the rotating assembly before starting. Once started, the motor provides its own oil circulation.

Instrumentation is provided to monitor motor cooling, bearing temperature, winding temperature, winding
7 differential current, and speed. Instrumentation is also provided for measuring shaft displacement and
7 frame velocity vibration.

In evaluating the design of the reactor coolant pump motor as it relates to the safety of the Reactor Coolant System, many items have been considered, namely: the overspeed of the motor; flywheel and shaft integrity; bearing design and system monitoring; seismic effects; and quality control and documentation.

An analysis of these considerations are given as follows as an indication of the safety and reliability that is integral with the motors:

5.4.4.1 Overspeed Considerations

The reactor coolant pump motors normally receive their electrical power from the nuclear generating unit through the unit's Auxiliary Electric System. On load rejection, the generating unit is designed to separate from the transmission network and remain in a standby operating condition carrying its own auxiliaries.

Figure 5-27 shows the turbine speed response following load rejection with the steam control valves wide open (VWO). On load rejection with VWO, the speed of the turbine-generator will increase under the control of the Normal Speed Governing Control System. The maximum speed attainable under the Normal Speed Governing Control System is less than 106 percent with the unit auxiliaries connected. This governing system is comprised of three independent control activities, namely: the speed control unit, power unbalance relay and the fast acting intercept valves all of which function to limit overspeed to below 106 percent.

As indicated in Figure 5-27 there are additional safety devices backing up the speed governing system, namely:

1. Mechanical overspeed trip which operates at 110 percent turbine-generator speed.
2. Generator overfrequency relay trip which is an electrical trip that operates at 111 percent turbine-generator speed.
3. Electrical back-up overspeed relay trip which operates at 112 percent turbine-generator speed.

In addition, each individual reactor coolant pump motor control circuit includes an overfrequency relay which trips the motor at 115 percent motor (or turbine-generator) speed. Therefore, it is evident that the reactor coolant pump motors speed will be limited to less than 115 percent.

5.4.4.2 Flywheel Design Consideration

For conservatism, the design of the flywheel on the reactor coolant pump motor is based on a design speed of 125 percent. The primary stress at the flywheel bore radius, with a speed of 125 percent, is 20,000 psi which is less than 50 percent of the 50,000 psi minimum yield strength of the flywheel material. This, therefore, yields a centrifugal stress design safety margin of 250 percent at 125 percent speed.

The Duke Power Company specification on the motor calls for 500 motor starts in forty years; the flywheels have been designed for 10,000 starts yielding a safety factor of 20. However, calculation based on the material used in the flywheel results in 400,000 cycles required for crack initiation which results in a flywheel fatigue design safety factor of 800.

9 The reactor coolant pump motors are large, vertical, squirrel cage, induction motors. The motors have
9 flywheels to increase rotational-inertia, thus prolonging pump coastdown and assuring a more gradual loss
9 of main coolant flow to the core in the event that pump power is lost. The flywheel is mounted on the
9 upper end of the rotor, below the upper radial bearing and inside the motor frame. The assumed
9 operation of the reactor coolant pumps was 500 motor starts over forty years. The aging effect of concern
9 is fatigue crack initiation in the flywheel bore key way from stresses due to starting the motor. Therefore,
9 this topic is considered to be a time- limited aging analysis for license renewal.

9 The flywheels have been designed for 10,000 starts that provide a safety factor of 20 over the original
9 operation assumptions. Reaching 10,000 starts in 60 years would require on average a pump start every
9 2.1 days. This conservative design is valid for the period of extended operation.

9 References for this section: Application [Reference 5] and Final SER [Reference 6]

5.4.4.3 Flywheel Material, Fabrication, Test and Inspection

5.4.4.3.1 Material

The flywheel is manufactured from vacuum degassed ASTM 533 steel.

5.4.4.3.2 Fabrication and Test

1. Flywheel blanks are flame cut from a plate with enough surplus material to allow for the removal of the flame affected metal.
2. At least three charpy tests are made on each plate parallel and normal to the rolling direction to determine that the blank meets specifications.
3. A complete 100 percent volumetric ultrasonic test is made on the blank and tension and bend tests are also made prior to shipment of a blank to Westinghouse Electric Company.
4. Following the machining of the flywheel at the Westinghouse plant, a complete 100 percent volumetric ultrasonic test is conducted on the fly wheel and a liquid penetrant test is conducted on the bore.
5. After the flywheel is installed and the motor is completely assembled, a 125 percent overspeed test for one minute is conducted on the assembled unit.
6. Following the overspeed test, a periphery sonic test is conducted on the flywheel through access holes in the motor frame.
7. To assure the original integrity of each flywheel during operation, the following inservice inspections will be performed.

   At approximately three-year intervals, the bore and keyway of each reactor coolant pump flywheel shall be subject to an in-place, volumetric examination. Whenever maintenance or repair activities necessitate flywheel removal, a surface examination of exposed surfaces and a complete volumetric examination shall be performed, if the interval measured from the previous such inspections is greater than 6 2/3 years. Results of the examination will be evaluated by the original acceptance criteria and compared with the original examination data to assure the absence of unacceptable defects.

5.4.4.4 Shaft Design and Integrity

The shear stress on the shaft in the vicinity of the flywheel is 5520 psi with short circuit torque on the motor. The minimum strength of the shaft material is 23,000 psi which results in a safety factor of four under the maximum torque condition. Because of the conservatism used in the design of the shaft, it is concluded that shaft failure is not credible.

5.4.4.5 Bearing Design and Failure Analysis

The motor pump assembly is supported by a Kingsbury type thrust bearing which consists of a runner and upper and lower thrust plates. The history of the Kingsbury type bearing design indicates that the device is highly reliable and has a non-locking failure mode.

Provided on the motor are a number of devices to warn the operator of bearing trouble and these devices are each independent in their operation. The thrust bearing monitoring devices are as follows:

1. Two thermocouples located diametrically opposite to each other in the upper thrust plates.

2. Two thermocouples located diametrically opposite to each other in the lower thrust plates.
3. One thermocouple in the upper oil pot.
4. Oil pot level alarm device.
7 5. Shaft displacement and frame velocity vibration devices.

7 These devices are arranged to provide alarm indications to the control room operator. If a thrust bearing
7 fails with the motor operating, the result would be melting of the bearing babbit and, finally automatic
7 tripout of the motor on overload. However, bearing degradation which would lead to this point would be
7 evident to the control room operators in at least one of the indicators discussed above and would be
7 mitigated by manually securing the pump. Therefore, since seizure of the bearing will not result from a
7 bearing failure, it is concluded that missiles will not be produced.

5.4.4.6 Seismic Effects

The pump motor units have been analyzed against the combination effects of mechanical and seismic loads including the gyroscopic effects of the flywheel to verify that the stress limits will not be exceeded and the pump motor unit will operate through the maximum hypothetical earthquake.

5.4.4.7 Documentation and Quality Assurance

The Duke Power Company and the motor supplier, Westinghouse Electric Corp., have a rigid quality assurance program directed at assuring the integrity of the reactor coolant pump motors.

7 A quality assurance folder was initially developed by Duke Power Company on each motor and included
7 the following:

1. Specifications and addendum
2. Description of the manufacturer's quality control organization and engineering order handling.
3. Copies of all inspection reports relating to the appropriate motor.
4. Samples of quality control drawings.
5. Copies of all test reports including flywheel material vendor test reports; Westinghouse motor test reports; bearing assembly reports; shaft tests; sonic test reports on the machined flywheel prior to assembly on the motor and following the 125 percent speed test; and certification on the motor test report that the overspeed test was conducted on the assembled motor.
6. Copies of the Duke Form QA-2 which is the manufacturer's certification to Duke Power Company Design Engineering that the motors were manufactured per specification and the Duke Power quality assurance program.
7. Copies of Duke Form QA-1 which the indication to the field quality control engineer that the motor described thereon was manufactured to the specification and the Duke quality assurance program.
8. Copies of Duke Form QC-31 which is the field receiving report on the motor.

7 A copy of each quality assurance folder was sent to the field quality control engineer and a copy was
7 placed in the Design Engineering Department file. See the applicable controlled Procurement Package
7 for RCP motor QA information.

5 Babcock & Wilcox has analyzed the reactor coolant pump assembly action resulting from postulated
Reactor Coolant System breaks. B&W Topical Report, BAW-10040, Reference 1, describes the

homologous pump model used for the speed calculations and presents results for the spectrum of breaks analyzed.

A discussion of the linear elastic fracture analysis to determine the structural failure speed of the reactor coolant pump motor flywheel assembly is also included.

5.4.5 REACTOR COOLANT EQUIPMENT INSULATION

7 The majority of the Reactor Coolant System components are insulated with metal reflective type
7 insulation. This insulation is supported by rings welded to weld pads on the components during field
installation of the insulation. The weld pads to which the holding rings are attached are added to the
7 components prior to final stress relief of the component. The remaining portion of the RCS is insulated
7 with approved removable blanket insulation, secured with velcro fasteners.

The insulation units are removable and are designed for ease of removal and installation in such areas as field welds, nozzles, and bolted closures. The insulation units permit free drainage of any condensate or moisture from within the insulation unit.

5.4.6 PRESSURIZER

The pressurizer general arrangement is shown in Figure 5-28 and principal design data are tabulated in Table 5-22.

The electrically heated pressurizer establishes and maintains the Reactor Coolant System pressure within prescribed limits, and provides a steam surge chamber and a water reserve to accommodate reactor coolant density changes during operation.

The pressurizer is a vertical cylindrical vessel with a bottom surge line penetration connected to the reactor coolant piping at the reactor outlet. The pressurizer contains removable electric heaters in its lower section and a water spray nozzle in its upper section. Heat is removed or added to maintain Reactor Coolant System pressure within desired limits. The pressurizer vessel is protected from thermal effects by a thermal sleeve in the surge line and by an internal diffuser located above the surge pipe entrance to the vessel.

During outsurges, as Reactor Coolant System pressure decreases, some of the pressurizer water flashes to steam, thus assisting in maintaining the existing pressure. Heaters are then actuated to restore the normal operating pressure. During insurges, as system pressure increases, water from the reactor vessel inlet piping is sprayed into the steam space to condense steam and reduce pressure. Spray flow and heaters are controlled by the pressure controller. The pressurizer water level is controlled by the level controller.

Since all sources of heat in the system, core, pressurizer heaters, and reactor coolant pumps, are interconnected by the reactor coolant piping with no intervening isolation valves, relief protection is provided on the pressurizer. Overpressure protection consists of two code safety valves and one electromatic relief valve.

To eliminate abnormal buildup or dilution of boric acid within the pressurizer, and to minimize cooldown of the coolant in the spray and surge lines, a bypass flow is provided around the pressurizer spray control valve. This continuously circulates approximately one gpm of reactor coolant from the heat transport loop. A sampling connection to the liquid volume of the pressurizer is provided for monitoring boric acid concentration. A steam space sampling line provides capability for monitoring of or venting accumulated gases.

During cooldown and after the decay heat system is placed in service, the pressurizer can be depressurized and cooled by circulating through a connection from the High Pressure Injection System to the pressurizer spray line.

Electroslag welding is utilized in the fabrication of the pressurizer, only in the longitudinal seams of the shell courses. A total of three individual electroslag welds are made in the fabrication of each pressurizer. The electroslag welding process and quality control is the same as described in Section 5.2.3.4, "Steam Generators."

Normal Reactor Coolant System pressure control is by the pressurizer steam cushion in conjunction with the pressurizer spray, electromatic relief valve, and heaters. The system is protected against overpressure by Reactor Protective System circuits such as the high pressure trip and by pressurizer relief valves located
7 on the top head of the pressurizer. The schematic arrangement of the relief valves is shown in Figure 5-1
7 and Figure 5-2. Reactor Coolant System pressure settings and relief valve capacities are listed in Table 5-1.

Reduction of pressure during Reactor Coolant System cooldown is accomplished by the pressurizer spray provided by the reactor coolant pump. Below a system temperature of approximately 250°F, the Low Pressure Injection System is used for system heat removal and the steam generators and reactor coolant pumps are removed from service. During this period, spray flow is provided by a branch line from one high pressure injection line to the pressurizer spray line for further pressure reduction or complete depressurization of the Reactor Coolant System.

5.4.6.1 Pressurizer Spray

The pressurizer spray line originates at the discharge of a reactor coolant pump in the same heat transport loop that contains the pressurizer. Pressurizer spray flow is controlled by a solenoid valve using an on-off control in response to the opening and closing pressure set points. An electric motor operated valve in series with the spray line is to provide for remote spray line isolation.

5.4.6.2 Pressurizer Heaters

The pressurizer heaters replace heat lost during normal steady state operation, raise the pressure to normal operating pressure during Reactor Coolant System heatup from the cooled down condition, and restore system pressure following transients. The heaters are grouped in four banks and are controlled by the pressure controller. The first bank utilizes proportional control and will normally operate at partial capacity to replace heat lost, thus maintaining pressure at the set point. On-off control is used for the remaining three banks. A low level interlock prevents the heaters from being energized with the heaters uncovered.

Based on B&W calculations and startup testing experience, a conservative value for the total heat loss
8 from the Reactor Coolant System, in MODE 3 with average Reactor Coolant temperature ≥525°F, is 107
kilowatts. On this basis, it was determined that a minimum of 126 kilowatts of pressurizer heaters, which corresponds to the smallest single bank of pressurizer heaters, should be available from an assured power
8 source within two hours after loss of offsite power in order to establish and maintain natural circulation in
8 MODE 3 with average Reactor Coolant temperature ≥525°F. This calculated heat loss is similar to heat loss estimates that have been accepted for other pressurized water reactors.

The pressurizer heaters for each unit are supplied from non-safety-related motor control centers (MCC). The MCC are in turn powered via load centers from the 4160-volt engineered safeguard buses. These buses are powered from a hydro station which is the emergency generation source (EGS) in the event of loss of offsite power. This emergency source has ample capacity to provide emergency power to all

pressurizer heaters and is capable of doing so promptly following an accident. The pressurizer heaters are divided among the three 4160 volt EGS buses such that the loss of one entire 4160 volt bus will not
8 preclude the capability to supply sufficient pressurizer heaters to maintain natural circulation in MODE 3
8 with average Reactor Coolant temperature ≥525°F.

Uncovering energized direct immersion heaters does not immediately harm the heaters. Three heaters, one for each bundle assembly, are tested in air to provide an accelerated life test as follows:

1. Tested for 100 hours at sheath temperature of 600°F to 1600°F with a watt density of 85 watt/in.^{2}.
2. Cycled 400 times with a cycle time of 15 minutes on and 15 minutes off with a watt density of 65 watt/in.^{2}.

The heaters successfully completed this test, which simulated a total of 200 hours "on" time for the heaters in an uncovered environment while in an energized condition. Moreover, the heater sheath is designed for 2500 psig and 670°F with the heater terminal also designed for these same conditions. Therefore, the heater sheath could fail and the pressurizer vessel integrity would be maintained. This conclusion has been substantiated in tests conducted by the heater vendor for a similar design.

5.4.6.3 Pressurizer Code Safety Valves

Two pressurizer code safety valves are mounted on individual nozzles on the top head of the pressurizer. The valves have a closed bonnet with bellows and supplementary balancing piston. The valve inlet and outlet is flanged to facilitate removal for maintenance or set point testing.

5.4.6.3.1 Safety Valve Testing and Qualification

During the EPRI Safety Valve testing, it was determined that the short inlet Dresser 31739A valve successfully met all the test requirements with the "reference" ring settings. The performance of the valve was determined to be dependent on the ring settings. Duke Power Company evaluated the safety impact of the inadequate safety ring settings and determined that for the limiting RCS overpressure transients the plant safety can be maintained. In October 1982, all the Oconee Nuclear Station safety valves were
9 adjusted to the recommended settings resulting from EPRI tests. Duke Power Company has committed
9 to the optimal ring settings for the Dresser 31739A safety valves, which are described in the corresponding
9 NRC Safety Evaluation Report (Reference 4).

5.4.6.4 Pressurizer Electromatic Relief Valve

The pressurizer electromatic relief valve, also called power operated relief valve (PORV), is mounted on a separate nozzle on the top head of the pressurizer. The main valve operation is controlled by the opening or closing of a pilot valve which causes unbalanced forces to exist on the main valve disc. The pilot valve is opened or closed by a solenoid in response to the pressure set points. Flanged inlet and outlet connections provide ease of removal for maintenance purposes.

The Power Operated Relief Valve (PORV) in each Oconee unit is actuated by a DC solenoid-operated pilot valve that is connected to a Class IE DC system. The block valve for the PORV is an AC motor operated valve and is connected to an AC emergency power supply. The power supplies for the PORV and its associated block valve are therefore independent and diverse.

5.4.6.4.1 PORV and Block Valve Testing and Qualification

Under the EPRI Test Program, for all tests applicable to Oconee, the Dresser PORV was opened and closed on demand. The functionability of the Dresser PORV has been shown for all expected operating and accident conditions applicable to Oconee Nuclear Station and the requirements of NUREG-0737, Item II.D.1.A have been met.

Under the EPRI PORV Block Valve Test Program, a Westinghouse motor-operated gate valve was tested on steam to full differential pressure conditions. Oconee Nuclear Station uses the same Westinghouse valve and LimiTorque operator for PORV block valve application. Based upon the successful EPRI tests for the valve-operator combination, the Oconee PORV block valves meet the intent of NUREG-0737 Item 11.D.1.B. The program test results were submitted to NRC in April 1982 and October 1985 (Reference 2 xxx form=numonly.).

8 With the initiation of NRC Generic Letter 89-10 (GL89-10) the Nuclear Industry and NRC have taken a
8 much more focused, rigorous approach to assuring Active Motor Operated Valves (MOVs) have sufficient
8 operating margin. The EPRI Performance Prediction Methodology (PPM) is a more conservative
8 calculation guideline for determining MOV operating margins. Based on the EPRI PPM and other
8 inputs, the operator for valves 1, 2, 3RC-4, the unit's PORV Block valves, have been upgraded from an
8 SB-00-15 to an SB-0-25 operator.

5.4.6.5 Relief Valve Effluent

Effluent from the pressurizer electromatic-relief and code safety valves discharges into the quench tank which condenses and collects the relief valve effluent. After the quench tank receives relief valve effluent, the tank contents are cooled to normal temperature by the component drain pump and quench tank cooler of the Coolant Storage System. The tank fluid is circulated from the tank through the cooler and returned to the tank by spraying into the tank vapor space. The quench tank is protected against overpressure by a rupture disc sized for the total combined relief capacity of the two pressurizer code safety valves and the pressurizer electromatic relief valve. The quench tank can be remotely vented to the Gaseous Waste Disposal System.

An Acoustical Monitoring System is installed on each unit. It is a reliable, single channel system, powered from a battery backed vital bus. It will provide the operator with positive indication of valve position and an annunciation of an open valve in the control room. The valve position indication components have been seismically and environmentally qualified as appropriate for conditions applicable to their location.

Backup valve position indication is provided by temperature sensors located downstream of the PORV and safety valves and by the quench tank level indicator.

5.4.7 INTERCONNECTED SYSTEMS

5.4.7.1 Low Pressure Injection

The Low Pressure Injection System provides the capability below about 250°F for cooling the Reactor Coolant System during plant cooldown. During this mode of operation, coolant is drawn from the Reactor Coolant System through a nozzle on the reactor outlet pipe, circulated through the low pressure injection coolers by the low pressure injection pumps and then injected back into the Reactor Coolant System through two nozzles on the reactor vessel into the inlet side of the core. The heat received by this system is rejected to the Low Pressure Service Water System. Components in these two systems are redundant for reliability purposes.

The Low Pressure Injection System also performs an emergency injection function for a loss of coolant accident and provides long term emergency core cooling; this is described in Chapter 6, "Engineered Safeguards."

5.4.7.2 High Pressure Injection

The High Pressure Injection System controls the Reactor Coolant System coolant inventory, provides the seal water for the reactor coolant pumps, and recirculates Reactor Coolant System letdown for water quality maintenance and reactor coolant boric acid concentration control. Letdown of reactor coolant is through a nozzle on the outlet coolant pipe from one steam generator. The discharge of the high pressure injection pumps connects to a nozzle on each of the reactor inlet pipes downstream of the reactor coolant pumps. The reactor coolant which is letdown is returned to the Reactor Coolant System through the nozzles in a different heat transport loop from the heat transport loop containing the letdown line.
7 Components are redundant for reliability purposes (Section 9.3.2, "High Pressure Injection System").

The High Pressure Injection System utilizes four injection nozzles in carrying out the high pressure emergency injection function after a loss of coolant accident.

The High Pressure Injection/Makeup Nozzle assemblies at Oconee incorporate a thermal sleeve to provide a thermal barrier between the cold HPI/MU Fluid and the HOT HPI Nozzle. In 1982, High Pressure Injection/Makeup Nozzle cracking problems were identified on several operating B&W plants. A task force formed by B&W owners group identified the root cause of the failures and undertook modifications, in consultations with NRC to eliminate such future failures.

Site inspections of Oconee 1, 2 and 3 were conducted. Oconee 2 and Oconee 3 were found to have nozzle cracking and thermal sleeve displacement problems. The radiographic and ultrasonic testing of the Oconee 1 indicated that no abnormal conditions were present in any of the nozzles; this is attributable to the unique double thermal sleeve design of the Oconee 1 nozzles.

The B&W owners task force studied the safe end nozzle cracking problems on a generic basis and reported its findings to the NRC (Reference 3). The task force concluded that all cracked safe ends of the HPI/MU nozzles were associated with loose thermal sleeves; the cracked safe ends were associated with the makeup nozzles only, and the cracks were propagated by thermal fatigue. B&W recommended modifications to the design and inspection of the HPI/MU nozzles. The modified design installs a hard rolled thermal sleeve which prevents thermal shock to the nozzle assembly and helps reduce flow induced vibrations more effectively. An in-service inspection program had been developed to provide early detection of the safe-end cracking problems. The Oconee 1 makeup nozzles did not require modifications but are now subject to an augmented ISI program.

5.4.7.3 Core Flooding System

The Core Flooding System floods the core in the event of a loss of coolant accident. Connection to the reactor vessel is through the two nozzles described above for low pressure injection. The low pressure injection and core flooding lines tie together and connect to the same nozzle on the reactor vessel.

2 The core flood nozzles have flow restrictors installed to minimize blowdown due to postulated core flood line break.

5.4.7.4 Secondary System

The principal Decay Heat Removal System interconnected with the Reactor Coolant System is the Steam and Power Conversion System. The Reactor Coolant System is dependent upon the Steam and Power Conversion System for decay heat removal at normal operating conditions and for all reactor coolant
7 operating temperatures above 250°F. The system is discussed in detail in Chapter 10, "Steam and Power
7 Conversion System."

The Turbine Bypass System routes steam to the condensers when the turbine has tripped or is shutdown and also during large plant load reduction transients when steam generation exceeds the demand. Overpressure protection for the secondary side of the steam generators is provided by the turbine bypass system and by safety valves mounted on the main steam lines outside of the Reactor Building. The
7 Emergency Feedwater System will supply water to the steam generators in the event that the Main Feedwater System is inoperative. The physical layout of the Reactor Coolant System provides natural circulation of the reactor coolant to ensure adequate core cooling following a loss of all reactor coolant pumps.

5.4.7.5 Sampling

A sample line from the pressurizer steam space to the Chemical Addition and Sampling System permits detection of non-condensible gases in the steam space. This sample line also permits a bleeding operation from the vapor space to the letdown line of the High Pressure Injection System to transport accumulated noncondensible gases in the pressurizer to the letdown storage tank.

5.4.7.6 Remote RCS Vent System

The Oconee design has the capability for venting post-accident non-condensible gases that, in sufficient quantities, could accumulate at high points in the RCS and impair natural circulation. Although such an event is highly unlikely, the remote RCS vents on the RCS hot legs and reactor vessels will enable venting of these gases.

The design of the RCS High Point Vent System consists of two valves installed in series in each of the following existing vent connections: steam generator piping high points, and reactor vessel head high point. The redundant valve in each vent line assures that venting operations can be terminated under postulated single failure. The three pairs of valves each receive electrical power from a different safety related power source. Vent valve position indication is provided by limit switches within each solenoid valve. The valves require power to open and fail close on loss of power. The existing power operated relief valve can be used to vent the pressurizer.

The reactor vessel head vent is attached to an existing Axial Power Shaping Rod motor tube and closure assembly. Two normally deenergized solenoid valves are installed in the vent line and controlled from the control room. The vent ties into a hot leg vent and discharges into the air stream from the Reactor Building Cooling Units when operated.

One independent remotely operated vent is provided at the high point of each 36-inch RCS hot leg line. Each vent makes use of the existing manual vent line. A tee has been added after the first manual valve and a new manual valve has been added after the tee. The first manual valve (1RC-19, 1RC-38, 2RC-19,
4 2RC-196, 2RC-38, 3RC-19, 3RC-38) is in the open position. The function of the first valve has been transferred to the second valve (1RC-168, 1RC-169, 2RC-168, 2RC-169, 3RC-168, 3RC-169). The new vent runs from the tee through two solenoid valves and discharges into the air stream from the Reactor Building Coolant Units. The solenoid valves are remotely controlled from the Control Room. The function of the manual vent is unaffected.

The reactor coolant vent system is acceptable to the NRC and in conformance with the requirements of 10CFR 50.44 paragraph (c)(3)(iii) and the guidelines of NUREG 0737 Item II.B.1, and NUREG-0800 Section 5.4.12.

5.4.8 COMPONENT FOUNDATIONS AND SUPPORTS

The supports for all major components listed in this section are analyzed in detail to insure adequate structural integrity for their intended function during normal operating, seismic, and accident conditions. Following calculation of sources of loading, stresses and motions at significant locations are computed and compared to applicable criteria. Details of this analysis are given in Chapter 3, "Design of Structures, Components, Equipment, and Systems."

5.4.8.1 Reactor Vessel

The reactor vessel is bolted to a reinforced concrete foundation designed to support and position the vessel and to withstand the forces imposed on it by a combination of loads including the weight of vessel and internals, thermal expansion of the piping, design basis earthquake (DBE), and dynamic load following reactor trip.

The foundation, in addition, restrains the vessel during the combined forces imposed by the circumferential rupture of a 36-inch reactor outlet line and a simultaneous maximum hypothetical earthquake (MHE).

The vessel foundation further is designed to provide accessibility for the installation and later inspection of incore instrumentation, piping, and nozzles; to contain ductwork and vent space for cooling air to remove heat losses from the vessel insulation; and to provide a sump and drainage line for leak detection.

5.4.8.2 Pressurizer

The pressurizer is supported on a structural steel foundation by eight lugs welded to the side of the vessel.

The foundation and supports are designed to withstand the loads imposed by thermal expansion of the pressurizer, the weight of the pressurizer including its contents and attached piping, relief valve reaction forces, and forces imposed by the design basis earthquake. In addition, the foundation and supports will restrain the vessel during the combined forces imposed by the circumferential rupture of the 10-inch surge line coupled with the MHE.

The foundation is also designed to permit accessibility to pressurizer surfaces for inspection.

5.4.8.3 Steam Generator

The steam generator foundation is designed to support and position the generator. The foundation is designed to accept the loads imposed by the generators and feedwater piping filled with water, the attached reactor coolant piping also filled with water, and steam lines under the MHE. The foundation is also designed to restrain the steam generator under the combined forces due to a circumferential rupture of a 28-inch coolant line and a simultaneous MHE.

Forces imposed on the generator by the rupture of a 36-inch coolant line are transferred to the shielding walls by a support structure located near the top of the generator.

5.4.8.4 Piping

The reactor coolant piping, inlet and outlet lines, are supported by the reactor vessel and steam generator nozzles. The piping will withstand the forces imposed on it by the MHE.

5.4.8.5 Pump and Motor

The reactor coolant pump casing, internals, and motor weight are supported by the 28 inch coolant lines and constant load hangers attached to the motor. In the cold condition, the coolant piping will support the coolant pump and motor without the hangers. The hangers are designed to withstand the forces imposed on them by the MHE.

5.4.8.6 LOCA Restraints

Each steam generator has a support located opposite the upper tube sheet and transfers forces from the generator into the shield walls in the event of a circumferential rupture of the 36-inch line.

Each 28-inch reactor coolant inlet line and 36-inch reactor coolant outlet line has a restraint located outside of and bolted to the primary shield to limit pipe motion in the event of a circumferential rupture of the piping inside the primary shield.

A detailed study of the primary loop was performed to determine potential pipe break locations which could possibly cause either fluid impingement or pipe impact forces on the Secondary System. The results of this evaluation indicated the most credible break locations which could cause either of these effects are:

1. A guillotine break at the pump discharge in the cold leg piping;
2. A longitudinal split in the vertical pump suction segment of the cold leg piping; or,
3. A longitudinal split in the vertical segment of the hot leg piping.

All of the above breaks could potentially affect the generator because of their proximity to it. The main steam lines, however, are shielded from the effects of pipe breaks by the generator.

The primary piping and steam generator were analyzed for each of the above breaks and supports provided to restrain the pipe from whipping into the generator. In addition, the stresses in the generator shell due to the fluid impingement forces were calculated and found to be within acceptable limits.

The restraints on the primary loop are shown in Figure 5-29 and Figure 5-30. The coolant pump is restrained by steel supports from the primary shield wall. The hot leg piping is restrained by the concrete support at the primary cavity penetration, an intermediate steel support from the primary wall, and another steel support near the generator upper tube sheet. The vertical segment of the cold leg piping is restrained by a steel support midway along its length, which would spread any rupture load over a larger area of the generator shell.

To verify the location and size of the piping supports, the piping was analyzed for rupture loads occurring at the worst point along its length. The rupture thrust force was assumed equal to P x A, where P is the coolant pressure and A the flow-sectional area of the pipe. The thrust was applied as an equivalent static force using a dynamic load factor of 2.0. Assuming the force to be a point load acting at the midpoint of the span between supports, the piping stresses were calculated using beam models. The supports are located so as to prevent the formation of plastic hinges in the piping, which would lead to an unstable linkage-type structure and possible impacting against the generator.

To evaluate the effect of fluid jet impingement on the generator, an equivalent static pressure load on the shell was calculated. A break of 14 ft^2 for the hot leg or 8.5 ft^2 for the cold leg was assumed. The maximum initial mass velocity was computed using the methods outlined in the report "Maximum Two-Phase Vessel Blowdown From Pipes, APED-4827," by F. J. Moody. It was assumed that the fluid leaves the break in a direction normal to the pipe and that its velocity undergoes a 90° change in direction upon impinging on the steam generator. The resulting shell pressure loading was calculated to be 1300 psi.

A shell analysis was performed on the steam generator to determine the stress intensity due to the above loading. A B&W proprietary digital computer code, which considers two-dimensional shells with asymmetric loading, was utilized. The loading distribution and stress model are shown in Figure 5-31 and Figure 5-32.

The maximum stress intensity was computed to be 38,600 psi. This is less than the allowable stress of 46,670 psi. Based on these results for the 36- inch i.d. pipe break, it was concluded that the steam generator shell could also withstand the reduced loading which would be generated by a 28-inch i.d. break.

5.4.9 REFERENCES

1. BAW-10040, Reactor Coolant Pump Assembly Overspeed Analysis

2. Letter from H. B. Tucker (Duke) to H. R. Denton (NRC) dated October 1, 1985.
Subject: Performance Testing of Relief and Safety Valves.

3. Babcock & Wilcox Owners Group Safe End Task Force Report on Generic HPI/MU Nozzle Component Cracking. B&W Document #77-1140611-00, Submitted to NRC on February 15, 1983.

7 4. Letter from L. A. Weins (NRC) to H. B. Tucker (Duke) dated July 19, 1989. Subject: Safety
7 Evaluation Report for NUREG-0737, Item II.D.1, Performance Testing of Relief and Safety Valves
7 for Oconee Units 1, 2, and 3 (TACS 44600, 44601, and 44602).

9 5. *Application for Renewed Operating Licenses for Oconee Nuclear Station, Units 1, 2, and 3*, submitted
9 by M. S. Tuckman (Duke) letter dated July 6, 1998 to Document Control Desk (NRC), Docket
9 Nos. 50-269, - 270, and -287.

9 6. NUREG-1723, *Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station,*
9 *Units 1, 2, and 3*, Docket Nos. 50-269, 50-270, and 50-287.

THIS IS THE LAST PAGE OF THE CHAPTER 5 TEXT PORTION.

Table 5-2 (Page 3 of 3). Transient Cycles for RCS Components Except Pressurizer Surge Line

Transient Number	Transient Description (ASME Category)	Design Cycles	Component Exceptions (See notes)
Component and General Flaw Location	Limiting Transient		Total Allowable Transient Cycles
Unit 1			
OTSG-Upper Head to Tubesheet	Heatup/Cooldown		207*
Unit 2			
None			
Unit 3			
None			

*-limiting evaluation

Note 2: In order to analytically demonstrate a usage factor of less than 1.0, certain welds associated with the Emergency HPI nozzles have been qualified for fewer than the design number of cycles of the two, noted transients. The analysis uses a total of 29 cycles for the combined number of occurrences (i.e. the sum of the number of occurrences of Manual Actuation of HPI System after Reactor Trip Transient 8) and the number of occurrences of Rapid Depressurizations (Transient 9) can not exceed 29.

The AOTC (Allowable Operating Transient Cycle) monitoring program keeps track of the number of occurrences on each unit. The number of allowed transients has been reduced in the AOTC log books to limit the sum of these two transients to 29 for each unit.

Table 5-15. Reactor Coolant Flow Distribution with Less than Four Pumps Operating

	Oconee 1 (10^6 lb/hr)	Oconee 2 or 3 (10^6 lb/hr)
3 Pumps		
Flow in loop with two pumps	68.92	71.1
Flow in loop with one pump	29.95	29.5
Flow of pump in one pump loop	43.50	43.6
Idle pump reverse flow	13.55	14.1
2 Pumps - 2 Loops		
Pump flow each loop	44.49	44.5
Steam generator flow each loop	32.67	32.6
Reverse flow each idle pump	11.82	11.9
2 Pumps - 1 Loop		
Operating loop flow	71.22	73.6
Idle loop reverse flow	10.82	11.9
1 Pump		
Operating pump flow	45.06	45.0
Operating loop idle pump reverse flow	10.65	10.6
Idle loop reverse flow	5.23	5.5

Note:

For the configurations with both loops in operation the temperature in the cold legs will be the core inlet temperature (about 554°F). The hot leg fluid will be at about 604°F.

9

9 The reactor will not be operated at power in the 2 pump - single loop configuration.

Table 5-24. Evaluation of Reactor Vessel Pressurized Thermal Shock Toughness Properties at 48 EFPY - Oconee Unit 1

Material Description				Chemical Composition								
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Initial RT_{NDT}	Chemistry Factor	Fluence, n/cm^2 Inside Surface	ΔRT_{NDT},F at 48 EFPY	Margin	RT_{PTS},F at 48 EFPY	Screening Criteria
10 CFR 50.61 (Tables)												
Lower Nozzle Belt Forging	AHR 54	ZV-2861	A 508 Cl. 2	0.16	0.65	+3	119.3	1.11E+18	52.2	70.7	126.0	270
Intermediate Shell Plate	C2197-2	C2197-2	SA-302 Gr. BM*	0.15	0.50	+1	104.5	1.18E+19	109.3	63.6	174.0	270
Upper Shell Plate	C3265-1	C3265-1	SA-302 Gr. BM*	0.10	0.50	+1	65.0	1.31E+19	69.9	63.6	134.5	270
Upper Shell Plate	C3278-1	C3278-1	SA-302 Gr. BM*	0.12	0.60	+1	83.0	1.31E+19	89.2	63.6	153.9	270
Lower Shell Plate	C2800-1	C2800-1	SA-302 Gr. BM*	0.11	0.63	+1	74.5	1.31E+19	80.0	63.6	144.7	270
Lower Shell Plate	C2800-2	C2800-2	SA-302 Gr. BM*	0.11	0.63	+1	74.5	1.31E+19	80.0	63.6	144.7	270
LNB to IS Circ. Weld (100%)	SA-1135	61782	ASA/Linde 80	0.23	0.52	-5	157.4	1.11E+18	69.0	68.5	132.4	300
IS Longit. Weld (Both 100%)	SA-1073	1P0962	ASA/Linde 80	0.21	0.64	-5	170.6	9.24E+18	166.8	68.5	[230.3]	270
IS to US Circ. Weld (ID 61%)	SA-1229	71249	ASA/Linde 80	0.23	0.59	+10	167.6	1.19E+19	175.7	56.0	241.7	300
US Longit. Weld (Both 100%)	SA-1493	8T1762	ASA/Linde 80	0.19	0.57	-5	152.4	1.12E+19	157.3	68.5	220.8	270
US to LS Circ. Weld (100%)	SA-1585	72445	ASA/Linde 80	0.22	0.54	-5	158.0	1.27E+19	168.5	68.5	232.0	300
LS Longit. Weld (100%)	SA-1426	8T1762	ASA/Linde 80	0.19	0.57	-5	152.4	1.08E+19	155.8	68.5	219.3	270
LS Longit. Weld (100%)	SA-1430	8T1762	ASA/Linde 80	0.19	0.57	-5	152.4	1.08E+19	155.8	68.5	219.3	270
10 CFR 50.61 (Surveillance Data)												
LNB to IS Circ. Weld (100%)	SA-1135	61782	ASA/Linde 80	0.23	0.52	-5	141.1	1.11E+18	61.8	48.3	105.1	300
US to LS Circ. Weld (100%)	SA-1585	72445	ASA/Linde 80	0.22	0.54	-5	145.2	1.27E+19	155.8	48.3	199.1	300

* - SA-302 Grade B modified by ASME Code Case 1339

[]- Controlling value of RT_{PTS} reference temperature

Table 5-25. Evaluation of Reactor Vessel Pressurized Thermal Shock Toughness Properties at 48 EFPY - Oconee Unit 2

Material Description				Chemical Composition								
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	NI wt%	Initial RT_{NDT}	Chemistry Factor	Fluence, n/cm^2 Inside Surface	ΔRT_{NDT},F at 48 EFPY	Margin	RT_{PTS},F at 48 EFPY	Screening Criteria
10 CFR 50.61 (Tables)												
Lower Nozzle Belt Forging	AMX77	123T382	A 508 Cl. 2	0.13	0.76	+3	95.0	1.19E+19	99.6	70.7	173.3	270
Upper Shell Forging	AAW 163	3P2359	A 508 Cl. 2	0.04	0.75	+20	26.0	1.28E+19	27.8	27.8	75.6	270
Lower Shell Forging	AWG 164	4P1885	A 508 Cl. 2	0.02	0.80	+20	20.0	1.27E+19	21.3	21.3	62.7	270
LNB to US Circ. Weld (100%)	WF-154	406L44	ASA/Linde 80	0.27	0.59	-5	182.6	1.19E+19	191.5	68.5	255.0	300
US to LS Circ. Weld (100%)	WF-25	299L44	ASA/Linde 80	0.34	0.68	-5	220.6	1.23E+19	233.3	68.5	[296.8]	300
10 CFR 50.61 (Surveillance Data)												
None												

[]- Controlling value of RT_{PTS} reference temperature

Table 5-26. Evaluation of Reactor Vessel Pressurized Thermal Shock Toughness Properties at 48 EFPY - Oconee Unit 3

Material Description				Chemical Composition								
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Cu wt%	Ni wt%	Initial RT_{NDT}	Chemistry Factor	Fluence, n/cm^2 Inside Surface	ΔRT_{NDT}, F at 48 EFPY	Margin	RT_{PTS}, F at 48 EFPY	Screening Criteria
10 CFR 50.61 (Tables)												
Lower Nozzle Belt Forging	4680	4680	A 508 Cl. 2	0.13	0.91	+3	96.0	1.14E+19	99.5	70.7	173.2	270
Upper Shell Forging	AWS 192	522314	A 508 Cl. 2	0.01	0.73	+40	20.0	1.26E+19	21.3	21.3	82.6	270
Lower Shell Forging	ANK 191	522194	A 508 Cl. 2	0.02	0.76	+40	20.0	1.26E+19	21.3	21.3	82.6	270
LNB to US Circ. Weld (100%)	WF-200	821T44	ASA/Linde 80	0.24	0.63	-5	178.0	1.14E+19	184.6	68.5	248.1	300
US to LS Circ. Weld (ID 75%)	WF-67	72442	ASA/Linde 80	0.26	0.60	-5	180.0	1.22E+19	190.0	68.5	[253.5]	300
10 CFR 50.61 (Surveillance Data)												
Upper Shell Forging	AWS 192	522314	A 508 Cl. 2	0.01	0.73	+40	36.0	1.26E+19	38.3	34.0	75.5	270
Lower Shell Forging	ANK 191	522194	A 508 Cl. 2	0.02	0.76	+40	17.4	1.26E+19	18.5	17.0	112.3	270
LNB to US Circ. Weld (100%)	WF-200	821T44	ASA/Linde 80	0.24	0.63	-5	158.3	1.14E+19	159.5	48.3	202.8	300

[]- Controlling value of RT_{PTS} reference temperature

Table 5-27. Evaluation of Reactor Vessel Extended Life (48EFPY) Charpy V-Notch Upper-Shelf Energy - Oconee Unit 1

Material Description: Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Copper Composition w/o	Initial CvUSE, ft-lbs	48 EFPY Fluence T/4 Location n/cm2	Estimated 48 EFPY CvUSE at T/4	48 EFPY % Drop at T/4
Regulatory Guide 1.99, Revision 2, Position 1								
Lower Nozzle Belt Forging	AHR-54	ZV-2861	A508 Cl.2	0.16	109	9.18E+17	94	14
Intermediate Shell Plate	C2197-2	C2197-2	SA-302 Gr. B M	0.15	81	6.22E+18	63	22
Upper Shell Plate	C3265-1	C3265-1	SA-302 Gr. B M	0.10	108	7.06E+18	90	17
Upper Shell Plate	C3278-1	C3278-1	SA-302 Gr. B M	0.12	81	7.06E+18	66	19
Lower Shell Plate	C2800-1	C2800-1	SA-302 Gr. B M	0.11	81	6.78E+18	66	18
Lower Shell Plate	C2800-2	C2800-2	SA-302 Gr. B M	0.11	119	6.78E+18	98	18
LNB to IS Circ. Weld (100%)	SA-1135	61782	ASA/Linde 80	0.25	70	9.18E+17	55	22
IS Longit. Weld (Both 100%)	SA-1073	1P0962	ASA/Linde 80	021	70	4.91E+18	50	29
IS to US Circ. Weld (61%ID)	SA-1229	71249	ASA/Linde 80	0.26	70	6.22E+18	45	36
IS to US Circ. Weld (39%OD)	WF-25	299L44	ASA/Linde 80	0.35	70	-----	--	--
US Longit. Weld (Both 100%)	SA-1493	8T1762	ASA/Linde 80	0.20	70	5.66E+18	49	30
US to LS Circ. Weld (100%)	SA-1585	72445	ASA/Linde 80	0.21	70	6.78E+18	48	32
LS Longit. Weld (100%)	SA-1430	8T1762	ASA/Linde 80	0.20	70	5.71E+18	49	30
LS Longit. Weld (100%)	SA-1426	8T1762	ASA/Linde 80	0.20	70	5.71E+18	49	30
LS to Dutch. Circ. Weld (100%)	WF-9	72445	ASA/Linde 80	0.21	70	3.95E+16	64	9
Regulatory Guide 1.99, Revision 2, Position 2								
Upper Shell Plate	C3265-1	C3265-1	SA-302 Gr. B M	0.10	108	7.06E+18	91	16
LNB to IS Circ. Weld (100%)	SA-1135	61782	ASA/Linde 80	0.25	70	9.18E+17	53	24
IS to US Circ. Weld (61%ID)	SA-1229	71249	ASA/Linde 80	0.26	70	6.22E+18	47	33
IS to US Circ. Weld (39%OD)	WF-25	299L44	ASA/Linde 80	0.35	70	-----	--	--
US to LS Circ. Weld (100%)	SA-1585	72445	ASA/Linde 80	0.21	70	6.78E+18	48	31
LS to Dutch. Circ. Weld (100%)	WF-9	72445	ASA/Linde 80	0.21	70	3.95E+16	64	9

Table 5-28. Evaluation of Reactor Vessel Extended Life (48 EFPY) Charpy V-Notch Upper-Shelf Energy - Oconee Unit 2

Material Description				Copper Composition w/o	Initial CvUSE, ft-lbs	48 EFPY Fluence T/4 Location n/cm^2	Estimated 48 EFPY CvUSE at T/4	48 EFPY % Drop at T/4
Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type					
Regulatory Guide 1.99, Revision 2, Position 1								
Lower Nozzle Belt Forging	AMX-77	123T382	A508 Cl.2	0.06	109	6.83E+18	94	14
Upper Shell Forging	AAW-163	3P2359	A508 Cl.2	0.04	133	7.78E+18	117	12
Lower Shell Forging	AWG-164	4P1885	A508 Cl.2	0.02	138	7.45E+18	124	10
LNB to US Circ. Weld (100%)	WF-154	406L44	ASA/Linde 80	0.31	70	6.83E+18	42	40
US to LS Circ. Weld (100%)	WF-25	299L44	ASA/Linde 80	0.35	70	7.45E+18	41	41
LS to Dutch. Circ. Weld (100%)	WF-112	406L44	ASA/Linde 80	0.31	70	4.36E+16	62	12
Regulatory Guide 1.99, Revision 2, Position 2								
Upper Shell Forging	AAW-163	3P2359	A508 Cl.2	0.04	133	7.78E+18	116	13
NB to US Circ. Weld (100%)	WF-154	406L44	ASA/Linde 80	0.31	70	6.83E+18	45	36
US to LS Circ. Weld (100%)	WF-25	299L44	ASA/Linde 80	0.35	70	7.45E+18	44	37
LS to Dutch. Circ. Weld (100%)	WF-112	406L44	ASA/Linde 80	0.31	70	4.36E+16	62	11

Table 5-29. Evaluation of Reactor Vessel Extended Life (48 EFPY) Charpy V-Notch Upper-Shelf Energy - Oconee Unit 3

Material Description: Reactor Vessel Beltline Region Location	Matl. Ident.	Heat Number	Type	Copper Composition w/o	Initial CvUSE, ft-lbs	48 EFPY Fluence T/4 Location n/cm^2	Estimated 48 EFPY CvUSE at T/4	48 EFPY % Drop at T/4
Regulatory Guide 1.99, Revision 2, Position 1								
Lower Nozzle Belt Forging	4680	4680	A508 Cl.2	0.13	109	6.66E+18	87	20
Upper Shell Forging	AWS-192	522314	A508 Cl.2	0.01	112	7.56E+18	102	9
Lower Shell Forging	ANK-191	522194	A508 Cl.2	0.02	144	7.28E+18	130	10
LNB to US Circ. Weld (100%)	WF-200	821T44	ASA/Linde 80	0.24	70	6.66E+18	46	35
US to LS Circ. Weld (75%ID)	WF-67	72442	ASA/Linde 80	0.24	70	7.28E+18	46	35
US to LS Circ. Weld (25%OD)	WF-70	72105	ASA/Linde 80	0.35	70	-----	--	--
LS to Dutch. Circ. Weld (100%)	WF-169-1	8T1554	ASA/Linde 80	0.18	70	4.23E+16	64	9
Regulatory Guide 1.99, Revision 2, Position 2								
Upper Shell Forging	AWS-192	522314	A508 Cl.2	0.01	112	7.56E+18	95	15
Lower Shell Forging	ANK-191	522194	A508 Cl.2	0.02	144	7.28E+18	111	23
NB to US Circ. Weld (100%)	WF-200	821T44	ASA/Linde 80	0.24	70	6.66E+18	55	21
US to LS Circ. Weld (25%OD)	WF-70	72105	ASA/Linde 80	0.35	70	-----	--	--

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

CHAPTER 6. ENGINEERED SAFEGUARDS

2 6.1 ENGINEERED SAFEGUARDS

Engineered safeguards are those systems and components designed to function under accident conditions to prevent or minimize the severity of an accident or to mitigate the consequences of an accident. During
8 accident conditions when reactor coolant is lost, or in the event of secondary system pipe breaks, the engineered safeguards act to provide emergency cooling to assure structural integrity of the core, to maintain the integrity of the Reactor Building, and to collect and filter Potential Reactor Building penetration leakage. Separate and independent engineered safeguards are provided for each of the three reactor units at Oconee. Special precautions are taken to assure high quality in the system design and components.

The engineered safeguards include provisions for:

a. High pressure injection.

b. Low pressure injection.

c. Core flooding.

d. Two types of Reactor Building cooling.

e. The collection and control of Reactor Building penetration leakage.

f. Reactor Building isolation.

Figure 6-1 and Figure 6-4 depict the portion of the Engineered Safeguards System related to core and building protection (see (a) through (d) above). A general description of the engineered safeguards provisions is presented below and a more detailed description is presented in the latter portion of this section. Since each reactor unit has the same arrangement of Emergency Safeguard Systems, the performance of the systems is described on a unit basis.

2 6.1.1 GENERAL SYSTEMS DESCRIPTION

The High and Low Pressure Injection Systems and the Core Flooding Tanks are designed to form collectively an overall Emergency Core Cooling System (ECCS), which is designed to prevent melting or physical disarrangement of the core over the entire spectrum of Reactor Coolant System break sizes. Figure 6-1 shows the Emergency Core Cooling Systems for one reactor unit. The High Pressure Injection System is arranged so that three pumps are available for emergency use. The Low Pressure Injection System is arranged to assure that two pumps are normally available and a third pump is installed but normally valved off. The Core Flooding System for each unit is composed of two separate pressurized tanks containing borated water at Reactor Building ambient temperature. These tanks automatically discharge their contents into the reactor vessel at a preset Reactor Coolant System pressure
8 without reliance on any actuating signal, any electrical power or any external actuated component.

8 Reactor Building integrity is assured by two pressure reducing systems operating on different principles; the Reactor Building Spray System and the Reactor Building Emergency Cooling System. (Refer to Figure 6-2 and Figure 6-3). These systems have the redundancy required to meet the single failure
2 criterion. These systems operate to lower Reactor Building pressure over the spectrum of Reactor Coolant System break sizes and to reduce the driving force for leakage of radioactive materials from the
8 Reactor Building. They also serve to reduce Reactor Building pressure and temperature in the event of a
8 main steam line break.

The Reactor Building Penetration Room Ventilation System shown on Figure 6-4 collects and filters air leakage to control and minimize the release of radioactive material from Reactor Building penetrations following an accident. Two full capacity filtering paths are provided.

2 6.1.2 EQUIPMENT OPERABILITY

Operability of engineered safeguards equipment is assured in several ways. Much of the equipment in these systems serves a function during normal reactor operation. In those cases where equipment is used for emergency functions only such as the Reactor Building Spray System, systems have been designed to permit meaningful periodic tests. Operational reliability is achieved by using proven component design,
5 and by conducting tests where either the component or its application was considered unique. In-house quality control procedures are imposed on the components of the Engineered Safeguards Systems. These procedures include use of accepted codes and standards as well as supplementary test and inspection requirements to assure that all components will perform their intended function under the design conditions following a loss-of-coolant accident.

The purpose of this section is to describe the physical arrangement, design, and operation of the Engineered Safeguards Systems as related to their safety function.

Reactor Building isolation is described in Section 6.2, "Containment Systems." Other sections of the report contain information which is pertinent to the Engineered Safeguards Systems. Chapter 7, "Instrumentation and Control" describes the actuation instrumentation of these systems. Chapter 15, "Accident Analyses" describes the analysis of the Engineered Safeguards Systems' capability to provide adequate protection during accident conditions. Chapter 9, "Auxiliary Systems" discusses functions performed by these systems during normal operation and gives further design details and descriptive information concerning those systems.

2 6.1.3 LEAKAGE AND RADIATION CONSIDERATIONS

The use of normally operating equipment for engineered safeguards functions and location of some of this equipment outside the Reactor Building require that consideration be given to direct radiation levels after fission products have accumulated in these systems and leakage from these systems.

The shielding for components of the engineered safeguards is designed to meet the following objectives in the event of a maximum hypothetical accident:

a. To provide protection for personnel to perform all operations necessary for mitigation of the accident.

b. To provide sufficient accessibility in all areas around the station to permit safe continued operation of the unaffected nuclear units.

Summary of Post-Accident Recirculation: Following a loss-of-coolant accident, flow is initiated in the Low Pressure Injection System from the borated water storage tank to the reactor vessel. Flow is also initiated by the Reactor Building Spray Systems to building spray headers. When the borated water
9 storage tank level reaches the emergency low level alarm, recirculation from the Reactor Building emergency sump is initiated by the operator for both the reactor core cooling flow and the Reactor Building sprays. The operator will maintain the 3,000 gal/min design flow rates of the Low Pressure Injection pumps, but will throttle the Reactor Building Spray pumps from the 1,500 gal/min design flow rate to 1,000 gal/min in order to ensure adequate NPSH. The post-accident recirculation system includes all piping and equipment both internal and external to the Reactor Building as shown on Figure 6-1, up to the stop and test line valves leading to the borated water storage tank.

The NPSH available to the Low Pressure Injection and Reactor Building Spray pumps during the post-LOCA recirculation phase has been calculated based on:

a. "As Built" piping drawings.

b. Pipe and fitting losses calculated using the information in Crane Technical Paper No. 410.

5 c. Total indicated flow in a single string (i.e., consisting of one Low Pressure Injection Pump and
one Reactor Building Spray pump served by a single sump suction line) is 4,000 gal/min. This
consists of 3,000 gal/min to the Low Pressure Injection pump and 1,000 gal/min to the Reactor
5 Building Spray pump. Instrument uncertainties have been applied to these values to provide
5 conservatism in the NPSH analysis.

5 d. Sump water temperatures and Reactor Building pressures were determined from analysis of hot
5 leg break with conservative building cooling assumptions (LPSW temperature, RBCU capacity,
9 etc.). RBCU capacity was varied from 60 million Btu/hr to 160 million Btu/hr to produce
9 bounding pressures for all sump temperatures.

9 e. Reactor Building Spray pump shaft center line at elevation 760 ft. 1 in.

f. Low Pressure Injection pump shaft center line at elevation 761 ft. 1 in.

9 g. Water level in the Reactor Building sump is at elevation 781.15 ft. based on the following
9 assumptions: (height above Reactor Building basement level is 4.27 ft.)

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

2) Some water is maintained in the Reactor Building atmosphere as vapor. The quantity was
9 determined using the results of a FATHOMS Computer Run for a 14.1 ft^2 break with 2
9 RBCUs and one Reactor Building Spray pump operating.

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

5 h. Unit 3 NPSH available to each pump in the "B" string, LP-P3B and BS-P3B. Calculations show
5 this to be the worst case flowpath of all possible units and trains.

5

9 Results of the NPSH analysis are presented in Table 6-33. Credit is taken for 2.2 psi of reactor building
9 overpressure in the calculation of available NPSH for the RBS and LPI pumps from approximately 3,000
9 seconds to approximately 30,000 seconds post-accident.

5 Available NPSH has been determined to meet or exceed the required NPSH for worst case accident
5 conditions with conservative inputs as identified above. Curves of total dynamic head and NPSH versus
5 flow are shown in Figure 6-5 for the Reactor Building Spray Pumps and in Figure 6-17 for the Low
5 Pressure Injection Pumps. These curves are representative in nature and are provided for information
5 only. They are not intended to constitute design commitments or performance requirements for the
5 pumps. Refer to the Inservice Test Program for actual performance requirements for BS and LPI pumps.

Bases of Leakage Estimates: While the reactor auxiliary systems involved in the recirculation complex are closed to the Auxiliary Building atmosphere, leakage is possible through component flanges, seals, instrumentation, and valves.

The leakage sources considered are:

a. Valves

1) Disc leakage when valve is on recirculation system boundary.

2) Stem leakage.

3) Bonnet flange leakage.

b. Flanges

c. Pump shaft seals

While leakage rates have been assumed for these sources, maintenance and periodic testing of these systems will preclude all but a small percentage of the assumed amounts. With the exception of the boundary valve discs, all of the potential leakage paths may be examined during periodic tests or normal
3 operation. Boundary valves which have been identified to have leakage paths are tested periodically. All
3 other valve disc leakage is retained in the other closed systems and, therefore, will not be released to the Auxiliary Building.

While valve stem leakage has been assumed for all valves, the manual valves in the recirculation complex are backseating and do not rely on packing alone to prevent stem leakage.

Leakage Assumptions

	Source		Quantities
a.	Valves - Process		
	1.	Disc Leakage	10 cc/hr./in. of nominal disc diameter
	2.	Stem leakage	1 drop/min.
	3.	Bonnet flange	10 drops/min.
b.	Valves - Instrumentation		
	Bonnet flange and stem		1 drop/min.
c.	Flanges		10 drops/min.
d.	Pump seals		50 drops/min.

For the analysis, it was assumed that the water leaving the Reactor Building was at 252 °F. This assumption is conservative as this peak temperature would only exist for a short period during the post-accident condition. Water downstream of the coolers was assumed to be 115 °F. The Auxiliary Building was assumed to be at 70 °F and 30 percent relative humidity. Under these conditions, approximately 22 percent of the leakage upstream of the coolers and 4 percent of the leakage downstream of the coolers would flash into vapor. For the analysis, however, it was assumed that 50 percent of the leakage upstream of the coolers would become vapor because of additional heat transfer from the hot metal.

Design Basis Leakage: The design basis leakage quantities are tabulated in Table 6-2.

Leakage Analysis Conclusions: It was concluded from this analysis (in conjunction with the discussion and analysis in Section 15.15.4, "Effects of Engineered Safeguards Systems Leakage") that leakage from Engineered Safeguards Systems outside the Reactor Building does not pose a public safety problem.

2 6.1.4 QUALITY CONTROL STANDARDS

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

2 6.1.5 PIPING DESIGN CONDITIONS

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

2 6.1.6 ENGINEERED SAFEGUARDS MATERIALS

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

6.2 CONTAINMENT SYSTEMS

6.2.1 CONTAINMENT FUNCTIONAL DESIGN

5 ### 6.2.1.1 Containment Structure

5 #### 6.2.1.1.1 Design Bases

The Reactor Building completely encloses the Reactor Coolant System to minimize release of radioactive material to the environment should a serious failure of the Reactor Coolant System occur. The structure provides adequate biological shielding for both normal operation and accident situations. The Reactor Building is designed for an internal pressure of 59 psig. The leakage rate will not exceed 0.25 percent by volume in 24 hours under the conditions of the maximum hypothetical accident as described below.

5 The Reactor Building is designed for an external pressure 3.0 psi greater than the internal pressure. The
5 design external pressure of 3.0 psi corresponds to a margin of 0.5 psi above the differential pressure that
5 could be developed if the building is sealed with an internal temperature of 120°F with a barometric
5 pressure of 29.0 inches of Hg and the building is subsequently cooled to an internal temperature of 80°F
5 with a concurrent rise in barometric pressure to 31.0 inches of Hg. The weather conditions assumed here
5 are conservative since an evaluation of National Weather Service records for this area indicates that from
5 1918 to 1970 the lowest barometric pressure recorded is 29.05 inches of Hg and the highest of 30.85 inches
5 of Hg.

The principal design basis for the structure is that it be capable of withstanding the internal pressure
5 resulting from a loss-of-coolant accident or a secondary line rupture with no loss of integrity. In a
5 LOCA, the total energy contained in the water of the Reactor Coolant System is assumed to be released
5 into the Reactor Building through a break in the reactor coolant piping. In a secondary line break event
5 the energy contained in the water in the secondary coolant system, as well as energy transferred across the
5 steam generator tubes from the Reactor Coolant System is assumed to be released into the Reactor
5 Building through a break in the steam line piping. However, in the case of a secondary line break, the
5 release of energy essentially stops when the faulted steam generator empties and is no longer being
5 supplied with feedwater. In either case, subsequent pressure behavior is determined by the building
volume, engineered safeguards, and the combined influence of energy source and heat sinks.

Energy is available for release into the containment structure from the following sources:

	LOCA	Secondary Line Break
5	Reactor Coolant Stored Energy	Secondary Coolant Stored Energy
5	Reactor Stored Energy	Secondary System Stored Energy
5	Reactor Decay Heat	Reactor Coolant Stored Energy
5	Metal-Water Reactions	Reactor Stored Energy
5	Secondary Coolant Stored Energy	Reactor Decay Heat
5	Secondary System Stored Energy	

5 **6.2.1.1.2 Design Features**

5 Since the design of the Engineered Safeguards Systems and their operation is discussed more fully in
5 Section 6.3, "Emergency Core Cooling System," only their relation to the basis of Reactor Building design is discussed below. The Engineered Safeguards Systems are provided to limit the consequences of an accident. Their energy removal capabilities limit the internal pressure after the initial peak so that Reactor Building design limits are not exceeded and the potential for release of fission products is minimized.

5 Following a LOCA, the Emergency Core Cooling Systems inject borated water into the Reactor Coolant System to remove core decay heat and to minimize metal-water reactions and the associated release of heat and fission products. Flashed primary coolant, Reactor Coolant System sensible heat, and core decay heat transferred to Reactor Building are removed by two engineered safeguards systems: the Reactor Building Spray and/or the Reactor Building Cooling Systems.

5 Following a secondary line break at power, main feedwater and turbine- driven emergency feedwater flow
5 to the faulted steam generator are isolated by the Feedwater Isolation System. Motor-driven emergency
5 feedwater flow to the faulted steam generator is isolated manually by the operator.

The Reactor Building Spray System removes heat directly from the Reactor Building atmosphere by cold water quenching of the Reactor Building steam.

The air recirculation and cooling systems remove heat directly from the Reactor Building atmosphere to the Service Water System with recirculating fans and cooling coils.

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

Section 3.8, "Design of Structures" provides a detailed description of the Reactor Building design.

5 **6.2.1.1.3 Design Evaluation**

5 *6.2.1.1.3.1 LOCA Short Term Containment Pressure Response*

5 This section provides analyses of the short-term (3 minutes) pressure response of the containment to a
5 spectrum of postulated Reactor Coolant System pipe ruptures. The break size and location of each
5 postulated loss-of-coolant accident is given in Table 6-21. The pressure and temperature response of the
5 four break location sensitivity studies, Cases 1A through 1D, are given in the following figures:

7 Figure 6-28 Containment pressure for a 14.1 ft^2 break at the reactor vessel outlet (1A)
7 Figure 6-29 Containment pressure for a 14.1 ft^2 break at the steam generator inlet (1B)
7 Figure 6-30 Containment pressure for a 8.55 ft^2 break at the RCP discharge (1C)
7 Figure 6-31 Containment pressure for a 8.55 ft^2 break at the RCP suction (1D)
7 Figure 6-32 Containment temperature for a 14.1 ft^2 break at the reactor vessel outlet (1A)
7 Figure 6-33 Containment temperature for a 14.1 ft^2 break at the steam generator inlet (1B)
7 Figure 6-34 Containment temperature for a 8.55 ft^2 break at the RCP discharge (1C)
7 Figure 6-35 Containment temperature for a 8.55 ft^2 break at the RCP suction (1D)

5 ***Analysis Method and Computer Codes***

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

5 ***Mass and Energy Release Rate Data***

5 The mass and energy release rate data used for the LOCA analyses described in this section are given in
5 Section 6.2.1.3, "Mass and Energy Release Analyses for Postulated Loss-of-Coolant Accidents."

5 ***Initial Condition Assumption Conservatisms***

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

6 1. The initial pressure assumption is adjusted by 0.3 psi above the upper Technical Specification limit.
5 2. The initial temperature assumption is conservatively low for full power operation. This maximizes
5 the initial containment air mass, which maximizes the air partial pressure contribution to the pressure
5 peak.
5 3. The nominal containment free volume is reduced by 2%
5 4. A low initial relative humidity is used to maximize the initial air mass.

5 The initial conditions used are tabulated in Table 6-22.

5 ***Containment Heat Removal Systems***

5 No credit is taken in the LOCA peak pressure analysis for either the Reactor Building cooling units or the
5 Reactor Building Spray System. The peak pressure occurs within the first 20 seconds after the postulated
5 break, prior to the assumed actuation of either of these heat removal systems.

5 ***Emergency Core Cooling Systems***

5 The emergency core cooling systems are not explicitly modeled in FATHOMS for the LOCA peak
5 pressure analysis, but are considered in the mass and energy releases discussed in Section 6.2.1.3, "Mass
5 and Energy Release Analyses for Postulated Loss-of-Coolant Accidents."

5 ***Single Failure***

5 A component single failure generally has little impact on the peak pressure analysis. This is because peak
5 pressures usually occur before the engineered safeguards equipment has time to activate and become
5 effective.

5 ***Structural Heat Sinks***

5 The structural heat sinks within containment are divided into nine groups for the purposes of containment
5 pressure and temperature response modeling. These nine structures are tabulated in Table 6-23. The
5 concrete and steel portions of the building cylinder, the building dome, and the building base are
5 combined in three structures of two materials each.

5 *6.2.1.1.3.2 LOCA Long-Term Containment Temperature Response*

5 This section provides analyses of the long-term (~1 day) pressure response of the containment to a
5 spectrum of postulated Reactor Coolant System pipe ruptures. The long-term large break containment
5 analysis considers only a single break size and location: a double-ended guillotine break located at the A1
5 cold leg pump discharge. There is no need to analyze a spectrum of large break locations since a suitably
5 bounding site can be chosen by inspection. The qualitative bases for this position is explained in the
5 following paragraphs. Reference 1 also extensively analyzed the mass and energy releases from and
5 containment response to small break LOCAs. The conclusion of these analyses is that small break

5 LOCAs are not more limiting than large break LOCAs with respect to challenging the containment
5 equipment qualification acceptance criteria.

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

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

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

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

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

5 Figure 6-36 Containment pressure (600 gpm RBS flow rate)
5 Figure 6-37 Containment atmosphere temperature (600 gpm RBS flow rate)
5 Figure 6-38 Containment sump water temperature (600 gpm RBS flow rate)
5 Figure 6-39 Containment pressure (800 gpm RBS flow rate)
5 Figure 6-40 Containment atmosphere temperature (800 gpm RBS flow rate)
5 Figure 6-41 Containment sump water temperature (800 gpm RBS flow rate)

8 A case is also plotted (Case 3-170) which shows the containment response at an elevated Reactor Building
8 initial temperature of 170°F. These cases represent the combinations of minimum and maximum LPSW
8 temperature and RBCU heat removal rate for each assumed Reactor Building Spray flow rate. All results
8 plotted are for the minimum tabulated LPI cooler heat removal rate, 93 million Btu/hour. The
8 equipment qualification criteria plotted on these figures are met for all cases analyzed.

5 *__Analysis Method and Computer Codes__*

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

5 *__Mass and Energy Release Rate Data__*

5 The mass and energy release rate data used for the LOCA analyses described in this section are given in
5 Section 6.2.1.3, "Mass and Energy Release Analyses for Postulated Loss-of-Coolant Accidents."

5 *__Initial Condition Assumption Conservatisms__*

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

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

5 The initial conditions used are tabulated in Table 6-22

5 *__Containment Heat Removal Systems__*

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

6 • Low Pressure Service Water (LPSW) temperature of 75°F
5 • Containment air temperature of 286°F
5 • Containment air mass characteristic of 110°F and 100% relative humidity

5 The performance of the two operating coolers (refer to single failure discussion below) is parameterized in
5 terms of the percentage of this reference heat removal rate and the LPSW temperature. A reduction
5 below 80 million Btu/hour reflects degradation in RBCU performance. A colder LPSW temperature is
5 used to enhance performance due to a higher ΔT across the RBCU. For example, one of the cases
6 assumes two RBCUs with a combined nominal capacity of 28 million Btu/hr and an LPSW temperature
5 of 55°F.

5 Low Pressure Injection (LPI) cooler test data at a heat removal rate of 102 million Btu/hour is used to
5 determine the relationships between cooler degradation (number of plugged tubes and amount of tube
5 surface fouling) and thermal performance parameters such as fluid flow rates and temperatures. These
5 relationships are then modeled with specified heat removal rates of 93, 97, and 102 million Btu/hour to
5 determine an LPI cooler overall heat transfer coefficient as a function of LPI temperature. Since assumed
5 cooler degradation, fluid flow rates, and LPSW temperature are constant during a simulation, and since
5 LPI temperature changes during the accident, this change determines LPI cooler heat removal rate for a
5 particular case. For example, the case referred to above also assumes a 93 million Btu/hr heat removal
5 rate at the following conditions:

5 • LPI temperature of 250°F

- LPI flow rate of 3000 gpm
- LPSW temperature of 90°F
- LPSW flow rate of 5000 gpm

The actual heat removal rate calculated by FATHOMS for this case is reduced by accounting for a 2685 gpm LPI flow rate (minimum actual flow for 3000 gpm indicated flow rate), increased by accounting for an assumed colder LPSW temperature of 55°F, and then varied as calculated LPI temperature varies. The preceding discussion applies to the recirculation phase. No credit is taken for heat removal by the LPI coolers during the injection phase.

The single operating (refer to single failure discussion below) Reactor Building Spray (RBS) pump is conservatively modeled with respect to flow rate and temperature. The injection phase flow rate used is the nominal flow rate of 1500 gpm per pump less an allowance of 143 gpm for flow indication uncertainty. The recirculation phase flow rate used is the nominal flow rate of 1000 gpm per pump with BS-15 and BS-20 closed, less an allowance of 72 gpm for operating in the recirculation phase with these valves open, less an additional allowance of 128 gpm for flow indication uncertainty. Some cases were also run with an additional 200 gpm reduction in recirculation phase RBS flow rate. The injection phase temperature used is a conservatively high for the borated water storage tank, the source of RBS water during the injection phase. The recirculation phase RBS temperature is the sump temperature calculated by FATHOMS. No credit is taken for aligning the RBS pumps to take suction from the outlet of the LPI coolers.

Assumed values for containment heat removal equipment performance parameters are given in Table 6-26.

Emergency Core Cooling Systems

The single operating Low Pressure Injection (LPI) pump (refer to single failure discussion below) is assumed in the FATHOMS computer code to be supplying a conservatively low flow rate to the reactor vessel. The flow rate assumed is the nominal value less an allowance for flow indication uncertainty. The injection phase temperature used is a conservatively high for the borated water storage tank, the source of LPI water during the injection phase. The recirculation phase temperature is calculated by FATHOMS based on the heat removal from the LPI coolers.

The two operating High Pressure Injection (HPI) pumps (refer to single failure discussion below) are assumed to be supplying a conservatively low flow rate to the cold legs. The HPI water injected into the broken cold leg is added directly to the containment sump. The injection phase temperature used is a conservatively high for the borated water storage tank, the source of HPI water during the injection phase. No credit is taken for HPI flow during the recirculation phase.

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

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

Single Failure

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

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

- One HPI pump
- One LPI pump
- One RBS pump
- One RBCU

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

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

Structural Heat Sinks

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

6.2.1.1.3.3 Steam Line Break Containment Pressure and Temperature Response

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

Analysis Method and Computer Codes

The analysis method used in this section is described in Reference 1 The computer codes used in this section are RETRAN-02/MOD5 (Reference 9) for calculating the steam line break mass and energy releases and FATHOMS (Reference 3) for calculating the containment pressure and temperature response.

Mass and Energy Release Rate Data

The mass and energy release rate data used for the steam line break analyses described in this section are given in Section 6.2.1.4, "Mass and Energy Release Analyses for Postulated Secondary System Pipe Ruptures Inside Containment."

5 ***Initial Condition Assumption Conservatisms***

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

6 1. The initial pressure assumption is adjusted by 0.3 psi above the upper Technical Specification limit
6 for cases in which high initial pressure is conservative.
5 2. The initial temperature assumption is conservatively high for full power operation. It is known from
5 the LOCA analyses described in the previous section that a lower initial temperature maximizes the
5 containment peak pressures due to a higher initial air mass. However, a higher initial temperature
5 reduces the cooling capacity of the structural heat sinks, outweighing the impact of a higher initial air
5 mass.
5 3. The nominal containment free volume is reduced by 2%
5 4. A low initial relative humidity is used to maximize the initial air mass.

5 The initial conditions used are tabulated in Table 6-22.

5 ***Containment Heat Removal Systems***

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

5 ***Single Failure***

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

5 ***Structural Heat Sinks***

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

5 *6.2.1.1.3.4 Functional Capability of Normal Containment Ventilation Systems*

5 Normal containment ventilation is provided by four Reactor Building auxiliary cooling units (RBACUs)
5 and two of the three RBCUs. The function of these units during normal operation is described in Section
5 9.4.6, "Reactor Building Cooling System." Upper and lower limits on containment pressure during
5 normal operation are maintained by complying with the Technical Specifications.

5 *6.2.1.1.3.5 Post-Accident Monitoring of Containment Conditions*

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

5 Reactor Building pressure
5 Reactor Building air temperature
5 Reactor Building normal sump level
5 Reactor Building emergency sump level
5 Reactor Building wide range sump level

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

6.2.1.2 Containment Subcompartments

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

The reactor compartment has a volume of 5520 ft^3, one 6 ft^2 vent flowpath, and concrete shield plugs with a total flow area of 69 ft^2. Only the vent flowpath is assumed to be available for pressure relief. Although the maximum break area within the compartment has been determined to be 3.0 ft^2, hot leg breaks of 8.0, 5.0, and 3.0 ft^2 were analyzed, as well as the maximum cold leg break of 8.55 ft^2. The CRAFT mass and energy release rates are given in Figure 6-44 and Figure 6-45. The resulting pressure differential across the compartment walls are shown in Figure 6-46. The peak pressure of 160 psi, which occurs for the 8.0 ft^2 hot leg break, is only 78 percent of the design differential pressure of 205 psi.

The west steam generator compartment has a volume of 61,750 ft^3 and a total vent flow area of 1333 ft^2. The east compartment has a volume of 60,400 ft^3 and a flow area of 1222 ft^2. The discharge coefficients for each of the flowpaths and the effective discharge coefficient calculated to result in the correct choked flow are given in Table 6-28 and Figure 6-47. The maximum hot leg break of 14.1 ft^2 was analyzed using the CRAFT mass and energy release rates in Figure 6-44 and Figure 6-45. The resulting pressure differentials across the compartment walls are shown in Figure 6-48. The structural integrity of the compartments is sufficient to withstand 130 percent of the peak differential pressure of 15 psi.

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

6.2.1.3.1 Mass and Energy Release Data

The short-term LOCA peak pressure mass and energy releases are given in Table 6-29. The long-term LOCA mass and energy releases calculated by RELAP5/MOD2-B&W for the first 1800 seconds are given in Table 6-30. The mass and energy releases used by the FATHOMS computer code for periods beyond 1800 seconds are given in Table 6-31.

6.2.1.3.2 Energy Sources

The generated energy sources considered in the LOCA mass and energy release calculations are fission power, fission product and actinide decay energy, and metal-water reaction. The analyzed (actual) core power level is conservatively assumed to be 2% above the licensed (indicated) power. The assumed core axial power distribution is chosen to maximize the amount of steam exiting the break. The short-term LOCA mass and energy release calculation conservatively determines fission power by modeling moderator density, fuel, temperature, and boron feedbacks as described in Reference 1. The long-term LOCA mass and energy release calculation uses the fission power given in Figure 6-49. Fission product and actinide decay power is calculated as a function of time based on the methodology from Reference 7. For conservatism an upper bound of two standard deviations above the mean value is used. The modeling of metal-water reaction energy is discussed in Section 6.2.1.3.5, "Metal-Water Reaction."

The stored energy sources considered in the LOCA mass and energy release calculations are fluid stored energy in the initial primary and secondary system inventories, stored energy in the primary and secondary structural metal components, stored energy in the fuel rods, and the energy content of the fluid added to the primary and secondary systems during the accident. The specific conservatisms used in modeling these stored energy sources are:

1. The nominal volume of the Reactor Coolant System calculated based on cold dimensions is increased by 1% to account for the increase in volume due to thermal expansion to operating temperatures.
2. Initial pressurizer liquid mass is increased by assuming the initial level is at the high-high level alarm setpoint of 315 inches.
3. The assumed reactor vessel average temperature is the nominal at power value, 579°F, plus a 2°F uncertainty allowance. This maximizes the stored energy in the Reactor Coolant System.
4. Reactor Coolant System pressure is at the nominal at power value, 2155 psig, plus a 30 psi uncertainty allowance. This maximizes the saturation temperature and therefore the energy content of the pressurizer fluid.
5. The value of steam generator total fluid mass used is the maximum which can be obtained with the RELAP5/MOD2-B&W computer code at the chosen operating conditions.
6. The core flood tank (CFT) temperature is assumed to be a conservatively high value of 120°F. This minimizes available sensible heat capacity of the CFT liquid and therefore maximizes the break steaming rate.
7. The CFT initial pressure used is the upper Technical Specification limit plus a 30 psi instrument uncertainty. This maximizes the amount of noncondensible gas released to containment and therefore the containment pressure.
8. The CFT liquid volume is the lower Technical Specification limit less an instrument uncertainty of 38 ft^3. This minimizes available sensible heat capacity of the CFT liquid and maximizes the amount of noncondensible gas released to containment, both of which, as explained above, tend to increase containment pressure and temperature.
9. The RCS flow rate assumed is 366,080 gpm. This conservatively low flow rate maximizes the temperature difference across the reactor vessel, lowering the cold leg temperature and raising the hot leg temperature. Since slightly more of the RCS volume is at the hot leg temperature, this increases the initial RCS liquid stored energy.
10. The initial structural metal temperature is assumed to be constant across the entire volume of the metal and in equilibrium with the adjacent primary or secondary coolant. This is conservative since there will be some temperature gradient across the metal with the hottest temperature at the coolant surface.
11. Some of the fuel rods are assumed to be at a higher than average temperature as described in Reference 1. No credit is taken for offsetting lower temperatures in other fuel rods, thereby effectively increasing the fuel rod stored energy above expected average for the assumed power level and coolant temperature.
12. The turbine stop valves are assumed to close coincident with the opening of the break. This maximizes the rate of secondary side heatup and therefore the rate of secondary to primary heat transfer and energy release to containment.
13. No credit is taken for opening of the turbine bypass valves. This maximizes the amount of hot secondary side fluid remaining in the steam generators and able to transfer its energy via the primary side to containment.
14. Except for the offsite power lost sensitivity study case in the short-term LOCA analysis, the main feedwater flow boundary condition assumed allows up to the nominal hot full power flow to be delivered to the steam generators. This is conservative since it maximizes the amount of higher energy secondary side inventory available to transfer heat to the containment via the primary side.
15. The BWST liquid level is the lower Technical Specification limit less a level uncertainty of 20.2 inches. Minimizing the amount of BWST inventory minimizes the available sensible heat capacity of the BWST liquid and therefore maximizes the break steaming rate. This is because the BWST water is colder than the sump liquid inventory.

16. The BWST temperature is assumed to be a conservatively high value of 115°F, which minimizes available sensible heat capacity of the BWST liquid and therefore maximizes the break steaming rate.
17. Main feedwater temperature is at the nominal at power value, 453°F, plus a 7°F uncertainty allowance. This maximizes the potential energy release to containment. For further conservatism this assumed main feedwater temperature is maintained during the analysis although actual temperature would decrease as bleed steam was lost due to the break.
18. Main feedwater flow is maximized by controlling flow to the higher natural circulation setpoint even before the reactor coolant pumps are tripped. The nominal level setpoint is increased by a 10.5% operating range allowance for instrument uncertainty. This is conservative since it maximizes the amount of higher energy secondary side inventory available to transfer heat to the containment via the primary side.
19. Emergency feedwater temperature is at a conservatively high value of 120°F. This maximizes the potential energy release to containment. This is only relevant for the peak pressure large break analyses since the long-term large break analyses conservatively use hotter main feedwater.

6.2.1.3.3 Description of Analytical Models

The mass and energy releases during the blowdown and core reflood periods of a postulated LOCA are calculated by the RELAP5/MOD2-B&W computer code (Reference 2). The methodology for applying this code is given in Reference 1. After the first 30 minutes of releases quasi-steady state conditions are reached. Beyond this point the BFLOW and FATHOMS (Reference 3) codes are used to calculate mass and energy releases for the remainder of the accident as detailed in Reference 1.

6.2.1.3.4 Single Failure Analysis

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

6.2.1.3.5 Metal-Water Reaction

The energy released by steam/cladding metal-water reaction is considered in the short-term LOCA mass and energy release calculation. Reference 1 provides the methodology for modeling this energy source. The energy from the metal-water reaction is not considered in the long-term LOCA mass and energy release analysis. Reference 1 gives a quantitative discussion of the significance of this energy source and the justification for omitting it.

6.2.1.4 Mass and Energy Release Analyses for Postulated Secondary System Pipe Ruptures Inside Containment

The limiting secondary system pipe rupture from a containment response point of view is the steam line break. This is because the feedwater exiting a steam line break will have been heated to a higher temperature inside the steam generator via heat transfer across the steam generator tubes. In contrast, the feedwater exiting a feedwater line break will only be as hot as the outlet of the last feedwater heater upstream of the break location. Therefore, only steam line breaks are evaluated in this section. The model used is adjusted as described in Reference 1 to prevent any predicted liquid entrainment from decreasing the break enthalpy below the enthalpy of dry steam. Only the results of double-ended guillotine breaks of the maximum steam line pipe area are presented. This break size is 6.305 ft^2. For peak containment pressure this is the worst case because the rate of mass and energy release to containment is maximized. For peak containment temperature the response depends on both steam mass flow rate and on steam enthalpy. The steam enthalpy depends on hot leg temperature, steam flow and steam generator pressure. A double-ended steam line break results in higher steam enthalpies than smaller breaks. The larger break flow increases the overcooling of the primary side, which increases the hot leg temperature through an increase in reactor power. The higher feedwater flow enhances the heat transfer between the primary and the secondary sides, and the low steam generator pressure occurring during a

5 large break tends to increase the steam enthalpy. Sensitivity studies were done in which the break flow
5 was reduced by factors of 0.9, 0.8 and 0.5. In all three cases a conservatively high exit enthalpy was used.
5 The results show that the peak containment temperature response is most severe for the double-ended
5 break.

5 6.2.1.4.1 Mass and Energy Release Data

9 The mass and energy release data for the limiting secondary line break analyzed, a 6.3 ft^2 (34″ ID pipe)
9 double-ended guillotine break of a main steam line near the steam generator outlet, is presented in
5 Table 6-32.

5 6.2.1.4.2 Single Failure Analysis

5 The failure of an emergency feedwater control valve is chosen as the single failure for the steam line break
5 mass and energy release analysis. Other potential single failures were considered:

5 • Failure of a 4160V switchgear was also analyzed. The failure of one of the three available switchgear
5 results in the loss of one train of LPI and one train of HPI. This failure also results in the loss of one
5 Reactor Building Cooling Unit and one Reactor Building Spray train as discussed in Section
5 6.2.1.1.3.2, "LOCA Long-Term Containment Temperature Response." Note that this single failure
5 continues to be limiting for and conservatively assumed in the containment response analysis.
5 • There are no steam line isolation valves at Oconee.
5 • Although the feedwater isolation valves receive a feedwater isolation signal, this is used only to
5 provide a redundant means of accomplishing the feedwater isolation function. The steam line break
5 mass and energy release analyses credit the faster closing feedwater control valves to provide the
5 feedwater isolation function. Therefore the failure of a feedwater isolation valve has no effect on these
5 analyses.
5 • It is assumed that failure of a feedwater control valve to close on a feedwater isolation signal is
5 beyond the licensing basis.
5 • Credit is taken for the trip of the main feedwater pumps in the mass and energy release analyses for
5 steam line breaks with automatic feedwater isolation available.

5 6.2.1.4.3 Initial Conditions

5 The criteria presented in Reference 8 are used as the bases for the choices of initial conditions in the steam
5 line break mass and energy release analyses. The specific conservatisms are:

5 1. End of core life conditions are chosen to maximize the energy addition to the primary system. The
5 initial fuel temperature used is 1072°F.
5 2. 102% power is assumed, corresponding to the licensed core thermal power plus a 2% measurement
5 uncertainty allowance. This maximizes the available generated energy and stored core energy for
5 release to the secondary side.
5 3. The assumed reactor vessel average temperature is the nominal at power value, 579°F, plus a 2°F
5 uncertainty allowance. This maximizes the stored energy in the Reactor Coolant System.
5 4. The assumed Reactor Coolant System pressure is the nominal value, 2155 psig, plus a 30 psi
5 uncertainty allowance. This maximizes time to reactor trip and thus the energy transferred to the
5 secondary system.
5 5. Steam line pressure is left at the nominal value rather than being increased to delay the generation of a
5 feedwater isolation signal. This is required so that RETRAN-02 model calculated steam generator
5 tube heat transfer areas correspond to the physical tube areas.
5 6. A conservatively large steam generator initial fluid mass is assumed to maximize the inventory
5 available for release through the break.

5 7. End of core life fuel and moderator temperature feedback is assumed to maximize positive reactivity
5 insertion from the cooldown. A low effective delayed neutron fraction and prompt neutron lifetime
5 are also chosen to maximize the positive reactivity added by the cooldown.
5 8. The control rods are assumed to be positioned such that a reactor trip inserts only the amount of
5 negative reactivity which produces and maintains the minimum shutdown margin required by the
5 Technical Specifications.
5 9. The core boron concentration is assumed to be zero, which is consistent with end of core life
5 conditions.

5 #### 6.2.1.4.4 Description of Blowdown Model

5 The RETRAN-02/MOD005 computer code, described in Reference 9 is used to generate the mass and
5 energy releases for steam line breaks inside containment. The models used for this calculation are
5 generally described in Reference 10 with modifications for the containment mass and energy release
5 calculations as described in Reference 1. The calculational methods for applying this code and model to
5 calculate mass and energy releases for steam line breaks are also described in Reference 1. Reference 1
5 also discusses and justifies the conservatisms in this calculational method. Reference 9 presents the heat
5 transfer correlations used to calculate the heat transferred from the steam generator tubes and shell and
5 justifies their application. No liquid entrainment is assumed in the break flow. The analysis methodology
5 used credits the Feedwater Isolation System in conjunction with operator action to manually isolate
5 motor-driven emergency feedwater at ten minutes after a postulated steam line break occurs.

5 ### 6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability
5 ### Studies on Emergency Core Cooling System

5 The pressure response of the containment to a LOCA is analyzed to determine the backpressure in the
5 containment as a boundary condition for the reflood analysis and the calculation of the peak clad
5 temperature. The assumptions used in these analyses result in a conservatively calculated minimum
5 containment pressure response. This method has been shown to result in the maximum peak clad
5 temperature.

7 The analysis of the minimum containment pressure response for the reflood analysis is performed using
7 the methodology detailed in Reference 11 with Oconee-specific inputs. The mass and energy release to
5 the Reactor Building and the resulting pressure response for the worst case LOCA, 8.55 ft^2 cold leg break
7 at the pump discharge, is shown in Figure 6-50 (mass releases), Figure 6-51 (energy releases), and
7 Figure 6-52 (pressure response).

9 ### 6.2.1.6 Coating Materials

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

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

9 The elements of the Oconee Coatings Program are documented in a Nuclear Generation Department
9 Directive. The Oconee Coatings Program includes periodic condition assessments of Service Level I

9 coatings used inside containment. As localized areas of degraded coatings are identified, those areas are
9 evaluated for repair or replacement, as necessary.

6.2.2 CONTAINMENT HEAT REMOVAL SYSTEMS

6.2.2.1 Design Bases

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

6.2.2.2 System Design

6.2.2.2.1 Piping and Instrumentation Diagrams

A schematic diagram of the Reactor Building Spray (BS) System is shown in Figure 6-2. The system serves no function during normal operation.

Removal of post-accident energy is accomplished by directing borated water spray into the Reactor
Building atmosphere. The system consists of two pumps, two Reactor Building Spray headers, isolation
valves, and the necessary piping, instrumentation and controls. The pumps and remotely operated valves
8 for each unit can be operated from the control room.

8 A high Reactor Building pressure signal of less than or equal to 15 psig (typical value is 10 psig) from the
Engineered Safeguards System (Channels 7 and 8) initiates operation of the BS system. The two pumps
start, taking suction initially from the borated water storage tank through the intertie with the Low
Pressure Injection System, and initiate building spray through the spray headers and nozzles. After the
9 water in the borated water storage tank reaches an emergency low level, the spray pump suction is
8 transferred to the Reactor Building sump manually when the operator places the Low Pressure Injection
System in the recirculation mode. The Reactor Building emergency sump water is cooled by the Low
Pressure Injection System as described in Section 6.3, "Emergency Core Cooling System."

This system shares borated water storage tank capacity with the Low Pressure Injection System and the High Pressure Injection System.

Figure 6-3 illustrates the Reactor Building Cooling Units (RBCU's). Each cooling unit consists of a fan,
cooling coils, and the required distribution duct work. The Reactor Building atmosphere is circulated past
cooling coils by fans and returned to the building. Cooling water for the cooling units is supplied by the
6 Low Pressure Service Water System. During normal operation these units serve to cool the Reactor
Building atmosphere. The Engineered Safeguards System (Channels 5 and 6) is actuated when the
Reactor Building pressure reaches 3 psig (4 psig Technical Specification Limit). Upon ES actuation, the
6 fan motors associated with the RBCU's operating at high speed change to low speed, and any idle unit(s)
6 are energized at low speed.

8 Performance of the cooling system is monitored by flow and temperature instrumentation in the service
8 water supply and return lines for each cooler; by relative humidity and temperature transmitters in the
8 RBCU ductwork; and by the Reactor Building temperature and pressure instrumentation.

6.2.2.2.2 Codes and Standards

8 BS System equipment is designed to the applicable codes and standards given in Table 6-3.

4 The cooling coils for the RBCU's are constructed in accordance with ASME Section III, Class 3
4 guidelines. The Low Pressure Service Water System is designed to USAS B31.1.

6.2.2.2.3 Materials Compatibility

All materials in the BS System are compatible with the reactor coolant. The major components of the system are constructed of stainless steel. Minor parts such as pump seals utilize other corrosion resistant materials.

The materials for the RBCU's have been selected to be compatible with the use of untreated service water
4 to minimize corrosion in accordance with ASME guidelines.

6.2.2.2.4 Component Design

<u>BS Pumps</u>

The Reactor Building Spray pumps are similar to those used in refinery service. These pumps are liquid-penetrant tested by methods described in the ASME Boiler and Pressure Vessel Code Section VIII and are hydrotested and qualified to be able to withstand pressures greater than 1.5 times the design pressure. The pumps are designed so that periodic testing may be performed to assure operability at all times.

8 Curves of total dynamic head and NPSH versus flow are shown in Figure 6-5. These curves are included
8 as representative information only and are not intended as performance commitments. For design
8 purposes, actual performance data should be obtained from manufacturer's certified performance test
8 curves.

<u>BS Valves</u>

The remotely operated valves of the Reactor Building Spray System are designed and manufactured to the same requirements as the valves in the Emergency Core Cooling Systems. Refer to Section 6.3, "Emergency Core Cooling System."

<u>RB Spray Headers and Nozzles</u>

120 full core spray nozzles are arranged on each of the two Reactor Building Spray headers. The spray nozzles are spaced in the headers to give uniform spray coverage of the Reactor Building volume above the operating floor.

<u>BS Piping</u>

Except for the sections of lines requiring flanged connections for maintenance, the entire system is welded construction. Table 6-4 lists the design conditions for this system.

<u>RBCU Coolers</u>

The cooling surface of the cooling units has been designed for and satisfactorily tested under simulated
4 post-accident conditions. A conservative design has resulted in a heat exchanger which has a design heat
transfer capability in excess of the expected heat transfer requirements.

The Reactor Building cooler is located in the discharge ducting for the fan. The air-steam mixture flows
across the tube bank, resulting in condensation of a portion of the steam and removal of sensible heat
4 from the air. Figure 6-6 shows the design heat transfer capability of each unit at various Reactor Building

temperature conditions. Figure 6-6 is based on a Low Pressure Service Water temperature of 75°F. Actually, the cooling water is drawn from a point near the bottom of the lake and the anticipated service water temperature would be in the range of 45 to 85°F. Therefore, the curve shown in Figure 6-6 is
7 conservative for most of the year. Figure 6-7 shows how the Reactor Building cooling rate varies with the air-steam mixture flow rate. It can be seen that even if the mixture flow rate decreases by 40 percent, the cooling capability decreases by less than 7 percent.

2

RBCU Fans

Circulation of the Reactor Building atmosphere under accident conditions is by the same fans used for
6 normal ventilation. Upon actuation by an engineered safeguards signal, the fan motor(s) switch from full
9 speed to half speed and the idle unit(s) is started at half speed (Section 6.2.2.2, "System Design"). The
9 fans are tested each refueling outage to verify they can pass the required air flow rate across the coils. The control circuitry of the RBCU fans has been modified to remain in the ES state after reset of the ES channels. This modification ensures that deliberate separate action is required to shutdown the RBCU's. This modification is made pursuant to the requirements of IE Bulletin 80-06.

6.2.2.2.5 Reliability Considerations

A failure analysis has been made on all active components of the BS System to show that the failure of any single active component will not prevent fulfilling the design function. This analysis is shown in Table 6-5.

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

The major equipment of the Reactor Building Cooling Units is arranged in three independent strings with three duplicate service water supply lines. In the unlikely event of a failure in one of the three cooling
6 units, half of the Reactor Building Spray System capacity combined with the remaining two cooling units, will provide cooling capacity in excess of that required. Fan-motor operation under design LOCA condition has been demonstrated by prototype test.

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

6.2.2.2.6 Missile Protection

BS System protection against missile damage is provided by direct shielding or by physical separation of duplicate equipment. The spray headers are located outside and above the primary and secondary concrete shield.

The RBCU's and associated piping are located outside the secondary concrete shielding. The ductwork required to operate during an accident is located outside of the secondary shielding.

6.2.2.2.7 System Actuation

The Reactor Building Spray System will be activated when Reactor Building pressure reaches a setpoint
6 of less than or equal to 15 psig (typical value is 10 psig). The system components may also be actuated by operator action from the control room for performance testing.

In the event of a loss-of-coolant accident, the RBCU's are initiated at a Reactor Building pressure of 3 psig (4 psig Technical Specification Limit). The cooling units are placed in operation as follows:

a. Valve LPSW-565 is closed, stopping water flow in the Reactor Building Auxiliary Cooling Units.
8 Valve LPSW-566 is opened, if not already full open, establishing flow to RBCU "B".

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

6 c. The idle cooling unit fan(s) is started; and the speed of all fans is switched to half speed to change the horsepower capacity, required by the denser building atmosphere.

d. The links that hold the fusible dropout plates in the duct work melt and drop off, assuring that a positive path for recirculation of the Reactor Building atmosphere is available.

e. Depending upon the severity of the accident, the blowout plates at the bottom of the downcomer are designed to be forced out by any shock wave, allowing attenuation of the wave before it reached the cooling coils. Analysis has shown this to be a highly unlikely scenario due to duct
9 deformation, and therefore the blowout plates are not needed for this function. In addition, the
9 blowout plates are not considered functional.

6.2.2.2.8 Environmental Considerations

8 None of the electrically operated active components of the Reactor Building Spray System are located within the Reactor Building, so none are required to operate in the steam-air environment produced by the accident.

Figure 6-8 depicts the Reactor Building post-accident steam-air conditions. The RBCU fans and motors are designed for operation in the post-accident conditions. Cooling capability of the coolers has been satisfactorily tested in this environment.

6.2.2.2.9 Quality Control

Quality standards for the Reactor Building Spray System components are given in Table 6-3.

6.2.2.3 Design Evaluation

6 The Reactor Building Spray System, acting with the Reactor Building Cooling System, is capable of
8 keeping the containment pressure and temperature within environmental qualification (EQ) limits after a
6 loss-of-coolant or steam line break accident. Assuming a single failure, the post-accident Reactor Building
6 cooling load is provided by two cooling units and the Reactor Building Spray System at one-half capacity.
8 The Reactor Building Spray System and Reactor Building Cooling Systems are designed for long term
6 post-accident operation.

8 Both the Reactor Building Spray System and the Reactor Building Cooling System, with either at full
8 capacity, are individually capable of maintaining the containment pressure below the design limit
8 following a LOCA or MSLB. This capability satisfies the requirements of the design criteria given in
8 Section 3.1.52, "Criterion 52 - Containment Heat Removal Systems (Category A)."

The Reactor Building Spray System will deliver 3,000 gal/min through the spray nozzles within 92
9 seconds after the Reactor Building pressure reaches the Reactor Building Spray System actuation setpoint
(typical value is 10 psig), although some flow would be expected at least 18 seconds earlier.

6 The Reactor Building Cooling System provides the design heat removal capacity with two of three coolers operating by continuously circulating the steam-air mixture past the cooling tubes to transfer heat from the containment atmosphere to the low pressure service water.

Building pressure is limited below the design pressure. The design heat load at these conditions is 240 x 10^6 Btu/hr. The design inlet cooling water is 75°F, although the expected cooling water range is 45 -
7 85°F. The design heat removal capacity for these units is shown in Figure 6-6. The safety analyses given
6 in Section 6.2.1, "Containment Functional Design" demonstrate system effectiveness.

6.2.2.4 Tests and Inspection

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

Reactor Building Spray Pumps

The delivery capability of one pump at a time can be tested by opening the valve in the line from the borated water storage tank, opening the corresponding valve in the test line, and starting the corresponding pump. Pump discharge pressure and flow indication demonstrate performance.

Borated Water Storage Tank Outlet Valves

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

Reactor Building Spray Injection Valves

8 With the pumps shut down and the pump suction valves closed, these valves can each be opened and
8 closed by operator action. These valves are required to be electrically operable for accident mitigation.

Reactor Building Spray Nozzles

With the Reactor Building Spray inlet valves closed, low pressure air or fog can be blown through the test connections. Visual observation will indicate flow paths are open.

During these tests, the equipment can be visually inspected for leaks. Valves and pumps will be operated and inspected following maintenance on the system to assure proper operation.

The RBCU equipment, piping, valves, and instrumentation are arranged so that they can be visually inspected. The cooling units and associated piping are located outside the secondary concrete shield. Personnel can enter the Reactor Building during power operations to inspect and maintain this equipment. The service water piping and valves outside the Reactor Building are inspectable at all times.
8 Operational tests and inspections are performed prior to initial startup after each refueling outage.

The cooling units will be tested periodically as follows:

8 a. The fans will be started and inspected for proper operation.

b. The return line service water valves will be opened, and the lines checked for flow.

Additional discussion of tests of the containment heat removal systems is provided in Section 3.8, "Design of Structures."

6.2.3 CONTAINMENT ISOLATION SYSTEM

6.2.3.1 Design Bases

3 The general design basis governing isolation requirements is:

> Leakage through all fluid penetrations not serving accident-consequence limiting systems is to be minimized by a double barrier so that no single, credible failure or malfunction of an active component can result in loss-of-isolation or intolerable leakage. The installed double barriers take the form of closed piping systems, both inside and outside the Reactor Building, and various types of isolation valves.

3 Reactor Building Essential and Non-essential Isolation occurs on an Engineered Safeguards signal of 3
3 psig (4 psig Technical Specification value) in the Reactor Building. Reactor Building Non- essential
8 Isolation occurs on an Engineered Safeguards signal of 1600 psig(1590 psig Technical Specification value)
3 in the Reactor Coolant System. For details on Reactor Building Essential and Non- essential Isolation,
3 refer to Section 7.3, "Engineered Safeguards Protective System" and Table 7-2 and Table 7-3. Valves
3 which isolate penetrations that are normally directly open to the Reactor Building (the Reactor Building
3 purge valves and sump drain valves) will also be closed on a high radiation signal. (The radiation monitor
3 signal is not an Engineered Safeguards signal). Although normally open to the Reactor Building, the
3 Reactor Building Gaseous Radiation Monitor penetrations are not closed on a high radiation signal; they
3 remain open (except during ES isolation) to provide continuous monitoring.

The isolation system closes all fluid penetrations, not required for operation of the engineered safeguards systems, to prevent the leakage of radioactive materials to the environment.

9 All remotely operated Reactor Building isolation valves that are active to close for containment isolation
9 have position limit indicators in the control room. All solenoid valves used in actuating pneumatic RB isolation valves are environmentally qualified to the requirements of the IE Bulletin 79-01B.

6.2.3.2 System Design

The fluid penetrations which require isolation after an accident may be classed as follows:

Type A. Each line connecting directly to the Reactor Coolant System has two Reactor Building isolation valves. One valve is inside and the other is outside the Reactor Building. These
3 valves may be either a check valve and an automatic remotely operated valve, two
3 automatic remotely operated valves, or two check valves, depending upon the direction of
3 normal flow (referred to as Type I penetrations in the DBD).

Type B. Each line connecting directly to the Reactor Building atmosphere has two isolation valves. At least one valve is outside and the other may be inside or outside the Reactor
3 Building. These valves may be either a check valve and an automatic remotely operated
5 valve, or one check valve and one, normally closed manual valve, or two automatic
3 remotely operated valves, or two check valves, depending upon the direction of normal
5 flow (referred to as Type II penetrations in the DBD). For piping not part of the
5 process flow, double isolation will be used. One or more of the isolations will be a
5 normally closed manual valve located on the vent, drain, or test connection. The other
5 isolation valve may be located on the process piping.

3 Type C. Each line not directly connected to the Reactor Coolant System or not open to the
3 Reactor Building atmosphere has at least one valve, either a check valve or an automatic
3 remotely operated valve. This valve is located outside the Reactor Building. A seismic
3 closed loop forms the inside barrier for most Type C penetrations. Since the Component
3 Cooling System has a non-seismic closed loop, penetrations for this system have an
3 additional automatic remotely operated valve or check valve located inside the Reactor
3 Building (referred to as Type III penetrations in the DBD).

3 Type D. Each line connected to either the Reactor Building atmosphere or the Reactor Coolant
3 System, but which is not normally open during reactor operation, has two isolation
3 valves. They may be manual valve(s) with provisions for locking in a closed position,
3 check valve(s), and/or remotely operated valve(s), depending upon the direction of the
3 normal flow (referred to as Type IV penetrations in the DBD).

There are additional subdivisions in each of these major groups. The individual system flow diagrams show the manner in which each Reactor Building isolation valve arrangement fits into its respective
3 system. For convenience, each different valve arrangement is shown in Table 6-7 and Figure 6-9 of this
3 section. The symbols on Figure 6-9 are described at the end of Table 6-7. This table lists the mode of actuation, the type of valve, its normal position and its position under Reactor Building isolation conditions. The specific system penetrations to which each of the arrangements is applied is also
3 presented. It may be noted that only electric motor-operated, manual normally closed, or check valves
3 are used inside the Reactor Building. Each valve will be tested periodically during normal operation or during shutdown conditions to assure its operability when needed.

3 Fluid penetrations which do not require isolation after an accident are also classified as Type A through
3 D, however the redundant containment isolation provisions described above are not applicable. Such
3 penetrations are identified on Figure 6-9 as "PA" for Post Accident.

There is sufficient redundancy in the instrumentation circuits of the engineered safeguards protective system to minimize the possibility of inadvertent tripping of the isolation system. Further discussion of this redundancy and the instrumentation signals which trip the isolation system is presented in Chapter 7, "Instrumentation and Control."

3 6.2.3.3 Periodic Operability Tests

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

6.2.4 CONTAINMENT LEAKAGE TESTING

6.2.4.1 Periodic Leakage Testing

Tests and surveillance are performed periodically to verify that leakage from the containment is maintained within acceptable limits. These tests include:

Integrated Leak Rate Tests

3 Local Leak Detection

2 These tests are discussed in detail in Section 3.8.1.7.4, "Leakage Monitoring."

6.2.4.2 Continuous Leakage Monitoring

No continuous Reactor Building leakage monitoring system is provided.

9

The comprehensive program for preoperational testing, inspection, and postoperational surveillance is
9 described in detail in Section 3.8, "Design of Structures."

9

6.2.5 REFERENCES

5 1. DPC-NE-3003-P-A, "Duke Power Company Oconee Nuclear Station Mass and Energy Release and
5 Containment Response Methodology", Revision 0, March, 1995.

6 2. BAW-10164-P-A, "RELAP5/MOD2-B&W--An Advanced Computer Program for Light Water
5 Reactor LOCA and Non-LOCA Transient Analysis", Revision 1, April, 1990.

5 3. "CAP--Containment Analysis Package (FATHOMS 2.4)", Numerical Applications, Inc., October 10,
5 1989.

5 4. BAW-10030, "CRAFT - Description of Model for Equilbrium LOCA Analysis Program", Revision 0,
5 October 1971.

5 5. BAW-10034, "Multinode Analysis of B&W's 2568 MWt Nuclear Plants During a LOCA", Babcock
5 & Wilcox, Revision 3, May 1972.

5 6. Letter from A. C. Thies (Duke) to A. Schwencer (NRC) dated June 28, 1973.

5 7. ANSI/ANS-5.1-1979, "Decay Heat Power in Light Water Reactors", American Nuclear Society.

5 8. ANSI/ANS-56.4-1983, "Pressure and Temperature Transient Analysis for Light Water Reactor
5 Containments", American Nuclear Society, December 1983.

5 9. NP-1850-CCM-A, "RETRAN-02--A Program for Transient Analysis of Complex Fluid Flow
5 Systems", Electric Power Research Institute, Revision 4, November 1988.

5 10. DPC-NE-3000-P-A, "Duke Power Company Nuclear Station Thermal-Hydraulic Transient Analysis
5 Methodology", Revision 1, December 1995.

7 11. BAW-10192P, "BWNT Loss of Coolant Accident Evaluation Model for Once-Through Steam
7 Generator Plants", February 1994.

7 12. Deleted Per 1997 Update

7 13. Deleted Per 1997 Update

9 14. OM 235-0517-001, "Oconee Nuclear Station, Reactor Building Cooling Coil Performance Analysis";
9 rev.1.

9 15. OSC-5683, "Verification of Aerofin's NUCK Program, RBCU Coil Heat Removal Under
9 Post-LOCA Conditions", rev.0.

9 16. M. S. Tuckman (Duke) letter dated November 11, 1998 to Document Control Desk (NRC),
9 "Response to Generic Letter 98-04: Potential Degradation of the Emergency Core Cooling System
9 and the Containment Spray System After a Loss-of-Coolant Accident Because of Construction and
9 Protective Coating Deficiencies and Foreign Material in Containment," Oconee Nuclear Station,
9 Units 1, 2, and 3, Docket Nos. 50-269, -270, and -287.

6.3 EMERGENCY CORE COOLING SYSTEM

0 **Note**

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

6.3.1 DESIGN BASES

The Emergency Core Cooling System (ECCS) is designed to cool the reactor core and provide shutdown capability following initiation of the following accident conditions:

1. Loss-of-coolant accident (LOCA) including a pipe break or a spurious relief or safety valve opening in the RCS which would result in a discharge larger than that which could be made up by the normal make-up system.
2. Rupture of a control rod drive mechanism causing a rod cluster control assembly ejection accident.
3. Steam or feedwater system break accident including a pipe break or a spurious relief or safety valve opening in the secondary steam system which would result in an uncontrolled steam release or a loss of feedwater.
4. A steam generator tube rupture.

The primary function of the ECCS is to remove the stored and fission product decay heat from the reactor core during accident conditions.

The ECCS provides shutdown capability for the accident above by means of boron injection. It is designed to tolerate a single active failure (short term) or single active or passive failure (long term). It can meet its minimum required performance level with onsite or offsite electrical power and under simultaneous Safe Shutdown Earthquake loading.

The Emergency Core Cooling System for one reactor unit is shown in Figure 6-1. The overall Emergency Core Cooling System is comprised of the following independent subsystems:

a. High Pressure Injection System

b. Low Pressure Injection System

c. Core Flooding System

The principal design basis for the Emergency Core Cooling System as described in the proposed AEC General Design Criterion 44 has been met. Protection for the entire spectrum of break sizes is provided. Two separate and independent flow paths containing redundant active components are provided in the
8 HPI and LPI portions of the ECCS. Redundancy in active components assures performing the required functions should a single failure occur in any of the active components. Separate power sources are provided to the redundant active component. Separate instrument channels are used to actuate the systems. The adequacy of the installed ECCS to prevent fuel and clad damage is discussed in Chapter 15, "Accident Analyses."

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

The ECCS is designed to operate in the following modes:

a. Injection of borated water from the borated water storage tank by the High Pressure Injection System.

b. Rapid injection of borated water by the Core Flooding System.

c. Injection of borated water from the borated water storage tank by the Low Pressure Injection System.

d. Long term core cooling by recirculation of injection water from the Reactor Building sump to the core by the Low Pressure Injection pumps.

e. Gravity drain from the reactor outlet piping to the Reactor Building emergency sump by the Low Pressure Injection System.

Although the high and low pressure emergency injection systems operate to provide full protection across the entire spectrum of break sizes, each system may operate individually and each is initiated independently. High pressure injection prevents uncovering of the core for small coolant piping leaks where high system pressure is maintained, and to delay uncovering of the core for intermediate-sized leaks. The core flooding and low pressure injection systems are designed to re-cover the core at intermediate-to-low pressures, and to assure adequate core cooling for break sizes ranging from intermediate breaks to the double-ended rupture of the largest pipe. The Low Pressure Injection System is also designed to permit boron concentration control and long-term core cooling in the recirculation mode after a LOCA. The injection and core flooding functions are subdivided so that there are two separate and independent strings, each including one high pressure pump, one low pressure pump, and
1 one core flooding tank.

Much of the equipment in these systems serves a function during normal reactor operation. In those
9 cases where equipment is used for emergency functions only, such as the Core Flood System, systems have been designed to permit meaningful periodic tests. Operational reliability is achieved by using proven component designs, and by conducting tests where either the component or its application was considered unique. Quality control procedures are imposed on the components of the engineered safeguards systems. These procedures include use of accepted codes and standards as well as supplementary test and inspection requirements to assure that all components will perform their intended function under the design conditions following a LOCA.

6.3.2 SYSTEM DESIGN

6.3.2.1 Schematic Piping and Instrumentation Diagrams

The schematic diagrams for the Emergency Core Cooling System are shown in Figure 6-1. Instrumentation is shown schematically in Chapter 7, "Instrumentation and Control."

6.3.2.2 ECCS Operation

6.3.2.2.1 High Pressure Injection System

During normal reactor operation, the High Pressure Injection System recirculates reactor coolant for purification and for supply of seal water to the reactor coolant circulating pumps. This normal operation mode is described in Chapter 9, "Auxiliary Systems." The High Pressure Injection System is initiated at:
8 (a) a low Reactor Coolant System pressure of 1,600 psig (1590 psig Technical Specification value) or (b) a Reactor Building pressure of 3 psig (4 psig Technical Specification value). Automatic actuation of the valves and pumps by the actuation signals switches the system from its normal operating mode to the emergency operating mode to deliver water from the borated water storage tank into the reactor vessel through the reactor coolant inlet lines. The following automatic actions accomplish this change:

a. The isolation valves in the purification letdown line and in the seal return lines close.

b. The high pressure injection pumps start.

3 c. The throttle valve in each high pressure injection line opens.

d. The valves in the lines connecting to the borated water storage tank outlet header open.

3 In addition to the automatic action described, the pumps and valves may be remote manually operated
from the control room. If any of the valve operators necessary to accomplish the above functions are
3 inoperable, the valves may be left in their emergency position during operation provided control of normal
3 parameters is not inhibited.

Operation of the High Pressure Injection System in the emergency mode will continue until the system
3 action is manually terminated. The HPI system is not designed to withstand a single passive failure since
3 the duration of system usage during an accident is not considered to be long term.

6.3.2.2.2 Low Pressure Injection System

The Low Pressure Injection System is designed to 1) maintain core cooling for larger break sizes and 2) control the boron concentration in the core while operating in the recirculation mode. The Low Pressure Injection System operates independently of and in addition to the High Pressure Injection System. A description of the normal reactor operation mode for the system is given in Chapter 9, "Auxiliary Systems."

Automatic actuation of the Low Pressure Injection System is initiated at: (a) Reactor Coolant System pressure of 550 psig (500 psig Technical Specification value) or (b) a Reactor Building pressure of 3 psig (4 psig Technical Specification value). Initiation of operation provides the following actions:

a. The valves in the lines connecting to the borated water storage tank outlet header open.

b. The low pressure injection pumps start on receipt of an engineered safeguards signal.

c. The inlet valves in the low pressure injection lines open.

7 d. Low pressure service water pumps start.

Low pressure injection is accomplished through two separate flow paths, each including one pump and one heat exchanger and terminating directly in the reactor vessel through core flooding nozzles located on opposite sides of the vessel.

9 The initial emergency operation of the Low Pressure Injection System involves pumping water from the
9 borated water storage tank into the reactor vessel. With all ES actuated pumps operating and assuming
the maximum break size, this mode of operation lasts for a minimum of about 30 minutes. When most
9 of the borated water storage tank inventory is exhausted, an emergency low water level alarm is

annunciated in the control room. At this time the operator will take action to open the suction valve from the Reactor Building emergency sump, permitting recirculation of the spilled reactor coolant and injection water from the Reactor Building emergency sump.

Following a large break LOCA located in the reactor inlet piping, the boric acid concentration within the core region will increase. Recrystallized boric acid could deposit on fuel assemblies and hinder heat
4 transfer. The LPI system provides two redundant gravity flow paths from the reactor outlet piping to the
4 Reactor Building Emergency Sump (RBES) to maintain continuous liquid flow through the core and
5 assure post-LOCA boric acid solubility. Additionally, the design of the reactor vessel and vessel internals
5 around the hot leg nozzles provides a third path that can assure post-LOCA boric acid solubility. At least
5 two of the three paths will always be available.

In the event of an accident where the Reactor Coolant System piping remains intact, then the Low Pressure Injection System will operate in the recirculation mode with suction being taken from the normal decay heat line. If in this mode of operation a decay heat isolation valve should fail closed, then a bypass line to the emergency sump would be opened. Recirculation would then take place with suction being taken from the emergency sump.

6.3.2.2.3 Core Flooding System

8 The Core Flooding System provides core protection continuity for intermediate and large Reactor Coolant System pipe failures. It automatically floods the core when the Reactor Coolant System pressure
8 drops below approximately 600 psig. The Core Flooding System is self-contained, self-actuating, and passive in nature. The combined coolant volume in the two tanks is sufficient to re-cover the core assuming no liquid remains in the reactor vessel following the loss-of-coolant accident.

The discharge pipe from each core flooding tank (CFT) is attached directly to a reactor vessel core flooding nozzle. Each core flooding line at the outlet of the CFT's contains an electric motor operated stop valve adjacent to the tank and two in-line check valves in series. The stop valves at the core flooding tank outlet are fully open during reactor power operation. Valve position indication is shown in the control room. During power operation when the Reactor Coolant System pressure is higher than the Core Flooding System pressure, two series check valves between the flooding nozzles and the CFT's prevent high pressure reactor coolant from entering the core flooding tanks.

The driving force to inject the stored borated water into the reactor vessel is supplied by pressurized nitrogen which occupies approximately one-third of the core flooding tank volume. Connections are provided for adding both borated water and nitrogen during power operation so that the proper level and pressure can be maintained. Each core flooding tank is protected from overpressurization by a relief valve installed directly on the tank. The size of these relief valves is based upon maximum water makeup rate to the tank. Redundant level and pressure indicators and alarms are provided in the control room for each tank.

6.3.2.3 Equipment and Component Descriptions

6.3.2.3.1 Piping

The high pressure injection and low pressure injection lines are designed for the normal operating conditions. The system temperature and pressure requirements are greater than those encountered during emergency operation. The Low Pressure Injection System piping and valves are subjected to more severe conditions during decay heat removal operation than during emergency operation and, therefore, operate well within the design conditions. Table 6-4 gives the design pressure and temperatures of these systems. To assure system integrity, major piping has welded connections except where flanges are dictated for maintenance reasons.

6.3.2.3.2 Pumps

The pumps used in the Emergency Core Cooling Systems are of proven design and have been used in many other applications. Pumps similar to the high pressure injection pumps have been used in boiler feed pump service and in high pressure makeup pump nuclear reactor service. Pumps similar to the low pressure injection pumps are used extensively in refinery service. The low pressure injection pump seals have been tested satisfactorily under the conditions which would be encountered during the loss-of-coolant accident. Both the high pressure and low pressure injection pump casings are liquid penetrant tested by methods described in the ASME Boiler and Pressure Vessel Code, Section VIII, and have been hydrotested and qualified to be able to withstand pressures as great or greater than 1.5 times the system design pressure. The pumps are designed so that periodic testing may be performed to assure operability and ready availability. The operating characteristics of each engineered safeguard pump are verified by shop testing before installation of the pumps.

6.3.2.3.3 Heat Exchangers

The low pressure injection system heat exchangers (decay heat removal coolers) are designed and manufactured to the requirements of the ASME VIII and the TEMA-R (Rigorous) Standards. In addition to these requirements, uniformity of the tubes is assured by eddy current testing, and the tubes are seal welded to the tube sheet to decrease the possibility of leakage. All tube welded ends are liquid penetrant tested to assure the absence of welding flaws. The heat exchangers have been fabricated with surface areas greater than those dictated by the most severe heat transfer conditions.

6.3.2.3.4 Valves

All remotely operated valves in the Emergency Core Cooling Systems are manufactured and inspected in
5 accordance with the intent of the ASME Nuclear Power Piping Code B31.7 or FSAR Section 3.2.2.2,
5 "System Piping Classifications" which provides allowances for substitute codes. Liquid penetrant,
radiography, ultrasonic, and hydrotesting are performed as the Code classification requires.

The seats and discs of these valves are manufactured from materials which will be free from galling and seizing. All valve material is certified to be in accordance with ASTM specifications. All remotely
3 operated valves in these systems are of the backseating type.

6.3.2.3.5 Coolant Storage

The letdown storage tank has a total coolant volume of 600 ft^3 and normally contains approximately 2,600 gallons of water. This tank provides water to the high pressure injection pumps until the borated water storage tank outlet valves are opened. The letdown storage tank is designed and inspected in accordance with the requirements of ASME III-C.

Each unit is provided with a borated water storage tank as described in Chapter 9, "Auxiliary Systems."

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

3 Each core flooding tank contains approximately 7,000 gallons of borated water with a boron
3 concentration maintained in accordance with the Core Operating Limits Report.

6.3.2.3.6 Pump Characteristics

Curves of total dynamic head and NPSH versus flow are shown in Figure 6-16 for the high pressure
3 injection pumps and in Figure 6-17 for the low pressure injection pumps. These curves are representative

3 in nature and are provided for information only. They are not intended to constitute design commitments
3 or performance requirements for the pumps. Refer to the Inservice Test Program for actual performance
3 requirements for HPI and LPI pumps.

6.3.2.3.7 Heat Exchanger Characteristics

The decay heat removal coolers are designed to remove the decay heat generated during a normal shutdown. In addition, each cooler is capable of cooling the injection water during the recirculation mode following a loss-of-coolant accident to provide for removal of decay heat which provides adequate core cooling. The heat transfer capability of a decay heat removal cooler as a function of recirculated water
0 temperature is illustrated in Figure 6-18. Note that this figure is representative in nature and is provided
3 for information only. It is not intended to constitute design commitments or performance requirements
3 for the coolers.

6.3.2.3.8 Relief Valve Settings

0 Relief valves are provided to protect the low pressure injection piping and components from overpressure.
7 On Units 1 and 2 these relief valves will be set at 370 psig, the system design pressure at 300°F for the "B"
3 LPI Coolers and at 515 psig, the system design pressure at 250°F for the "A" LPI Coolers. On Unit 3 the
3 relief valves will be set at 505 psig, the system design pressure at 250°F.

6.3.2.3.9 Component Data

Component data for each ECCS System is given in the following tables:

1. High Pressure Injection System - Table 6-8
2. Low Pressure Injection System - Table 6-9
3. Core Flooding System - Table 6-10

6.3.2.3.10 Quality Control

Quality Standards for the Emergency Core Cooling System components are given in Table 6-3.

6.3.2.4 Applicable Codes and Classifications

The High Pressure Injection, Low Pressure Injection, and Core Flooding Systems are designed and
5 manufactured to the Codes and Standards in Table 6-3 or FSAR Section 3.2.2.2, "System Piping
5 Classifications" which allows use of substitute codes.

6.3.2.5 Material Specifications and Compatibility

All components with surfaces in contact with water containing boric acid are protected from corrosion and deterioration. The High Pressure Injection System, which operates continuously with borated reactor
9 coolant, is constructed of stainless steel, for those portions in contact with borated water. With the exception of the borated water storage tank, the major components in low pressure injection are
9 constructed of stainless steel, for those portions in contact with borated water. The borated water storage tank is carbon steel with an interior phenolic coating. The core flooding piping and valves are stainless steel and the tanks are constructed of stainless clad carbon steel.

6.3.2.6 System Reliability

System reliability is assured by the system functional design including the use of normally operating equipment for safety functions, testability provisions, and equipment redundancy; by proper component selection; by physical protection and arrangement of the system; and by compliance with the intent of the AEC General Design Criteria. There is sufficient redundancy in the Emergency Core Cooling System to assure that no credible single failure can lead to significant physical disarrangement of the core. This is demonstrated by the single failure analysis presented in Table 6-11. This analysis was based on the assumption that a major loss-of-coolant accident had occurred and coincidentally an additional malfunction or failure occurred in the Engineered Safeguards System. For example, the analysis included malfunctions or failures such as electrical circuit or motor failures, valve operator failures, etc. It was considered incredible that valves would change to the opposite position by accident if they were in the required position when the accident occurred. Table 6-11 also presents an analysis of possible malfunctions of the core flooding tanks that could reduce their post-accident availability. It is shown that these malfunctions result in indications that will be obvious to the operators so appropriate action can be taken. In general, failures of the type assumed in this analysis are considered highly improbable since a program of periodic testing will be incorporated in the station operating procedures. The adequacy of
0 equipment sizes in the ECCS is demonstrated by the post-accident performance analysis described in
0 Chapter 15, "Accident Analyses."

6.3.2.6.1 High Pressure Injection Operability

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

6.3.2.6.2 Core Flood Tank Valve Operability

To assure that the Core Flood Tank isolation valves will not be accidentally closed while the reactor is at power, the circuit breaker supplying power to these valves will be kept open and under administrative control. Power to the starter controls comes from this same circuit breaker through a control transformer and will be disconnected when the circuit breaker is open.

Lights in the control room indicate valve position (open or closed). These lights have a power supply separate from the circuit breaker serving the Core Flood Tank isolation valves and are operated from limit switches on the valve operator. Another limit switch on the valve operator will cause an annunciator alarm in the control room anytime a Core Flood Tank isolation valve is away from the wide open position. The annunciator system has a power supply separate from that used to operate the valve or indicating lights.

6.3.2.6.3 Active Valve Operability

9 On January 2, 1973, the AEC requested that Duke Power Company determine an acceptable program
9 that would demonstrate operability of active valves. The testing was to simulate conditions associated
9 with normal system operation as well as loading conditions that would appropriately demonstrate seismic
9 and accident vibratory responses. The AEC request was further clarified by stating the "the test program
9 may be based upon selectively testing a representative number of active valves in the piping system
9 according to valve type, seismic and accident load level, size, etc. on a prototype basis". On May 1, 1973,
9 Duke Power Company responded by adding a supplement (Supplement 15) to the FSAR which described
9 various testing (environmental, vibrational, life cycle, etc.) on a subset of active valves. From a historical

9 perspective, the request by the AEC predated the formalization of the established programs and testing
9 requirements which are currently in place that ensure that active valves properly function during normal
9 and post accident conditions. Such programs include, but are not limited to, the following examples:
9 Environmental Qualification (EQ) program (10CFR50.49), MOV Diagnostic program (GL 89-10),
9 Inservice Testing/Inspection Program (10CFR50.55a), Containment Leakage Program (10CFR50
9 Appendix J), and Quality Assurance Program (10CFR50 Appendix B). Prior to the formulation and
9 development of such programs, the AEC's request was relevant. However, with the current programs in
9 place, the AEC's 1973 request is deemed historical in nature. The final intent of the request, which was
9 assurance of the operability of active valves, is deemed to be included within current requirements
9 associated with the design, maintenance, and testing of active components.

9 *6.3.2.6.3.1 Deleted per 1999 Update*

9 *6.3.2.6.3.2 Deleted per 1999 Update*

9 *6.3.2.6.3.3 Deleted per 1999 Update*

9 *6.3.2.6.3.4 Deleted per 1999 Update*

6.3.2.7 Protection Provisions

6.3.2.7.1 Seismic Design

Components in the Emergency Core Cooling System are designated as Class I equipment and are
8 designed to maintain their functional integrity during an earthquake discussed in Section 2.5.2.6,
8 "Maximum Hypothetical Earthquake (MHE)."

6.3.2.7.2 Missile Protection

Protection against missile damage is provided by either direct shielding or by physical separation of duplicate equipment. For most of the routing inside the Reactor Building, the ECCS Piping will be outside the primary and secondary shielding, and hence, protected from missiles originating within these areas. The portions of the injection lines located between the primary reactor shield and the reactor vessel wall are not subject to missile damage because there are no credible sources of missiles in this area.

The high pressure injection lines enter the Reactor Building via penetrations on opposite side of the building. Each injection line splits into two lines inside the Reactor Building, but outside the secondary (missile) shield, to provide four injection paths to the Reactor Coolant System. The four connections to the Reactor Coolant System are located between the reactor coolant pump discharge and the reactor inlet nozzles. There are four injection lines penetrating the missile shield, minimizing the effect on injection flow in the unlikely event of missile damage to the injection lines inside the secondary shield.

Protection from missiles is given to the low pressure injection lines within the Reactor Building. The portion of the Low Pressure Injection System located in the Reactor Building consists of two redundant injection lines which are connected to injection nozzles located on opposite sides of the vessel. Both redundant suction lines from the sump are missile protected. The sump suction is located outside of the secondary shielding and is additionally protected by a grating.

The entire Core Flooding System is located within the Reactor Building. The core flooding tanks and two of the three valves in each core flooding line are located outside of the secondary shield.

6.3.2.8 Post-Accident Environmental Consideration

The major operating components of the Emergency Core Cooling System are external to the Reactor Building and will not be exposed to the post-accident building environment.

Electrical and mechanical equipment within the Reactor Building which are required to be operable during and subsequent to a LOCA and/or steam line break are:

a. Reactor Coolant System pressure transmitters.

b. Reactor Building isolation valves and associated position indications. For all containment isolation valves inside containment, the valves may be left in their closed position if the operator is not functional.

c. Reactor Building air cooling unit fans and cooling coils.

d. Instrument cables for pressure transmitters, level, and valve position indication.

e. Power cables for the Reactor Building fan motors and isolation valves.

f. Isolation valves and flow verification instrumentation in the gravity flow path from the reactor outlet piping to the Reactor Building emergency sump.

Non-nuclear instrumentation (Item a) in the Reactor Protection System and the Engineered Safeguards System located inside the Reactor Building are qualified in accordance with Criteria for Nuclear Power Plant Protection Systems, IEEE No. 279, dated August 30, 1969, to establish the adequacy of equipment performance in the LOCA environment.

A valve operator similar to those being used (Item b) was satisfactorily tested for performance under conditions expected to exist in the Reactor Building after the LOCA. The operator was tested in accordance with Level 4 of the Standard Draft, dated June 7, 1968, prepared by Sub-Committee 2 (Equipment Qualification Testing) of the IEEE/NSG/Technical Committee for Standards.

Table 6-12 provides an analysis for valve motors which may become submerged following a LOCA.

A scaled down Reactor Building cooling coil unit (a 24 x 24 inch section identical in construction with the full-size unit) has been satisfactorily tested under post-accident condition. The maximum test conditions were 70 psig, 286 °F and 100 percent relative humidity.

Other equipment and components located in the primary containment or elsewhere in the plant must be operable during and subsequent to a loss-of-coolant or steam-line-break accident and are as follows:

a. Equipment Outside Containment

   Safety related equipment and components which are located outside the containment and which therefore are not subject to the abnormal environmental conditions present within the containment during an accident are given operational performance tests on either the actual equipment or prototype units. A list of the equipment located outside the containment is tabulated in Table 6-13.

b. Equipment Inside Containment

   Safety related equipment located within the containment is qualified for the application by tests to demonstrate operability under the accident environment. A list of this equipment along with a brief description of the qualification tests is tabulated in Table 6-14.

Instrument transmitters and electric motor valve operators in the Reactor Protection System and the
Engineered Safeguards System located inside the Reactor Building were designed to withstand the
potential effects of radiation due to normal and accident conditions. Non-metallic materials and
lubricants were selected on the basis of their susceptibility to radiation damage demonstrated by
irradiation tests. The instrument transmitters were successfully irradiation tested at the Babcock & Wilcox
Nuclear Development Center (NDC). The transmitters with two dosimeters attached to each were placed
in a sealed aluminum box and positioned near fuel elements in the NDC storage pool. The test was
conducted in two parts; the first part simulated the environmental dose to the transmitters associated with
the 40-year design plant lifetime, and the second part simulated the maximum expected dose to the
9 transmitters associated with a LOCA. The non-metallic materials originally selected for the electric motor
9 valve operators based on irradiation testing were: melamine used in the limit switches (all plastic material
used is melamine), viton for all seals, Humble Nebula EP #1 as the lubricant, and Class "H" insulation for
9 the motor. Current material qualification for these components is addressed by the Environmental
9 Qualification (EQ) program discussed in Section 3.11, "Environmental Design of Mechanical and
9 Electrical Equipment."

6.3.3 PERFORMANCE EVALUATION

In establishing the required component redundancy for the Emergency Core Cooling System, several factors related to equipment availability were considered:

a. The probability of a major Reactor Coolant System failure is very low; i.e., the probability that the equipment will be needed to serve its emergency function is low.

b. The fractional part of a given component lifetime for which the component is unavailable due to maintenance is estimated to be very small. On this basis, the probability that a major Reactor Coolant System accident would occur while a component from the Emergency Core Cooling System was out of service for maintenance is several orders of magnitude below the low basic accident probability.

c. The maintenance period for important equipment can usually be scheduled for a period of time when the reactor is shut down. Where maintenance of an engineered safety feature component is required during operation, the periodic test frequency of the similar redundant components can be increased to insure availability.

d. Where the systems are designed so that the components serve a normal function in addition to the emergency function or where meaningful periodic tests can be performed, there is also a low probability that the required emergency action would not be performed when needed; i.e., equipment reliability is improved by using the equipment for other than emergency functions.

6.3.3.1 High Pressure Injection System (HPI)

One high pressure injection string can deliver 450 gal/min at 585 psig reactor vessel pressure. The safety
0 analysis in Chapter 15, "Accident Analyses" has shown that two high pressure injection pumps through
0 two injection trains are sufficient to prevent core damage for those smaller leak sizes which do not allow
the Reactor Coolant System pressure to decrease rapidly to the point where the Low Pressure Injection
System is initiated.

4 After receiving an actuation signal, the HPI system valves for injection will open sufficiently to admit the
4 required flow within 14 seconds and the HPI pumps will reach full speed within 6 seconds. One of the
three high pressure injection pumps is normally in operation and a positive static head of water assures
that all pipe lines are filled with coolant. The high pressure injection lines contain thermal sleeves at their
connections into the reactor coolant piping to prevent over stressing the pipe juncture.

Operation of this system does not depend on any portion of another engineered safety feature. The system can be operated in conjunction with the Low Pressure Injection System if the HPI System must be operated in the recirculation mode.

6.3.3.2 Low Pressure Injection and Core Flooding Systems

Two pumps will deliver 6,000 gal/min to the reactor vessel through two separate injection lines. One pump can deliver approximately 3,000 gal/min to the reactor vessel at 100 psig.

After receiving an actuation signal, the low pressure injection valves will reach full open within 15 seconds and the low pressure injection pumps will reach full speed within 8 seconds.

Injection response of the Core Flooding System is dependent upon the rate of reduction of Reactor Coolant System pressure. For the maximum pipe break (14.1 ft^2), the Core Flooding System is capable of reflooding the core to the hot spot in less than 25 seconds after a rupture has occurred.

Special attention has been given to the design of core flooding nozzles to assure that they will take the differential temperature imposed by the accident condition. Special attention has also been given to the ability of the injection lines to absorb the expansion resulting from the recirculating water temperature.

The gravity flow path from the reactor outlet piping to the Reactor Building emergency sump will maintain a minimum core flow in excess of 40 gal/min to assure boric acid solubility. The flow path is
3 open within 9 hours following a large LOCA.

The Low Pressure Injection System is connected with other safeguards systems in three respects, i.e., (1) the High Pressure and Low Pressure Injection Systems and the Reactor Building Spray System take their suction from the borated water storage tank; (2) the low pressure injection pumps and the Reactor Building spray pumps share common suction lines from the Reactor Building sump during the coolant recirculation mode; and (3) the Low Pressure Injection System and the Core Flooding System utilize common injection nozzles on the reactor vessel.

2 6.3.3.2.1 Boron Precipitation Evaluation

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

4 In the initial long term phase of post-LOCA heat removal, a natural circulation flowpath from the core
4 through the vent valves to the downcomer occurs which sufficiently circulates the coolant through the
4 core. At some point in time the flowpath through the vent valves will no longer be available as the decay
4 heat becomes insufficient to drive the flow. In addition, natural circulation flow through the gaps between
4 the reactor vessel hot leg nozzles and the reactor internals has also been evaluated to be available.
5 Operator action must be taken to initiate at least one of the two gravity flow paths to provide further
5 assurance that flow is established and post-LOCA boric acid solubility is maintained. The method for
2 performing this function is by means of a drain line from the hot leg to the Reactor Building sump which
2 draws coolant from the top of the core, thereby inducing core circulation. The system has been designed
2 with redundant drain lines and has been shown to be single failure proof. The boron concentration of the
2 liquid leaving through the drain line is equal to the core boron concentration. Most of the core decay
2 heat is removed by steam flow through the vent valves. ECCS pump flow will continue to be provided to

2 the RCS cold legs and will preclude any boron concentration buildup in the vessel for breaks in the hot
2 leg.

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

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

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

6.3.3.3 Loss of Normal Power Source

Following a loss-of-coolant accident assuming a simultaneous loss of normal power sources to the LOCA
7 unit, the emergency power source and the Low Pressure Injection Systems will be in full operation within
7 53 seconds after actuation, even assuming a single failure, and the High Pressure Injection System will be
7 in full operation within 48 seconds after actuation. The electrical power system design is based on the
1 assumption that ESG actuation in one unit occurs simultaneously with a loss of offsite power to all three
1 units. However, accident scenarios in FSAR Section Chapter 15, "Accident Analyses" assume loss of
7 offsite power to the LOCA unit only. Except for large break LOCA (as described in UFSAR Section
7 15.14.3.3.6, "ECCS Performance and Single Failure Assumption"), all calculations for the Oconee Units
3 have assumed a 48 second delay from receipt of the actuation signal to start of flow for both the HPI and
LPI Systems. Upon loss of normal power sources including the startup source and initiation of an
engineered safeguards signal, the 4160 volt engineered safeguards power line is connected to the
underground feeder from Keowee hydro (Section 8.3.1, "AC Power Systems"). The Keowee hydro unit
will start up and accelerate to full speed in 23 seconds or less. An analysis has shown that by energizing
8 the HPI and LPI valves (which have opening times of 14 seconds, to deliver required flow, and 15
7 seconds respectively at normal bus voltage) and pumps after a 5 second swapover time (required by the
single failure), the design injection flow rate (HPI - 450 gal/min, LPI - 3000 gal/min) will be obtained
7 within 48 and 53 seconds, respectively.

6.3.3.4 Single Failure Assumption

7 UFSAR Section 15.14.3.3.6, "ECCS Performance and Single Failure Assumption" discusses ECCS
7 performance and the single failure assumption.

6.3.4 TESTS AND INSPECTIONS

6.3.4.1 ECCS Performance Tests

Table 6-15 summarizes performance testing for the Emergency Core Cooling System.

6.3.4.2 Reliability Tests and Inspections

All active components, listed in Table 6-15, of the Emergency Injection System will be tested periodically to demonstrate system readiness. The High Pressure Injection System will be inspected periodically during normal operation for leaks from pump seals, valve packing, and flanged joints. During operational testing of the low pressure injection pumps, the portion of the system subjected to pump pressure will be inspected for leaks. Items for inspection will be pump seals, valve packing, flange gaskets, heat exchangers, and safety valves for leaks to atmosphere.

6.3.5 INSTRUMENTATION REQUIREMENTS

The High Pressure Injection System is actuated automatically by a low Reactor Coolant System pressure
8 of 1,600 psig (1590 Technical Specification value) or by a Reactor Building pressure of 3 psig (4 psig
Technical Specification value). All of the pumps and valves can also be remotely operated from the
control room. In the event valve operators are not functional for ES valves on the HPI pump suction,
3 letdown or seal return, these valves may be left in their ES position during operation provided control of
3 normal plant parameters is not inhibited. Flow instrumentation is available in each HPI train during an
3 accident.

The Low Pressure Injection System is automatically actuated by a low Reactor Coolant System pressure of 550 psig (500 psig Technical Specification value) or Reactor Building pressure of 3 psig (4 psig Technical Specification value). All of the pumps and automatic valves can also be remotely operated from the control room. In the event valve operators are not functional for ES valves on the LPI pump suction, these valves may be left in their ES position during operation.

The Core Flooding System is actuated at a Reactor Coolant System pressure of 600 psig. At this point the differential pressure across the inline check valves allows them to open releasing the contents of the tanks into the reactor vessel.

The Engineered Safeguards Actuation instrumentation for the Emergency Core Cooling System is provided with redundant channels and signals as described in Chapter 7, "Instrumentation and Control." The control room layout is arranged so that all indicators and alarms are grouped in one sector at a convenient location for viewing. Switches and controls are also located conveniently.

6.3.6 REFERENCES

1. Qualification test of Limitorque valve operator, motor brake, and other units in a simulated reactor containment post-accident environment, Final Report F-C3327, July, 1972.

2. Qualification test of Limitorque valve operators in a simulated reactor containment post-accident steam environment, Final Report F-C3441, September 1972.

7 3. Deleted Per 1997 Update

7 4. Deleted Per 1997 Update

4 5. Instruction Manual for Rotork Valve Actuators, OM-245-1023.

7 6. DPC-NE-3003-P-A, "Duke Power Company Oconee Nuclear Station Mass and Energy Release and
7 Containment Response Methodology," Revision 0, March, 1995.

7 7. Jones, R. C., Biller, J. R., Dunn, B. M., ECCS Analysis of B&W's 177-FA Lowered-Loop NSSS,
7 Babcock & Wilcox, BAW-10103 Rev. 3, July 1977.

8 8. Qualification Test Report for Two Valve Operators (11NAZT1 and 90NAZT1) for Rotork Controls,
8 Inc., Report No. 43979-1, Revision A, December 1978.

6.4 HABITABILITY SYSTEMS

6.4.1 DESIGN BASES

Oconee Nuclear Station's design pre-dates General Design Criterion 19 (GDC-19) of Appendix A to 10
8 CFR 50, however control room habitability was a design consideration at Oconee as discussed in Section
8 3.1.11, "Criterion 11 - Control Room (Category B)."

The Oconee Nuclear Station control rooms are located in the Auxiliary Building. Oconee 1 and 2 have a shared control room while Oconee 3 has a separate control room. Figure 6-19 shows the location of the two control rooms with regard to other major structures of the station. Figure 6-20 and Figure 6-21 show the Oconee 1 and 2 and Oconee 3 control room general arrangement, respectively.

The facility is provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection is provided to permit access, even under accident conditions, to equipment in the control room or other areas as necessary to shut down and maintain safe control of the facility without radiation exposures of personnel in excess of 10CFR20 limits. The control room shielding meets the NUREG-0578 requirements. It is possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost.

6.4.2 SYSTEM DESIGN

6.4.2.1 Definition of Control Room Envelope

The control room envelope includes the control room and all rooms the control room personnel may require access to during emergency plant operation. This envelope is designated as the Control Room
7 Zone and is comprised of the Control Room, Offices, Computer Rooms, Operator's Break Area, and Operator's Toilet Room.

All controls and displays necessary to bring the plant to a safe shutdown condition are included within the control room envelope.

6.4.2.2 Ventilation System

0 The Control Room Ventilation System is described in detail in Section 9.4.1, "Control Room Ventilation." The ventilation system was designed and installed in accordance with HVAC Industry Standards and practices for commercial and industrial systems.

6.4.2.3 Leak Tightness

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

0

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

6.4.2.4 Interaction With Other Zones and Pressure-Containing Equipment

The control room envelope is bounded on the north, south, and west by the Auxiliary Building and on the east by the Turbine Building. The Ventilation Systems serving the Auxiliary Building and Turbine
0 Buildings are separate from the Control Room Ventilation System.

3 Interaction with other areas is minimal as air for pressurizing the Control Room Zone is taken from outside and is filtered through charcoal filters to eliminate airborne radioactive contaminants.

Pressure retaining equipment generally is not permitted in the control room zone. Exceptions to this are several hand held fire extinguishers for local fire control and several self-contained breathing apparatus with additional bottles of replenishment air.

6.4.2.5 Toxic Gas Protection

Chlorine gas is used in the Water Treatment System. Other gases used on site are Ammonia, Hydrazine, Hydrogen, Liquid Nitrogen, and welding gases. No potential sources of toxic gas releases were identified off site. Protection of control room operators against potential toxic gas release accidents has been found to be adequate by the NRC (Reference 1).

Self-contained type breathing apparatus are available to operator personnel. The Oconee 1 and 2 Control Room has six apparatus with twelve refill bottles and the Oconee 3 Control Room has three apparatus with six refill bottles.

6.4.3 TESTING AND INSPECTION

0 The Control Room Ventilation System is normally operable and is accessible for periodic inspection. The
0 pressurization portion of the system is tested periodically to demonstrate its readiness and operability as required by the Technical Specifications.

6.4.4 INSTRUMENTATION REQUIREMENTS

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

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

6.4.5 REFERENCES

1. J. F. Stolz (NRC) to H. B. Tucker (Duke) November 24, 1986.

6.5 FISSION PRODUCT REMOVAL AND CONTROL SYSTEMS

The systems addressed below reduce accidental release of fission products following a design basis accident.

6.5.1 ENGINEERED SAFEGUARDS (ES) FILTER SYSTEMS

Included in this section is a discussion of the Reactor Building Penetration Room Ventilation System.

6.5.1.1 Design Bases

The Reactor Building Penetration Room Ventilation System (PRVS) is designed to collect and process potential Reactor Building penetration leakage to minimize environmental activity levels resulting from post-accident Reactor Building leaks. Experience (Reference 1) has shown that Reactor Building leakage is more likely at penetrations than through the liner plates or weld joints.

The design basis for filtration was a requirement to remove 25 percent of the core iodine inventory. The 25 percent was derived using the standard assumption that during a Maximum Hypothetical Accident 50 percent of the halogens are released from the core and that 50 percent of the iodine released plates out within the Reactor Building. The initial inventory of the individual isotopes in terms of Curies/MWt is given in Table 6-18.

6.5.1.2 System Design

This section addresses the design only as related to fission product removal. More details of system design and operation are addressed in Section 9.4.7.2, "System Description."

The system schematic and characteristics are shown on Figure 6-4 and Figure 6-22, respectively. Figure 6-23 and Figure 6-24 show penetration and opening locations in the penetration rooms. Mechanical openings, electrical openings, and construction details are illustrated in Figure 6-25, Figure 6-26, and Figure 6-27, respectively.

Penetration rooms are formed adjacent to the outside surface of each Reactor Building by enclosing the area around the majority of the penetrations.

Each unit's penetration room is provided with two fans and two filter assemblies. Both fans, discharging
1 through a single line to the unit vent, may be controlled from the main control room.

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

Particulate filtration is achieved by a medium efficiency pre-filter and a high efficiency (HEPA) filter. Adsorption filtration is accomplished by an activated charcoal filter. When the system is in operation, a
0 negative pressure will be maintained in the penetration room to ensure inleakage. Penetration room pressure is displayed in the control room and excessive and insufficient vacuum are annunciated. It can be assumed that no pressure differentials exist in the room, so that an instrument string sensing pressure at a single point can be used. This is because the communicative paths between various parts of the

penetration room are very large in comparison with the minute leakage that might exist due to imperfect seals.

From time to time the system will be activated to purge the filters of any moisture that may accumulate. The air will be taken from the penetration room where it will be sufficiently warm to accomplish this purpose. Dampers are placed in the system inlets to prevent moisture from being carried through by natural circulation.

The only penetrations which do not pass through the penetration rooms are:

5 1. Reactor Building Fuel Transfer Tube and Reactor Coolant SSF Makeup (Penetration No. 11a).

5 2. Reactor Building Fuel Transfer Tube and Reactor Coolant SSF Letdown (Penetration No. 12a).

5 3. One Main Steam Line per unit --1B (Penetration No. 28) and 2A & 3A (Penetration No. 26).

5 4. Normal Personnel Access Lock (Penetration No. 90).

5 5. Permanent Equipment Hatch which contains a double-gasketed closure (Penetration No. 91).

5 6. Emergency Personnel Access Lock (Penetration No. 92).

9 7. Reactor Building Normal Sump Drain/Hydrogen Recombiner Drain (Penetration No. 5a).

5 8. Reactor Coolant Post-Accident Liquid Sample Lines (Penetration No. 5b).

5 9. Reactor Coolant Quench Tank Drain (Penetration No. 29).

5 10. Reactor Building Emergency Sump Recirculation A (Penetration No. 36).

5 11. Reactor Building Emergency Sump Recirculation B (Penetration No. 37).

5 12. Reactor Building Emergency Sump Drain (Penetration No. 40).

5 13. Reactor Coolant Decay Heat Drop Line and Post-Accident Boron Dilution Line (Penetration No. 62
5 -- Units 2 and 3 only).

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

8 The above lines, including the main steam lines, are not considered a source of significant leakage because
8 they are welded to the liner plate. The access openings can be tested during normal operation and are not considered sources of significant leakage. There are double seals at each of these access openings, and the space between these double seals is connected to the penetration room. The refueling tube is equipped with a blind flange which is only opened during shutdown for transfer of fuel to the spent fuel pool.

6.5.1.3 Design Evaluation

A single failure analysis of the various portions of this system is presented in Table 6-19.

Redundant fans, cross connected piping, and locked open filter inlet valves render incredible a loss of
3 cooling air flow to the filters. However, even if air flow is lost through a filter, Calculation OSC-4024
3 (Performed for PIR #4-090-0057) has shown charcoal ignition temperature will not be reached.

3

Adequate instrumentation is provided to detect loss of air flow through either filter. Reduction in air flow below a preset minimum would result in low Penetration Room vacuum and cause an alarm in the
9 control room. Flow indication with readout outside the penetration filter area is furnished for each filter.

The Reactor Building penetration room is maintained at a negative pressure of greater than 0.06 in. H_20
1 with respect to the outside atmosphere when the penetration room fans are in operation.

Even in the event of unfiltered leakage of all the iodine input to the penetration room due to high wind velocity, the improvement in atmospheric dilution more than compensates for bypassing of the penetration room filter by this portion of the iodine. At a wind velocity of greater than 8.1 mph, the improvement in X/Q compensates for the complete loss of the filtering system in the calculation of offsite dose. A wind velocity of 8.1 mph will cause a reduction in pressure of .032 in. H_20 along the penetration room wall. (This assumes that wind velocity is exactly parallel to the wall which is the worst case assumption). By maintaining the penetration room at a negative pressure of 0.06 in. H_20, a conservative margin of pressure is established.

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

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

Absolute Filter- The basic design criteria for this filter is set forth in AEC Health and Safety Bulletin 212 (6-25-65) which incorporates U.S. Military Specification MIL-F-51068A captioned "Filter, Particulate, High Efficiency, Fire Resistant".

In addition, the dust holding capacity is determined by utilizing the test procedures of AFI "Code of Testing Air Cleaning Devices Used in General Ventilation", Section I (1952).

Adsorptive (Carbon) Filter- The specified ignition temperature of the carbon is checked using the
6 methodology of ASTM D-3803-1989. This test is conducted on one sample from each lot of carbon.

Fans- Fan performance is determined by prototype test according to procedures set forth by the Air Moving and Conditioning Association (AMCA) 1960 Standard Test Code.

6.5.1.4 Tests and Inspections

The Reactor Building PRVS may be actuated during normal operation for testing and inspection. The high efficiency particulate air (HEPA) filters and the charcoal iodine filters are tested to ensure that they are able to remove airborne materials from penetration leakage.

Sight glasses in the PRVS drain lines and humidity sensors are available for monitoring the penetration room humidity. (Procedures are implemented to monitor the humidity, and prompt action is to be taken
3 to reduce the humidity to less than 80 percent when these values are exceeded.) External carbon sample canisters are installed on the filters to facilitate sampling. Provision is made to check penetration room negative pressure relative to either the Auxiliary Building or the outside.

Testing and inspection of the system is as required by the Technical Specifications.

6.5.1.5 Instrumentation Requirements

Instrumentation is used only to monitor system performance and has no control function other than to guide the operator in adjusting the final control elements.

Penetration room pressure and humidity and loss of air flow through either filter are monitored.

6.5.1.6 Materials

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

6.5.2 CONTAINMENT SPRAY SYSTEMS

4 No credit is taken for this system for fission product removal or control in LOCA analysis (see 15.14.7,
4 "Environmental Evaluation"). Credit is taken for this system for fission product removal in the MHA
4 off-site dose analyses only. (see 15.15.1, "Identification of Accident").

6.5.3 FISSION PRODUCT CONTROL SYSTEMS

5 Credit is taken for fission product control by the Containment Hydrogen Recombiner System which is
5 addressed in Section 15.16, "Post-Accident Hydrogen Control"; however, this system is not considered an
5 Engineered Safeguards System.

6.5.4 REFERENCES

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

6.6 INSERVICE INSPECTION OF CLASS 2 AND 3 COMPONENTS

6.6.1 COMPONENTS SUBJECT TO EXAMINATION

Class 2 and 3 components, indicated in the Oconee Inservice Inspection Plan, are equivalent to Quality Groups B and C respectively of Regulatory Guide 1.26. These components will be examined in accordance with the provisions of the ASME Boiler and Pressure Vessel Code Section XI in effect as specified in 10CFR50.55a(g) to the extent practical. Requests for relief from inservice inspection requirements determined to be impractical will be submitted to the NRC for review in accordance with NRC guidelines for submitting such requests.

6.6.2 ACCESSIBILITY

Class 2 and 3 systems at Oconee were installed before any inservice inspection requirements existed for these systems. In most cases adequate clearance is available to perform the inspection required by Section XI. In cases where adequate clearance is not available, the use of alternate inspection techniques will be investigated. If no alternate techniques appear practical, relief will be requested.

6.6.3 EXAMINATION AND PROCEDURES

The examination techniques to be used for inservice inspection include radiographic, ultrasonic, magnetic particle, liquid penetrant, eddy current, and visual examination methods. For all examinations, both remote and manual, specific procedures will be prepared describing the equipment, inspection technique, operator qualifications calibration standards, flaw evaluation, and records. These techniques and procedures will meet the requirements of the Section XI edition in effect as stated in Section 6.6.1, "Components Subject to Examination."

6.6.4 INSPECTION INTERVALS

The inservice inspection interval for ASME Class 2 and 3 components is 10 years. The inspection schedule will be developed in accordance with IWC-2400 and IWD-2400. Detailed inspection listings and scheduling will be contained in the Oconee Inservice Inspection Plan.

6.6.5 EXAMINATION CATEGORIES AND REQUIREMENTS

The examination categories to be used are those listed in Tables IWC-2500-1 and IWD-2500-1 of ASME Section XI. Specific examinations will be identified by an Item Number, composed of the Item Number assigned in Tables IWC-2500-1 and IWD-2500-1 of ASME Section XI, plus an additional number to uniquely identify that examination.

6.6.6 EVALUATION OF EXAMINATION RESULTS

Evaluation of examination results shall be in accordance with the Section XI in effect as stated in Section 6.6.1, "Components Subject to Examination" where these evaluation standards are contained in Section XI. For examination where evaluation standards are not contained in Section XI, evaluation shall be performed in accordance with the original construction code.

6.6.7 SYSTEM PRESSURE TESTS

Classes 2 and 3 system pressure testing complies with Section XI Articles IWC-5000 and IWD-5000 in effect as stated in Section 6.6.1, "Components Subject to Examination."

6.6.8 AUGMENTED INSERVICE INSPECTION TO PROTECT AGAINST POSTULATED PIPING FAILURES

Class 2 high energy fluid piping systems will be inspected in accordance with Article IWC-2000 of Section XI up to the isolation valve outside containment. The examination areas, methods, extent, and frequency will be as specified in Article IWC-2000. Those lines requiring augmented inservice inspection will be contained in the Oconee Nuclear Station Inservice Inspection Plan.

THIS IS THE LAST PAGE OF THE CHAPTER 6 TEXT PORTION

APPENDIX 6. CHAPTER 6 TABLES AND FIGURES

5 **Table 6-1. Deleted per 1995 Update**

Table 6-2. Leakage Quantities to Auxiliary Building

	Leakage Leakage Source	No. of Sources	Estimated Quantities: Leakage Per Source (drops/min)	Estimated Quantities: Total (cc/hr)
	Low Pressure Injection System			
a.	Pump Seals			
	Low Pressure Injection Pump	2	50	300
	Spray Pump	2	50	300
9 9 b.	Flanges	17 (Unit 2) 14 (Units 1&3)	10	510 (Unit 2) 420 (Units 1&3)
c.	Process Valves	28	1	84
d.	Instrumentation Valves	40	1	120
e.	Valve Seats at Boundaries	31	(**)	1070 (Units 1&2) 950 (Unit 3)
7 7			Total	2294 (Units 1&2) 2174 (Unit 3)

Note:

7

(**)Assuming 10 cc/hr/in. of nominal disc diameter.

9 Ref ONOE-14087 (Unit 2)

Table 6-4 (Page 1 of 2). Engineered Safeguards Piping Design Conditions

			Temp (F)	Press. (psig)
	1.	High Pressure Injection System		
7	a.	From the pump discharge to upstream of the check valves inside the secondary shielding.	200/150	3,040/3120
3	b.	High pressure injection pump.	200/150	3,040/3,120
7	c.	From upstream of the check valves to the reactor inlet line.	650	2,500
	2.	Low Pressure Injection System		
	a.	From the borated water storage tank to upstream of the borated water storage tank outlet valves.	150	Static
9 9	b.	From upstream of the borated water storage tank outlet valve to upstream of the electric motor operated valves in the borated water feed lines. (Unit 2 only, is to the check valves in the borated water feed lines)	200	100
9 9	c.	From upstream of the electric motor operated valves in the borated water feed lines to upstream of the valves at the pump inlets. (Unit 2 only, is from the check valves in the borated water feed lines)	300	200
9 9	d.	From upstream of the system inlet valves at the pump inlets to the pump inlet.		
9		Trains 1A, 1C, 2A, 2C	300/250	470/505
9		Trains 1B, 2B	300	370
9 9	e.	From the pump outlet to upstream of the throttle valves at the cooler discharge.		
9		Trains 1A, 1C, 2A, 2C (Note 1)	300/250	470/515
9		Trains 1B, 2B	300	370
9 9	f.	From upstream of the throttle valves at the cooler discharge to upstream of the LPI Header isolation valves.		
9		Trains 1A, 2A	300/250	470/515
9		Trains 1B, 2B	300	470/505
	g.	From upstream of the system inlet valves to upstream of the check valves in the core flooding lines.	300	2,500
	h.	From upstream of the check valves in the core flooding lines to the reactor vessel.	650	2,500
	i.	From the Reactor Building emergency sump to upstream of the valves in the recirculation lines.	300	59
	3.	Reactor Building Spray System		
	a.	From downstream of the pump inlet valves to downstream of the Reactor Building valves.	300	495
	b.	From downstream of the inlet valves through the nozzles.	300	200

Table 6-4 (Page 2 of 2). Engineered Safeguards Piping Design Conditions

			Temp (F)	Press. (psig)
	4.	<u>Low Pressure Service Water System</u>		
	a.	Condenser circulating water crossover to low pressure service water pump suction.	100	50
	b.	Pump discharge	100	100
		(OCONEE 3 ONLY)		
	1.	Low Pressure Injection System		
	a.	From the borated water storage tank to upstream of the borated water storage tank outlet valves.	150	Static
	b.	From upstream of the borated water storage tank outlet valve to upstream of the check valves in the borated water feed lines.	200	100
	c.	From upstream of the check valves to upstream of the motor operated valves in the borated water feed lines.	300	200
	d.	From upstream of the electric motor operated valves in the borated water feed lines to upstream of the valves at the pump inlets.	300	388
9	e.	From upstream of the system inlet valves at the pump inlets to upstream of the LPI Header Isolation valves.	300/250	470/505
	f.	From upstream of the system inlet valves to upstream of the check valves in the core flooding lines.	300	2,500
	g.	From upstream of the check valves in the core flooding lines to the reactor vessel.	650	2,500
	h.	From the Reactor Building emergency sump to upstream of the valves in the recirculation lines.	300	59

9 **Note:**

9 1. For the C Train Connection to the B Train Cooler, design conditions are 300°F and 370 psig
9 beginning downstream of the cross over valve.

Table 6-5. Single Failure Analysis Reactor Building Spray System

Component		Malfunction	Comments
1.	Reactor Building the spray pump.	Fails to start.	Since each of the two strings of Reactor Building Spray System is equally sized, the remaining string will provide heat removal capability at a reduced rate. In combination with the Reactor Building Cooling System, heat removal capability in excess of the requirements will be provided.
2.	Building isolation valve.	Fails to open.	(Same as above.)
3.	Check valve in suction or discharge line.	Fails to open.	(Same as above.)

Table 6-6. Single Failure Analysis For Reactor Building Cooling System

	Component		Malfunction	Comments
6 6	1.	Circulating fan	Fails to operate.	The cooling capacity of the cooling units is reduced; however, 2 of 3 cooling units provide the required cooling.
6 6 6	2.	Cooler service water outlet valve. (LPSW-18, -21, -24)	Fails to open fully	Valve will normally be partially open. If the valve fails to open fully, the unit will operate under reduced heat removal capability. The required cooling load will be met by 2 of 3 cooling units.
0 0 6 6	3.	Cooler service water inlet valve. (LPSW-16, -19, -22)	Inadvertently left closed.	The flow through this string will be unavailable for cooling. It is unlikely that this condition would occur during an accident since the position and flow are monitored during normal operation. The required cooling load will be met by 2 of 3 cooling units.
0 0	4.	Service water pump (1A, 1B, 1C).	Fails to operate.	The two remaining pumps will provide full low pressure service water flow to all components.
0 0	5.	Service water pump (3A, 3B).	Fails to operate.	The one remaining pump will provide full low pressure service water flow to all components.

Table 6-7 (Page 1 of 4). Reactor Building Penetration Valve Information

			Inside Penetration Valve Data									Outside Penetration Valve Data								
								Valve Position									Valve Position			
Pen	Description	Vlv Arrg	Qty	Size	Type	Oper	Signal	Norm	Fail	Post Acc	Indication	Qty	Size	Type	Oper	Signal	Norm	Fail	Post Acc	Indication
1[16]	Pressurizer Sample	1	2	¼	SH	EMO	ES	OP	AI	CL	YES	1	½	SH	AIR	ES	OP	CL	CL	YES
2	OTSG A Sample	8	Closed Loop Inside Containment									1	½	SH	AIR	ES	CL	CL	CL	YES
3	Component Cooling Inlet	3	1	6	CK	-	-	OP	-	CL	NO	1	6	CK	-	-	OP	-	CL	NO
4	OTSG B Drain	7	Closed Loop Inside Containment									1	4	SH	EMO	ES	OP	AI	CL	YES
5A	Hydrogen Recombiner Drain	24	None									4	1	CK	-	-	CL	-	CL[2]	NO
												4	1	SH	MAN	-	CL	-	CL[2]	NO
												1	2	SH	EMO	ES	OP	AI	CL	YES
												1	2	SH	AIR	ES	OP	CL	CL	YES
5B	Post Accident Liquid Sample	13	None									2[5]	1	SH	SOL[5]	-	CL	CL[5]	CL[15]	YES[5]
6	RC Letdown	1	2	2	SH	EMO	ES	OP	AI	CL	YES	1	2	SH	AIR	ES	OP	CL	CL	YES
7	RC Pump Seal Return	4	1	3, 4[7]	SH	EMO	ES	OP	AI	CL	YES	1	2	SH	AIR	ES	OP	OP[1]	CL	YES
8A	Pressurizer Aux Spray	3	1	1, 1½[11]	CK	-	-	OP	-	CL	NO	1	1, 1½[11]	CK	-	-	OP	-	CL	NO
8B	Loop A Nozzle Warming	3	1	1, 1½[11]	CK	-	-	OP	-	CL	NO	1	1, 1½[11]	CK	-	-	OP	-	CL	NO
9	HP Injection Loop A	2	1	4	CK	-	-	OP	-	OP	NO	None								
10A	RC Pump Seal Injection	3	1	1	CK	-	-	OP	-	CL	NO	1	1, 1½[6]	CK	-	-	OP	-	CL	NO
10B	RC Pump Seal Injection	3	1	1	CK	-	-	OP	-	CL	NO	1	1	CK	-	-	OP	-	CL	NO
11	Fuel Transfer Tube	19	Special Closure (Flange)									None								
			1	3	SH	EMO	-	CL	AI	CL	YES									
			1	4	SH	EMO	-	CL	AI	CL	YES									
12	Fuel Transfer Tube	18	Special Closure (Flange)									None								
			1	3	SH	EMO	-	CL	AI	CL	YES									
			1	1½	SH	EMO	-	CL	AI	CL	YES									
			2	1	SH	EMO	-	CL	AI	CL	YES									
13	RB Spray Inlet	2	1	8	CK	-	-	CL	-	OP	NO	None								
14	RB Spray Inlet	2	1	8	CK	-	-	CL	-	OP	NO	None								
15	LPI Inlet	2	1	10	CK	-	-	CL	-	OP	NO	None								
16	LPI Inlet	2	1	10	CK	-	-	CL	-	OP	NO	None								
17	OTSG B EFW Injection	5	Closed Loop Inside Containment									None								
18	Quench Tank Vent	4	1	2	SH	EMO	ES	OP	AI	CL	YES	1	2	SH	AIR	ES	OP	CL	CL	YES
19	RB Purge Inlet	4	1	48	SH	EMO	ES	CL	AI	CL	YES	1	48	SH	AIR	ES	CL	CL	CL	YES
20	RB Purge Outlet	4	1	48	SH	EMO	ES	CL	AI	CL	YES	1	48	SH	AIR	ES	CL	CL	CL	YES

9 Table 6-7 (Page 2 of 4). Reactor Building Penetration Valve Information

				Inside Penetration Valve Data									Outside Penetration Valve Data								
9									Valve Position									Valve Position			
9	Pen	Description	Vlv Arrg	Qty	Size	Type	Oper	Sig-nal	Norm	Fail	Post Acc	Indi-cation	Qty	Size	Type	Oper	Sig-nal	Norm	Fail	Post Acc	Indi-cation
9	21	LPSW to RCP Coolers	7	Closed Loop Inside Containment									1	10	SH	EMO	ES	OP	AI	CL	YES
9	22	LPSW from RCP Coolers	7	Closed Loop Inside Containment									1	10	SH	EMO	ES	OP	AI	CL	YES
9	23A	RC Pump Seal Injection	3	1	1	CK	-	-	OP	-	CL	NO	1	1	CK	-	-	OP	-	CL	NO
9	23B	RC Pump Seal Injection	3	1	1	CK	-	-	OP	-	CL	NO	1	1	CK	-	-	OP	-	CL	NO
9	24A	RB Hydrogen Analyzer	10	None									1	½, ⅜[12]	SH	SOL	-	CL	CL	CL[2]	YES
9	24B	RB Hydrogen Analyzer	10	None									1	½, ⅜[12]	SH	SOL	-	CL	CL	CL[2]	YES
9	25	OTSG B Feedwater Line	6	Closed Loop Inside Containment									1	24	CK	-	-	OP	-	CL	NO
9	26	OTSG A Main Steam Line	5	Closed Loop Inside Containment									None								
9	27	OTSG A Feedwater Line	6	Closed Loop Inside Containment									1	24	CK	-	-	OP	-	CL	NO
9	28	OTSG B Main Steam Line	5	Closed Loop Inside Containment									None								
9	29	Quench Tank Drain	4	1	4	SH	EMO	ES	OP	AI	CL	YES	1	2	SH	AIR	ES	OP	CL	CL	YES
9	30	LPSW to RBCU	5	Closed Loop Inside Containment									None								
9	31	LPSW to RBCU	5	Closed Loop Inside Containment									None								
9	32	LPSW to RBCU	5	Closed Loop Inside Containment									None								
9	33	LPSW from RBCU	5	Closed Loop Inside Containment									None								
9	34	LPSW from RBCU	5	Closed Loop Inside Containment									None								
9	35	LPSW from RBCU	5	Closed Loop Inside Containment									None								
9	36	RB Emergency Sump Recirc	12	None									1	14	SH	EMO	-	CL	AI	OP	YES
9	37	RB Emergency Sump Recirc	15	None									1	14	SH	EMO	-	CL	AI	OP	YES
9													1	1	SH	MAN	-	CL	-	OP	NO
9	38	Quench Tank Cooler Inlet	3	1	1½	CK	-	-	OP	-	CL	NO	1	1½, 2[9]	CK	-	-	OP	-	CL	NO
9	39a[17]	Core Flood Tank Vent	16	1	1	SH	MAN	-	CL	-	CL	NO	2	1	SH	MAN	-	CL	-	CL	NO
9	39b	HP Nitrogen Supply	20	1	1	CK	-	-	CL	-	CL	NO	1	½	SH	MAN	-	CL	-	CL	NO
9													1	1	CK	-	-	CL	-	CL	NO
9													1	1	SH	MAN	-	CL	-	CL	NO
9	40	RB Emergency Sump Drain	14	None									2	2	SH	MAN	-	CL	-	CL	NO
9	41	Instrument Air Supply	9	1	3	SH	MAN	-	CL	-	CL	NO	1	3	SH	MAN	-	CL	-	CL	NO
9	42A	RB Hydrogen Analyzer	10	None									1	½, ⅜[12]	SH	SOL	-	CL	CL	CL[2]	YES
9	42B	RB Hydrogen Analyzer	10	None									1	½, ⅜[12]	SH	SOL	-	CL	CL	CL[2]	YES
9	43	OTSG A Drain	7	Closed Loop Inside Containment									1	4	SH	EMO	ES	OP	AI	CL	YES
9	44	Component Cooling to CRD	3	1	2½	CK	-	-	OP	-	CL	NO	1	2½	CK	-	-	OP	-	CL	NO

Table 6-7 (Page 3 of 4). Reactor Building Penetration Valve Information

Pen	Description	Vlv Arrg	Inside Penetration Valve Data									Outside Penetration Valve Data								
								Valve Position									Valve Position			
			Qty	Size	Type	Oper	Sig-nal	Norm	Fail	Post Acc	Indi-cation	Qty	Size	Type	Oper	Sig-nal	Norm	Fail	Post Acc	Indi-cation
45A	Leak Rate Test Line	9	1	½	SH	MAN	-	CL	-	CL	NO	1	½	SH	MAN	-	CL	-	CL	NO
45B	Leak Rate Test Line	9	1	½	SH	MAN	-	CL	-	CL	NO	1	½	SH	MAN	-	CL	-	CL	NO
45C[17]	Leak Rate Test Line	9	1	½	SH	MAN	-	CL	-	CL	NO	1	½	SH	MAN	-	CL	-	CL	NO
48	Breathing Air Supply to RB	9	1	2	SH	MAN	-	CL	-	CL	NO	1	2	SH	MAN	-	CL	-	CL	NO
49[16]	LP Nitrogen Supply to RB	22	1	1½	CK	-	-	CL	-	CL	NO	2	1	SH	MAN	-	CL	-	CL	NO
50	OTSG A EFW Injection	5	Closed Loop Inside Containment									None								
51	LRT Supply and Exhaust	11	Special Closure (Flange)									1	8	SH	AIR	-	CL	AI	CL	NO
												1	½, 1[10]	SH	MAN	-	CL	-	CL	NO
52	HP Injection B Loop	2	1	4	CK	-	-	CL	-	OP	NO	None								
53A	HP Nitrogen to CFT	20	1	1	CK	-	-	CL	-	CL	NO	1	½, 1[8]	SH	MAN	-	CL	-	CL	NO
												1	1	SH	MAN	-	CL	-	CL	NO
												1	1	CK	-	-	CL	-	CL	NO
53B[17]	LP Nitrogen Supply to RB	21	1	1½	CK	-	-	CL	-	CL	NO	1	2	SH	MAN	-	CL	-	CL	NO
54	Component Cooling Outlet	4	1	8	SH	EMO	ES	OP	AI	CL	YES	1	8	SH	AIR	ES	OP	OP[1]	CL	YES
55	Demin Water Supply	9	1	4	SH	MAN	-	CL	-	CL	NO	1	4	SH	MAN	-	CL	-	CL	NO
56	Spent Fuel Canal Fill/Drain	9	1	8	SH	MAN	-	CL	-	CL	NO	1	8	SH	MAN	-	CL	-	CL	NO
57[16]	DHR Return Line	12	None									1	10	SH	EMO	-	CL	AI	CL[14]	YES
58A[17]	Pressurizer Sample Line	1	2	¼, 1[13]	SH	EMO	ES	OP	AI	CL	YES	1	½	SH	AIR	ES	OP	CL	CL	YES
58B	OTSG B Sample	8	Closed Loop Inside Containment									1	½	SH	AIR	ES	CL	CL	CL	YES
59	Core Flood Tank Sample	17	2	1	SH	EMO	-	CL	AI	CL	YES	2	1	SH	MAN	-	CL	-	CL	NO
60	RB Sample Line (Outlet)	23	1	2	SH	EMO	ES	OP	AI	CL[2]	YES	1	2	SH	AIR	ES	OP	OP[1]	CL[2]	YES
			1	2, 3[4]	SH	EMO	-	CL	AI	CL[2]	YES	1	½, 1[3]	SH	MAN	-	CL	-	CL[2]	NO
61	RB Sample Line (Inlet)	25	1	2	SH	EMO	ES	OP	AI	CL[2]	YES	1	2	SH	AIR	ES	OP	OP[1]	CL[2]	YES
			1	2	SH	EMO	-	CL	AI	CL[2]	YES									
62[17]	DHR Return Line	12	None									1	12	SH	EMO	-	CL	AI	CL[14]	YES

Table 6-7 (Page 4 of 4). Reactor Building Penetration Valve Information

			Inside Penetration Valve Data										Outside Penetration Valve Data							
								Valve Position									Valve Position			
Pen	Description	Vlv Arrg	Qty	Size	Type	Oper	Sig-nal	Norm	Fail	Post Acc	Indi-cation	Qty	Size	Type	Oper	Sig-nal	Norm	Fail	Post Acc	Indi-cation

REACTOR BUILDING PENETRATION VALVE INFORMATION LEGEND & NOTES

LEGEND

Valve Arrgt - Refer to Figure 6-9.
Qty - Quantity of comparable penetration valves shown
Type - Valve types:

SH (Shut Off Valve) - gate, globe, ball, plug, butterfly, diaphragm or other type on/off valve with the ability to shut off flow.
CK (Check Valve) - stop check, swing check, tilting disc check, lift check, or other type of check valve whose function is to prevent flow in the reverse direction.

Size - Valve size in inches
Oper - Valve operator types

MAN - manual
EMO - electric motor operator
AIR - diaphragm operator
HYD - hydraulic operator
SOL - solenoid

Signal - Noted "ES" if the valve receives an Engineered Safeguards signal. Refer to Section 7.3 for further discussion of ES signals.
Valve Positions - Norm: position during normal operation, Fail: position without operator motive force, Post Acc: desired post-accident position

OP - Open
CL - Closed
AI - As Is

Indication - remote valve position indication

NOTES

1. Penetrations 7, 54, 60, and 61 outboard isolation valves fail open on loss of ES power to their associated solenoid valves provided air is available to open the valves. With ES power energizing the solenoid valves, the isolation valves close with or without supply air.
2. Although initially closed for Reactor Building isolation, valves associated with penetrations 5a (Hydrogen Recombiner Portion), 24, 42, 60, and 61 can be opened for post-accident hydrogen control.
3. For Penetration 60, the Unit 3 outside, manual penetration valve is ½ inch while the Unit 1 and Unit 2 outside, manual penetration valve is 1 inch.
4. For Penetration 60, the Unit 3 inside, non-ES actuated penetration valve is 3 inch while the Unit 1 and Unit 2 inside, non-ES actuated penetration valve is 2 inch.
5. For Penetration 5b, there are two outside penetration solenoid valves for Units 1 and 2. There are two outside penetration manually operated valves for Unit 3. Only the solenoid valves on Units 1 & 2 fail closed and have remote position indication.
6. For Penetration 10a, the Unit 2 outside penetration valve is 1.5 inches while the Unit 1 & 3 outside penetration valves are 1 inch.
7. For Penetration 7, the Unit 2 inside penetration valve is 3 inch while the Unit 1 and Unit 3 inside penetration valves are 4 inch.
8. For Penetration 53a, the Unit 1 outside penetration valve is 1 inch while the Unit 2 and Unit 3 outside penetration valves are ½ inch.
9. For Penetration 38, the Unit 1 outside penetration valve is 1.5 inches while the Unit 2 and Unit 3 outside penetration valves are 2 inches.
10. For Penetration 51, the Unit 1 outside penetration valve is 1 inch while the Unit 2 and Unit 3 outside penetration valves are ½ inch.
11. For Penetration 8a & 8b, the Unit 1 inside and outside penetration valve sizes are 1.5 inches. The Units 2 and 3 inside and outside penetration valve sizes are 1 inch.
12. For Penetration 24 & 42, the Unit 1 and 3 outside penetration valve sizes are ½ inch while the Unit 2 outside penetration valve sizes are ⅜ inch.
13. For Penetration 58a, the two Unit 2 inside penetration valve sizes are 1 inch while the two Unit 3 inside penetration valve sizes are ¾ inch and 1 inch.
14. Although initially closed for Reactor Building isolation, valves associated with penetrations 57 and 62 (DHR Drop Line) can be opened for post-accident boron dilution.
15. Although initially closed for Reactor Building isolation, valves associated with penetration 5b may be opened post-accident for post accident liquid samples (PALS).
16. Penetration number applies to Unit 1 only.
17. Penetration number applies to Unit 2 and Unit 3 only.

Table 6-15. Emergency Core Cooling Systems Performance Testing

High Pressure Injection Pumps	One of two pumps operates continuously. The other pump will be operated periodically.
High Pressure Injection Line Valves	The remotely operated stop valves in each line are opened partially one at a time. The flow monitors will indicate flow through the lines.
High Pressure Injection Pump Suction Valves	The valves are opened and closed individually and console lights monitored to indicate valve position.
Low Pressure Injection Pumps	Pumps are used in normal service for shutdown cooling. These pumps are tested singly for operability by opening the borated water storage tank outlet valves and the bypass valves in the borated water storage tank fill line. This allows water to be pumped from the borated water storage tank through each of the injection lines and back to the tank.
Borated Water Storage Tank Outlet Valves	The operational readiness of these valves is established in completing the pump operational test discussed above. During this test, each valve is tested separately.
Low Pressure Injection Valves	With pumps shut down and borated water storage tank outlet valves close, these valves can be opened and reclosed by operator action.
Sump Recirculation Suction Valves	With low pressure injection pumps shut down, operation of these valves can be checked.
Check Valves in Core Flooding Injection	With the reactor shut down, the check valves in each core flooding line Lines are checked for operability by closing the isolation valves, reducing the Reactor Coolant System pressure to provide ΔP slightly above the check valve opening pressure, and opening the isolation valves. Check valve operability is shown by tank pressure and level changes.

9 **Table 6-16. Deleted Per 1999 Update**

9 **Table 6-17. Deleted Per 1999 Update**

5 **Table 6-23. Containment Structural Heat Sink Data**

	Heat Sink	Painted Material Thickness (ft)	Unpainted Metal Thickness Group	Unpainted Metal Exposed Surface Area (ft^2)	Unpainted Metal Total Mass (lbm)	Total Surface Area (ft^2)
7	Carbon Steel Building Cylinder	0.0208				61,353
7	Concrete Building Cylinder	3.75				61,353
9	Carbon Steel Building Dome	0.0208				16,320
9	Concrete Building Dome	3.25				16,230
9	Carbon Steel Building Base	0.0208				8890
9	Concrete Building Base	8.5				8890
5	Internal Concrete	1.76				66,231
5	Internal Carbon Steel	0.0316				165,400
5	Internal Carbon Steel		1	63,727	300,000	63,727
5	Internal Stainless Steel		2	8628	258,000	8628
5	Internal Aluminum		1	9892	3828	9892
5	Internal Copper		4	727	23,268	727

5 **Table 6-29 (Page 1 of 3). Peak Pressure Mass and Energy Release Data**

	Time (sec)	Gas Mass Flow (lbm/sec)	Gas Enthalpy (Btu/lbm)	Reactor Vessel Side Liquid Mass Flow (lbm/sec)	Liquid Enthalpy (Btu/lbm)	Pressure (psia)	Gas Mass Flow (lbm/sec)	Gas Enthalpy (Btu/lbm)	Steam Generator Side Liquid Mass Flow (lbm/sec)	Liquid Enthalpy (Btu/lbm)	Pressure (psia)
5	0.00	3271.4	1223.6	48383.5	583.1	1933.3	3550.6	1224.4	51140.0	582.4	1925.4
5	0.10	3271.4	1223.6	48383.5	583.1	1933.3	3550.6	1224.4	51140.0	582.4	1925.4
5	0.30	3643.2	1222.8	43916.5	574.1	1560.9	3341.9	1229.0	43030.0	578.3	1541.0
5	0.55	3709.0	1225.9	38740.0	569.0	1502.8	4403.3	1251.3	31366.7	550.7	1486.3
5	0.80	3811.5	1221.7	38435.0	569.1	1459.2	5324.5	1279.3	28600.0	513.5	1452.3
5	1.00	4027.0	1220.9	36905.0	568.2	1426.2	4687.5	1287.6	28530.0	504.4	1423.1
5	1.25	4762.0	1223.8	33393.3	561.1	1384.4	3858.3	1282.7	30653.3	502.2	1380.9
5	1.50	6174.5	1245.1	28600.0	532.5	1342.7	3203.5	1273.5	32845.0	503.3	1338.3
5	1.75	6775.0	1257.6	25976.7	522.5	1301.0	2251.0	1265.5	33950.0	514.6	1297.4
5	2.00	7173.5	1267.9	22770.0	511.0	1256.3	1775.0	1259.2	34090.0	520.0	1253.9
5	2.25	7163.3	1271.3	21260.0	505.8	1207.6	1815.0	1253.4	33380.0	519.4	1208.2
5	2.50	7405.0	1278.9	18345.0	498.2	1161.0	1917.5	1249.0	32185.0	518.4	1163.3
5	2.75	7466.7	1285.3	16163.3	491.2	1117.2	2020.3	1244.0	31046.7	517.2	1115.9
5	3.00	7270.0	1288.9	15345.0	486.2	1085.9	2135.0	1243.6	29785.0	515.9	1083.3
5	3.25	6973.3	1287.8	15593.3	483.5	1061.9	2271.7	1239.9	28266.7	513.0	1058.6
5	3.50	6615.0	1284.2	16150.0	481.1	1037.1	2405.0	1241.2	26750.0	511.0	1030.1
5	3.75	6236.7	1281.1	16906.7	479.5	1018.3	2836.7	1245.6	27066.7	497.9	1012.2
5	4.05	5930.0	1279.4	17200.0	476.6	994.5	3596.7	1255.8	26266.7	483.6	990.5
5	4.30	5795.0	1277.8	17000.0	474.4	975.0	3790.0	1261.2	24450.0	477.5	970.6
5	4.55	5713.3	1280.1	16533.3	472.8	954.5	3886.7	1264.2	22933.3	473.7	950.3
5	4.80	5665.0	1278.9	16050.0	468.5	934.2	3985.0	1266.0	21400.0	468.7	930.1
5	5.05	5613.3	1282.1	15466.7	467.2	917.9	4056.7	1272.0	19900.0	465.0	914.4
5	5.30	5590.0	1281.8	15150.0	464.0	900.2	4115.0	1278.3	18500.0	460.0	896.8
5	5.55	5463.3	1281.3	14600.0	461.9	876.0	4163.3	1278.6	17266.7	455.8	871.9
5	5.80	5310.0	1282.5	14350.0	457.8	853.0	4185.0	1282.0	16000.0	451.3	848.3
5	6.05	5116.7	1280.1	14400.0	455.8	833.0	4233.3	1287.4	14633.3	445.3	827.4
5	6.30	5005.0	1279.7	14100.0	454.6	812.3	4305.0	1293.8	13100.0	441.2	806.4
5	6.55	5083.3	1282.6	12966.7	450.1	789.0	4363.3	1299.5	11666.7	435.7	783.4

5 **Table 6-29 (Page 2 of 3). Peak Pressure Mass and Energy Release Data**

	Time (sec)	Gas Mass Flow (lbm/sec)	Gas Enthalpy (Btu/lbm)	Reactor Vessel Side Liquid Mass Flow (lbm/sec)	Liquid Enthalpy (Btu/lbm)	Pressure (psia)	Gas Mass Flow (lbm/sec)	Gas Enthalpy (Btu/lbm)	Steam Generator Side Liquid Mass Flow (lbm/sec)	Liquid Enthalpy (Btu/lbm)	Pressure (psia)
5	6.80	5240.0	1286.3	11500.0	444.8	763.9	4370.0	1305.5	10450.0	432.5	758.8
5	7.05	5380.0	1291.8	9966.7	442.5	737.6	4356.7	1308.3	9433.3	425.8	732.8
5	7.30	5465.0	1296.4	8650.0	439.9	710.3	4320.0	1313.7	8450.0	421.3	705.7
5	7.55	5473.3	1299.0	7666.7	433.9	681.4	4273.3	1315.9	7566.7	417.6	676.8
5	7.80	5415.0	1302.9	6800.0	433.8	652.0	4205.0	1318.7	6850.0	413.1	647.7
5	8.05	5293.3	1301.6	6200.0	426.3	616.1	4136.7	1319.1	6166.7	408.1	613.5
5	8.30	5115.0	1301.1	5650.0	420.4	573.9	4075.0	1320.3	5400.0	402.8	574.7
5	8.55	4873.3	1300.3	5066.7	414.5	535.0	4023.3	1323.1	4400.0	398.5	537.8
5	8.80	4610.0	1299.4	4550.0	408.8	500.1	3950.0	1330.4	3350.0	395.5	502.8
5	9.05	4360.0	1301.2	4100.0	402.4	465.5	3786.7	1338.0	2633.3	383.5	467.4
5	9.30	4115.0	1302.6	3650.0	398.6	432.0	3535.0	1340.9	2250.0	382.2	433.2
5	9.55	3890.0	1299.1	3300.0	386.9	400.2	3270.0	1337.4	2166.7	370.8	401.3
5	9.80	3675.0	1300.7	2800.0	389.3	367.6	2995.0	1333.9	2100.0	361.9	368.8
5	10.05	3446.7	1300.8	2300.0	375.4	333.8	2710.0	1322.3	2200.0	359.1	336.1
5	10.30	3135.0	1298.3	2100.0	369.1	302.9	2630.0	1319.4	1900.0	347.4	305.3
5	10.55	2770.0	1288.8	2233.3	355.2	268.6	2476.7	1309.6	1700.0	349.0	269.3
5	10.80	2490.0	1289.2	2100.0	350.0	247.7	2340.0	1312.0	1650.0	345.5	254.3
5	11.05	2333.3	1288.6	1900.0	347.4	240.0	2360.0	1309.3	1233.3	343.2	251.7
5	11.30	2140.0	1287.4	1900.0	336.8	223.5	2400.0	1318.8	700.0	342.9	230.6
5	11.55	2013.3	1283.1	1800.0	331.5	209.0	2420.0	1338.8	300.0	333.3	211.6
5	11.80	1860.0	1279.6	1700.0	326.5	194.4	2160.0	1321.8	100.0	400.0	196.6
5	12.05	1680.0	1269.8	1600.0	320.8	165.7	1990.0	1291.5	200.0	333.3	169.6
5	12.30	1420.0	1264.1	1500.0	310.0	143.0	1885.0	1270.6	150.0	333.3	149.5
5	12.55	1226.7	1241.9	1433.3	295.4	134.6	1720.0	1255.8	66.7	300.0	136.5
5	12.80	1050.0	1233.3	1350.0	292.6	114.6	1550.0	1248.4	50.0	400.0	118.9
5	13.05	916.7	1232.7	1266.7	289.5	106.1	1366.7	1241.5	33.3	200.0	113.5
5	13.30	855.0	1216.4	1250.0	288.0	102.8	1210.0	1231.4	50.0	200.0	107.3
5	13.55	736.7	1203.6	1066.7	284.4	88.1	996.7	1220.7	33.3	500.0	93.3

5 **Table 6-29 (Page 3 of 3). Peak Pressure Mass and Energy Release Data**

	Time (sec)	Gas Mass Flow (lbm/sec)	Gas Enthalpy (Btu/lbm)	Reactor Vessel Side Liquid Mass Flow (lbm/sec)	Liquid Enthalpy (Btu/lbm)	Pressure (psia)	Gas Mass Flow (lbm/sec)	Gas Enthalpy (Btu/lbm)	Steam Generator Side Liquid Mass Flow (lbm/sec)	Liquid Enthalpy (Btu/lbm)	Pressure (psia)
5	13.80	570.0	1193.0	800.0	275.0	73.2	750.0	1200.0	100.0	200.0	76.6
5	14.05	400.0	1166.7	600.0	277.8	64.4	410.0	1178.9	66.7	250.0	66.9
5	14.30	310.0	1161.3	450.0	300.0	60.3	15.0	1333.3	0.0	180.0	61.4
5	14.55	230.0	1188.4	400.0	258.3	55.9	0.0	1150.0	0.0	180.0	57.6
5	14.80	180.0	1138.9	250.0	300.0	53.9	0.0	1150.0	0.0	180.0	55.8
5	14.90	180.0	1138.9	250.0	300.0	53.9	0.0	1150.0	0.0	180.0	55.8
5	14.91	445.7	1188.3	715.7	279.7	87.6	13.3	1191.2	0.0	180.0	80.9
5	17.45	445.7	1188.3	715.7	279.7	87.6	13.3	1191.2	0.0	180.0	80.9
5	22.50	774.0	1215.3	1196.0	285.5	112.9	11.2	1214.3	0.0	180.0	105.1
5	27.50	543.4	1207.2	1358.0	280.9	93.5	15.6	1230.8	0.0	180.0	93.1
5	32.50	219.0	1190.9	1948.0	277.2	79.4	10.6	1245.3	0.0	180.0	78.8
5	37.50	121.6	1187.5	1384.0	276.7	78.2	15.2	1223.7	0.0	180.0	78.0
5	42.50	98.6	1188.6	2390.0	276.5	78.2	19.8	1252.5	0.0	180.0	78.4
5	47.50	49.8	1184.7	980.0	276.1	75.7	18.6	1268.8	0.0	180.0	75.7
5	50.00	49.8	1184.7	980.0	276.1	75.7	18.6	1268.8	0.0	180.0	75.7
5	50.01	27.6	1183.6	320.0	277.1	75.6	20.9	1258.0	0.0	180.0	75.6
5	57.50	27.6	1183.6	320.0	277.1	75.6	20.9	1258.0	0.0	180.0	75.6
5	73.00	14.4	1186.2	47.5	276.3	75.8	21.3	1261.8	0.0	180.0	75.8
5	89.00	12.6	1183.2	34.4	272.7	77.2	22.3	1261.2	0.0	180.0	77.2
5	105.00	24.8	1189.4	994.4	276.6	78.5	26.8	1268.7	0.0	180.0	78.3
5	121.00	14.4	1187.0	324.4	275.5	77.1	19.3	1263.0	0.0	180.0	76.9
5	137.00	6.5	1182.7	0.0	180.0	77.8	23.2	1264.2	0.0	180.0	77.6
5	153.00	15.9	1192.2	610.6	276.4	80.1	28.9	1272.1	0.0	180.0	79.8
5	169.00	18.8	1186.1	845.6	277.2	78.5	22.8	1261.0	0.0	180.0	78.2
5	178.55	2.9	1222.2	0.0	180.0	76.4	19.4	1283.3	0.0	180.0	76.2
5	180.10	2.9	1222.2	0.0	180.0	76.4	19.4	1283.3	0.0	180.0	76.2

	Table 6-30 (Page 1 of 4). RELAP5 Long-Term Mass and Energy Release Data										
9	Time (sec)	Gas Mass Flow (lbm/sec)	Gas Enthalpy (Btu/lbm)	Reactor Vessel Side Liquid Mass Flow (lbm/sec)	Liquid Enthalpy (Btu/lbm)	Pressure (psia)	Gas Mass Flow (lbm/sec)	Gas Enthalpy (Btu/lbm)	Steam Generator Side Liquid Mass Flow (lbm/sec)	Liquid Enthalpy (Btu/lbm)	Pressure (psia)
5	0.50	2.0	1190.4	47261.0	557.6	1683.3	456.0	1194.1	25759.0	541.4	1636.0
5	1.50	43.4	1189.9	41385.0	555.6	1099.5	684.5	1193.4	25238.0	544.5	998.8
5	2.50	271.0	1189.4	32454.0	558.7	1103.1	1152.0	1196.1	19507.0	539.7	934.8
5	3.50	757.9	1190.5	27480.0	557.0	1085.1	1511.2	1198.5	14977.0	528.6	810.8
5	4.50	1335.2	1192.3	22590.0	552.9	1036.0	2419.4	1206.0	11285.0	492.7	690.1
5	5.50	1554.8	1193.7	19020.0	546.1	964.5	2607.0	1208.6	8904.0	478.0	611.6
5	6.50	1544.2	1195.9	18040.0	531.6	899.2	2553.9	1209.5	8150.0	471.2	562.4
5	7.50	1551.3	1197.8	16760.0	520.9	843.6	2743.0	1212.5	5980.0	457.7	506.4
5	8.50	1712.8	1200.1	14220.0	512.7	777.7	2724.0	1213.3	4970.0	447.9	461.5
5	9.50	2075.4	1205.1	13030.0	491.9	694.1	2607.0	1214.0	4340.0	439.2	421.9
5	10.50	2120.0	1207.1	10960.0	478.1	611.3	2504.0	1215.3	3500.0	427.7	377.5
5	11.50	2014.0	1208.5	9230.0	459.4	514.9	2383.0	1216.1	2610.0	414.6	329.4
5	12.50	985.0	1207.1	11760.0	420.1	393.8	2216.0	1217.1	1830.0	403.3	281.8
5	13.50	770.0	1205.2	12220.0	392.0	289.6	2042.0	1217.9	1220.0	386.9	239.4
5	14.50	459.0	1198.3	10560.0	365.5	216.8	1849.0	1217.4	720.0	372.2	198.6
5	15.50	220.0	1195.5	8950.0	336.3	165.5	1664.0	1220.0	240.0	370.8	160.4
5	16.50	143.0	1202.8	8690.0	316.5	151.6	1428.0	1217.1	120.0	333.3	130.7
5	17.50	222.0	1193.7	8660.0	308.3	145.8	1182.0	1215.7	50.0	340.0	105.3
5	18.50	227.0	1185.0	8620.0	277.3	104.7	983.0	1211.6	30.0	266.7	84.7

5 Table 6-30 (Page 2 of 4). RELAP5 Long-Term Mass and Energy Release Data

		Reactor Vessel Side					Steam Generator Side				
	Time (sec)	Gas Mass Flow (lbm/sec)	Gas Enthalpy (Btu/lbm)	Liquid Mass Flow (lbm/sec)	Liquid Enthalpy (Btu/lbm)	Pressure (psia)	Gas Mass Flow (lbm/sec)	Gas Enthalpy (Btu/lbm)	Liquid Mass Flow (lbm/sec)	Liquid Enthalpy (Btu/lbm)	Pressure (psia)
5	19.50	258.0	1189.9	7090.0	273.6	71.2	744.0	1205.7	40.0	325.0	72.7
5	20.50	84.0	1178.6	7750.0	242.6	55.9	527.0	1195.5	130.0	284.6	67.3
5	21.50	23.0	1130.4	6440.0	201.9	42.2	437.0	1187.6	170.0	270.6	65.6
5	22.50	14.0	1214.3	6100.0	177.1	43.4	387.0	1191.2	180.0	272.2	64.9
5	23.50	11.0	1090.9	5400.0	168.5	46.2	352.0	1190.3	180.0	266.7	64.2
5	24.50	9.0	1333.3	4730.0	169.1	48.7	308.0	1188.3	170.0	288.2	63.7
5	25.50	8.0	1250.0	4120.0	172.3	48.2	257.0	1206.2	140.0	264.3	63.4
5	26.00	8.0	1250.0	4120.0	172.3	48.2	257.0	1206.2	140.0	264.3	63.4
5	26.01	0.5	1000.0	179.0	173.2	53.0	53.4	1282.8	8.0	250.0	62.7
5	31.00	0.5	1000.0	179.0	173.2	53.0	53.4	1282.8	8.0	250.0	62.7
5	41.00	210.6	1184.2	1163.0	221.0	61.4	240.9	1210.1	88.0	269.3	61.9
5	51.00	213.1	1181.1	278.0	266.2	61.6	193.5	1195.4	182.0	264.8	61.1
5	61.00	132.5	1182.6	173.0	265.9	60.4	152.0	1194.1	157.0	264.3	60.2
5	71.00	111.0	1182.9	141.0	255.3	59.4	132.8	1192.0	166.0	261.5	59.3
5	81.00	72.7	1182.9	93.0	268.8	58.8	113.2	1188.2	179.0	262.6	58.7
5	91.00	58.1	1184.2	57.0	263.2	58.6	101.6	1188.0	157.0	260.5	58.5
5	101.00	37.5	1186.7	37.0	243.2	58.6	91.5	1185.8	144.0	261.8	58.5
5	111.00	66.8	1187.1	101.0	267.3	58.8	93.2	1185.6	137.0	260.6	58.5
5	121.00	177.1	1183.5	230.0	260.9	59.4	121.3	1183.0	200.0	261.0	58.6

5 Table 6-30 (Page 3 of 4). RELAP5 Long-Term Mass and Energy Release Data

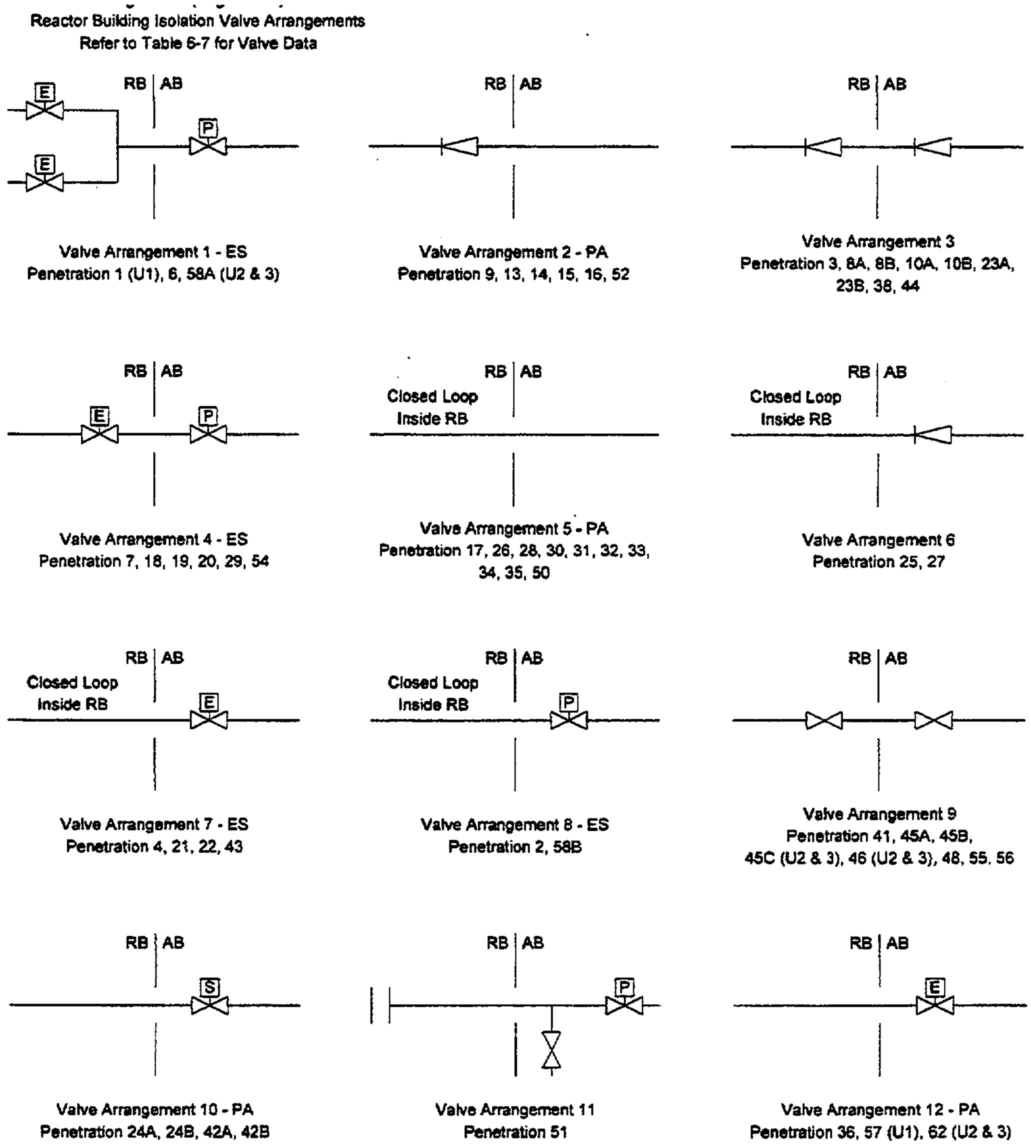
	Time (sec)	Gas Mass Flow (lbm/sec)	Gas Enthalpy (Btu/lbm)	Reactor Vessel Side Liquid Mass Flow (lbm/sec)	Liquid Enthalpy (Btu/lbm)	Pressure (psia)	Gas Mass Flow (lbm/sec)	Gas Enthalpy (Btu/lbm)	Steam Generator Side Liquid Mass Flow (lbm/sec)	Liquid Enthalpy (Btu/lbm)	Pressure (psia)
5	131.00	166.7	1180.6	209.0	263.2	59.2	115.7	1185.0	148.0	262.8	58.6
5	141.00	114.1	1183.2	81.0	259.3	58.5	85.7	1182.0	131.0	259.5	58.5
5	151.00	97.2	1186.2	108.0	259.3	58.5	79.1	1183.3	141.0	261.7	58.5
5	161.00	95.2	1180.7	110.0	254.6	58.4	74.9	1180.2	149.0	261.1	58.5
5	171.00	52.4	1185.1	32.0	281.3	58.4	57.2	1181.8	134.0	260.5	58.4
5	176.00	52.4	1185.1	32.0	281.3	58.4	57.2	1181.8	134.0	260.5	58.4
5	176.01	29.4	1182.2	114.1	257.7	57.2	18.7	1206.7	4.8	260.3	57.2
5	236.05	29.4	1182.2	114.1	257.7	57.2	18.7	1206.7	4.8	260.3	57.2
5	356.10	21.5	1180.0	336.5	240.7	54.6	8.9	1188.7	8.3	256.6	54.7
5	476.10	23.4	1177.3	452.8	217.6	52.0	9.3	1179.7	22.3	253.6	52.0
5	596.10	23.8	1179.9	505.9	214.5	49.4	4.5	1187.5	1.8	247.6	49.4
5	716.10	12.2	1176.9	520.3	202.3	47.2	3.8	1186.4	0.8	260.0	47.2
5	836.10	10.5	1177.6	538.9	186.6	45.7	2.8	1188.8	0.1	100.0	45.8
5	956.10	9.3	1178.6	543.3	185.3	44.6	2.2	1193.8	0.0	180.0	44.6
5	1077.10	8.4	1175.8	529.0	180.2	43.5	1.5	1196.8	0.0	180.0	43.5
5	1198.10	7.9	1174.6	548.0	178.7	42.4	1.1	1209.0	0.0	180.0	42.4
5	1318.10	6.3	1173.5	547.3	178.2	41.2	1.3	1206.5	0.0	180.0	41.2
5	1438.10	8.0	1173.8	527.3	171.7	40.1	0.9	1213.6	0.0	180.0	40.1
5	1558.10	13.6	1170.9	595.8	195.4	39.0	16.8	1183.7	35.3	235.4	39.0

5 **Table 6-30 (Page 4 of 4). RELAP5 Long-Term Mass and Energy Release Data**

	Time (sec)	Gas Mass Flow (lbm/sec)	Gas Enthalpy (Btu/lbm)	Reactor Vessel Side Liquid Mass Flow (lbm/sec)	Liquid Enthalpy (Btu/lbm)	Pressure (psia)	Gas Mass Flow (lbm/sec)	Gas Enthalpy (Btu/lbm)	Steam Generator Side Liquid Mass Flow (lbm/sec)	Liquid Enthalpy (Btu/lbm)	Pressure (psia)
5	1678.35	6.4	1172.3	429.9	187.6	37.9	2.5	1207.9	0.1	100.0	37.9
5	1769.35	35.0	1171.1	897.6	192.0	38.0	37.7	1208.5	51.9	233.2	37.1
5	1800.10	35.0	1171.1	897.6	192.0	38.0	37.7	1208.5	51.9	233.2	37.1

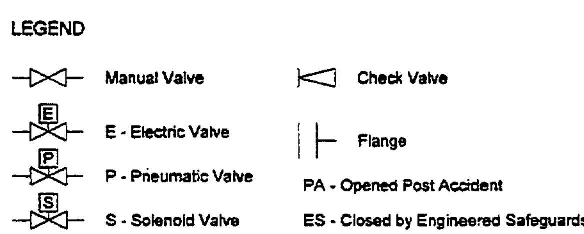
9 **Table 6-33. NPSH Available and Required for LPI and BS Pumps (Limiting Flow Case)**

		Flow	NPSHr	NPSHa
9		**Flow**	**NPSHr**	**NPSHa**
9	BS	1150 gpm	17 ft	19.76 ft
9	LPI	3310 gpm	13 ft	18.48 ft



NOTES

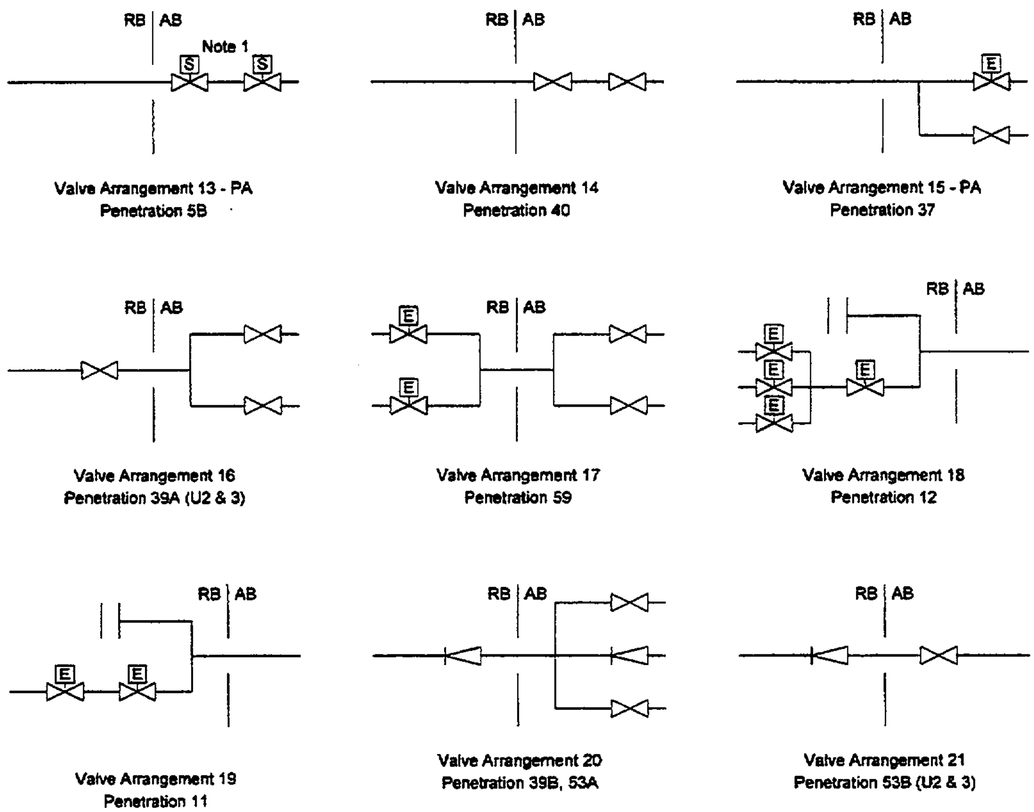
Note 1: For Penetration 5B, the drawing shown represents Units 1 & 2. Unit 3 has double manual valves rather than double solenoid valves.
General Note: Branch lines are not shown to normally closed valves for vents, drains and miscellaneous services (including relief valves).



9
9

Figure 6-9 (Part 1 of 3). Reactor Building Isolation Valve Arrangements

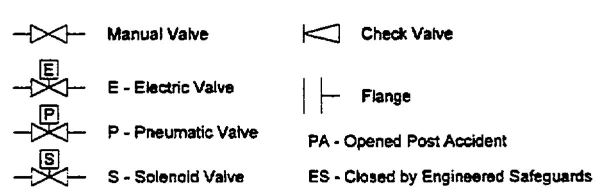
Reactor Building Isolation Valve Arrangements
Refer to Table 6-7 for Valve Data



NOTES

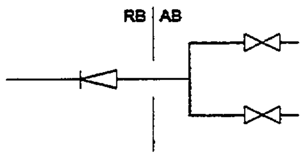
General Note: Branch lines are not shown to normally closed valves for vents, drains and miscellaneous services (including relief valves).

LEGEND

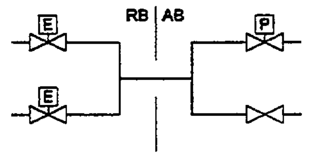


9 **Figure 6-9 (Part 2 of 3).**
9 **Reactor Building Isolation Valve Arrangements**

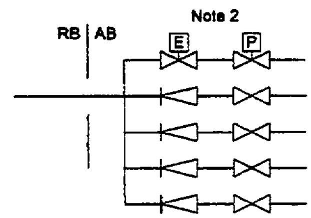
Reactor Building Isolation Valve Arrangements
Refer to Table 6-7 for Valve Data



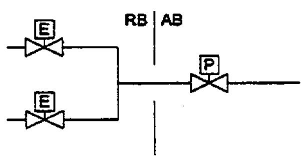
Valve Arrangement 22
Penetration 49 (U1)



Valve Arrangement 23 - ES, PA
Penetration 60



Valve Arrangement 24 - ES, PA
Penetration 5A



Valve Arrangement 25 - ES, PA
Penetration 61

NOTES

Note 2: For Penetration 5A, the drawing shown represents Units 2 & 3. For Unit 1, the electric and pneumatic valves are reversed.

General Note: Branch lines are not shown to normally closed valves for vents, drains and miscellaneous services (including relief valves).

LEGEND

- Manual Valve
- E - Electric Valve
- P - Pneumatic Valve
- S - Solenoid Valve
- Check Valve
- Flange
- PA - Opened Post Accident
- ES - Closed by Engineered Safeguards

9
9

Figure 6-9 (Part 3 of 3).
Reactor Building Isolation Valve Arrangements

3 Figure 6-10.
3 Deleted per 1993 Update

3 Figure 6-11.
3 Deleted per 1993 Update

3 Figure 6-12.
3 Deleted per 1993 Update

9 Figure 6-13.
9 Deleted per 1999 Update

9 Figure 6-14.
9 Deleted per 1999 Update

1 Figure 6-15.
1 Deleted Per 1991 Update