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4.0 Reactor

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4.1 Summary Description

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

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

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

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 control rod assemblies and burnable poison rod assemblies. There are 2 axial power shaping rod (APSR) designs. See Table 4-22 for information on compatibility of the 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 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.

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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 and Section 4.2.2.

4.2.1 Design Bases - Fuel System Design

The fuel is designed to meet the performance objectives specified in Section 4.2.1.1 without exceeding the limits of design and operation specified in Section 4.2.1.2.

4.2.1.1 Fuel System Performance Objectives

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

The fuel rod cladding is designed to maintain its integrity for the anticipated operating transients 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

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

1. The core will have sufficient reactivity to produce the design power level and lifetime without exceeding the control capacity or shutdown margin.
2. Fuel assemblies have been designed for the maximum burnups shown in Table 4-2.
3. Power histories must be bounded by those assumed within generic mechanical and thermal hydraulic (fuel assembly) analyses. If they are not bounded, acceptable reanalyses shall be performed.
4. The maximum feed fuel enrichment is constrained by the maximum allowed in the Technical Specifications (Spent Fuel Pool storage requirements).
5. Values of important core safety parameters predicted for the cycle have been verified to be conservative with respect to their values assumed in the Chapter 15 safety/accident (and any other pertinent) analyses. If they are not conservative, acceptable reanalyses shall be performed.

   Controlled reactivity insertion rates due to a single CRA group withdrawal shall be limited to a maximum value assumed within the Section 15.3 Rod Withdrawal Accident at Rated Power, and within the Section 15.2 Startup Accident. Controlled reactivity insertion rates due to soluble boron removal shall be limited to a maximum value assumed within the Section 15.4 Moderator Dilution Accident.

   The overall power coefficient is negative in the power operating range. However, as described within Chapter 15, the control system is capable of compensating for reactivity changes resulting from either positive or negative nuclear coefficients.
6. Reasonable and permissive reactor control and maneuvering procedures during nominal operation and during transients will not produce unacceptable peak-to-average power

distributions. This, along with criteria 7 and 8, below, preserves the LOCA linear heat rate, linear heat rate to melt (LHRTM), and DNBR limits.

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

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

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

4.2.1.2.2 Reactivity Control Limits

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

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

2. A shutdown margin of at least 1.0% $\Delta\rho$ shall be maintained throughout core life with the most reactive CRA stuck in the fully withdrawn position. However, for shutdown margin calculations with all control rods verified fully inserted by two independent means, it is not necessary to account for a stuck rod in the shutdown margin calculation. (Reference 24)

3. CRA withdrawal rate (as listed within Chapters 7 and 15) shall limit the maximum reactivity insertion rate to that assumed within the Section 15.3 Rod Withdrawal Accident at Rated Power, and within the Section 15.2 Startup Accident.

4. Boron dilution rate (as listed within Chapter 15) shall limit the maximum reactivity insertion rate to that assumed within the Section 15.4 Moderator Dilution Accident.

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

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

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

8. During Refueling (Mode 6), shutdown margin is assured by administrative means in various procedures. This is consistent with Duke's response to NRC Bulletin 89-03, "Potential Loss of Required Shutdown Margin During Refueling Operations" (References 22, 23).

For more detail, refer to Section 4.3.

4.2.1.2.3 Thermal and Hydraulic Limits

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

1. The fuel pin must be designed so that the maximum fuel temperature does not exceed the fuel melting limit at any time during core life. The TACO3 computer program is used to verify heat rate capacity (Reference 2).
2. The minimum allowable DNBR during steady-state operation and anticipated transients is: (a) 1.18 with the BWC correlation (Reference 6), (b) 1.19 with the BWU-Z correlation with FB11 multiplication factor (Reference 19), or (c) a proprietary value with the BHTP correlation (Reference 21).
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.

4.2.1.2.4 Mechanical Limits

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

Spacer grids, located along the length of the fuel assembly, position fuel rods in a square array, and are designed to maintain fuel rod spacing during core operation, handling, and shipping. Spacer-grid to fuel-rod contact loads are established to minimize fretting, but 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. The fuel rods are designed to meet the following mechanical limits:

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

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

5. Fuel Thermal Analysis

   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 pin pressure is maintained below the value which would cause clad lift off. Refer to Section 4.2.3.1.3 for design evaluations of the fuel thermal analyses.

6. The cladding oxide thickness for the highest burnup rod in each sub-batch is limited to 100 μm as 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 location. The reactivity of the core is controlled by 61 control rod assemblies (CRAs) and 8 axial power shaping rod assemblies (APSRAs), a variable number of burnable poison rod assemblies (BPRAs), and soluble boron in the coolant. APSRAs are 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 that the net effect of the power Doppler and moderator temperature coefficients at power will be negative through core lifetime. The mechanical and geometric configuration of the CRAs and BPRAs permit full interchangeability in any fuel assembly.

Deleted paragraph(s) per 2005 update.

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

Fuel assembly designs (References 17, 18, and 21) are limited to those that have been analyzed with the applicable NRC approved codes and methods. A limited number of lead test assemblies (LTAs) that have not completed representative testing may be placed in non-limiting core locations.

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

Cladding, fuel pellets, end caps, and fuel support components form a “fuel rod”. Two hundred and eight fuel rods, sixteen control rod guide tubes, one instrumentation tube assembly, eight spacer grids, and two end fittings make up the basic “Fuel Assembly” (Figure 4-37). Some fuel assembly designs prior to MK-B-HTP had seven segmented spacer sleeves. The guide tubes, spacer grids, and end fittings form a structural cage to arrange the rods and tubes in a 15 x 15 array. The center position in the assembly is reserved for instrumentation. Control rod guide tubes are located in 16 locations of the array. Use of similar material in the guide tubes and fuel

rods results in minimum differential thermal expansion. Fuel assembly components, materials, and dimensions are tabulated in Table 4-2. Fuel assembly design descriptions are depicted in Table 4-23.

4.2.2.1.2 Fuel Rod

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

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

There are spring spacers located both above and below the pellet stack in the MK-B10D and MK-B10E 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 spring. The MK B-10F and higher fuel assembly designs do not contain a bottom plenum spring.

The fuel rods within an assembly may have differing enrichments radially. Axially, the fuel rods may be of a constant enrichment, or they may be blanketed, which means that a portion of the top and bottom of the fuel stack has a lower enrichment. The function, behavior, and analysis of fuel rods containing axial 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 cladding, and into the plenum spring space. Fuel rod data are given in Table 4-2, and a typical fuel rod is 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 walls of each square opening are integrally punched in the strips.

4.2.2.1.4 Lower End Fittings

The lower end fitting positions the assembly in the lower grid rib section. During fabrication, the lower 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 assembly. The lower end fittings are of an anti-straddle design which will prevent the fuel assembly from being improperly seated on the lower grid 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.

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

4.2.2.1.6 Guide Tubes

The Zircaloy and M5 guide tubes provide continuous guidance for the control rod assemblies when inserted in the fuel assembly and provide the structural continuity for the fuel assembly. On the MK-B10D to MK-B10F through MK-B-11A designs, the upper guide tube nut is held secure by a crimped locking cup. On the MK B-10G, each guide tube is designed to engage with a locking device on the upper end fitting. MK-B-HTP fuel has a recon crimp top hat nut to secure the upper end fitting to the guide tube. Transverse location of the guide tubes is provided by the spacer grids. The guide tube hole size is 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. MK-B-HTP fuel no longer utilizes spacer sleeves.

4.2.2.2 Lead Test Assembly Programs

The effort to continually improve fuel performance often necessitates use of lead test assemblies (LTA). Per Technical Specifications, Duke is allowed to operate cores with a limited number of LTAs that have not completed representative testing as long as they are located in non-limiting core regions (locations). The use of LTAs allows demonstration of the acceptability of different or improved fuel designs. Demonstration programs provide the fundamental engineering data used to develop computer codes and analytical methods. Demonstration programs also provide representative testing to ensure that a fuel design complies with all fuel safety design bases.

4.2.2.2.1 Current Demonstration Programs

Westinghouse-177 (WH-177)

Beginning with Oconee 3 Cycle 22, Duke installed four WH-177 LTAs in the reactor core. The WH-177's were designed by Westinghouse Corporation and are similar to LTAs previously used at TMI-1 except that they have an improved mid-grid design and three added intermediate support grids. These FAs have a 15 X 15 fuel rod array and are fully compatible with the other Mk-B design FAs in the core.

With regard to interaction with the plant fuel handling equipment, the WH-177 LTAs do require use of a special handling tool in order to receive them into each SFP.

Since the primary purpose of this LTA program is to demonstrate the mechanical and thermal hydraulic compatibility of the WH-177 LTAs at Oconee, these assemblies are subject to a detailed post irradiation examination (PIE) program to verify acceptable performance and validation of the design.

For details on differences between the Westinghouse and AREVA designs, see Table 4-24 and Figure 4-38.

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. Thermal hydraulic design evaluation is presented in Section 4.4.3.

4.2.3.1 Fuel Rod

The basis for the design of the fuel rod is discussed in Section 4.2.1.2. Materials testing and actual operation in reactor service with Zircaloy cladding have demonstrated that Zircaloy-4 and M5 material have sufficient corrosion resistance and mechanical properties to maintain the integrity and serviceability required for design burnup.

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

Fuel assemblies found with damaged or leaking fuel rods, can be reconstituted in order to replace damaged rods. One replacement option is a fuel rod that contains pellets of naturally enriched uranium dioxide (UO_2). Aside from enrichment, this rod is similar in design and behavior to a standard fuel rod and is analyzed using standard approved methods. Another replacement option is a solid filler rod made of stainless steel, Zircaloy or M5. Solid filler rods are useful when grid damage exists. A maximum of 10 such filler rods can be substituted into a single fuel assembly. Fuel assemblies with severe structural damage or with failed pins that can not be completely removed may be recaged or discharged. A recage operation entails transferring all of the sound fuel rods from the damaged cage to a new fuel assembly cage. This new fuel assembly will function the same as the assembly which it replaces. A safety evaluation (generic or core specific) is performed for repaired fuel assemblies to ensure acceptable nuclear, mechanical, and thermal-hydraulic performance. While the focus of this section has been on the replacement of damaged or leaking rods, a sound rod can be replaced. For example, a sound rod may be sent to a hot cell for detailed examination.

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

4.2.3.1.1 Clad Stress and Strain

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

1. Cladding Stress Analysis

   The cladding stress analysis 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 most reload fuel cycle designs.

The static stress analysis uses design stress intensity limits on mechanical properties based on the requirements of ASME code Article III-2000. The design stress intensity value, SM, for Zircaloy-4 and M5 is 2/3 of the minimum non-irradiated yield strength at operating temperature as specified in Reference 2 for Zircaloy-4 and Reference 20 for M5.

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

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

The internal pressure of the peak fuel rod in the reactor will be limited to a proprietary value below that which would cause (1) the fuel-clad gap to increase due to outward cladding creep during steady-state operation and (2) extensive DNB propagation to occur. (Section 4.2.3.1.3)

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:

a. 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.

b. Conservative cladding dimensions with regard to stress.

c. 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.)

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

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

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%.

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

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

$$\text{CladStrain}\,\% = \frac{\text{CladID}_{\text{transient}} - \text{CladID}_{\text{transient beginning}}}{\text{CladID}_{\text{transient beginning}}} \text{ x } 100$$

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

4.2.3.1.2 Cladding Collapse

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

The CROV (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 CROV (Reference 1), are calculated by TACO3 using a (conservative) generic radial power history, and a typical axial flux shape.

The conservative fuel rod geometry and conservative power history are used to predict the number of EFPH (or equivalent burnup) required for complete cladding collapse. To demonstrate acceptability, the maximum 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.

4.2.3.1.3 Fuel Thermal Analysis

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

1. Fuel Pin Pressure Analysis

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

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

a. Maximum Internal Pin Pressure: The fuel rod internal pressure is limited to a proprietary value above the nominal system pressure.

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

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

2. Linear Heat Rate Capability

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

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

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

3. ECCS Analysis Interface Criteria

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

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

4.2.3.1.4 Cladding Corrosion

The cladding oxide thickness, for the highest burnup rod in each sub-batch, is limited to 100 μm on a best estimate basis. References 12, 14, and 21 define the corrosion analysis methodology. If an assembly contains a rod whose predicted oxide thickness is over 100 μm, it can be designated a lead corrosion assembly and continue to operate. Corrosion measurements will be taken on these assemblies after they have been discharged from the core. The total number of lead corrosion assemblies is limited to 8 per cycle. The 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 control rod drive fuel assembly system, the drive was stroked and tripped to duplicate the expected 20-year 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.

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

Refer to Reference 1 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 Areva Fuel Surveillance Program

Areva` 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 Areva fuel design. The operation of all Areva fuel will continue to be closely monitored using activities such as manufacturing reviews, coolant chemistry monitoring, post irradiation examinations, etc. to ensure continued safe and reliable fuel performance. Post irradiation examinations typically perform tests such as visual inspections, fuel assembly growth measurements, spacer grid position determination, fuel assembly bow measurements, shoulder gap measurement, water channel measurements, spring preload, verification of the quick disconnect upper end fitting operation, and other non-destructive testing.

Fuel with suspected defective fuel rods are typically examined and tested for leakage. Leakage verification may utilize an ultrasonic test rig, a vacuum can sipping system, or an in-mast sipping system. The ultrasonic technique looks for water inside the fuel rod. The vacuum and in-mast sipping techniques pull a liquid or gas sample from above the assembly and pipe the sample to a shielded detector in order to look for elevated gaseous fission products emanating from the leaking fuel rod. If these methods indicate that a fuel pin is defective, then the fuel assembly will either be repaired or evaluated for acceptability for use in future cycle designs.

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.

Deleted paragraph(s) per 2005 update .

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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

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

1. The core will have sufficient reactivity to produce the design power level and lifetime without exceeding the control capacity or shutdown margin.
2. Fuel assemblies have been designed for the maximum burnups shown in Table 4-2.
3. Power histories must be bounded by those assumed within generic mechanical and thermal hydraulic (fuel assembly) analyses. If they are not bounded, acceptable reanalyses shall be performed.
4. The maximum feed fuel enrichment is constrained by the maximum allowed in the Technical Specifications (Spent Fuel Pool storage requirements).
5. Values of important core safety parameters predicted for the cycle have been verified to be conservative with respect to their values assumed in the Chapter 15 safety/accident (and any other pertinent) analyses. If they are not conservative, acceptable reanalyses shall be performed.

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

   The overall power coefficient is negative in the power operating range. However, as described within Chapter 15, the control system is capable of compensating for reactivity changes resulting from either positive or negative nuclear coefficients.
6. Reasonable and permissive reactor control and maneuvering procedures during nominal operation and during transients will not produce unacceptable peak-to-average power distributions. This, along with criteria 7 and 8, below, preserves the LOCA linear heat rate, linear heat rate to melt (LHRTM), and DNBR limits.

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

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

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

4.3.2 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).

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 fuel cycle's excess reactivity at various conditions.

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

4.3.2.2 Reactivity Control

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

Soluble Boron

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

Burnable Poison Rod Assemblies (BPRAs)

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

The BPRA loadings and placement are chosen to shape radial power peaks and to decrease initial soluble boron concentration to a level where the BOC moderator temperature coefficient is non-positive above 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 for a physical description of the BPRAs. See the appropriate reload design change report for actual BPRA loadings for any particular cycle.

Control Rod Assemblies

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

Reactivity Control During Refueling

Core subcriticality during refueling operations (Mode 6) is maintained through plant fuel handling procedures that ensure adequate shutdown margin is maintained. (References 20, 21 responses to NRC Bulletin 89-03, “Potential Loss of Required Shutdown Margin During Refueling Operations”).

4.3.2.3 Reactivity Shutdown Analysis

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

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

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

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

4.3.2.4 Reactivity Coefficients

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

4.3.2.4.1 Doppler Coefficient

The Doppler coefficient reflects the change in reactivity as a function of fuel temperature. The Doppler coefficient of reactivity is due primarily to Doppler broadening of the U-238 resonances with increasing fuel temperature. A rise in fuel temperature results in an increase in the effective absorption cross section of the fuel and a corresponding reduction in neutron production. A typical range for the Doppler coefficient under operating conditions would be -1.1×10^{-5} to -1.7×10^{-5} $(\Delta\rho)$/deg F.

4.3.2.4.2 Moderator Void Coefficient

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

4.3.2.4.3 Moderator Pressure Coefficient

The moderator pressure coefficient relates the change in moderator density, resulting from a reactor coolant pressure change, to the corresponding effect on neutron production. This coefficient is opposite in sign and considerably smaller when compared to the moderator temperature coefficient. A typical range of pressure coefficients over a life cycle would be -1.4×10^{-7} to $+4 \times 10^{-6}$ $(\Delta\rho)$/psi.

4.3.2.4.4 Moderator Temperature Coefficient

The moderator temperature coefficient relates a change in neutron multiplication to the change in reactor coolant temperature. Reactors using soluble boron as a reactivity control have a less negative moderator temperature coefficient than do cores controlled solely by movable or fixed CRA. The major temperature effect on the coolant is a change in density. An increasing coolant temperature produces a decrease in water density and an equal percentage reduction in boron concentration. The boron concentration change results in a positive reactivity component by reducing the absorption in the coolant. The magnitude of this component is proportional to the total reactivity held by soluble boron. Distributed poisons (burnable poison rods or inserted control rods) have a negative effect on the moderator coefficient for a specified boron concentration. That is, the moderator coefficient for a system with 1200 ppm boron in the coolant and 1% rod worth inserted will be more negative than for a system with 1200 ppm boron and no rods inserted. Depending on the core size, core loading, and power density, a plant may or may not require additional distributed poisons to yield the appropriate moderator temperature coefficient as determined by the safety analysis and the stability analysis of the core. An example of this, as pertaining to the first cycle, is illustrated in Table 4-7.

Items 4d 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.

The Oconee reactors operate above approximately 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

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

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

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

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

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

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

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

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

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

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

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 ^7Li dilution (starting with 0.5 ppm ^7Li). 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 or the Integrated Control System.

4.3.2.5 Reactivity Insertion Rates

Figure 4-10 displays a typical integrated rod worth of three overlapping rod banks as a function of distance withdrawn. The indicated groups are those used in the core during power operation. Using an assumed nominal of 1.5% $\Delta\rho$ CRA groups and an assumed 30 in./min CRA drive speed in conjunction with the reactivity response given in Figure 4-10 yields a maximum reactivity insertion rate of 1.09 x 10^{-4} ($\Delta\rho$)/sec. The maximum reactivity insertion rate for soluble boron removal, using an assumed boron 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

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

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

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

1. Operate to the scheduled end-of-cycle (EOC) plus a long window for operational uncertainty, potentially using reduction in average coolant temperature and/or coast down in power for reactivity addition and reduction in initial enrichment requirements.
2. The U235 fuel enrichment must be bounded by that listed within the Technical Specifications (Spent Fuel Pool storage requirements).
3. Maximum pin burnup must be bounded by the appropriate limit for a fuel type.
4. Maximum assembly average burnup must be bounded by the appropriate limit for a fuel type.
5. The power histories must be bounded by those used in generic analyses, or provide acceptable results when specifically analyzed.
6. For the current bypass flow assumptions, the typical number of 44 BPRAs gives sufficient margin.

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

1. Moderator temperature coefficient
2. Doppler coefficient
3. Ejected rod worth
4. Dropped rod worth
5. Total/maximum CRA group worth
6. Kinetics parameters
7. Shutdown margin
8. Maximum reactivity insertion rates (due to controlled rod withdrawal and boron dilution)
9. Differential boron worth
10. Boron concentrations

The purpose of a maneuvering analysis is to generate three dimensional power distributions, rod positions, and imbalances for a variety of reasonable and permissive rod positions, xenon distributions, and power levels. The maneuvering analysis can be described as four discrete phases. The first is the nominal fuel cycle depletion performed at a nominal rod index (typically, rod index = 292 and APSRs at 35% withdrawn) to establish a fuel depletion history. The

second is the power maneuver performed at BOC (4 EPFD), at EOC (with appropriate adjustments to ensure critical conditions), and at least one other point in between; APSRs are positioned as necessary to maintain xenon control and to maintain predetermined imbalance limits. The third is to perform control rod and APSR scans at the most severe times of the power maneuver. The fourth step is to perform selected control rod and APSR scans at various nominal depletion steps. Each of these phases involves running multiple three dimensional cases and generation of three dimensional power distributions, rod positions, and imbalances for each case. The data is processed by utility codes to calculate margins to LHRTM, DNBR, and LOCA limiting criteria, and to produce 'fly-speck' plots. Application of appropriate calculational conservatisms are described within References 2, 4, 18, and 22. Note that the derivations of the LHRTM, DNBR, and LOCA limiting criteria have been bounded by limiting power distribution listed within Table 4-1.

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

1. Shutdown margin at HZP, BOC to EOC ≥ 1.0% $\Delta\rho$ (with the most reactive CRA stuck in the fully withdrawn position).
2. Maximum ejected rod worth at HZP, NoXe, BOC and EOC, as bounded by that assumed within the Chapter 15 Rod Ejection Accident.
3. Maximum ejected rod worth at HFP, EqXe, BOC and EOC, as bounded by that assumed within the Chapter 15 Rod Ejection Accident.
4. Maximum dropped rod worth at HFP, NoXe, EOC, as bounded by that assumed within the Chapter 15 Control Rod Misalignment Accident.
5. Maximum dropped rod worth HFP, EqXe, EOC, as bounded by that assumed within the Chapter 15 Control Rod Misalignment Accident.

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

1. In-core detector system (observability and variability) and in-core monitoring software uncertainties
2. Out-of-core to In-core calibration/correlation uncertainty
3. Control rod position uncertainties
4. Flux-flow ratio adjustment
5. Reactor protective system hardware uncertainties
6. Boron concentration and volume uncertainties

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

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

Table 4-9 for a typical reload design, generated with SIMULATE-3, shows that this presumption is valid. The moderator temperature coefficient (MTC), when changing conditions from BOC, HZP, NoXe, and ARO to BOC, 95%fp, NoXe, and ARO, becomes negative; note that the 95% no xenon condition is conservative. Given the change, and startup physics test program criteria of +0.5 x 10^{-4} $\Delta\rho$/deg F, the MTC at 95%fp is negative (approximately -0.7 x 10^{-4} $\Delta\rho$/deg F).

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

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

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

4.3.3.1 Analytical Models

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

4.3.3.1.1 CASMO-3 or CASMO-4/SIMULATE-3-Based Methodology

The CASMO-3/SIMULATE-3-based calculational methods for nuclear design have been reviewed and approved by the NRC in Reference 18. This methodology was first applied during the reload design analysis of Unit 1 Cycle 16. The CASMO-4/SIMULATE-3 based calculation methods for nuclear design have been reviewed and approved by the NRC in Reference 22. This methodology was first applied for the reload design analysis of Unit 2 Cycle 26.

Verification to Measured Data

The verification of the CASMO-3 and CASMO-4/SIMULATE-3-based methods for nuclear design are documented in Reference 18 and Reference 22, respectively.

4.3.3.1.2 Control of Power Distributions

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

During startup testing an out-of-core detector correlation test is performed to calibrate the imbalance as 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.

The out-of-core neutron flux detectors each 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 may 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 sometimes are analysis dependent and are tabulated in References 18 and 22.

4.3.3.1.4 Power Maldistributions

Misaligned Control Rods

The reactor has a control function to protect against a rod out of step with its group. The position of each rod is compared to the average of the group. If an assymetric fault is detected at power levels greater than 60% of rated power, a rod withdrawal inhibit is activated and the Integrated Control System (ICS) runs the plant back to 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 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.

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

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

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

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

Azimuthal Xenon Oscillations

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

Fuel Misloading

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

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

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

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 x 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 x 10^{-6} Δρ/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 x 10^{-4} Δρ/°F_m. Including compounded errors from Figure 4-15, the threshold moderator coefficient is approximately +1 x 10^{-4} Δρ/°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 x 10^{-6} Δρ/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 x 10^{-6} Δρ/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 x 10^{-4} Δρ/°F_m. When compounded errors are considered as in Figure 4-17 the threshold moderator coefficient is approximately +2.5 x 10^{-4} Δρ/°F_m.

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

1. The minimum moderator temperature coefficient (MTC) threshold value, as listed within Table 4-7, is +1.0 x 10^{-4} Δρ/degF. The most positive moderator temperature coefficient assumed within Chapter 15 safety/accident analysis is less than the threshold value.
2. There is considerable margin for a BPRA core (i.e., Oconee 2, Cycle 1 within Reference 5) between the Table 4-7 threshold MTC and the calculated threshold MTC, even when compounded errors are taken into account.
3. Current nuclear design bases require that the overall power coefficient be negative in the power operating range. As such, any azimuthal oscillations within current cores are self-damping by virtue of 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

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

1. Pre-critical Test Phase
    a. Control Rod Drop Time
2. Zero Power Physics Test Phase
    a. Critical Boron Concentration
    b. Moderator Temperature Coefficient
    c. Control Rod Worth
3. Power Escalation Test Phase
    a. Low Power Testing (5-30% FP)
    b. Intermediate Power Testing (40-75% FP)
    c. Full Power Testing (90-100% FP)

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

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

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

4.3.5 Pre-Critical Test Phase

4.3.5.1 Control Rod Drop Time

4.3.5.1.1 Plant Conditions

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

4.3.5.1.2 Procedure

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

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

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

4.3.5.1.3 Follow-Up Actions

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

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

4.3.6 Zero Power Physics Test Phase

4.3.6.1 Critical Boron Concentration

4.3.6.1.1 Plant Conditions

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

4.3.6.1.2 Procedure

The ARO critical boron concentration is measured by establishing an equilibrium RCS boron concentration near the predicted ARO critical boron concentration. Control Rod Groups 1 through 7 are fully withdrawn. Control Rod Group 8 is maintained at the nominal designed position. A sample of the equilibrium boron concentration is taken and analyzed to determine the critical boron concentration. Since it may not be practical to establish critical equilibrium conditions with Group 7 fully withdrawn, the small amount of inserted worth of Group 7 or worth of Group 8 (from its nominal designed position) is measured by a reactivity calculation or Reactimeter. This reactivity is then used to adjust the boron 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} Δk/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} Δk/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} Δk/k/°F acceptance criterion is exceeded and the maxium positive MTC is less than 0.5×10^{-4} Δk/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 corrective actions approved by representatives from Oconee Nuclear Station Nuclear Engineering and Regulatory Compliance, and General Office Nuclear Engineering prior to 100% FP.

4.3.6.3 Control Rod Worth

4.3.6.3.1 Plant Conditions

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

4.3.6.3.2 Procedure

The measurement of regulating rod group worths begin from a critical steady state condition with all regulating rod groups withdrawn as far as possible (i.e., within 0.12% Δk/k of ARO). From this point a boron concentration necessary to deborate control rod Groups 7 and 6 to fully inserted is calculated (See Reference 19). The resulting reactivity change during deboration is compensated for by discrete insertion of control rods with both signals being recorded by a reactivity calculation or Reactimeter. Integral rod worths are calculated by summing the differential rod worths for each control rod group.

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

4.3.6.3.3 Follow-Up Actions

If the difference between the measured and predicted total worth of control rod Groups 6 and 7 exceeds 10%, then, following calculation of the minimum control rod position for which the worth of the control rods withdrawn would equal 1% Δk/k, additional control rod group worths will be measured. The worths of additional control rod groups will be measured in sequence from Group 5 to Group 2, until either the difference between the measured and predicted total worth of all control rod groups measured does not exceed 10%, or the calculated minimum control rod position is reached. In the latter case, control rod worth testing will halt. The results will be reviewed by cognizant engineers to determine the appropriate additional corrective actions required to resolve the discrepancy. This review will be completed with the results and the 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 difference between the measured and predicted control rod worths of any of the individual control rod groups exceeds 15%, the results will be reviewed by cognizant engineers to determine the appropriate corrective actions required to resolve the discrepancy. This review will be completed prior to reaching 100% FP.

4.3.7 Power Escalation Test Phase

4.3.7.1 Low Power Testing

4.3.7.1.1 Plant Conditions

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

4.3.7.1.2 Procedure

Once the unit is between 5 and 30% FP, the output of the plant OAC reactor calculations program is analyzed. This program processes the signals from fixed incore detectors and

provides a relative core power distribution as output. The incore detector outputs are checked in order to identify malfunctioning detectors. After these have been eliminated, the results for corrected assembly power in functioning instrumented symmetric core locations are compared.

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

4.3.7.1.3 Follow-Up Actions

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

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

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

4.3.7.2 Intermediate Power Testing

4.3.7.2.1 Plant Conditions

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

4.3.7.2.2 Procedure

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

The results are reviewed by the Test Coordinator. If, for each assembly location with normalized measure power greater than 1.0, the measured radial peaking factor does not exceed the predicted radial peaking factor by more than 12.0% of the predicted radial peaking factor, and if, for each assembly location with normalized measured power greater than 1.0, the measured total peaking factor does not exceed the predicted total peaking factor by more than 15.0% of the predicted total peaking factor, and if the RMS difference between predicted and measured radial peaking factors is less than 0.075, then the results are acceptable.

4.3.7.2.3 Follow-Up Actions

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

Also, the observed parameter 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 any further escalation of power.

If any acceptance criteria are exceeded, 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 escalation to 100% FP.

4.3.7.3 Full Power Testing

4.3.7.3.1 Plant conditions

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

4.3.7.3.2 Procedure

Once the unit is between 90 and 100% FP with Xenon equilibrium, the output of the plant OAC reactor calculations program is analyzed. This program processes the signals from fixed incore detectors and provides a relative core power distribution as output. The incore detector outputs are checked, in order to identify malfunctioning detectors. After these have been eliminated, the radial and total peaking factors obtained from the OAC are compared with the values calculated as part of the reload design process on an eighth-core basis. The results are reviewed by the Test Coordinator. If, for each assembly location with normalized measure power greater than 1.0, the measured radial peaking factor does not exceed the predicted radial peaking factor by more than 12.0% of the predicted radial peaking factor, and if, for each assembly location with normalized measured power greater than 1.0, the measured total peaking factor does not exceed the predicted total peaking factor by more than 15.0% of the predicted total peaking factor, and if the RMS difference between predicted and measured radial peaking factors is less than 0.075, then the results are acceptable.

4.3.7.3.3 Follow-Up Actions

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

Also, the observed parameter 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 the recommended corrective actions approved by representatives from Oconee Nuclear Station Nuclear Engineering and Regulatory Compliance, and General Office Nuclear Engineering prior to any escalation of power.

If any acceptance criteria are exceeded, 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 any escalation of power.

4.3.7.4 Reactivity Anomaly

4.3.7.4.1 Plant Conditions

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

4.3.7.4.2 Procedure

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

The results are reviewed by the Site Nuclear Engineering Supervisor and are compared with the normalized predicted ARO boron concentration for the time in the cycle at which the measurement was taken.

The ARO boron concentration procedure is also used to ensure that the curve to maintain boron concentration for SSF operability is conservative. This is done by ensuring that the Measured ARO Boron minus Predicted ARO Boron is ≥ -25 ppmB. 25 ppmB is used because shutdown boron concentrations for SSF operability supplied by Nuclear Design contain a 25 ppmB analytical uncertainty. If the difference between measured and predicted ARO boron concentration values does not exceed 50 ppm or < -25 ppm boron for SSF subcriticality (see Section 9.6.1), then the results are acceptable.

4.3.7.4.3 Follow-Up Actions

If the acceptance criterion (±50 ppmb) is not met and the difference between measured and predicted ARO boron concentration is less than 100 ppm Boron, the results will be reviewed by cognizant engineers to determine the appropriate corrective action 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 within 14 days.

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

If the acceptance criteria for SSF subcriticality (≥ -25 ppmb) is not met the results will be reviewed by cognizant engineers to determine the appropriate corrective action required to resolve the discrepancy and ensure the SSF subcriticality function (Section 9.6.1) is met.

Oconee Startup Physics Test Program (OSPTP) Summary

TEST	PLANT CONDITIONS	ACCEPTANCE CRITERIA
1. Control Rod Trip Time Test	RCS Full Flow	1.66 seconds
2. All Rods Out Critical Boron	HZP	± 50 ppmB
3. Moderator Temperature Coefficient	HZP	± 0.3×10^{-4} $\Delta\rho$/°F
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 (assembly location > 1.0 normalized measured power) ±15.0%/±12.0.0% RMS (Radial) < 0.075%
7. Full Power Testing	90 - 100 %FP	Total/Radial Peaking (assembly location > 1.0 normalized measured power) ±15.0%/±12.0.0% RMS (Radial) < 0.075%
8. Reactivity Anomaly	HFP	All Rods Out Critical Boron ± 50 ppmB

4.3.8 References

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4. Oconee Nuclear Station Reload Design Methodology II, DPC-NE-1002-A, Revision 4, SE dated July 21, 2011.
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.

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12. Deleted Per 2009 0Update

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

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19. Framatome ANP Topical Report, BAW-10242 (NP) – A, Revision 0, “ZPPT Modifications for B&W Designed Reactors”, November 2003.

20. Duke Power Company, Letter from H.B. Tucker to NRC, January 26, 1990, “Oconee Nuclear Station, Units 1, 2 and 3; Docket Nos 50-269, 270, and 287, McGuire Nuclear Station, Units 1 and 2; Dockets Nos 50-369 and 370, Catawba Nuclear Station, Units 1 and 2; Docket Nos 50-412 and 414, Response to NRC Bulletin No. 89-03, Potential Loss of Required Shutdown Margin During Refueling Operations.”

21. Nuclear Regulatory Commission, Letter from D.B. Matthews to H.B. Tucker (DPC), March 5, 1990, “Response to Bulletin 89-03 – Catawba, McGuire and Oconee Nuclear stations (TACS 75413, 75414, 75343, 75434, 75439, 75440, and 75441).”

22. Oconee Nuclear Design Methodology using CASMO-4/SIMULATE-3, DPC-NE-1006-PA, SER dated August 2, 2011.

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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 for a discussion of fuel melting temperature.
2. The minimum allowable DNBR during steady-state operation and anticipated transients for Mark-BZ, Mark-B11, Mark-B11A, and Mark-B-HTP fuel are:
    a. Mark-BZ fuel is established as 1.18 with the BWC correlation (Reference 1) for non-SCD analyses and 1.43 for SCD analyses (Reference 13).
    b. MK-B11 and Mark-B11A fuel is established as 1.19 with the BWU-Z correlation with the FB11 multiplicative factor for non-SCD analyses and 1.33 for SCD analyses (Reference 13).
    c. MK-B-HTP fuel is established with BHTP correlation as a proprietary value for NON-SCD analysis and 1.34 for SCD analysis (Reference 19).

    These limits on MDNBR ensure a 95 percent confidence level that there is a 95 percent probability DNB will not occur.
3. Although generation of net steam is allowed in the hottest core channels, flow stability is required during all steady-state and operational transient conditions.

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

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

4.4.2 Description of Thermal and Hydraulic Design of the Reactor Core

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 DPC-NE-2003P-A (Reference 2) and DPC-NE-2005P-A (Reference 13), and DPC-NE-2015P-A (Reference 19).

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

1. Heat transfer, critical heat flux equations, and data correlations.
2. Nuclear peaking factors.
3. Engineering hot channel factors.
4. Core flow distribution hot channel factors.
5. Design reactor power.
6. Thermal hydraulic analysis computer codes.

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

Critical heat flux (CHF) calculations are performed with the Areva BWC or BWU-Z with FB11 multiplication factor, or BHTP correlations. Items 1 through 5 on the above list are explained in Chapters 5 and/or 6 of DPC-NE-2003P-A (Reference 2), and DPC-NE-2015P-A (Reference 19). 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 significantly lower power than the design overpower.

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

4.4.3 Thermal and Hydraulic Evaluation

4.4.3.1 Introduction

A summary of the characteristics of the reactor core design is given in Section 4.1. The methodology of the thermal and hydraulic design analysis is presented in DPC-NE-2003P-A (Reference 2), DPC-NE-2005P-A (Reference 13), and DPC-NE-2015P-A (Reference 19).

4.4.3.2 Deleted Per 1990 Update

4.4.3.3 Evaluation of the Thermal and Hydraulic Design

4.4.3.3.1 Hot Channel Coolant Conditions

The NRC approved VIPRE-01 code is used to calculate the reactor coolant enthalpy, mass flow, vapor void, and DNBR distributions within the core for all expected operating conditions. The VIPRE-01 code is described in detail in (Reference 3), and the models and empirical correlations that are used are discussed in (References 2, 13 and 19).

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

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 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 than the operating conditions have not exhibited unstable characteristics (Reference 9).

4.4.3.3.3 Reactor Coolant Flow System

Another significant variable to be considered in evaluating the design is the total reactor coolant system (RCS) flow. Conservative values for system and reactor pressure drop have been determined to insure 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.

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

4.4.3.3.4 Deleted Per 1990 Update

4.4.3.3.5 Core Flow Distribution

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

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

4.4.3.3.6 Mixing Coefficient

The flow distribution within the hot assembly is calculated using the VIPRE-01 code which allows for the interchange of momentum and heat between channels. The turbulent mixing model incorporated in the VIPRE-01 code and used for all core thermal-hydraulic analyses is discussed in Reference 2. A conservative mixing coefficient of 0.01, based on predictions of mixing tests, is used for DNB analyses for Mark-BZ fuel assembly design. A conservative mixing coefficient of 0.038, per Reference 13, is used for DNB analyses for Mark-B11 fuel assembly design due to the presence of mixing vane grids. For Mark-B-HTP fuel DNB analyses, proprietary turbulent mixing factors, per Reference 21, are used for spans containing HTP or HMP spacer grids.

4.4.3.3.7 Deleted Per 1990 Update.

4.4.3.3.8 Hot Channel Factors

Hot channel factors are included in the calculation of the statistical core design DNBR limit (Reference 13) to account for possible deviations of several parameters from their design values. The power hot channel factor, F_q, accounts for variations in average pin power caused by differences in the absolute number of grams of U_{235} per rod. F_q is applied to the heat generation rate of the hot pin of the hot subchannel. The value of F_q used is given in Reference 13. References 14 and 15 have shown that small local heat flux spikes (which result from power spikes due to flux depressions at the spacer grids and local variations in pellet enrichment/weight) have no effect on the critical heat flux. The hot channel flow area is also reduced when calculating the SCD limit to account for manufacturing tolerances.

4.4.3.3.9 Rod Bow Effects and Penalty

The mechanisms and resulting effects of fuel rod bow are discussed in Areva topical report 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 conservatively predicts the rod bow behavior of Mark-BZ fuel, Mark-B11 fuel and MK-B-HTP 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 reactor. The simulated core was designed to maintain dynamic similarity between the model and production reactor.

Each of the 177 simulated fuel assemblies contained a calibrated flow nozzle. The test loop is capable of supplying cold water (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

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 assemblies have been tested at the Alliance Research Center. Also, 5x5 rod bundle sections have been tested at the Columbia University Heat Transfer Laboratory. This test work substantiates the thermal design of the reactor core. The multiple rod CHF tests are briefly discussed below.

4.4.4.2.1 Deleted Per 1990 Update

4.4.4.2.2 Multiple-Rod Fuel Assembly Heat Transfer Tests

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

BWC CHF Correlation

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

Note: The following conditions were revised in 1998 update.

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

The BWC correlation was developed from 17 x 17 Mark-C CHF data. The BWC correlation was shown to conservatively represent the Mark-BZ CHF data with a 95/95 DNBR limit of I.18 (Reference 1).

The BWC correlation was developed by Areva's using the LYNX2 computer code (Reference 12). To verify use of the BWC correlation with the VIPRE-01 code, the Mark-BZ CHF data was predicted and compared with Areva's LYNX2 results. As discussed in Reference 2, the VIPRE-01 BWC results show that a DNBR limit of 1.18 will provide 95% probability of precluding DNB at a 95% confidence level.

BWU-Z CHF Correlation, With FB11 Multiplicative Factor

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

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

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

BHTP CHF Correlation

The BHTP correlation was developed by Areva using the LYNXT computer code (Reference 20). To verify use of the BHTP correlation with the VIPRE-01 code, the BHTP CHF data was predicted and compared with Areva's LYNXT results. As discussed in Reference 19, VIPRE-01 BHTP results show that the proprietary DNBR limit in Reference 19 will provide 95% probability of precluding DNB at a 95% confidence level. This CHF correlation is applicable to the following range of variables:

Pressure	$1,385 \leq P \leq 2,425$ psia
Local Mass Velocity	$0.492 \leq G \leq 3.549$ Mlbm/hr-ft^2
Local Quality	$X_{loc} \leq 0.512$

The BWU-Z correlation with Mark-B11 multiplier was developed by FCF using the LYNX2 computer code (Reference 12). To verify use of the BWU-Z correlation with the Mark-B11 multiplier with the VIPRE-01 code, the Mark-B11 CHF data was predicted and compared with FCF's LYNX2 results. As discussed in Reference 13, the VIPRE-01 BWU-Z with Mark-B11 multiplier results show that a DNBR 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.

Mark-B11 Fuel Assembly Flow Tests

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

4.4.5 References

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4.5 Reactor Materials

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.

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

Lateral deflection and torsional rotation of the lower end of the core support assembly is limited in order to prevent excessive deformation resulting from seismic disturbance thereby assuring insertion of control rod assemblies (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).

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.

<u>Allowable Stresses</u>

Section 3.9.2.4 describes the stress analysis for fuel assemblies under faulted conditions. Section 3.9.3.1 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).

<u>Methods of Load Analysis to be Employed for Reactor Internals and Fuel Assembly.</u>

Section 3.9.2.4 describes the methods used to analyze fuel assemblies under faulted conditions. Section 3.9.3.1 describes the analysis of the reactor internals.

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

Time-limited aging analyses applicable to the Oconee reactor vessel internals, along with the results of their review for license renewal, are as follows: (1) flow-induced vibration endurance limit assumptions - A review of the existing analysis showed conservatism in the original design, and no further action is needed in the period of extended operation to assure validity of the design; (2) transient cycle count assumptions for the replacement bolting - The ongoing programmatic actions under the Thermal Fatigue Management Program (See Section 5.2.1.4) will assure the validity of the design assumptions in the period of extended operation; and (3) reduction in fracture toughness - The actions developed as a part of the Reactor Vessel Internals Inspection (See Section 18.3.20) will assure the validity of the design assumptions in the period of extended 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, control rod drives in Section 4.5.3, and core instrumentation in Section 7.7.3.

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.

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.

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. 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 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.

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.

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 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.

There are 2 APSR designs which are not fully interchangeable between fuel assembly designs, because of the difference in hold down spring designs and APSR drive mechanisms. Table 4-22 depicts the APSR 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 creep life of the gray APSR. Minimum design cladding thickness is 25 mils.

The gray APSRs are prepressurized to extend their lifespan.

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

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

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

4.5.2.4 Burnable Poison Rod Assembly (BPRA)

Each BPRA (Figure 4-1) has 16 burnable poison rods, a stainless steel spider, and a coupling mechanism. The coupling mechanism and the 16 rods are attached to the spider. The BPRA is inserted into the fuel assembly guide tubes through the upper end fitting. Retention is provided by the feet on the BPRA spider, which rest upon the fuel assembly holddown spring retainer

ring. Thus the BPRA is pinned between this retainer ring and the reactor's upper grid pads. All Oconee fuel which is of the Mk B5 (or later) design, uses this BPRA design.

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).

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

4.5.3 Control Rod Drives

Oconee Units 1, 2 and 3 uses the Type C control rod drive mechanism. The control rod drive mechanisms are sealed, reluctance motor-driven screw units.

4.5.3.1 Type C 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 snubber assembly which is attached to the lower end of the torque tube. 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

   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 speed of the mechanism shall be 30 in./min for both insertion and withdrawal in the "Run" mode of control. The withdrawal speed shall be limited to not exceed 25 percent overspeed in the event of speed control fault. The speed of the mechanism shall be 3 in./min for both insertion and withdrawal 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 Type C shim safety drive mechanism consists of a motor tube which houses a lead screw and 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.

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 snubber assembly within the torque tube 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 belleville spring assembly. 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 torque taker assembly.

Two of the six phase stator housing windings are energized to maintain the control rod position when the drive is in the holding mode.

4.5.3.1.4 CRDM Subassemblies

The CRDM is shown in Figure 4-34. 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 martensitic stainless steel to present a small air gap to the motor. 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 varnish impregnated 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 hydraulically tensioning the closure insert and is retained by the closure nut.

   Deleted paragraph(s) per 2005 update

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 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

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, 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 programmable logic controller that generates a signal proportional to the position demand for the rod, as derived from counting the number and sequence of power pulses sent to the rod drive moter stator windings.

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 Deleted Per 2002 Update

4.5.3.2.1 Deleted Per 2002 Update

4.5.3.2.2 Deleted Per 2002 Update

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. 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 described in Section 4.2.4. 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. The internals surveillance specimen holder tubes and the material irradiation program are described in Section 5.2.3.13.

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:

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.

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.

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.

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.

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.

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.

4.5.4.2.7 Production Valve Testing

Each production valve will be subjected to tests described in Sections 4.5.4.2.2 and 4.5.4.2.3 except that no additional analysis will be performed in conjunction with the test described in Section 4.5.4.2.2.

The valve disc, hinge shaft, shaft journals (bushings), disc journal receptacles, and valve body journal receptacles are designed to withstand without failure the internal and external differential pressure loadings 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.

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 provided to engage with the valve disc exercise lug described in Item 7 of Section 4.5.1.3.2. With the aid of this tool, the valve disc will be manually exercised to evaluate the disc freedom. The hinge design incorporates special features, as described in Item 7 of Section 4.5.1.3.2 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

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2. BAW-10008, Part 1, Rev. 1, Reactor Internals Stress and Deflection due to Loss-of-Coolant Accident and Maximum Hypothetical Earthquake, June 1970.
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.
6. *Demonstration of the Management of Aging Effects for the Reactor Vessel Internals*, BAW-2248A, The B&W Owners Group Generic License Renewal Program, March 2000.
7. *Application for Renewed Operating Licenses for Oconee Nuclear Station, Units 1, 2, and 3*, submitted by M. S. Tuckman (Duke) letter dated July 6, 1998 to Document Control Desk (NRC), Docket Nos. 50-269, 50-270, and 50-287.
8. NUREG-1723, *Safety Evaluation Report Related to the License Renewal of Oconee Nuclear Station, Units 1, 2, and 3*, Docket Nos. 50-269, 50-270, and 50-287.

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