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3.0 REACTOR

NOTE: Fuel assembly design information in Section 3.1 through Section 3.4 is partially historical because cores are currently designed using only 422 VANTAGE + fuel.

The reactor utilizes a multi-region cycled core design, with fuel assemblies containing slightly enriched uranium dioxide (UO_2) fuel clad with ZIRLO® or Optimized ZIRLO™ tubing. Nuclear design data is summarized in Table 3.2-1. Thermal-hydraulic design parameters are provided in Table 3.2-4. Reactor mechanical design information is presented in Table 3.2-5.

3.1 DESIGN BASIS

3.1.1 PERFORMANCE OBJECTIVES

The construction permit for each Point Beach Unit was issued for an initial reactor power of 1396 MWt with an ultimate rating of 1518.5 MWt. In 2002, a measurement uncertainty recapture (MUR) power uprate was approved by the NRC resulting in an increased rated thermal power of 1540 MWt (Reference 3). Subsequently an extended power uprate (EPU) was approved (Reference 4) for 1800 MWt. This power operation is the basis, except where specifically noted, for all the safety evaluations in this report. Most of the Chapter 14 safety analyses bound operation at 1800 MWt, except where specifically noted. The reactor core fuel loading and programming is designed to yield an equilibrium cycle nominal burnup of approximately 19000 MWD/MTU.

In November 1984, Unit 2 began operating in its eleventh reload cycle with its first region of optimized fuel assemblies (OFA) and in June 1985, Unit 1 began operating in its thirteenth reload cycle with its first region of OFA fuel. Point Beach Unit 1 operated in Cycle 17 with its first region of upgraded OFA fuel. Point Beach Unit 2 operated Cycle 16 with its first region of upgraded OFA fuel. Natural enrichment axial blankets and Integrated Fuel Burnable Absorbers (IFBAs) began implementation with Unit 1 Cycle 19 in May 1991 and Unit 2 Cycle 18 in November 1991. Starting with Unit 1 Cycle 27 and Unit 2 Cycle 25, 14x14 0.422" VANTAGE+ assemblies, referred to as 422V+ fuel, were loaded as feed assemblies. Designs were no longer based on annual reload cycles operating with OFA fuel, but were based on 18 month cycles with OFA and 422V+ fuel. Since Unit 1 Cycle 30 and Unit 2 Cycle 28, cores have been designed using only 422V+ fuel. The reactor core can still utilize either OFA fuel, upgraded OFA fuel, or any combination of previously burned OFA, previously burned upgraded OFA, and 422V+ fuel assemblies. The original Low-Parasitic (LOPAR) fuel, also known as STD fuel, can no longer be used.

Based on the analyzed departure from nucleate boiling ratio (DNBR) limits and associated reactor control and protection system settings, the RCS must be operated at a nominal pressure of 2235 psig while 422V+ fuel assemblies are in the core.

The control rods provide sufficient control rod worth to shut the reactor down ($k_{eff} \leq 0.99$) from the hot condition at any time during cycle life with the most reactive control rod stuck in the fully withdrawn position. Redundant equipment is provided to add a soluble neutron absorber to the reactor coolant to ensure a similar shutdown capability when the reactor coolant is cooled to ambient temperatures.

Experimental measurements from critical experiments or operating reactors, or both, are used to validate the methods employed in the design. During design, nuclear parameters are calculated for every phase of operation of the respective core cycle and, where applicable, are compared with design limits to show that an adequate margin of safety exists.

In the thermal hydraulic design of the core, the maximum fuel and cladding temperatures during normal reactor operation and at design thermal overpower are evaluated conservatively and found to be consistent with safe operating limitations.

3.1.2 PRINCIPAL DESIGN CRITERIA

3.1.2.1 Reactor Core Design

Criterion: The reactor core with its related controls and protection systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated. (GDC 6)

The reactor core, with its related control and protection system, is designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The core design, together with reliable process and decay heat removal systems, provides for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations, including the effects of the loss of reactor coolant flow (Section 14.1.8), likelihood of turbine generator unit overspeed (Section 14.1.12), loss of normal feedwater (Section 14.1.10), and loss of external electric load (Section 14.1.9).

The reactor control and protection system is designed to actuate a reactor trip for any anticipated combination of plant conditions, when necessary, to ensure a departure from nucleate boiling ratio (DNBR) equal to or greater than the limits specified for STD, OFA, upgraded OFA, or 422V+ fuel, as applicable.

The integrity of the fuel cladding is ensured by preventing excessive cladding overheating and excessive cladding stress and strain. This is achieved by designing the core so that the following conservative limits are not exceeded during normal operation or any anticipated transient condition:

1. The minimum DNBR is equal to, or greater than, the safety limit DNBR values specified for STD fuel, OFA fuel, upgraded OFA fuel, or 422V+ fuel, as applicable.

2. Fuel center temperature below melting point of UO_2.

3. The internal pressure of the lead rod in the reactor is limited to a value below that which could cause:

   a) The diametral gap to increase due to outward cladding creep during steady state operations, and

   b) Extensive departure from nucleate boiling (DNB) propagation to occur.

4. Cladding stresses less than the Zircaloy, ZIRLO® or Optimized ZIRLO™ yield strength

5. Cladding strain less than 1%.

6. Cumulative strain fatigue cycles less than 100% of design strain fatigue life.

The ability of fuel designed and operated to these criteria to withstand postulated normal and abnormal service conditions is shown by the analyses described in Chapter 14 to satisfy the demands of plant operation well within applicable regulatory limits.

The reactor coolant pumps provided for the plant are supplied with sufficient rotational inertia to maintain an adequate flow coastdown in the event of a simultaneous loss of power to all pumps. The flow coastdown inertia is sufficient such that the reduction in heat flux obtained with a low flow reactor trip prevents core damage.

In the unlikely event of a turbine trip from full power without an immediate reactor trip, the subsequent reactor coolant temperature increase and volume insurge to the pressurizer results in a high pressurizer pressure trip and thereby prevents fuel damage for this transient. A loss of external electrical load of 50% of full power or less is normally controlled by rod cluster insertion together with a controlled steam dump to the condenser to prevent a large temperature and pressure increase in the reactor coolant system and thus prevent a reactor trip. In this case, the overpower-overtemperature protection would guard against any combination of pressure, temperature, and power which during the transient could result in a DNBR less than the safety limit DNBR values specified.

In neither the turbine trip nor the loss-of-flow events do the changes in coolant conditions provoke a nuclear power excursion because of the large system thermal inertia and relatively small void fraction. Protection circuits actuated directly by the coolant conditions identified with core limits are, therefore, effective in preventing core damage.

3.1.2.2 Suppression of Power Oscillations

Criterion: The design of the reactor core with its related controls and protection systems shall ensure that power oscillations, the magnitude of which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed. (GDC 7)

The potential for possible spatial oscillations of power distribution for this core has been reviewed. In summary, the review concludes that the only potential spatial instability is the xenon-induced axial instability which may be a nearly free-running oscillation with little or no inherent damping. Initially, part-length control rods were provided to suppress these oscillations; however, experience demonstrated that full-length control rods were effective in controlling these oscillations and the part-length rods were removed and replaced with thimble plugging devices. These thimble plugs were used on STD fuel only. Out-of-core instrumentation is provided to obtain necessary information concerning axial distributions. This instrumentation is adequate to enable the operator to monitor and control xenon induced oscillations. In-core instrumentation is used to periodically calibrate and verify the axial flux information provided by the out-of-core instrumentation. The analysis, detection, and control of these oscillations is discussed in Reference 1.

The moderator temperature coefficient in the power operating range is maintained less than or equal to +5 pcm/°F below 70% power, and below zero or negative above 70% power by inclusion of IFBAs and burnable absorber rods, as needed, in the core loadings.

3.1.2.3 Redundancy of Reactivity Control

Criterion: Two independent reactivity control systems, preferably of different principles, shall be provided. (GDC 27)

Two independent reactivity control systems are provided, one involving rod cluster control assemblies (RCCAs) and the other involving the injection of a soluble poison.

3.1.2.4 Reactivity Hot Shutdown Capability

Criterion: The reactivity control systems provided shall be capable of making and holding the core subcritical from any hot standby or hot operating condition. (GDC 28)

The reactivity control systems provided are capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes. This includes the maximum excess reactivity expected for the core, which occurs for the cold, clean condition at the beginning of life (BOL) of the initial core.

The RCCAs are divided into two categories comprising control and shutdown rod groups. The control group, used in combination with soluble poison, provides reactivity control throughout the life of the core at power conditions. This group of RCCAs is used to compensate for short term reactivity changes at power from variations in reactor power requirements or coolant temperature. The soluble poison control is used to compensate for the more slowly occurring changes in reactivity throughout core life such as those due to fuel depletion and fission product buildup and for load-follow.

Upon demand for the hot shutdown condition, insertion of both the control and shutdown groups of RCCAs will immediately make the reactor subcritical from any hot standby or hot operating condition. Subsequent injection of soluble poison can be used to assure continuation of the hot shutdown condition under all circumstances.

3.1.2.5 Reactivity Shutdown Capability

Criterion: One of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margin should assure subcriticality with the most reactive control rod fully withdrawn. (GDC 29)

The reactor core, together with the reactor control and protection system, is designed so that the minimum allowable DNBR is at least equal to the limits specified for STD, OFA, upgraded OFA, or 422V+ fuel, as applicable, and there is no fuel melting during normal operation, including anticipated transients.

The shutdown groups are provided to supplement the control group of RCCAs to make the reactor at least 1% subcritical (k_{eff}= 0.99) following a trip from any credible operating condition to the hot, zero power condition assuming the most reactive RCCA remains in the fully withdrawn position.

Sufficient shutdown capability is also provided to maintain the core subcritical for the most severe anticipated cooldown transient associated with a single active failure, e.g., accidental opening of a steam bypass or safety valve stuck fully open.

The criteria of GDC 28 and 29 are met fast enough to prevent exceeding acceptable fuel damage limits, even with the most reactive control rod fully withdrawn.

3.1.2.6 Reactivity Holddown Capability

Criterion: The reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public. (GDC 30)

The reactivity control systems provided are capable of making and holding the core subcritical, under accident conditions, in a timely fashion with appropriate margins for contingencies. Normal reactivity shutdown capability is provided within 2.2 seconds following a trip signal by control rods with soluble neutron absorber (boric acid) injection used to compensate for the long term xenon decay transient and for plant cooldown. Any time that the reactor is at power, the quantity of boric acid retained in the boric acid storage tanks and/or the refueling water storage tank (RWST) and ready for injection always exceeds that required for the normal cold shutdown. This quantity also exceeds that required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay.

Boric acid may be pumped from the boric acid tanks by one of two boric acid transfer pumps (or via gravity feed from the RWST) to the suction of one of three charging pumps which inject boric acid into the reactor coolant. Any charging pump and either boric acid transfer pump can be operated from diesel generator power on loss of outside power. Boric acid can be injected by one charging pump supplied by one boric acid transfer pump at a rate which shuts the reactor down hot with no rods inserted in less than 120 minutes. In 120 additional minutes, enough boric acid can be injected to compensate for xenon decay, although xenon decay below the equilibrium operating level does not begin until approximately fifteen hours after shutdown. If two boric acid transfer pumps are available, these time periods are reduced. Additional boric acid injection is employed if it is desired to bring the reactor to cold shutdown conditions.
On the basis of the above, the injection of boric acid is shown to afford backup reactivity shutdown capability independent of RCCAs which normally serve this function in the short term situation. Shutdown for long term and reduced temperature conditions can be accomplished with boric acid injection using redundant components, thus achieving the measure of reliability implied by the criterion.

Alternatively, boric acid solution at lower concentration can be supplied from the RWST. This solution can be transferred directly by the charging pumps. The reduced boric acid concentration lengthens the time required to achieve equivalent shutdown. For added flexibility, the safety injection pumps can also be supplied with boric acid solution from either the boric acid storage tanks or the RWST.

3.1.2.7 Reactivity Control Systems Malfunction

Criterion: The reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits. (GDC 31)

The reactor protection systems are capable of protecting against any single anticipated malfunction of the reactivity control system by limiting reactivity transients so as to avoid exceeding acceptable fuel damage limits.

Reactor shutdown with rods is completely independent of the normal rod control functions since the trip breakers completely interrupt the power to the latch type rod mechanisms regardless of existing control signals.

Details of the effects of continuous withdrawal of a control rod are described in Section 14.1.1 and Section 14.1.2. Details of the effects of continuous boron dilution are described in Section 14.1.4.

3.1.2.8 Maximum Reactivity Worth of Control Rods

Criterion: Limits, which include reasonable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to lose capability of cooling the core. (GDC 32)

Limits, which include considerable margin, are placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large reactivity change cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals so as to lose capability to cool the core.

The reactor control system employs control rod clusters, approximately half of which are fully withdrawn during power operation, serving as shutdown rods. The remaining rods comprise the controlling group which are used to control load and reactor coolant temperature. The rod cluster drive mechanisms are wired into preselected groups and are, therefore, prevented from being withdrawn in other than their respective groups. The rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel.

The maximum positive reactivity insertion rate assumed in the detailed plant analysis is greater than that for the simultaneous withdrawal of the combination of the two sequential control banks having the greatest combined worth at maximum speed. The resultant reactivity insertion rates are well within the capability of the overpower-overtemperature protection circuits to prevent core damage.

No credible mechanical or electrical control system malfunction can cause a rod cluster to be withdrawn at a speed greater than 72 steps per minute (45 inches per minute).

3.1.3 SAFETY LIMITS

The reactor is capable of meeting the performance objectives throughout core life under both steady state and transient conditions without violating the integrity of the fuel cladding. Thus, the release of unacceptable amounts of fission products to the coolant is prevented.

The limiting conditions for operation specify the highest functional capability or performance levels permitted to assure safe operation of the facility.

Design parameters which are established by safety limits are specified below for the nuclear, reactivity control, thermal and hydraulic, and mechanical aspects of the design.

3.1.3.1 Nuclear Limits

At a full power level of 1800 MWt, the nuclear heat flux hot channel factor, $F^N{}_q$, specified in Table 3.2-4 is not exceeded.

The nuclear axial peaking factor, $F^N{}_Z$, and the nuclear enthalpy rise hot channel factor, $F^N{}_{\Delta H}$, are limited in their combined relationship so as not to exceed the $F^N{}_q$ or DNBR limits. The effects of fuel densification and rod bow are taken into account.

The limiting nuclear hot channel factors are higher than those calculated at full power for the range from all control rods fully withdrawn to maximum allowable control rod insertion. Control rod insertion limits as a function of power are delineated in the Technical Specifications to ensure that despite differences in control rod insertion the DNBR is always greater at part power than at full power.

Axial xenon oscillations are monitored and controlled with the control rods to preclude adverse core conditions. The protection system ensures that the nuclear core limits are not exceeded.

3.1.3.2 Reactivity Control Limits

The control system and the operational procedures provide adequate control of the core reactivity and power distribution. The following control limits are met:

1. Sufficient control is available to produce a hot shutdown margin of at least that required in the Core Operating Limits Report (COLR), (Reference 2).

2. The shutdown margin is maintained with the most reactive RCCA stuck in the fully withdrawn position.

3. The shutdown margin is maintained at ambient temperature by the use of soluble neutron absorber.

3.1.3.3 Thermal and Hydraulic Limits

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

1. The minimum allowable DNBR during normal operation, including anticipated transients, is the DNBR for which DNB will not occur with a 95% probability at a 95% confidence level.

2. No fuel melting during any anticipated normal operating condition.

To maintain fuel rod integrity and prevent fission product release, it is necessary to prevent cladding overheating under all operating conditions. This is accomplished by preventing a DNB which causes a large decrease in the heat transfer coefficient between the fuel rods and the reactor coolant, resulting in high cladding temperatures.

Considering plant parameter uncertainties, there must be at least a 95 percent probability that the minimum DNBR of the limiting power rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The DNBR limit for the correlation is established based on the variance of the correlation such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the calculated DNBR is at the DNBR limit.

DNB is not, however, an observable parameter during reactor operation. Therefore, the observable parameters, reactor power, reactor coolant temperature, and pressure have been related to DNB through the W3 DNB correlation for STD fuel and the WRB-1 DNB correlation for OFA fuel, upgraded OFA fuel, and 422V+. Curves presented in Reference 2 represent the loci of points of reactor power, reactor coolant pressure, and average temperature for which the DNBR is less than the limit specified for STD, OFA, or 422V+ fuel, as applicable. The area of safe operation is the lower average temperatures and higher reactor coolant pressures limited by one specified curve of the reactor power parameter family of curves shown. The parameters used in the development of the curves were checked in the course of initial startup tests and are modified as necessary.

3.1.3.4 Mechanical Limits-Reactor Internals

The reactor internal components are designed to withstand the stresses resulting from startup, steady state operation with any number of pumps running, and shutdown conditions. No damage to the reactor internals occurs as a result of loss of pumping power.

Lateral deflection and torsional rotation of the lower end of the core barrel are limited to prevent excessive movements resulting from seismic disturbances and thus prevent interference with RCCAs. Core drop in the event of failure of the normal supports is limited so that the RCCAs do not disengage from the fuel assembly guide thimbles.

The structural internals are designed to maintain their functional integrity in the event of a major loss-of-coolant accident. The dynamic loading resulting from the pressure oscillations because of a loss-of-coolant accident does not prevent RCCA insertion even during an earthquake.

3.1.3.5 Fuel Assemblies

The fuel assemblies are designed to perform satisfactorily throughout their lifetime. The loads, stresses, and strains resulting from the combined effects of flow induced vibrations, earthquakes, reactor pressure, fission gas pressure, fuel growth, thermal strain, and differential expansion during both steady state and transient reactor operating conditions have been considered in the design of the fuel rods and fuel assembly. The assembly is also structurally designed to withstand handling and shipping loads prior to irradiation and to maintain sufficient integrity at the completion of design burnup to permit safe removal from the core and subsequent handling during cooldown, shipment, and fuel reprocessing or storage.

The fuel rods are supported at several locations along their length within the fuel assemblies by grid assemblies which are designed to maintain control of the lateral spacing between the rods throughout the design life of the assemblies. The magnitude of the support loads provided by the grids are established to minimize possible fretting without overstressing the cladding at the points of contact between the grids and fuel rods. The grid assemblies also allow axial thermal expansion of the fuel rods without imposing restraint of sufficient magnitude to result in buckling or distortion of the rods.

The fuel rod cladding is designed to withstand operating pressure loads without collapse or rupture and to maintain encapsulation of the fuel throughout the design life.

3.1.3.6 Rod Cluster Control Assemblies (RCCAs)

The criteria used for the design of the cladding on the individual absorber rods in the RCCAs are similar to those used for the fuel rod cladding. The stainless steel cladding is designed to be free standing under all operating conditions and will maintain encapsulation of the absorber material throughout the absorber rod design life. Allowance for wear during operation is included in the RCCA cladding thickness. The EP-RCCA (Enhanced Performance RCCA) has all the features described and also has full length chrome plating to reduce guide card wear and reduced tip absorber diameter to alleviate tip swelling and cracking.

Adequate clearance is provided between the absorber rods and the guide thimbles, which position the rods within the fuel assemblies so that coolant flow along the length of the absorber rods is sufficient to remove the heat generated, thereby preventing overheating of the absorber cladding. The clearance is also sufficient to compensate for any misalignment between the absorber rods and guide thimbles and to prevent mechanical interference between the rods and guide thimbles under any operating conditions.

3.1.3.7 Control Rod Drive Assembly

Each control rod drive assembly is designed as a hermetically sealed unit to prevent leakage of reactor coolant water. All pressure containing components are designed to meet the requirements of the ASME Code, Section III, Class 1, 1998 Edition through 2000 Addenda.

The control rod drive assemblies provide RCCA insertion and withdrawal rates consistent with the required reactivity changes for reactor operational load changes. This rate is based on the worths of the various rod groups which are established to limit power peaking flux patterns to design values. The maximum reactivity addition rate is specified to limit the magnitude of a possible nuclear excursion resulting from a control system or operator malfunction.

Also, the control rod drive assemblies provide a fast insertion rate during a "trip" of the RCCAs which results in a rapid shutdown of the reactor for conditions that cannot be handled by the reactor control system. This rate is based on the results of various reactor emergency analyses, including instrument and control delay times and the amount of reactivity that must be inserted before deceleration of the RCCA occurs.

REFERENCES

1. "Nuclear Design of Westinghouse Pressurized Water Reactor with Burnable Poison Rods," WCAP-9000 (Proprietary), 1968.

2. Point Beach Nuclear Plant Technical Requirements Manual (TRM) 2.1, Core Operating Limits Reports (COLRs) for Unit 1 and Unit 2."

3. NRC Safety Evaluation dated November 29, 2002, "Issuance of Amendments Re: Measurement Uncertainty Recapture Power Uprate (TAC Nos. MB4956 and MB4957)."

4. NRC Safety Evaluation dated May 3, 2011, "Issuance of License Amendments Regarding Extended Power Uprate (TAC Nos. ME1044 and ME1045)."

3.2 REACTOR DESIGN

3.2.1 NUCLEAR DESIGN AND EVALUATION

This section presents the nuclear characteristics of the core and an evaluation of the characteristics and design parameters which are significant to design objectives. The capability of the reactor to achieve these objectives while performing safely under operational modes, including both transient and steady state, is demonstrated.

Four fuel designs are considered in this section: the standard (STD) 14x14 fuel assembly; the 14x14 optimized fuel assembly (OFA); the upgraded OFA 14x14 assembly; and the 14x14 422 VANTAGE+ fuel assembly (422V+). The reload core may contain part-length hafnium absorber rods in peripheral assemblies to reduce the fast neutron flux at the reactor vessel walls. The upgraded OFA assembly includes a removable top nozzle (RTN) with high burnup enhancements and a debris filter bottom nozzle (DFBN), and may include the addition of Integral Fuel Burnable Absorber (IFBA) fuel rods and six-inch axial blankets, utilizing natural UO_2 at the top and bottom of the fuel stack. The 422V+ assembly includes the same features as the upgraded OFA assembly. Key differences between the upgraded OFA and 422V+ assemblies are the increased fuel rod and instrumentation tube OD (0.422") and a slight reduction (0.75") in the enriched (non-blanketed) portion of the fuel pellet stack. Additionally, the 422V+ assemblies include ZIRLO® or Optimized ZIRLO™ cladding and structural material, mid-enriched annular pellets in axial blankets of between 6 and 8 inches in length, and an increased B-10 loading for the IFBA fuel rods. These upgrade features, along with a low-low leakage loading pattern, maintain radial and axial neutron leakage and improve fuel economy. Since Unit 1 Cycle 30 and Unit 2 Cycle 28 cores have been designed using only 422V+ fuel.

Burnable absorber rods in RCC guide thimble tubes are no longer used and description of their design and use is retained for historical purposes.

3.2.1.1 Reactivity Control

Reactivity control is provided by:

1. A soluble chemical neutron absorber, boric acid, in the reactor coolant (chemical shim).
2. Movable neutron absorbing control rods, or rod cluster control assemblies (RCCAs).
3. Fixed burnable or non-burnable absorber rods, as specified in the respective cycle core design.

For the upgraded OFA assemblies, introduced into the Unit 1 Cycle 17 core and Unit 2 Cycle 16 core, and 422V+ assemblies, introduced into the Unit 1 Cycle 27 core and Unit 2 Cycle 25 core, the nuclear design analyses and evaluations allow the use of IFBA rods and axial blankets.

The concentration of boric acid is varied as necessary during the life of the core to compensate for:

1. Changes in reactivity which occur with the change in temperature of the reactor coolant from cold shutdown to the hot operating, zero power conditions.

2. Changes in reactivity associated with changes in concentration of the fission product absorbers xenon and samarium.

3. Reactivity losses associated with the depletion of fissile inventory and buildup of long-lived fission product absorbers other than xenon and samarium.

4. Changes in reactivity due to burnable absorber depletion.

5. Load-follow operation.

The control rods provide reactivity control for:

1. Fast shutdown.

2. Reactivity changes associated with changes in the average coolant temperature above hot zero power (core average coolant temperature is increased with power level).

3. Reactivity associated with any void formation.

4. Reactivity changes associated with the power coefficient of reactivity.

The rods are divided into two categories according to their function. The rods which compensate for changes in reactivity due to variations in operating conditions of the reactor, such as power or temperature, comprise the control group of rods. The other rods provide additional shutdown reactivity and are termed shutdown rods. The total shutdown worth of all the rods is specified to provide adequate shutdown with the most reactive rod stuck out of the core.

The burnable absorber rods provide control of part of the excess reactivity available. By using specific placement, fresh and depleted burnable absorber rods serve to reduce peaking factors and maintain the moderator temperature coefficient within limits. IFBA rods contain a stack of fuel pellets coated with a thin boron absorber compound. The IFBA is described and evaluated in Sections 2.4 and 2.5 of Reference 7.

3.2.1.2 Nuclear Design Data - Core Reactivity Characteristics

A summary of nuclear design data including core reactivity characteristics for full STD cores, reload OFA and upgraded OFA cores, and reload 422V+ cores is presented in Table 3.2-1. Discussion of the table is facilitated by numbering the lines. In addition, a summary of reactivity requirements and control rod worths is given in Table 3.2-2 and Table 3.2-3 which may be used in conjunction with Table 3.2-1.

A tabulation of general structural characteristics for 422V+, OFA and STD fuel is given in lines 1 through 10 of Table 3.2-1, while performance characteristics are listed in lines 11 through 19. Values of effective neutron multiplication constants and critical boron concentrations for the first core, equilibrium cycle OFA cores, and equilibrium cycle 422V+ cores are listed for specified conditions in lines 20 through 37. Several of these items, such as shim control, are discussed in

greater detail below. The values provided are typical. Values for key parameters are determined for each reload design.

Control to render the reactor subcritical at temperatures below the operating range is provided by chemical shim. The boron concentration during refueling, reported in line 28 of Table 3.2-1, together with the control rods, provides approximately a 5% shutdown margin for these operations. The concentration is also sufficient to maintain the core subcritical (k = 0.99) without any RCCAs during refueling. For cold shutdown at the beginning of core life, Table 3.2-1 line 36 shows a concentration sufficient for a 1% shutdown margin with all but one stuck rod inserted. The boron concentration for refueling is equivalent to less than 2% by weight boric acid (H_3BO_3) and is well within solubility limits at ambient temperature. This concentration is comparable to the range of boron concentration maintained in the spent fuel pool. The effects of different concentrations are acceptable even when the reactor coolant is directly connected with the refueling canal during refueling operations (Reference 12).

The initial and equilibrium core full power boron concentration without equilibrium xenon and samarium is shown in Table 3.2-1 line 33. As these fission product poisons are built up, the boron concentration is reduced. The initial boron concentration is that which permits the positioning of the control banks at their operational limits. The xenon-free, zero power shutdown (k = 0.99) with all but one stuck rod inserted, must be maintained with the boron concentrations shown in lines 36 and 37 for the cold and hot conditions, respectively.

The chemical shim concentrations discussed above are those used when burnable absorber rods are present in the initial core, as listed in Table 3.2-1 lines 38, 39 and 40. Likewise, kinetic characteristics are dependent upon boron concentrations, presence of burnable absorbers, and control rods. Equivalent values are calculated for each reload core design.

3.2.1.3 Kinetic Characteristics

The response of the reactor core to plant conditions or operator adjustments during normal operation, as well as the response during abnormal or accidental transients, is evaluated by means of a detailed plant simulation. In these calculations, reactivity coefficients are required to couple the response of the core neutron multiplication to the variables which are set by conditions external to the core. Since the reactivity coefficients change during the life of the core, a range of coefficients is established to determine the response of the plant throughout life and to establish the design of the reactor control and protection system.

3.2.1.4 Moderator Temperature Coefficient of Reactivity

The moderator temperature coefficient in a core controlled by chemical shim is less negative than the coefficient in an equivalent rodded core. One reason is that control rods contribute a negative increment to the coefficient and, in a chemical shim core, the rods are only partially inserted. Also, the chemical absorber density is decreased with the water density upon an increase in temperature. This gives rise to a positive component of the moderator temperature coefficient due to the boron being removed from the core. This component is directly proportional to the amount of reactivity controlled by the soluble absorber.

To reduce the dissolved absorber requirement for control of excess reactivity, burnable absorber rods can be incorporated in the core design. The result is that changes in the coolant density will have less effect on the density of absorber and the moderator temperature coefficient will be more negative.

The fixed discrete burnable absorber is in the form of borated pyrex glass rods clad in stainless steel. Clusters of these rods are distributed throughout the core in vacant rod cluster control guide thimbles. As an example, the initial core pattern is shown in Figure 3.2-5 and Figure 3.2-6 on a gross core and assembly-wise basis, respectively. Information regarding research, development, and nuclear evaluation of the discrete burnable absorber rods can be found in Reference 1 and Reference 2. The number of rods and the corresponding reactivity worths for the initial core are indicated in lines 38, 39, and 40 of Table 3.2-1.

The IFBA fuel rods are distributed in selective upgraded OFAs and 422V+ assemblies to control peaking factors and to reduce the moderator temperature coefficient. The length of the boron burnable absorber coating in the enriched fuel stack may vary from assembly to assembly and cycle to cycle. As a part of the contraints to ensure sub-criticality in the spent fuel pool, as discussed in Section 9.4.1, there are some restrictions in the IFBA patterns used for reactivity control during the core design process. Allowable IFBA patterns of 52 or less IFBA pins are identified in Reference 51, Figure 3-5. IFBA patterns of 52 or less IFBA pins other than those shown in Reference 51 require a 10 CFR 50.59 evaluation to validate that the conclusions from the criticality analysis remain unchanged. Such an evaluation was performed in Reference 52 and Reference 53 to document the acceptability of additional IFBA patterns, less than 52 pins, which can be credited for storage in the Point Beach spent fuel pool. Any IFBA loadings with more than 52 pins per assembly up to 120 are allowed with no IFBA pattern restrictions (Reference 54). In addition, allowable IFBA length must be 120 inches or greater and poison loading must be equal to or greater than 1.0X IFBA (e.g., 1.5X, 2.0X, etc), as identified in Reference 51.

The moderator temperature coefficient becomes more negative with increasing burnup, resulting from buildup of plutonium and fission products and dilution of the boric acid concentration. The reactivity loss due to equilibrium xenon is also controlled by soluble boron. As xenon builds up, boron is taken out. The range of the calculated net unrodded moderator temperature coefficient is shown in Table 3.2-1 line 41.

The control rods provide a negative contribution to the moderator coefficient as illustrated in Figure 3.2-17.

3.2.1.5 Moderator Pressure Coefficient of Reactivity

The moderator pressure coefficient is positive at plant operating conditions. Its effect on core reactivity and stability is small because of the small magnitude of the pressure coefficient.

3.2.1.6 Moderator Density Coefficient of Reactivity

A uniform moderator density coefficient is defined as a change in the neutron multiplication per unit change in moderator density. The range of the moderator density coefficient from BOL to EOL is specified in Table 3.2-1, line 43.

3.2.1.7 Fuel Temperature (Doppler) Coefficient

The fuel temperature (Doppler) coefficient is defined as the change in reactivity per degree change in effective fuel temperature and is primarily a measure of the Doppler broadening of

U-238 and Pu-240 resonance absorption peaks. Doppler broadening of other isotopes such as U-236, Np-237 etc. are also considered but their contributions to the overall Doppler effect is negligible. An increase in fuel temperature increases the effective resonance absorption cross sections of the fuel and produces a corresponding reduction in reactivity.

The fuel temperature coefficient is calculated by performing two-group three dimensional calculations using the ANC code (Reference 11). Moderator temperature is held constant and the power level is varied. Spatial variation of fuel temperature is taken into account by calculating the effective fuel temperature as a function of power density.

The Doppler temperature coefficient is shown in Figure 3.2-18 as a function of the effective fuel temperature (at beginning-of-life (BOL) conditions). The effective fuel temperature is lower than the volume averaged fuel temperature since the neutron flux distribution is non-uniform through the pellet and gives preferential weight to the surface temperature. The Doppler-only contribution to the power coefficient, defined later, is shown in Figure 3.2-19 as a function of relative core power.

3.2.1.8 Power Coefficient

The combined effect of moderator temperature and fuel temperature change as the core power level changes is called the total power coefficient and is expressed in terms of reactivity change per percent power change. The typical power coefficient at BOL conditions is given in Figure 3.2-20.

It becomes more negative with core life reflecting the combined effect of moderator and fuel temperature coefficients with fuel depletion.

3.2.1.9 Summary of Control Rod Requirements

Control rod reactivity requirements at BOL and EOL are summarized in Table 3.2-2. The installed worth of the control rods is shown in Table 3.2-3. The difference is available for excess shutdown upon reactor trip.

The control rods are required to provide sufficient reactivity to account for the power defect from full power to zero power and to provide the required shutdown margin. The reactivity addition resulting from power reduction consists of contributions from Doppler, variable average moderator temperature, flux redistribution, and reduction in void content as discussed below.

3.2.1.10 Doppler

The Doppler effect arises from the broadening of U-238 and Pu-240 resonance peaks with an increase in effective pellet temperature. This effect is most noticeable over the range of zero power to full power due to the large pellet temperature increase with power generation.

3.2.1.11 Variable Average Moderator Temperature

When the core is shutdown to the hot, zero power condition, the average moderator temperature changes from the equilibrium full load value determined by the steam generator and turbine characteristics (steam pressure, heat transfer, tube fouling, etc.) to the equilibrium no load value, which is based on the steam generator shell side design pressure. The design change in temperature is conservatively increased by 4°F to account for the control dead band and measurement errors.

Since the moderator coefficient is usually negative (may be positive up to 70% power at or near BOL), there is a reactivity addition with power reduction. The moderator coefficient becomes more negative as the fuel depletes because the boron concentration is reduced. This effect is the major contributor to the increased requirement for control rod reactivity at EOL.

3.2.1.12 Redistribution

During full power operation the coolant density decreases with core height, and this, together with partial insertion of control rods, results in less fuel depletion near the top of the core. Under steady state conditions, the relative power distribution will be slightly asymmetric towards the bottom of the core. On the other hand, at hot zero power conditions, the coolant density is uniform up the core, and there is no flattening due to Doppler. The result will be a flux distribution which at zero power can be skewed toward the top of the core. The reactivity insertion due to the skewed distribution is calculated with an allowance for the most adverse effects of xenon distribution.

3.2.1.13 Void Content

A small void content in the core is due to nucleate boiling at full power. The void collapse coincident with power reduction makes a small reactivity contribution.

3.2.1.14 Rod Insertion Allowance

At full power, the control bank is operated within a prescribed band of travel to compensate for small periodic changes in boron concentration, changes in temperature and very small changes in the xenon concentration not compensated for by a change in boron concentration. When the control bank reaches either limit of this band, a change in boron concentration is required to compensate for additional reactivity changes. Since the insertion limit is set by a rod travel limit, a conservatively high calculation of the inserted worth is made which exceeds the normally inserted reactivity.

3.2.1.15 Xenon Stability Control

This 121-assembly core is too small to experience azimuthal, radial, or diametral xenon oscillations. Although minimal xenon oscillations may be experienced in the axial direction, experience has demonstrated these oscillations can be controlled with the normal control rods. Consequently, no extra rods are needed or provided to mitigate such spatial transients.

3.2.1.16 Excess Reactivity Insertion Upon Reactor Trip

The control requirements are nominally based on providing an amount of excess reactivity insertion upon a reactor trip sufficient to obtain the shutdown margin required by the COLR.

3.2.1.17 Calculated Rod Worths

The compliment of 33 control rods is arranged as shown in Figure 3.2-1. Table 3.2-3 lists the calculated worths of this rod configuration for BOL and EOL. In order to be sure of maintaining a conservative margin between calculated and required rod worths, the calculated reactivity worths listed are decreased in the design by 7 to 10% (as defined by the reload specific design) to account for any errors or uncertainties in the calculation. This worth is established for the condition that the highest worth rod is stuck in the fully withdrawn position. A comparison between calculated and measured rod worths in operating reactors has shown the calculations to be well within the allowed uncertainty of 7%.

3.2.1.18 Reactor Core Power Distribution

In order to meet the performance objectives without violating safety limits, the peak to average power density must be within the limits set by the nuclear hot channel factors. For the peak power point in the core at rated power, the nuclear heat flux hot channel factor, F^N_q, was established as specified in Table 3.2-1, line 18. For the hottest channel at rated power, the nuclear enthalpy rise hot channel factor, $F^N_{\Delta H}$, was established as specified in Table 3.2-1, line 19.

Power capability of a PWR core is determined largely by consideration of the power distribution and its interrelationship to limiting conditions involving:

1. The linear power density.
2. The fuel cladding integrity.
3. The enthalpy rise of the coolant.

To determine the core power capability, local as well as gross core neutron flux distributions have been determined for various operating conditions at different times in core life. Allowance for the effect of fuel densification has been made in the design (Reference 2, Reference 3, and Reference 32). The effects of rod bow on DNB have been taken into account and appropriate design penalties have been imposed.

The presence of control rods, burnable absorber rods, and chemical shim concentration all play significant roles in establishing the fission power distribution, in addition to the influence of thermal-hydraulic and temperature feedback considerations. The computer programs used to determine neutron flux distributions include a model to simulate nonuniform water and chemical shim density distributions.

Thermal-hydraulic feedback considerations are especially important late in cycle life where the magnitude of the flux redistribution and reactivity change with change in core power or rod movement are strongly influenced by enthalpy rise up the core and by the fuel burnup distribution. Consequently, extensive X, Y and Z power distribution analyses have been performed to evaluate fission power distributions. In-core instrumentation is employed to evaluate the core power distributions throughout core lifetime to assure that the thermal design criteria are met.

The control system for axial power distribution control is based on manual operation and the use of Constant Axial Offset Control (CAOC) analysis methodology (Reference 8, Reference 9 and Reference 10). Administrative procedures, alarms, and automatic rod stops guide the operator in performing these tasks.

The out-of-core nuclear instrumentation system supplies the necessary information for the operator to control the core power distribution within the limits established for the protection system design. This information consists of recorders for the long ion chambers which display the upper and lower ion chamber currents and indicators which give the difference in these two currents for each long ion chamber. The ion chamber currents to the recorders and indicators are calibrated against the in-core power distribution generated in the adjacent section of the core as obtained from the movable detector system. This essentially divides the core into eight sections,

four in the upper half and four in the lower half, and the operator manually positions the rods to maintain a prescribed relationship between the power generated in the upper and lower sections of the core.

The relationship between core power distribution and out-of-core nuclear instrumentation readings was established during the startup testing program. In-core flux measurements were made over the range of relative positions between control banks for reactor power in the range of 25% to 100%. These measurements, together with long ion chamber currents, were processed to yield the relationships between core average axial power generation, the axial peak factor, and axial offset as indicated by the out-of-core nuclear instrumentation. These relationships were then checked during operation to assess the effect of core burnup on the sensitivity between in-core power distribution and out-of-core readings.

The reactor core is subject to axial xenon oscillations at the end of a fuel cycle life. The axial instability is due principally to the negative moderator temperature coefficient of reactivity which exists at EOL. Since the moderator coefficient at BOL is small, the core is stable with respect to axial oscillations at BOL.

Figure 3.2-2 through Figure 3.2-4 show the radial power distributions in various planes of one quarter of the initial core at BOL. Figure 3.2-7 illustrates a typical reload pattern (OFA or STD assemblies) with four fuel regions. Figure 3.2-8 shows a typical BOL assembly burnup distribution for a low-low leakage loading pattern with an upgraded OFA reload. The location and number of IFBA rods is shown. Figure 3.2-9 through Figure 3.2-12 show the radial power distributions in various planes of one quarter of a typical reload core with a full OFA loading at BOL and EOL conditions. For the full upgraded OFA Core, Figure 3.2-13 through Figure 3.2-14a show typical radial power distributions at BOL and EOL conditions. For a full 422V+ core, Figure 3.2-15 and Figure 3.2-16 illustrate a typical core loading power distribution. The upgraded OFA and 422V+ cores may contain IFBAs, axial blankets, and part-length hafnium rods in peripheral assemblies.

A more detailed discussion of the background, and both the analytical and experimental data which forms the basis for this approach is given in Reference 4.

3.2.1.19 Analytical Methods

Calculations required in nuclear design consist of three distinct types which are performed in sequence:

1. determination of effective fuel temperatures for Doppler cross section calculation
2. generation of macroscopic few-group parameters
3. space-dependent, few-group diffusion calculations

These calculations have been performed using the PHOENIX-P and ANC computer codes (Reference 11 and Reference 33). Beginning with Unit 1 Cycle 37, PARAGON (Reference 55) computer code was implemented in the reload design analysis. PARAGON is a two-dimensional transport theory based code that calculates lattice physics constants. These are the same methods and models that have been used in several Westinghouse reload cycle designs. PARAGON can be used as a standalone or as a direct replacement for the previously licensed Westinghouse PWR PHOENIX-P lattice codes as approved by the NRC in Reference 55.

3.2.1.20 Fuel Temperature (Doppler) Calculations

Temperatures vary radially within the fuel rod, depending on the heat generation rate in the pellet, the conductivity of the materials in the pellet, gap, and cladding, and the temperature of the coolant.

Calculation of fuel pellet temperatures for Doppler cross section calculations is performed by the FIGHT-H computer code. PHOENIX-P and PARAGON incorporates, in their depletion, the same FIGHT-H fuel temperature calculational model used in the present Westinghouse design methodology. The FIGHT-H model includes radial variations of heat generation rate, thermal conductivity, and thermal expansion in the fuel pellet, elastic deflection of the cladding, and a pellet-clad gap conductance which depends on the kind of initial fill gas, the hot open gap dimension, and the fraction of the pellet circumference over which the gap is effectively closed due to pellet cracking. The steady-state radial temperature distribution in the fuel rod is calculated at a specified burnup, given the local value of the linear heat generation rate in the pellet and the moderator temperature and flow rate. The effective resonance temperatures of U-238 and Pu-240 are obtained by appropriate radial weighting of the temperature distribution, and used by PHOENIX-P and PARAGON in their depletion calculations.

3.2.1.21 Macroscopic Group Constants

Macroscopic few-group constants and analogous microscopic cross sections (needed for feedback and microscopic depletion analysis) are generated by PHOENIX-P or PARAGON (Reference 33 and Reference 55). PHOENIX-P is a two dimensional, multi-group transport theory code which has been approved by the USNRC. The nuclear cross section library used by PHOENIX-P contains cross section data based on multiple energy-group structure. The solution of the flux distribution is divided into two major steps in PHOENIX-P:

1. Solve for two-dimensional, multiple energy-group nodal fluxes which couple individual subcell regions (pellet, clad, moderator) as well as surrounding pins using a method based on collision probabilities and heterogeneous response flux.

2. Solve for a coarse energy-group flux distribution using a standard S4 discrete ordinates calculation and use these fluxes to normalize the detailed multiple energy group nodal fluxes from step 1.

PARAGON is a two dimensional, multi-group neutron (and Gamma) transport theory code which has been approved by the NRC. PARAGON contains cross-section data based on multiple energy-group structure. The PARAGON flux solver is based on Collision Probability theory and interface current coupling methods. The code uses the cross-section library group structure in all calculation steps (resonance self-shielding, flux solution, homogenization, and burnup calculation) to generate the multi-group data which will be used by a core simulator code. In the flux solution and depletion steps, the exact heterogeneity of the assembly is preserved in the calculation schemes.

PHOENIX-P or PARAGON is capable of modeling all cell types necessary for PWR design applications. Nodal group constants (two group) are obtained by flux-volume homogenization of the fuel cell (including IFBA pins), guide thimbles, instrumentation thimbles, and inter-assembly

gaps using the PHOENIX-P or PARAGON multi-group flux distribution. Group constants for control rods are calculated in a similar manner. Validation of the cross section method is based on analysis of critical experiments, isotopic data, and plant critical boron values at HZP and at HFP conditions as a function of burnup as discussed in detail in Reference 33 and Reference 55. Control rod worth measurements are also discussed in the reference.

Confirmatory critical experiments on burnable absorbers are described in Reference 6.

3.2.1.22 Spatial Few-Group Diffusion Calculations

The ANC code (Reference 11) is used in two-dimensional and three dimensional core calculations. ANC can be used in safety analysis calculations, and to determine critical boron concentrations, control rod worths, and reactivity coefficients.

Axial calculations are used to determine differential control rod worth curves (reactivity versus rod insertion) and axial power shapes during steady state and transient xenon conditions (flyspeck curve). Group constants and the radial buckling used in the axial calculation are obtained from the ANC radial calculation, in which group constants in annular rings representing the various material regions in the X-Y plane are homogenized by flux-volume weighting. Two-group axial calculations utilize APOLLO, an updated version of the PANDA code (Reference 35).

Validation of the spatial codes for calculating power distributions involves the use of incore and excore detectors and is discussed in Reference 5, Reference 33 and Reference 55.

Based on comparison with measured data it is estimated that the accuracy of current analytical methods is:

±0.2% $\Delta\rho$ for Doppler defect

±2 x 10^5 $\Delta k/k/°F$ for moderator coefficient

±50 ppm for critical boron concentration with depletion

±3% for power distributions

±0.2% $\Delta\rho$ for rod bank worth

±4 pcm/step for differential rod worth

±0.5 pcm/ppm for boron worth

±0.1% $\Delta\rho$ for moderator defect

3.2.2 THERMAL AND HYDRAULIC DESIGN AND EVALUATION

This section presents an evaluation of the characteristics and design parameters which are significant to the thermal-hydraulic design objectives. The capability of the reactor to achieve these objectives while performing safely under operational modes, including both transient and steady-state, is demonstrated in this section.

The thermal and hydraulic design parameters are given in Table 3.2-4

3.2.2.1 Thermal Hydraulic Design Basis

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

A. There is at least a 95% probability that DNB will not occur on the most limiting fuel rods during MODES 1 and 2, operational transients, or any conditions of moderate frequency at a 95% confidence level.

B. No fuel melting during any anticipated normal operating condition, operational transients, or any conditions of moderate frequency.

3.2.2.1.1 Departure from Nucleate Boiling Design Basis

There shall be at least a 95% probability (at a 95% confidence level) that departure from nucleate boiling (DNB) will not occur on the most limiting fuel rod during normal operation, operational transients, and any transient conditions arising from faults of moderate frequency (Condition I and II events).

By preventing DNB, adequate heat transfer is assured between the fuel cladding and the reactor coolant, thereby preventing cladding damage as a result of inadequate cooling. Maximum fuel rod surface temperature is not a design basis as it is within a few degrees of coolant temperature during operation in the nucleate boiling region. Limits provided by the nuclear control and protection systems are such that this design basis will be met for transients associated with Condition I1 events, including overpower transients.

3.2.2.1.2 Fuel Temperature Design Basis

For Condition I and II events, the fuel design and overpower protection system setpoints are designed to assure that a calculated centerline fuel temperature does not exceed the fuel melting temperature. The melting temperature of U02 is taken to be 5080°F (un-irradiated) and decreases by 58°F per 10,000 MWD/MTU of fuel burnup (Reference 45). The fuel temperatures have been evaluated by the same methods used for all Westinghouse fuel designs. Rod geometries, thermal properties, heat fluxes, and temperature differences are modeled to calculate the temperature at the surface and centerline of the fuel pellet. To preclude fuel melting, the peak local power experienced during Condition I and II events can be limited to a maximum value which is sufficient to ensure that the fuel centerline temperatures remain below the melting temperature at all burnups.

3.2.2.2 Thermal Hydraulic Design Analysis

3.2.2.2.1 Departure from Nucleate Boiling

3.2.2.2.1.1 DNBR Correlations

Departure from nucleate boiling (DNB) is predicated upon a combination of hydrodynamic and heat transfer phenomena and is affected by the local and upstream conditions including the heat flux distribution.

W-3

The W-3 DNB correlation (Reference 14) incorporates both local and system parameters in predicting the local DNB heat flux. The W-3 correlation was developed from tests with flow in tubes and rectangular channels. Good agreement is obtained when the correlation is applied to test data for rod bundles. This correlation includes the nonuniform axial heat flux effect. The W-3 correlation has been extensively validated against test data and shown to be conservative for the prediction of DNB in fuel rod bundles with and without mixing vane grids.

The W-3 DNBR correlation is used where the WRB-1 correlation is not applicable. The WRB-1 correlation was developed based on mixing vane data and therefore is only applicable in the heated rod spans above the first mixing vane grid. In addition, the W-3 correlation is applied in the analysis of accident conditions where the system pressure is below the range of the WRB-1 correlation. For system pressures in the range of 500 psia to 1000 psia, the W-3 correlation limit is 1.45 (Reference 42). For system pressures greater than 1000 psia, the W-3 correlation limit is 1.30.

WRB-1

The WRB-1 DNB correlation is based entirely on rod bundle data and takes credit for improvements in the accuracy of the critical heat flux predictions over previous DNB correlations. This correlation, based on local fluid conditions, represents the rod bundle data with better accuracy over a wide range of variables than the previous correlation used in design. This correlation accounts directly for both typical and thimble cold wall cell effects, uniform and non-uniform heat flux profiles, and variations in rod heated length and in grid spacing (Reference 18, Reference 19). A DNB evaluation based upon criteria presented in Reference 36, the Westinghouse Fuel Criteria Evaluation Process (FCEP), for the 422V+ fuel design concludes that the WRB-1 DNB correlation with a limit of 1.17 can be conservatively applied to this design.

3.2.2.2.1.2 DNBR Analysis

In conjunction with the WRB-1 correlation, the design method employed to meet the DNB design basis is “Revised Thermal Design Procedure” (RTDP) (Reference 22). With RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes, and DNB correlation predictions are considered statistically to obtain DNB uncertainty factors. Based on the DNB uncertainty factors, RTDP design limit DNBR values are determined such that there is at least 95-percent probability at a 95-percent confidence level that DNB will not occur on the most limiting fuel rod during normal operation and operational transients and during transient conditions arising from faults of moderate frequency. Only the random portion of the plant operating parameter uncertainties is included in the statistical combination. Instrumentation bias is treated as a direct DNBR penalty. Since the parameter uncertainties are considered in determining the RTDP design limit DNBR values, the safety analyses are performed using input parameters at their nominal values.

The RTDP design limit DNBR values are 1.24 and 1.23 for typical and thimble cells, respectively. To maintain DNBR margin to offset DNB penalties such as those due to fuel rod bow, transition cores, and instrumentation biases, the safety analyses are performed to DNBR limits higher than the design limit DNBR values. The difference between the design limit DNBRs and the safety analysis limit DNBRs results in available DNBR margin. The net DNBR margin, after consideration of all penalties, is available for operating and design flexibility.

The Standard Thermal Design Procedure (STDP) is used for those analyses where RTDP is not appropriate. In STDP, the parameters used in analysis are treated in a conservative way from a DNBR standpoint. The parameter uncertainties are applied directly to the plant safety analysis input values to give the lowest minimum DNBR. The DNBR limit for STDP is the appropriate DNB correlation limit increased by sufficient margin to offset the applicable DNBR penalties.

Prior to the power uprate to 1800 MWt, the THINC-IV code (Reference 20 and Reference 21) was used for the core thermal design. Commencing with the power uprate, the VIPRE-01 code is used for the core thermal design. VIPRE-01 is a three-dimensional subchannel code that has been developed to account for hydraulic and nuclear effects on the enthalpy rise in the core and hot channels (Reference 43 and Reference 44). VIPRE-01 modeling of a PWR core is based on one-pass modeling approach. In the one-pass modeling, hot channels and their adjacent channels are modeled in detail, while the rest of the core is modeled simultaneously on a relatively coarse mesh. The behavior of the hot assembly is determined by superimposing the power distribution upon inlet flow distribution while allowing for flow mixing and flow distribution between flow channels. Local variations in fuel rod power, fuel rod and pellet fabrication, and turbulent mixing are also considered in determining conditions in the hot channels. Conservation equations of mass, axial and lateral momentum, and energy are solved for the fluid enthalpy, axial flow rate, lateral flow, and pressure drop. The VIPRE-01 model has been demonstrated in Reference 44 to be equivalent to the THINC-IV code.

3.2.2.2.2 Hot Channel Factors

The total hot channel factors for heat flux and enthalpy rise are defined as the maximum to-core average ratios of these quantities. The heat flux factors consider the local maximum at a point (the “hot spot”), and the enthalpy rise factors involve the maximum integrated value along a channel (the “hot channel”).

Each of the total hot channel factors is a function of a nuclear hot channel factor describing the neutron flux distribution and an engineering hot channel factor to allow for variations in flow conditions and fabrication tolerances. The engineering hot channel factors are made up of subfactors accounting for the influence of the variations of fuel pellet diameter, density, enrichment and eccentricity; fuel rod diameter; pitch and bowing; inlet flow distribution; flow redistribution; and flow mixing. These engineering hot channel factors are described below.

Heat Flux Engineering Hot Channel Factor, $F^E Q$

The heat flux engineering hot channel factor is used to evaluate the maximum heat flux. This subfactor is determined by statistically combining the tolerances for the fuel pellet diameter, density, enrichment, eccentricity and the fuel rod diameter, and has a value of 1.03. Measured manufacturing data on recent Westinghouse fuel were used to verify that this value was not exceeded for 95 percent of the limiting fuel rods at a 95 percent confidence level. Thus, it is expected that a statistical sampling of the fuel assemblies of the reference plant will yield a value no larger than 1.03. This factor is used in kW/ft analyses. An additional factor of 0.003 is added for DNB analysis to account for a nonuniform azimuthal heat flux distribution.

Enthalpy Rise Engineering Hot Channel Factor, $F^{E}DH$

The effect of variations in flow conditions and fabrication tolerances on the hot-channel enthalpy rise is directly considered in the core thermal subchannel analysis under any reactor operating condition. The items considered contributing to the enthalpy rise engineering hot channel factor are discussed below:

1. Pellet Diameter, Density and Enrichment and Fuel Rod Diameter, Pitch and Bowing Design values employed in the VIPRE analysis related to the above fabrication variations are based on applicable limiting tolerances such that these design values are met for 95 percent of the limiting channels at a 95 percent confidence level. Measured manufacturing data cited above show the tolerances used in this evaluation are conservative. These fabrication variations are considered statistically in establishing the DNBR limit.

2. Inlet Flow Maldistribution

   Studies performed on 1/7 scale hydraulic reactor models indicate that a conservative design basis is to consider a 5% reduction in the flow to the hot fuel assembly under isothermal conditions.

3. Flow Redistribution

   The flow redistribution accounts for the reduction in flow in the hot channel resulting from the high flow resistance in the channel due to the local or bulk boiling. The effect of the non-uniform power distribution is inherently considered in the VIPRE analysis for every operating condition which is evaluated.

4. Flow Mixing

   Mixing vanes have been incorporated into the spacer grid design. These vanes induce flow mixing between the various flow channels in a fuel assembly and also between adjacent assemblies. This mixing reduces the enthalpy rise in the hot channel resulting from local power peaking or unfavorable mechanical tolerances.

3.2.2.2.3 Hydraulic Analysis

Pressure Drop and Hydraulic Forces

The total pressure drop across the reactor vessel, including the inlet and outlet nozzles, and the pressure drop across the core are listed in Table 3.2-4. These values include a 10% uncertainty factor. The hydraulic forces are not sufficient to lift a control rod cluster during normal operation even if the rod is not attached to its coupling.

Hydrodynamic and Flow Power Coupled Instability

Boiling flows may be susceptible to thermohydrodynamic instabilities (Reference 46). These instabilities are undesirable in reactors since they may cause a change in thermohydraulic conditions that may lead to a reduction in the DNB heat flux relative to that observed during a steady flow condition or to undesired forced vibrations of core components.

Two (2) specific types of flow instabilities are considered for Westinghouse PWR operation. These are the Ledinegg of flow excursion type of static instability and the density wave type of dynamic instability.

Ledinegg instability involves a sudden change in flow rate from one steady-state to another. This instability occurs when the slope of the reactor coolant system pressure drop-flow rate curve ($\delta\Delta P/\delta G$ internal) becomes algebraically smaller than the loop supply (pump head) pressure drop-flow rate curve ($\delta\Delta P/\delta G$ external). The Westinghouse pump head pressure drop-flow rate curve has negative slope ($\delta\Delta P/\delta G$ external <0) whereas the reactor coolant system pressure drop-flow rate curve has a positive slope ($\delta\Delta P/\delta G$ internal > 0) over the Condition I and II operational ranges. Thus, the Ledinegg instability will not occur.

The mechanism of density wave oscillations in a heated channel has been described by Lahey and Moody (Reference 47). However, since the total pressure drop across the core is maintained by the characteristics of the fluid system external to the core, then the two-phase pressure drop perturbation feeds back to the single phase region. These resulting perturbations can be either attenuated or self-sustained.

A simple method has been developed by Ishii (Reference 48) for parallel closed channel systems to evaluate whether a given condition is stable with respect to the density wave type of dynamic instability.

The application of the method of Ishii to Westinghouse reactor designs is conservative due to the parallel open channel feature of Westinghouse PWR Cores. For such cores, there is little resistance to lateral flow leaving the flow channels of high power density. This coupling with cooler channels has led to the opinion that an open channel configuration is more stable than the above closed channel analysis under the same boundary conditions.

Flow instabilities which have been observed have occurred almost exclusively in closed channel systems operating at low pressures relative to the Westinghouse PWR operating pressures. Kao, Morgan, and Parker (Reference 49) analyzed parallel closed channel stability experiments simulating a reactor core flow. These experiments were conducted at pressures up to 2200 psia. The results showed that for flow and power levels typical of power reactor conditions, no flow oscillations could be induced above 1200 psia.

Additional evidence that flow instabilities do not adversely affect thermal margin is provided by the data from the rod bundle DNB tests.

In summary, it is concluded that thermohydrodynamic instabilities will not occur under Condition I and II modes of operation for Westinghouse PWR reactor designs. A large power margin, greater than doubling rated power, exists to predicted inception of such instabilities. Analysis has been performed which shows that minor plant to plant differences in Westinghouse reactor designs such as fuel assembly arrays, core power to flow ratios, fuel assembly length, etc. will not result in gross deterioration of the above power margins.

3.2.2.2.4 Fuel Temperature Analysis

Fuel rod thermal evaluations (fuel centerline, average and surface temperatures) are performed at several times in the fuel rod lifetime (with consideration of time dependent densification) to determine the maximum fuel temperatures using the PAD 4.0 code (Reference 50). The fuel rod behavior is evaluated utilizing a semi-empirical thermal model which considers, in addition to the thermal aspects, such items as cladding creep, fuel swelling, fission gas release, release of

absorbed gases, cladding corrosion and elastic deflection, and helium solubility. To preclude fuel melting, the peak local power experienced during Condition I and II events can be limited to a maximum value which is sufficient to ensure that the fuel centerline temperatures remain below the melting temperature at burnups less than 62,000 MWD/MTU.

3.2.2.2.5 Effects of DNB on Neighboring Rods

DNB has never been observed to occur in a group of neighboring rods in a rod bundle as a result of DNB in one rod in the bundle.

3.2.2.2.6 DNB with Physical Burnout

Westinghouse has conducted DNB tests in a 25-rod bundle where physical burnout occurred with one rod (Reference 15). After this occurrence, the 25-rod test section was used for several days to obtain more DNB data from the other rods in the bundle. The burnout and deformation of the rod did not affect the performance of neighboring rods in the test section during the burnout or the validity of the subsequent data points. No occurrences of flow instability or other abnormal operation were observed.

3.2.2.2.7 DNB with Return to Nucleate Boiling

Additional DNB tests have been conducted by Westinghouse (Reference 16) on 19 and 21 rod bundles. In these tests, DNB without physical burnout was experienced more than once on single rods in the bundles for short periods of time. Each time, a reduction to power of approximately 10% was sufficient to re-establish nucleate boiling on the surface of the rod. During these and subsequent tests, no adverse effects were observed on this rod or any other rod in the bundle as consequences of operating in DNB.

3.2.2.2.8 Rod Bow As Applied to DNBR Analysis

DNBR reduction as a result of rod bow is calculated by:

$$MDNBR_B = MDNBR_{NB} (1-\delta_B)$$

where:

MDNBR = minimum DNBR
$MDNBR_{NB}$ = MDNBR for non-bowed fuel
$MDNBR_B$ = MDNBR for bowed fuel
δ_B = rod bow penalty, fractional reduction in MDNBR due to bowing

Westinghouse's detailed methodology for calculating fuel rod bowing and its MDNBR effect is given in Reference 23.

δ_B is given as a function of assembly average burnup. The value of δ_B is less than 3.5% for low flow. This value bounds the full flow value of the rod bow penalty and is used for the full and reduced flow calculations. This rod bow penalty is representative for an average assembly burnup of 24,000 MWD/MTU (Reference 24).

While the amount of rod bowing increases beyond this exposure, the fuel is not capable of achieving limiting peaking factors due to the decrease in fissionable isotopes and buildup of fission product inventory. The physical burndown effect is greater than the rod bowing effects which would be calculated based on the amount of bow predicted at those burnups.

Therefore, for the purpose of evaluating effects of rod bow on Westinghouse fuel, 24,000 MWD/MTU represents the maximum burnup of concern.

The reduction in MDNBR is accounted for by taking a DNBR penalty for all conditions at which the rod bow penalty applies.

3.2.3 MECHANICAL DESIGN AND EVALUATION

The reactor core and reactor vessel internals are shown in cross-section in Figure 3.2-34 and in elevation in Figure 3.2-35. The core, consisting of the fuel assemblies and control rods, provides and controls the heat source for the reactor operation. Source rods, thimble plugging devices and burnable absorber rods in RCC guide thimbles are no longer being used. The internals, consisting of the lower core support structure, upper core support assembly, in-core instrumentation support structures, core barrel and thermal shield, are designed to support, align, and guide the core components, direct the coolant flow to and from the core components, and to support and guide the in-core instrumentation. The laws of the State of Wisconsin require American Society of Mechanical Engineers (ASME) Code construction on the reactor vessel. A listing of the core mechanical design parameters for the initial core as well as reload fuel, is given in Table 3.2-5.

The fuel assemblies are arranged in a roughly circular cross-sectional pattern. The assemblies are all mechanically compatible and similar in design, but contain fuel of different enrichments depending on the location of the assembly within the core. Each reload core is designed to utilize fresh and previously burned fuel in a low leakage loading pattern.

The fuel is in the form of slightly enriched uranium dioxide (UO_2) ceramic pellets. In STD and OFA fuel assemblies, the pellets are stacked to an active height of 144 inches within Zircaloy-4 tubular cladding which is plugged and seal welded at the ends to encapsulate the fuel. In the 422V+ fuel, the active height of the pellet stack has been reduced to 143.25 inches within the ZIRLO® or Optimized ZIRLO™ tubular cladding which is plugged and seal welded at the ends to encapsulate the fuel. Heat generated by the fuel is removed by demineralized borated light water which flows upward through the fuel assemblies and acts as both moderator and coolant.

The control rods, designated as rod cluster control (RCC) assemblies, consist of groups of individual absorber rods which are held together by a spider at the top end and actuated as a group. In the inserted positions, the absorber rods fit within hollow guide thimbles in the fuel assemblies. The guide thimbles are an integral part of the fuel assemblies and occupy locations within the regular fuel rod pattern where fuel rods have been deleted. In the withdrawn position, the absorber rods are guided and supported laterally by guide tubes which form an integral part of the upper core support structure. Figure 3.4-1 shows a typical RCCA, and Section 3.4 describes the functional operation of the RCCAs.

As shown in Figure 3.2-35, the fuel assemblies are positioned and supported vertically in the core between the upper and lower core plates. The core plates are provided with pins which index into closely fitting mating holes in the fuel assembly top and bottom nozzles. The pins maintain the

fuel assembly alignment, which permits free movement of the control rods from the fuel assembly into the guide tubes in the upper support structure without binding or restriction between the rods and their guide surfaces.

Operational or seismic loads imposed on the fuel assemblies are transmitted through the core plates to the upper and lower support structures and ultimately to the internals support ledge at the pressure vessel flange in the case of vertical loads, or to the lower radial support and internals support ledge in the case of horizontal loads. The internals also provide a form fitting baffle surrounding the fuel assemblies which confines the upward flow of coolant in the core area to the fuel bearing region.

3.2.3.1 Reactor Internals Design Description

The reactor internals are designed to support and orient the reactor core fuel assemblies and control rod assemblies, absorb the control rod dynamic loads and transmit these and other loads to the reactor vessel flange, provide a passageway for the reactor coolant, and support in-core instrumentation. The reactor internals are shown in Figure 3.2-35.

The internals are designed to withstand the forces due to weight, preload of fuel assemblies, control rod dynamic loading, vibration, possible blowdown forces, and earthquake acceleration. These internals are analyzed in a manner similar to Connecticut Yankee, San Onofre, Zorita, Saxton, and Yankee. Under the loading conditions, including conservative effects of design earthquake loading, the structure satisfies stress values prescribed in Section III, ASME Nuclear Vessel Code. The dynamic criteria for design and stress levels of the internals in this plant are similar to those in Connecticut Yankee.

The reactor internals are equipped with bottom-mounted in-core instrumentation supports. These supports are designed to sustain the applicable loads outlined above.

In the event of downward vertical displacement of the internals, energy absorbing devices limit the displacement by contacting the vessel bottom head. The load is transferred through the energy absorbing devices to the vessel. The energy absorbers, cylindrical in shape, are contoured on their bottom surface to the reactor vessel bottom head geometry. Their number and design are determined so as to limit the forces imposed to a safe fraction of yield strength. Assuming a downward vertical displacement, the potential energy of the system is absorbed mostly by the strain energy of the energy absorbing devices (see Figure 3.2-37).

The free fall in the hot condition is on the order of 1/2 inch, and there is an additional strain displacement in the energy absorbing devices of approximately 3/4 inch. Alignment features in the internals prevent cocking of the internals structure during this postulated drop. The control system, as designed, provides assurance of control rod insertion capabilities under these assumed drop conditions. The drop distance of about 1-1/4 inch is not enough to cause the tips of the shutdown group of RCCAs to come out of the guide tubes in the fuel assemblies.

The components of the reactor internals are divided into three parts consisting of the lower core support structure (including the entire core barrel and thermal shield), the upper core support structure, and the in-core instrumentation support structure.

Lower Core Support Structure

The major containment and support member of the reactor internals is the lower core support structure shown in Figure 3.2-37. This support structure assembly consists of the core barrel, the core baffle, the lower core plate and support columns, the thermal shield, the intermediate diffuser plate, and the bottom support plate which is welded to the core barrel. All the major material for this structure is Type 304 stainless steel. The core support structure is supported at its upper flange from a ledge in the reactor vessel head flange and its lower end is restrained in its transverse movement by a radial support system attached to the vessel wall. Within the core barrel are axial baffle and former plates which are attached to the core barrel wall and form the enclosure periphery of the assembled core. The lower core plate is positioned at the bottom level of the core below the baffle plates and provides support and orientation for the fuel assemblies.

The lower core plate provides the necessary flow distributor holes for each fuel assembly. Fuel assembly locating pins (two for each assembly) are also inserted into this plate. Columns are placed between this plate and the bottom support plate of the core barrel in order to provide stiffness to this plate and transmit the core load to the bottom support plate. Intermediate between the support plate and lower core support plate is positioned a perforated plate to uniformly diffuse the coolant flowing into the core.

Irradiation baskets, into which material samples can be inserted and irradiated during reactor operation, are attached to the thermal shield. The irradiation capsule basket supports are welded to the thermal shield. There is no extension of this support above the thermal shield as was done in the older designs. Thus, the basket has been removed from the high flow disturbance zone. The welded attachment to the shield extends the full length of the support except for small interruptions about one inch long. This type of attachment has an extremely high natural frequency. The specimens are held in position within the baskets by a stop on the bottom and a slotted cylindrical spring at the top which fits against a relief in the basket. The specimen does not extend through the top of the basket and thus is protected by the basket from the flow.

The lower core support structure, and principally the core barrel, serve to provide passageways and control for the coolant flow. Inlet coolant flow from the vessel inlet nozzles proceeds down the annulus between the core barrel and the vessel wall, flows on both sides of the thermal shield, and then into a plenum at the bottom of the vessel. It then turns and flows up through the lower support plate, passes through the intermediate diffuser plate, and then through the lower core plate. The flow holes in the diffuser plate and the lower core plate are arranged to give a very uniform entrance flow distribution to the core. After passing through the core, the coolant enters the area of the upper support structure and then flows generally radially to the core barrel outlet nozzles and directly through the vessel outlet nozzles.

A small amount of water flows between the baffle plates and core barrel to provide additional cooling of the barrel. Similarly, a small amount of the entering flow is directed into the vessel head plenum and exits through the vessel outlet nozzles.

Vertically downward loads from weight, fuel assembly preload, control rod dynamic loading, and earthquake acceleration are carried by the lower core plate partially through the lower core plate support flange on the core barrel shell and partially through the lower support columns to the bottom support plate and thence through the core barrel shell to the core barrel flange supported by the vessel head flange. Transverse loads from earthquake acceleration, coolant cross flow, and

vibration are transmitted to the core barrel shell to be shared between the lower radial support and the vessel head flange. Transverse acceleration of the fuel assemblies is transmitted to the core barrel shell by direct connection of the lower core support plate to the barrel wall and by a radial support type connection of the upper core plate to slab sided pins pressed into the core barrel.

The main radial support system of the core barrel is accomplished by "key" and "keyway" joints to the reactor vessel wall. At equally spaced points around the circumference, an Inconel block is welded to the vessel inner face. Another Inconel block is bolted to each of these blocks, and has a "keyway" geometry. Opposite each of these is a "key" which is attached to the internals. At assembly, as the internals are lowered into the vessel, the keys engage the keyways in the axial direction. With this design, the internals are provided with a support at the farthest extremity and may be viewed as a beam fixed at the top and simply supported at the bottom.

Radial and axial expansions of the core barrel are accommodated, but transverse movement of the core barrel is restricted by this design. With this system, cycle stresses in the internal structures are within the ASME Section III limits.

Upper Core Support Assembly

The upper core support assembly shown in Figure 3.2-38 consists of the upper support plate, deep beam sections, and upper core plate between which are contained support columns and guide tube assemblies. The support columns which establish the spacing between the upper support plate, deep beam sections, and the upper core plate are fastened at top and bottom to these plates and beams. The support columns transmit the mechanical loadings between the two plates and serve the supplementary function of supporting thermocouple guide tubes. The guide tube assemblies shown on Figure 3.2-39 sheath and guide the control rod drive shafts and control rods and provide no other mechanical functions. They are fastened to the upper support plate and are guided by pins in the upper core plate for proper orientation and support. Additional guidance for the control rod drive shafts is provided by the control rod shroud tube which is attached to the upper support plate and guide tube.

The upper core support assembly, which is removed as a unit during refueling operation, is positioned in its proper orientation with respect to the lower support structure by flat-sided pins pressed into the core barrel which in turn engage in slots in the upper core plate. At an elevation in the core barrel where the upper core plate is positioned, the flat-sided pins are located at equal angular positions. Slots are milled into the upper core plate at the same positions. As the upper support structure is lowered into the main internals, the slots in the plate engage the flat-sided pins in the axial direction. Lateral displacement of the plate and of the upper support assembly is restricted by this design. Fuel assembly locating pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper assembly is lowered into place. Proper alignment of the lower core support structure, the upper core support assembly, the fuel assemblies, and control rods is thereby assured by this system of locating pins and guidance arrangement. The upper core support assembly is restrained from any axial movements by a large circumferential spring which rests between the upper barrel flange and the upper core support assembly, and is compressed by the reactor vessel head flange.

Vertical loads from fuel assembly preload are transmitted through the upper core plate via the support columns to the deep beams and upper support plate and then through the circumferential

spring to the reactor vessel head. Transverse loads from coolant cross flow, earthquake acceleration, and possible vibrations are distributed by the support columns to the upper support plate and upper core plate. The upper support plate is particularly stiff to minimize deflection.

In-Core Instrumentation Support Structures

The in-core instrumentation support structures consist of an upper system to convey and support thermocouples penetrating the vessel through the head and a lower system to convey and support flux thimbles penetrating the vessel through the bottom.

The upper system utilizes the reactor vessel head penetrations. Instrumentation port columns are slip-connected to in-line columns that are in turn fastened to the upper support plate. These port columns protrude through the head penetrations. The thermocouples are routed through these port columns and across the upper support plate to positions above their readout locations. The thermocouple conduits are supported from the columns of the upper core support system. The thermocouple conduits are sealed stainless steel tubes.

In addition to the upper in-core instrumentation, there are reactor vessel bottom port columns which carry the retractable, cold worked stainless steel flux thimbles that are pushed upward into the reactor core. Conduits extend from the bottom of the reactor vessel down through the concrete shield area and up to a thimble seal table. The minimum bend radius is about 90 inches and the trailing ends of the thimbles (at the seal table) are extracted approximately 13 feet during refueling of the reactor in order to avoid interference within the core. The thimbles are closed at the leading ends and serve as the pressure barrier between the reactor pressurized water and the nuclear detector.

Mechanical seals between the retractable thimbles and the surrounding conduits are provided to seal the reactor coolant from the containment atmosphere. Thus, primary system pressure exists up to the seal table. During normal operation, the retractable thimbles are stationary in the core and are moved only during refueling or for maintenance. Section 7.6 contains more information on the arrangement of the in-core instrumentation system.

The in-core instrumentation support structure is designed for adequate support of instrumentation during reactor operation and is rugged enough to resist damage or distortion under the conditions imposed by handling during the refueling sequence.

Evaluation of Core Barrel and Thermal Shield

The internals design is based on analysis, test, and operational information. Problems in previous Westinghouse PWRs have been evaluated and information derived has been considered in this design. For example, the Point Beach design uses a one-piece thermal shield which is attached rigidly to the core barrel at one end and flexured at the other. The earlier designs that malfunctioned were multi-piece thermal shields that rested on vessel lugs and were not rigidly attached at the top.

Early core barrel designs that have malfunctioned in service, now abandoned, employed threaded connections such as tie rods joining the bottom support to the bottom of the core barrel, and a bolted connection that tied the core barrel to the upper barrel. The malfunctioning of the core barrel designs in earlier service was believed to have been caused by the thermal shield which was

oscillating, thus creating forces on the core barrel. Other forces were induced by unbalanced flow in the lower plenum of the reactor. In the Point Beach RCCA design there are no fuel followers to necessitate a large bottom plenum in the reactor. The elimination of these fuel followers enabled Westinghouse to build a shorter core barrel.

The Connecticut Yankee reactor and the Zorita reactor core barrels are of the same construction as the Point Beach reactor core barrel. Deflection measuring devices employed in the Connecticut Yankee reactor during the hot-functional test, and deflection and strain gauges employed in the Zorita reactor during the hot-functional test have provided important information that has been used in the design of the later internals, including that for Point Beach. When the Connecticut Yankee thermal shield was modified to the same design as for Southern California Edison, it, too, operated satisfactorily as was evidenced by the examination after the hot-functional test. After these hot-functional tests on all of these reactors, a careful inspection of the internals was provided. All the main structural welds were examined, nozzle interfaces were examined for any differential movement, upper core plate inside supports were examined, the thermal shield attachments to the core barrel, including all lock welds on the devices used to lock the bolt were checked; no malfunctions were found.

Substantial scale model testing was performed at Westinghouse Atomic Power Division. This included tests which involved a complete full-scale fuel assembly which was operated at reactor flow, temperature, and pressure conditions. Tests were run on a 1/7 scale model of the Indian Point 1 reactor. Measurements taken from these tests indicate very little shield movement, on the order of a few mils when scaled up to Point Beach. Strain gauge measurements taken on the core barrel also indicate very low stresses. Testing to determine thermal shield excitation due to inlet flow disturbances have been included. Information gathered from these tests was used in the design of the thermal shield and core barrel. It can be concluded from the testing program and the analyses and with the experience gained that the design as employed on the Point Beach Nuclear Plant is adequate.

Point Beach Nuclear Plant Units 1 and 2 achieved initial criticality in November 1970 and in May 1972, respectively. The reactor vessels were constructed prior to the existence of many of the present materials requirements. Accordingly, analyses have been performed to evaluate the fracture toughness of the reactor vessels and vessel internals and tests have been conducted on the materials surveillance capsules to verify the adequacy of the original design.

The core barrel support pads thermal, mechanical, and pressure stresses are calculated at various locations on the pad and at the vessel wall. Mechanical stresses are calculated by the flexure formula for bending stress in a beam; pressure stresses are taken from the analysis of the vessel to bottom head juncture; and thermal stresses are determined by the conservative method of skin stresses. The stresses due to the cyclic loads are multiplied by a stress concentration factor where applicable and used in the fatigue evaluation.

In the event of a loss-of-coolant accident and subsequent operation of the emergency core cooling system, cold water is injected from the accumulators, through the nozzles and downcomer, to the core. Thermal gradients through the core support components will originate transient thermal stresses. Analysis shows that the worst thermal stress case occurs to the core barrel. The barrel is affected by the cold water in the downcomer and the somewhat hotter water in the compartments

between barrel and baffle, producing a thermal gradient across barrel wall. The lower support structure is cooled more uniformly because of the large and numerous flow holes, and consequently, thermal stresses are lower.

The method used to obtain the maximum barrel stresses is as follows:

1. Temperature distribution across the barrel wall is computed as a function of time taking into consideration water temperatures and film coefficients.

2. Assuming that the obtained thermal gradients are axisymmetrically distributed, which is conservative for stresses, maximum thermal stresses are computed in the barrel considered as an infinite cylinder.

3. Thermal stresses are added to primary stresses, including seismic, in order to obtain the maximum stress state of the barrel.

Results of studies performed for different conditions show that maximum thermal stresses in the barrel wall are well below the allowable criteria given for design by Section III of the ASME Code.

Interaction Analyses

The following discussion is applicable to the original reactor vessel components and has been retained without revision.

Areas of discontinuity or stress concentration of the following components of the reactor pressure vessel have been analyzed in detail through systematic analytical procedures. The reactor vessel areas and discontinuity geometries are presented in Figure 3.2-43. A summary description of the stress analysis is provided.

An interaction analysis is performed on the CRDM housing. The flange is assumed to be a ring and the tube is assumed to be a long cylinder. The different values of Young's Modulus and coefficients of thermal expansion of the tubes are taken into account in the analysis. The local flexibility is considered at appropriate locations. The closure head is treated as a perforated spherical shell with modified elastic constants. The effects of redundants on the closure head are assumed to be local only. Using the mechanical and thermal stresses from this analysis, a fatigue evaluation is made for the J weld.

The closure head, closure head flange, vessel flange, vessel shell, and closure studs are all evaluated in the same analysis. An analytical model is developed by dividing the actual structure into different elements such as sphere, ring, long cylinder, and cantilever beam, etc. An interaction analysis is performed to determine the stresses due to mechanical and thermal loads. These stresses are evaluated in light of the strength and fatigue requirements of the ASME Boiler and Pressure Vessel Code Section III. A similar analysis is performed for the vessel flange to vessel shell juncture and main closure studs.

For the analysis of nozzle and nozzle to shell juncture, the loads considered are internal pressure, operating transients, thermally induced and seismic pipe reactions, static weight of vessel, earthquake loading, and expansion and contraction, etc. A combination of methods is used to evaluate the stresses due to mechanical and thermal loads and external loads resulting from seismic pipe reactions, earthquake, and pipe break, etc.

For fatigue evaluation, peak stresses resulting from external loads and thermal transients are determined by concentrating the stresses as calculated by the above described methods. Combining these stresses enables the fatigue evaluation to be performed. Method of analysis for outlet nozzle and vessel supports is the same as described above.

Vessel wall transition is analyzed by means of a standard interaction analysis. The thermal stresses are determined by the skin stress method where it is assumed that the inside surface of the vessel is at the same temperature as the reactor coolant and the mean temperature of the shell remains at the steady state temperature. This method is considered conservative.

For the bottom head to shell juncture, the standard interaction analysis and skin stress methods are employed to evaluate the stresses due to mechanical and thermal stresses, respectively. The fatigue evaluation is made on a cumulative basis where superposition of all transients is taken into consideration.

For the bottom head instrument penetrations, an interaction analysis is performed by dividing the actual structure into an analytical model composed of different structural elements. The effects of the redundants on the bottom head are assumed to be local only. It is also assumed that for any condition where there is interference between the tube and the head no bending at the weld can exist. Using the mechanical and thermal stresses from this analysis, a fatigue evaluation is made for the J weld.

The location and geometry of the areas of discontinuity and/or stress concentration are shown in Figure 3.2-43.

For reactor vessels, the maximum thermal stress due to gamma ray heating occurs in the cylindrical portion of the vessel adjacent to the core and its value is about 2200 psi. This additional thermal stress does not augment the stress intensity values considerably. The maximum stress intensity values under steady state and transient operating conditions are still far below the allowable limits of N-414 of ASME Boiler and Pressure Vessel Code Section III. The effect of gamma ray heating on the cumulative usage factor is negligible.

The following pressure or strength bearing stainless steel component parts in the reactor vessel became furnace sensitized during the fabrication sequence:

1. Four primary nozzle safe ends - weld metal buttering

2. Two safety injection nozzle safe ends - forgings

3. Bottom instrumentation safe ends - forgings

Follow-up nondestructive examinations show no loss of integrity of the materials in the furnace sensitized areas.

<u>Interaction Analyses for Replacement Reactor Vessel Closure Heads and CRDMs</u>

The original reactor vessel closure heads, along with their respective CRDMs, have been replaced. The following discussion applies to the replacement components.

The pressure-retaining portions of the replacement CRDMs are evaluated to the requirements of Section III of the ASME Boiler and Pressure Vessel Code. The analysis is performed using two-dimensional axisymmetric finite element models of critical portions of the pressure housings, including models of the lower portion of the latch housing, the upper portion of the latch housing and the lower portion of the rod travel housing including the full penetration weld, and the upper portion of the rod travel housing. Stresses and fatigue usages are calculated for critical locations in the lower and upper portions of the latch housing, as well as the weld to the rod travel housing. The rod travel housing region is qualified by comparison with the upper latch housing region, which is concluded to bound the critical locations of the rod travel housing for stress and fatigue considerations.

The replacement vessel closure head and closure head flange have been evaluated in a separate analysis to the stress and fatigue requirements of Section III of the ASME Boiler and Pressure Vessel Code. Stresses are calculated using a finite element analysis approach. Several analysis models are used to envelope all locations of the replacement closure head. These models include the reactor vessel closure head, closure head flange, CRDM head adapter (including a portion of the head) for the centermost and outermost penetrations, the bimetallic weld joint between the head adapters and CRDM adapter flange tubes, the vent pipe to shell junction, and the instrument port head adapter flange for the core exit thermocouple nozzle assembly. Lift lugs on the closure head were also analyzed, but this analysis used empirical (not finite element) analysis techniques.

3.2.3.2 Core Components Design Description

Fuel Assembly

All of the fuel assemblies in the core are of similar design. The overall configuration of the fuel assemblies is shown in Figure 3.2-41 Sheets 1 to 7. The assemblies are square in cross section, nominally 7.761 inches on a side, and have an overall height (excluding hold down springs) of 159.975 inches for STD and OFA and 159.775 inches for upgraded OFA and 422V+.

The Westinghouse 14x14 422V+ fuel assembly design is a 14x14 array with a 0.422 inch fuel rod design. The 14x14 422V+ fuel assembly incorporates and adapts many of the current Westinghouse advanced fuel features.

The 14x14 422V+ fuel assembly features include: reconstitutable top nozzles (RTNs), reduced rod bow (RRB) inconel top grid, ZIRLO OFA-type mid-grids, high burnup inconel bottom grid, skirted debris filter bottom nozzle (DFBN), ZIRLO guide thimble and instrumentation tubes and fuel rods with zirconium dioxide coated cladding.

Although the 14x14 422V+ fuel assembly design incorporates and adapts many new Westinghouse fuel features, the parameters and dimensions are designed to be identical or compatible with STD or OFA Point Beach Unit 1 and 2 fuel. Figure 3.2-41 Sheet 6 illustrates the overall height and grid elevation dimensions of the Westinghouse 14x14 422V+ fuel assembly. The principle differences between the 14x14 422V+ fuel assembly and the 14x14 OFA fuel designs are:

1. 0.422 inch fuel rod outer diameter versus 0.400 inches for the 14x14 OFA
2. Thin strap ZIRLO OFA-type mid-grids versus thick-strap Zircaloy-4 OFA mid-grid
3. 0.200 inch lower top nozzle adapter plate and top grid elevation

The fuel rods in a fuel assembly are arranged in a square array with 14 rod locations per side and a nominal centerline-to-centerline pitch of 0.556 inch between rods. Of the total possible 196 rod locations per assembly, 16 are occupied by guide thimbles for the RCC rods or burnable absorber rods and one for in-core instrumentation. The remaining 179 locations contain fuel rods. However, limited substitutions of Zircaloy-4, ZIRLO® or Optimized ZIRLO™, or stainless steel filler rods for fuel rods, in accordance with NRC approved applications of fuel rod configurations, may be used when fuel assembly reconstitution is required due to leaking fuel rods. This methodology is addressed in detail in Reference 37. In addition to fuel rods, a fuel assembly is composed of a top nozzle, a bottom nozzle, 7 grid assemblies, 16 absorber rod guide thimbles, and one instrumentation thimble. Figure 3.2-40 shows a typical fuel assembly and control cluster cross section.

The guide thimbles, in conjunction with the grid assemblies and the top and bottom nozzles, comprise the basic structural fuel assembly skeleton. The top and bottom ends of the guide thimbles are fastened to the top and bottom nozzles, respectively. The grid assemblies, in turn, are fastened to the guide thimbles at each location along the height of the fuel assembly at which lateral support for the fuel rods is required. Within this skeletal frame-work, the fuel rods are contained and supported and the rod-to-rod centerline spacing is maintained along the assembly. Figure 3.2-41 shows a typical fuel assembly outline and detail.

Bottom Nozzle

The bottom nozzle is a square pedestal structure which controls the coolant flow distribution to the fuel assembly and functions as the bottom structural element of the fuel assembly. The nozzle, which is square in cross section, is fabricated from 304 stainless steel parts consisting of a perforated plate and 4 pads or feet. The legs are welded to the plate to form a plenum space for the inlet coolant to the fuel assembly. The perforated plate serves as the bottom end support for the fuel rods. The bottom support surface for the fuel assembly is formed under the plenum space by the four pads which are welded to the corner angles.

Coolant flow to the fuel assembly is directed from the plenum in the bottom nozzle, upward to the interior of the fuel assembly and to the channel between assemblies.

A stainless steel debris filter bottom nozzle (DFBN) has been introduced into the Point Beach Unit 1 Region 19 (Cycle 17 feed) upgraded OFAs and into the Unit 2 Region 18 (Cycle 16 feed) upgraded OFAs to reduce the possibility of fuel damage due to debris-induced fretting. The 14 x 14 DFBN design is based upon the VANTAGE 5 bottom nozzle design (Reference 7). The re-designed top plate of the nozzle includes a revised pattern of smaller flow holes that are sized to minimize passage of flow-entrained debris particles large enough to cause damage while providing sufficient flow area, comparable pressure drop, and continued structural integrity of the nozzle. A typical revised bottom nozzle flow hole pattern is illustrated in Figure 3.2-36. The DFBN design also incorporates the “extended burnup,” or low profile, geometry of the VANTAGE 5 bottom nozzle. This low profile nozzle, which has a reduced nozzle height and a thinner top plate, is designed to accommodate longer fuel rods and provide greater fuel rod growth room within the 14 x 14 upgraded OFA and 422V+ assemblies. Increased fuel rod plenum volumes and rod growth gaps accommodate the increased fission gas releases and fuel rod growths associated with extended discharge fuel burnups.

The RCC guide thimbles, which carry axial loads imposed on the assembly, are fastened to the bottom nozzle end plate. These loads, as well as the weight of the assembly, are distributed through the nozzle to the lower core support plate. Indexing and positioning of the fuel assembly in the core is controlled through two holes in diagonally opposite pads which mate with locating pins in the lower core plate. Lateral loads imposed on the fuel assembly are also transferred to the core support structures through the locating pins.

Top Nozzle

The top nozzle is a box-like structure which functions as the fuel assembly upper structural element and forms a plenum space where the heated coolant leaves the fuel assembly and is directed toward the flow holes in the upper core plate. The nozzle is comprised of an adapter plate, enclosure, top plate, two clamps, four double leaf springs, and assorted hardware. All parts with the exception of the springs and their hold-down bolts are constructed of Type 304 stainless steel. The springs are made from age hardened Inconel 718. The spring screws are made from peened Inconel 718 or Inconel 600.

The adapter plate is square in cross section and is perforated by machined slots to provide for coolant flow through the plate. At assembly, the top ends of the control guide thimbles are fastened to the adapter. Thus, the adapter plate acts as the fuel assembly top end plate and provides a means of distributing evenly among the guide thimbles any axial loads imposed on the fuel assemblies.

The nozzle enclosure is a square thin walled shell which forms the plenum section of the top nozzle. The bottom end of the enclosure is pinned and welded to the periphery of the adapter plate and the top end is welded to the periphery of the top plate.

The top plate is square in cross section with a central hole. The hole allows clearance for the RCC absorber rods to pass through the nozzle into the guide thimbles in the fuel assembly and for coolant exit from the fuel assembly to the upper internals area. Two pads containing axial through-holes which are located on diametrically opposite corners of the top plate provide a means of positioning and aligning the top of the fuel assembly. As with the bottom nozzle, alignment pins in the upper core plate mate with the holes in the top nozzle plate.

Hold-down forces of sufficient magnitude to oppose the hydraulic lifting forces on the fuel assembly are obtained by means of the double leaf springs which are mounted on the top plate. The springs are fastened in pairs to the top plate at the two corners where alignment holes are not used and radiate out from the corners parallel to the sides of the plate. Fastening and locking of springs is accomplished with a clamp which fits over the ends of the springs and two bolts (one per spring) which pass through the clamp and spring and thread into the top plate. At assembly, the spring bolts are torqued sufficiently to preload against the maximum spring load and then lockwelded to the clamp. The clamp is locally welded to the top plate to retain its position, prior to the spring bolt being lockwelded to the clamp.

The spring load is obtained through deflection of the spring set by the upper core plate. The spring form is such that it projects above the fuel assembly and is depressed by the core plate when the internals are loaded into the reactor. The free end of the spring is bent downward and captured in a key slot in the top plate. The free end of the lower spring is captured by the bent

down leg of the upper spring. This is done to guard against loose parts in the reactor in the event (however remote) of spring fracture. In addition, the fit between the upper spring and key slot and between the spring set and the mating slot in the clamp are sized to prevent rotation of either end of the spring set into the control rod path in the event of spring fracture.

In addition to its plenum and structural functions, the nozzle provides a protective housing for components which mate with the fuel assembly. In handling a fuel assembly with a control rod inserted, the control rod spider is contained within the nozzle. During operation in the reactor, the nozzle protects the absorber rods from coolant cross flows in the unsupported span between the fuel assembly adapter plate and the end of the guide tube in the upper internals package. Plugging devices which can be used to fill the ends of the fuel assembly thimble tubes at unrodded core locations and the spiders which support the source rods and burnable absorber rods are all contained within the fuel top nozzle.

Reconstitutable top nozzles (RTN) have been introduced into the Point Beach upgraded OFA reloads beginning with Unit 1 Region 19 (Cycle 17 feed) and Unit 2 Region 18 (Cycle 16 feed). The reconstitutable top nozzle for this upgraded fuel assembly differs from the OFA/STD design in that a groove is provided in each thimble through-hole in the nozzle plate to facilitate removal. To remove the top nozzle, a tool is first inserted through a lock tube and expanded radially to engage the bottom edge of the tube. An axial force is then exerted on the tool which overrides local lock tube deformations and withdraws the lock tube from the insert. After the lock tubes have been withdrawn, the nozzle is removed by raising it off the upper slotted ends of the nozzle inserts which deflect inwardly under the axial lift load.

With the top nozzle removed, direct access is provided for fuel rod examinations or replacement. Reconstitution is completed by the remounting of the nozzle and the re-insertion of the lock tubes. Additional details of this design feature, the design bases and evaluation of the reconstitutable top nozzle are given in Section 2.3.2 in Reference 7.

Guide Thimbles

The control rod guide thimbles in the fuel assembly provide guided channels for the absorber rods during insertion and withdrawal of the control rods. They are fabricated from a single piece of Zircaloy-4 or ZIRLO tubing which is drawn to two different diameters. The larger inside diameter at the top provides a relatively large annular area for rapid insertion of a withdrawn control rod during a reactor trip and to accommodate a small amount of upward cooling flow during normal operations. The bottom portion of the guide thimble is of reduced diameter to produce a dashpot action when the absorber rods are near the end of travel in the guide thimbles during a reactor trip. The transition zone at the dashpot section is conical in shape so that there are no rapid changes in diameter in the tube.

Flow holes are provided just above the transition of the two diameters to permit the entrance of cooling water during normal operation and to accommodate the outflow of water from the dashpot during reactor trip.

The dashpot is closed at the bottom by means of a welded end plug. The end plug is fastened to the bottom nozzle during fuel assembly fabrication.

Grids

The spring clip grid assemblies consist of individual slotted straps which are assembled and interlocked in an "egg crate" type arrangement and then furnace brazed or welded to permanently join the straps at their points of intersection. Details such as spring fingers, support dimples, mixing vanes, and tabs are punched and formed in the individual straps prior to assembly.

Two types of grid assemblies are used in the fuel assembly. One type of grid, having mixing vanes which project from the edges of the straps into the coolant stream, is used in the high heat region of the fuel assemblies to promote mixing of the coolant. A grid of this type is shown in Figure 3.2-42. The other type of grid, located at the bottom and top ends of the assembly, are of the nonmixing type. They are similar to the mixing type with the exception that they contain no mixing vanes on the internal straps.

The outside straps on all grids contain mixing vanes which, in addition to their mixing function, aid in guiding the grids and fuel assemblies past projecting surfaces during handling or loading and unloading the core. Additional small tabs on the outside straps and the irregular contour of the straps are also for this purpose.

Inconel 718 is chosen for the grid material on STD fuel assembly design and top and bottom grids of the OFA and 422V+ designs, because of its corrosion resistance and high strength properties. After the combined brazing and solution annealing temperature cycle, the grid material is age hardened to obtain the material strength necessary to develop the required grid spring forces. The OFA mixing vane grids are made from Zircaloy-4 and the 422V+ mixing vane grids are made from ZIRLO.

Fuel Rods

The fuel rods consist of UO_2 ceramic pellets in a slightly cold worked and partially annealed Zircaloy-4 or ZIRLO® or Optimized ZIRLO™ tubing which is plugged and seal welded at the ends to encapsulate the fuel. Sufficient void volume and clearances are provided within the rod to accommodate fission gases released from the fuel, differential thermal expansion between the cladding and the fuel, and fuel swelling due to accumulated fission products without overstressing of the cladding or seal welds. Shifting of the fuel within the cladding is prevented during handling or shipping prior to core loading by a stainless steel helical compression spring which bears on the top of the fuel.

At assembly, the pellets are stacked in the cladding to the required fuel height. The compression spring is then inserted into the top end of the fuel and the end plugs pressed into the ends of the tube and welded. All fuel rods are internally pressurized with helium at a pressure in the range of one to three hundred pounds during the welding process. A hold-down force in excess of the weight of the fuel is obtained by compression of the spring between the top end plug and the top of the fuel pellet stack.

The fuel pellets are right circular cylinders consisting of slightly enriched UO_2 powder which has been compacted by cold pressing and then sintered to the required density. The ends of each pellet are dished slightly to allow the greater axial expansion at the center of the pellets to be taken up within the pellets themselves and not in the overall fuel length. Reload fuel contains pellets with a small chamfer around the outer cylindrical surface on the pellet ends. This reduces the potential for pellet chipping during the fabrication process.

To prevent the possibility of mixing enrichments during fuel manufacture and assembly, meticulous process control is exercised. The UF_6 gas is received from the supplier in sealed containers, the contents of which are fully identified both by descriptive tagging and preselected color coding. A single enrichment only is received per shipment. Upon receipt, an additional Westinghouse identification tag completely describing the contents is affixed to the containers before transfer to segregated storage, where containers of different enrichment are never mixed.

The UF_6 is then converted to UO_2 powder by a series of highly controlled processes. The UO_2 powder is also placed in segregated storage according to enrichment. Powder withdrawal from storage can be made by one authorized group only who directs the powder to the correct pellet production line. All pellet production lines are physically separated from each other and pellets of only a single enrichment and density are produced in a given production line.

Finished pellets are placed on trays having the same color code as the powder containers and transferred to segregated storage racks. Physical barriers prevent mixing of pellets of different densities and enrichments in this storage area. Unused powder and substandard pellets to be reprocessed are returned to storage in the original color coded containers.

Loading of the pellets into the cladding is again accomplished in isolated production lines and again only one density and enrichment is loaded on a line at a time.

At the time of loading, the top fuel tube end plug identification character is checked with the density and enrichment identification of the color code of the pellet storage tray. After each fuel tube is seal welded, it is given the same color coding as has been carried throughout the previous processes. The fuel tube remains color coded until just prior to installation in the fuel assembly. The color coding, therefore, provides a cross reference of the fuel contained in the fuel rods.

At the time of installation into an assembly, the color coding is removed. After the fuel rods are installed, an inspector verifies that the top nozzle to be used on the assembly carries the correct identification character describing the fuel enrichment and density for the core region being fabricated. The top nozzle identification then becomes the permanent description of the fuel contained in the assembly.

The identification numbers on the fuel assembly top nozzles will then maintain the enrichment identity and ensure that the assemblies with the correct enrichment are loaded into the proper core region.

Each assembly will be assigned a core loading position prior to insertion. A record will then be made of the core loading position, serial number, and enrichment. Prior to core loading, independent checks will be made to ensure that this assignment is correct.

During initial core loading and subsequent refueling operations, detailed written handling and checkoff procedures will be utilized throughout the sequence. Current reload cores utilize loading patterns which are consistent with low-low leakage fuel management.

Neutron Source Assemblies

Neutron sources can be used to provide at least a required minimum count rate during startup operations. Four neutron source assemblies were initially utilized in the core; two secondary

source assemblies with four secondary source rods each and two primary source assemblies comprised of one combination primary and secondary rod, three secondary rods, and twelve burnable absorber rods. Currently, source assemblies are not utilized in Unit 1 and Unit 2. When sources are used, source rods are fastened to a spider at the top end similar to the RCC spiders.

In the core, the neutron source assemblies are inserted into the RCC guide thimbles in fuel assemblies at unrodded locations.

The primary and secondary source rods utilize the same type of cladding material as the absorber rods (cold-worked Type 304 stainless steel tubing, 0.432 in. OD, 0.019 in. thick walls). All secondary source rods contain Sb-Be pellets stacked to a height of 121.754 inches. The primary source for Unit 1 contained capsules of Po-Be source material 6 inches long and Sb-Be pellet material to fill the remainder of the rod height. The primary source rods for Unit 2 contained Pu-Be source. The active material was encased in a length of 24 inches maximum and was enclosed in a custom fabricated capsule. The remainder of the rod length was void. Design criteria similar to those for the fuel rods are used for the design of the source rods; i.e., the cladding is free standing, internal pressures are always less than reactor operating pressure, and internal gaps and clearances are provided to allow for differential expansion between the source material and cladding.

Plugging Devices

When necessary to limit bypass flow through the RCC guide thimbles in fuel assemblies which do not contain either control rods, source assemblies, or burnable absorbers, the fuel assemblies at those locations can be fitted with plugging devices. When utilized, the plugging devices would consist of a flat spider plate with short rods suspended from the bottom surface and a spring pack assembly attached to the top surface. At installation in the core, the plugging devices fit within the fuel assembly top nozzles and rest on the adapter plate. The short rods project into the upper ends of the thimble tubes to reduce the bypass flow area. The spring pack is compressed by the upper core plate when the upper internals package is lowered into place. Similar short rods are also used on any insert assembly which does not have sixteen rods. The plugging rods can be used to fill the upper ends of any vacant guide thimbles in a fuel assembly.

All components in the plugging device, except for the springs, are constructed from Type 304 stainless steel. The springs (one per plugging device) are wound from an age hardenable nickel base alloy to obtain higher strength.

Although the core bypass flow through the thimble tubes increases with the removal of thimble plugging devices, all thermal hydraulic criteria and safety limits are satisfied.

3.2.3.3 Evaluation of Core Components

Fuel Rod Evaluation

The fission gas release and the associated buildup of internal gas pressure in the fuel rods are calculated by the PAD code based on experimentally determined rates. The increase of internal pressure in the fuel rod due to these phenomena is included in the determination of the maximum cladding stresses at the end of core life when the fission product gap inventory is a maximum.

The maximum allowable strain in the cladding, considering the combined effects of internal fission gas pressure, external coolant pressure, fuel pellet swelling, and cladding creep is limited to less than 1% throughout core life. The associated stresses are below the yield strength of the material under all normal operating conditions.

To assure that manufactured fuel rods meet a high standard of excellence from the standpoint of functional requirements, many inspections and tests are performed both on the raw material and the finished product. These tests and inspections include chemical analysis, elevated temperature tensile testing of fuel tubes, dimensional inspection, x-ray of both end plug welds, ultrasonic testing, and helium leak tests.

In the event of cladding defects, the high resistance of UO_2 fuel pellets to attack by hot water protects against fuel deterioration or decrease in fuel integrity. Thermal stress in the pellets, while causing some fracture of the bulk material during temperature cycling, does not result in pulverization or gross void formation in the fuel matrix. As shown by operating experience and experimental work in the industry, the thermal design parameters conservatively account for any changes in the thermal performance of the fuel element due to pellet fracture.

The consequences of a breach of cladding are greatly reduced by the ability of uranium dioxide to retain fission products including those which are gaseous or highly volatile. This retentiveness decreases with increasing temperature or fuel burnup, but remains a significant factor even at full power operating temperature in the maximum burnup element.

A survey of high burnup UO_2 fuel element behavior indicates that for an initial UO_2 void volume, which is a function of the fuel density, it is possible to conservatively define the fuel swelling as a function of burnup (Reference 25). The fuel swelling model considers the effect of burnup, temperature distribution, and internal voids. It is an empirical model which has been checked with data from Bettis, Yankee, CVTR, Saxton, and others. Region 3 was retained through three initial cycles of reactor operation and Region 2 through two initial cycles. The initial pellet density was 92% in Region 2 and 91% in Region 3 for Unit 1 to accommodate the effects of increased burnup. For Unit 2, pellet densities were 94% in Region 1, 93% in Region 2, and 92% in Region 3.

Experience with the earlier fuel regions mentioned above provided information which was later used to improve fuel rod designs so as to reduce the effects of fuel pellet densification and eliminate clad collapse during the useful life of the fuel assemblies (Reference 29, and Reference 30). The integrity of fuel rod cladding so as to retain fission products or fuel material is directly related to cladding stress and strain under normal operating and overpower conditions. Design limits (cladding perforation) in terms of stress and strain are as follows:

	Damage Limit	Design Limit
Stress	Ultimate Strength 65,000 psi minimum	Yield Strength 52,000 psi minimum
Strain	1.7%	1.0%

The stress damage and design limits given above are minimum values. Actual damage and design limits depend upon cladding temperature, neutron exposure and normal variation of material properties and are greater than these minimum limits.

For most of the fuel rod life the actual stresses and strains are considerably below the design limits. Thus, significant margin exists between actual operating conditions and the damage limits. The other parameters having influence on cladding stress and strain and the relationship of these parameters to the damage limits are as follows:

1. <u>Internal Gas Pressure</u>

The internal pressure of the lead fuel rod in the reactor will be limited to a value below that which cause (1) the diametral gap to increase due to outward clad creep during steady state operation, and (2) extensive DNB propagation to occur.

The rod internal pressure for the Point Beach Unit 1 and Unit 2 fuel rods has been evaluated by modeling the gas inventories, gas temperature, and rod internal volumes through the rods' life. The resulting rod internal pressure is compared to the design limit on a case-by-case basis of current operating conditions to EOL. Reload evaluations show that the rod internal pressure satisfies the design limit.

The second part of the rod internal pressure design basis precludes extensive DNB propagation and associated fuel failure. The basis for this criterion is that no significant additional fuel failures, due the DNB propagation, will occur in cores which have fuel rods operating with rod internal pressure in excess of system pressure. The design limit for Condition II events is that DNB propagation is not extensive, i.e., the process is shown to be self limiting and the number of additional rods in DNB due to propagation is relatively small. For Condition III/IV events, it is shown that the total number of rods in DNB, including propagation effects, is consistent with the assumptions used in radiological dose calculations for the event under consideration.

2. <u>Cladding Temperature</u>

The strength of the fuel cladding is temperature dependent. The minimum ultimate strength reduces to the design yield strength at an average cladding temperature of approximately 850°F.

For Condition I and II events, the fuel and reactor protection systems are designed to assure that a calculated centerline fuel temperature does not exceed the fuel melting temperature criterion. The intent of this criterion is to avoid a condition of gross fuel melting which can result in severe duty on the clad. The concern here is based on the large volume increase associated with the phase change in the fuel and the potential for loss of clad integrity as a result of molten fuel/clad interaction.

The temperature of the fuel pellets was evaluated by modeling the fuel rod geometry, thermal properties, heat fluxes, and temperature differences in order to calculate fuel surface, average, and centerline temperatures of the fuel pellets.

3. <u>Swelling and Cladding Strain</u>

Fuel burnup results in fuel swelling which produces cladding strain. The strain damage limit is conservatively set at a 1.0% strain limit from the unirradiated condition for all Westinghouse fuel during steady-state operation. An evaluation performed for the 14x14 422V+ design has demonstrated that the cladding strain criterion has been met for this design.

4. Fuel Temperature and kW/ft

At zero burnup, cladding damage for fuel rods is calculated to occur at 31 kW/ft based upon cladding strain reaching the damage limit.

At this power rating, 17% of the pellet central region is expected to be in the molten condition. The maximum thermal output at rated power is 16.0 kW/ft.

Effects of Vibration and Thermal Cycling on Fuel Assemblies

Analyses of the effect of cyclic deflection of the fuel rods, grid spring fingers, RCC control rods, and burnable absorber rods due to hydraulically induced vibrations and thermal cycling show that the design of the components is good for an infinite number of cycles.

In the case of the fuel grid spring support, the amplitude of a hydraulically induced motion of the fuel rod is extremely small (~0.001) and the stress associated with the motion is significantly small (<100 psi).

Likewise, the reactions at the grid spring due to the motion is much less than the preload spring force and contact is maintained between the fuel cladding and the grid spring and dimples. Fatigue of the cladding and fretting between the cladding and the grid support have not normally been experienced and are not anticipated.

The effect of thermal cycling on the grid-cladding support is merely a slight relative movement between the grid contact surfaces and the cladding, which is gradual in nature during heatup and cooldown. Since the number of cycles of the occurrence is small over the life of a fuel assembly (up to 4 years), negligible wear of the mating parts is expected.

In-core operation of assemblies in the Yankee Rowe and Saxton reactors using similar cladding support have verified the calculated conclusions. Additional test results under a simulated reactor environment in the Westinghouse Reactor Evaluation Center also support these conclusions.

The dynamic deflection of the full length control rods, source rods, and the burnable absorber rods is limited by their fit with the inside diameter of either the upper portion of the guide thimble or the dashpot (0.074 inch diametral clearance at guide thimble for the STD fuel and 0.061 for the OFA and 422V+ fuel; 0.0155 inch diametral clearance at the dashpot). With this limitation, the occurrence of truly cyclic motion is questionable. However, an assumed cyclic deflection through the available clearance gap results in an insignificantly low stress in either the cladding tubing in the joint at the spider or retainer plate. The above consideration assumes the rods are supported as cantilevers from the spider or the retainer plate in the case of the burnable absorber rods.

A calculation assuming the rods are supported by the surface of the dashpots and the upper end by the spider or retainer results in a similar conclusion.

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Table 3.2-1 NUCLEAR DESIGN DATA

(Sheet 1 of 4)

		Cores With All Standard Assemblies (STD)*	Reloads of OFA and Upgraded OFA*	Reloads of 422V+** W/O EPU	Reloads of 422V+ W/EPU***
STRUCTURAL CHARACTERISTICS					
1.	Fuel Weight (UO_2), lbs.	118,729	107,430	120,047	120,047
2.	Zircaloy (STD & OFA) or ZIRLO® (422V+) or Optimized ZIRLO™ (422V+) Weight, lbs.	24,260	26,380	27,429	27,429
3.	Core Diameter, inches	96.5	96.85	96.5	96.5
4.	Core Height, inches	144	144	143.25	143.25
	Reflector Thickness and Composition				
5.	Top Reflector - (Water Plus Steel), inches	~10	~10	~10	~10
6.	Bottom Reflector - (Water Plus Steel), inches	~10	~10	~10	~10
7.	Side Reflector - (Water Plus Steel), inches	~15	~15	~15	~15
	Fuel				
8.	H_2O/U Volume Ratio (cold)	1.9	2.27	2.0	2.0
9.	Number of Fuel Assemblies	121	121	121	121
10.	UO_2 Rods per Assembly	179	179	179	179
PERFORMANCE CHARACTERISTICS					
11.	Total Core Heat Output, MW_t (initial rating)	1518.5	1518.5	1540	1800
12.	Total Primary Heat Output, MW_t (maximum calculated turbine rating)	1524	1524	1546	1806
13.	Fuel Burnup, First Cycle MWD/MTU	15,100			
	Equilibrium Cycles (Nominal Cycle)	9,500	10,800 10,500 (1)	17,200	19,000
	Region Average Discharge	33,000	40,000 45,000 (1)	52,000	52,000
	Lead Rod Average Burnup			62,000 (3)	62,000 (3)

* These parameter values are typical for a nominal 12 month fuel cycle length. Reload designs for a nominal 18 month fuel cycle length may be different.
** These parameter values are typical for a nominal 18 month fuel cycle length.
*** These parameter values are typical for a nominal 18 month fuel cycle at uprated conditions.

Table 3.2-1 NUCLEAR DESIGN DATA

(Sheet 2 of 4)

		Cores With All Standard Assemblies (STD)*	Reloads of OFA and Upgraded OFA*	Reloads of 422V+** W/O EPU	Reloads of 422V+ W/EPU***
PERFORMANCE CHARACTERISTICS					
14.	Region 1 Enrichment, w/o	2.27			
15.	Region 2 Enrichment, w/o	3.03			
16.	Region 3 Enrichment, w/o	3.40			
17.	Equilibrium Enrichment, w/o	3.2 - 3.6	3.2 - 3.6 3.8 - 4.0 (1)	4.4 - 4.95	4.4 - 4.95
18.	Nuclear Heat Flux Hot Channel Factor, F^N_q	2.32	2.50	2.60 (V+) 2.50 (OFA)	2.60
19.	Nuclear Enthalpy Rise Hot Channel Factor, $F^N_{\Delta H}$	1.58	1.70	1.77 (V+) 1.70 (OFA)	1.68
CONTROL CHARACTERISTICS					
	Effective Multiplication (Beginning-of-Life)				
	Boron Free, Rods Out				
20.	Cold, No Power, Xenon Free	1.211	1.232	1.200	1.200
21.	Hot, No Power, Xenon Free	1.167	1.171	1.156	1.156
22.	Hot, Full Power, Xenon Free	1.151	1.146	1.131	1.131
23.	Hot, Full Power, Xe and Sm Equilibrium	1.113	1.109	1.097	1.097
	Rod Cluster Control Assemblies				
24.	Material	5% Cd; 15% In; 80% Ag			
25.	Number of Full Length RCC Assemblies	33	33	33	33
26.	Number of Absorber Rods per RCC Assembly	16	16	16	16
27.	Rod Worth		See Table 3.2-3		

* These parameter values are typical for a nominal 12 month fuel cycle length. Reload designs for a nominal 18 month fuel cycle length may be different.
** These parameter values are typical for a nominal 18 month fuel cycle length.
*** These parameter values are typical for a nominal 18 month fuel cycle at uprated conditions.

Table 3.2-1 NUCLEAR DESIGN DATA

(Sheet 3 of 4)

		Cores With All Standard Assemblies (STD)*	Reloads of OFA and Upgraded OFA*	Reloads of 422V+** W/O EPU	Reloads of 422V+ W/EPU***
	BOL Boron Concentrations				
28.	Refueling Shutdown; Rods in (k =0.95)	<1800 ppm	1800 ppm	2100 ppm	2502
29.	Cold Shutdown (k =0.99) with All Rods Inserted, Xenon Free	1015 ppm	928 ppm	1450 ppm	1777
30.	Hot Shutdown (k=0.99) with all Rods Inserted, Xenon Free	667 ppm	677 ppm	1575 ppm	1405
31.	Cold Shutdown (k=0.99) with No Rods Inserted, Xenon Free	1581 ppm	1426 ppm	2200 ppm	2480
32.	Hot Shutdown (k=0.99) with No Rods Inserted, Xenon Free	1613 ppm	1419 ppm	2300 ppm	2576
	Boron Concentrations To Control at Hot Full Power, Rods Inserted, k= 1.0 (With Burnable Absorber Rods)				
33.	Xenon Free	1348 ppm	1184 ppm	1817 ppm	2101
34.	Xenon	1023 ppm	931 ppm	1435 ppm	1673
35.	Xenon and Samarium	970 ppm	880 ppm	1383 ppm	1621
36.	Cold Shutdown (k=0.99) All But Most Reactive Rod Inserted, Xenon Free	1117 ppm	1034 ppm	1550 ppm	1796
37.	Hot Shutdown (k=0.99), All but Most Reactive Rod Inserted, Xenon Free	771 ppm	773 ppm	1650 ppm	1496
BURNABLE ABSORBER RODS					
38.	Number/Material	Discrete 704/ Borated Pyrex Glass	ZrB_2 IFBA	ZrB_2 IFBA	ZrB_2 IFBA
39.	Worth Hot, Δk/k	7.4%	N/A	N/A	N/A
40.	Worth Cold, Δk/k	5.8%	N/A	N/A	N/A

* These parameter values are typical for a nominal 12 month fuel cycle length. Reload designs for a nominal 18 month fuel cycle length may be different.

** These parameter values are typical for a nominal 18 month fuel cycle length.

*** These parameter values are typical for a nominal 18 month fuel cycle at uprated conditions.

Table 3.2-1 NUCLEAR DESIGN DATA

(Sheet 4 of 4)

		Cores With All Standard Assemblies (STD)*	Reloads of OFA and Upgraded OFA*	Reloads of 422V+** W/O EPU	Reloads of 422V+ W/EPU***
KINETIC CHARACTERISTICS					
(Equilibrium Cycle Design)					
41.	Moderator Temperature Coefficient, (% Δk/k/ºF)		$+3.0\times10^{-3}$ to $-25.0\times10^{-3(2)}$	$+1.0\times10^{-3}$ to -36.0×10^{-3}	$+2.0\times10^{-3}$ to -37.0×10^{-3}
42.	Moderator Pressure Coefficient, (% Δk/k/psi)		-0.3×10^{-4} to $+2.8\times10^{-4(2)}$	-0.1×10^{-4} to $+4.5\times10^{-4}$	-0.2×10^{-4} to $+4.5\times10^{-4}$
43.	Moderator Density Coefficient, (% Δk/k/gm/cm^3)		-3.0 to +22$^{(2)}$	-1.0 to +36.0	-2.0 to +36.0
44.	Doppler Coefficient (% Δk/k/ºF)		-2.9×10^{-3} to $-1.4\times10^{-3(2)}$	-2.5×10^{-3} to -1.0×10^{-3}	-2.90×10^{-3} to -0.91×10^{-3}
45.	Delayed Neutron Fraction, (%)		0.58 to 0.51$^{(2)}$	0.45 to 0.70	0.43 to 0.72
46.	Prompt Neutron Lifetime, (sec.)		1.9×10^{-5} to $2.1\times10^{-5(2)}$	1.0×10^{-5} to 1.7×10^{-5}	1.0×10^{-5} to 1.7×10^{-5}

(1) Upgraded OFA Value

(2) Typical Values for a Full OFA/Upgraded OFA Core

(3) Per NRC letter to Westinghouse (Reference 41), the lead rod average burnup limit for WCAP-12610-P-A, "Vantage + Fuel Assembly Reference Core Report," (Reference 38) can be increased to a maximum of 62,000 MWD/MTU provided the evaluation of the fuel design performance is performed with PAD 4.0 (Reference 30). The NRC letter also concludes that the use of the FCEP (Reference 36) is also valid up to a burnup limit of 62,000 MWD/MTU provided the change process uses PAD 4.0 to evaluate the effect of any proposed design change on the fuel.

* These parameter values are typical for a nominal 12 month fuel cycle length. Reload designs for a nominal 18 month fuel cycle length may be different.
** These parameter values are typical for a nominal 18 month fuel cycle length.
*** These parameter values are typical for a nominal 18 month fuel cycle at uprated conditions.

Table 3.2-2 REACTIVITY REQUIREMENTS FOR CONTROL RODS ΔK/K (%)

	BOL (Cycle 1)	EOL (Cycle 1)	EOL Typical STD Fuel Core*	EOL Typical OFA Fuel Core (1) *	EOL Typical 422V+ Fuel Core **	EOL Typical 422V+ Fuel Core w/ EPU ***
Reactivity Defects (Combined Doppler, T_{avg}, Void and Redistribution Effects)	2.09	3.22	2.80	2.57	2.50	3.10
Rod Insertion Allowance	0.50	0.50	0.50	0.50	0.40	0.40
Total Control			3.30	3.07	2.90	3.50
Worth of 32 Rods Less 10%	2.59	3.72	6.88	6.78	6.02	5.91
Shutdown Margin	7.12	6.49	3.58	3.71	3.12	2.41
Shutdown Margin Requirement	2.59	3.72	2.77	2.77	2.77	2.00
Excess Shutdown Margin	4.53	2.77	0.81	0.94	0.35	0.41

(1) Includes OFA Upgrade

* These parameter values are typical for a nominal 12 month fuel cycle length. Reload designs for a nominal 18 month fuel cycle length may be different.

** These parameter values are typical for a nominal 18 month fuel cycle length.

*** These parameter values are typical for a nominal 18 month fuel cycle at uprated conditions.

Table 3.2-3 CALCULATED [1] ROD WORTHS, ΔK/K(%)

Core Condition	Rod Configuration	Worth	Less 10%[1]	Design Reactivity Requirements	Shutdown Margin
BOL, HFP Cycle 1	33 Rods In	9.42			
BOL, HZP Cycle 1	32 Rods In Highest Worth Rod Stuck Out	7.91	7.12	2.59	4.53
EOL, HFP Cycle 1	33 Rods In	9.41			
EOL, HZP Cycle 1	32 Rods In Highest Worth Rod Stuck Out	7.21	6.49	3.72	2.77
EOL, HFP Equilibrium Cycle	33 Rods In	8.38[2]* 7.44[3]***			
EOL, HZP Equilibrium Cycle	32 Rods In Highest Worth Rod Stuck Out	7.54 6.57***	6.78 5.91***	3.07 3.50***	3.71[2]* 2.41[3]***

BOL = Beginning-of-Life HFP = Hot Full Power

EOL = End-of-Life HZP = Hot Zero Power

[1] Calculated rod worth is reduced by 10% to allow for uncertainties.
[2] Typical Full OFA/Upgraded OFA Core.
[3] Typical Full 422V+ Core.

* These parameter values are typical for a nominal 12 month fuel cycle length. Reload designs for a nominal 18 month fuel cycle length may be different.
**These parameter values are typical for a nominal 18 month fuel cycle length.
*** These parameter values are typical for a nominal 18 month fuel cycle at uprated conditions.

Table 3.2-4 THERMAL AND HYDRAULIC DESIGN PARAMETERS

(Sheet 1 of 2)

	Pre-EPU Reloads of 422V+ Fuel	EPU Reloads with 422V+ Fuel
Total Primary Heat Output, MW_t	1546	1806
Total Reactor Coolant Pump Heat Output, MW_t	6.0	6.0
Total Core Heat Output, MW_t	1540	1800
Total Heat Output, Btu/hr	5.181×10^9	6.142×10^9
Heat Generated in Fuel, % of Total Core Heat Output	97.4	97.4
Maximum Thermal Overpower, %	21.1	20.0
Nominal System Pressure, psia	2250	2250
Nuclear Heat Flux Hot Channel Factor, F^N_Q (1)	2.60	2.60
Nuclear Enthalpy Rise Hot Channel Factor, $F^N_{\Delta H}$	1.77	1.68
Coolant Flow (2)		
Total Flow Rate (Thermal Design Flow), gpm	178,000	178,000
Total Flow Rate (Thermal Design Flow), lbm/hr	6.76×10^7	6.76×10^7
Design Bypass Flow	6.5%	6.5%
Average Velocity Along Fuel Rods, ft/sec	14.6	13.7
Average Mass Velocity, lbm/hr-ft^2	2.34×10^6	2.34×10^6
Coolant Temperature, °F (2)		
Nominal Inlet	542.5	542.9
Average Rise in Vessel	63.0	68.2
Average Rise in Core	67.0	72.4
Average in Core	577.5	581.0
Average In Vessel	574.0	577.0
Heat Transfer (2)		
Active Heat Transfer Surface Area, ft^2	28,507	28,507
Average Heat Flux, Btu/hr-ft^2	177,075	209,584
Maximum Heat Flux, Btu/hr-ft^2	460,395	545,620
Maximum Thermal Output, kw/ft	14.9	17.7
Peak Fuel Centerline Temperature for Prevention of Centerline Melt, °F	4700	4700

Table 3.2-4 THERMAL AND HYDRAULIC DESIGN PARAMETERS

(Sheet 2 of 2)

	Pre-EPU Reloads of 422V+ Fuel	EPU Reloads with 422V+ Fuel
DNB Ratio		
Minimum DNB Ratio at Nominal Operation Conditions	2.15 (3)	1.95(3)
Pressure Drop, psi		
Across Core	20.9 (4)	25.0(4)
Across Vessel, Including Nozzles	44	47.3

(1) Includes a nuclear uncertainty of 1.05 and an engineering uncertainty of 1.03

(2) Based on thermal design flow and 2250 psia system pressure.

(3) Minimum DNBR reflects RTDP Methodology.

(4) Reflects elimination of thimble plugs. Based on RCS flow rate of 201,200 gpm.

Table 3.2-5 CORE MECHANICAL DESIGN PARAMETERS (1)

(Sheet 1 of 2)

		STD Fuel	OFA Fuel	422V+ Fuel
Active Portion of the Core				
Equivalent Diameter, in.		96.5	96.5	96.5
Active Fuel Height, in.		144.0	144.0	143.25
Length-to-Diameter Ratio		1.495	1.495	1.495
Total Cross Section Area, ft^2		50.8	50.8	50.8
Fuel Assemblies				
Number		121	121	121
Rod Array		14 x 14	14 x 14	14 x 14
Rods per Assembly		179 (2)	179 (2)	179 (2)
Rod Pitch, in.		0.556	0.556	0.556
Nominal Assembly Envelope at Bottom Nozzle		7.761 x 7.761	7.761 x 7.761	7.761 x 7.761
Fuel Weight (as UO_2), pounds		118,729	108,078	120,047
Total Weight, pounds		154,519	137,335	151,250
Number of Grids per Assembly		7 (Inconel)	2 (Inconel) Ends 5 (Zircaloy) Middle	2 (Inconel) Ends 5 (ZIRLO) Middle
Guide Thimble Diameters				
(in. above dashpot)	I D	0.505	0.492	0.492
	OD	0.539	0.526	0.526
(in. below dashpot)	I D	0.4465	0.4465	0.4465
	OD	0.4805	0.4805	0.4805
Fuel Rods				
Number		21,659	21,659	21,659
Outside Diameter, in.		0.422	0.400	0.422
Diametrical Gap, in.		0.0075	0.0070	0.0075
Clad Thickness, in.		0.0243	0.0243	0.0243
Clad Material		Zircaloy-4	Zircaloy-4	ZIRLO® or Optimized ZIRLO™
Overall Length, in.		151.850	151.850 (3)	152.563
Fuel Pellets				
Material		UO_2 sintered	UO_2 sintered	UO_2 sintered
Density (% of Theoretical Initial Cores)		95	95	96
Diameter, in.		0.3659	0.3444	0.3659
Length, in.		0.600	0.565 (4) 0.413 (5)	0.4390

Table 3.2-5 CORE MECHANICAL DESIGN PARAMETERS [1]

(Sheet 2 of 2)

Rod Cluster Control Assemblies	
Neutron Absorber	5% Cd, 15% In, 80% Ag
Cladding Material	Type 304 SS - Cold worked
Clad Thickness, in.	0.019
Number of Full-Length Clusters	33
Number of Control Rods per Cluster	16
Weight in 60°F Water	
Full-Length, pounds	114
Length of Control Rod, in.	158.454 (Overall)
	150.954 (Insertion Length)
Length of Absorber Section, in.	142.00 (Full-Length)
Core Structure	
Core Barrel, in.	
ID	109.0
OD	112.5
Thermal Shield, in.	
ID	115.3
OD	122.5
Burnable Absorber Rods	
Material	Borosilicate Glass
Outside Diameter, in.	0.431
Inner Tube, OD, in.	0.2365
Clad Material	SS
Inner Tube Material	SS
Boron Loading (Natural)	
gm/cm of glass rod	0.0429

(1) All dimensions for cold conditions
(2) Seventeen rods are omitted; sixteen to provide passage for control rods and one to contain in-core instrumentation
(3) 152.285 for upgraded OFA
(4) Up to Region 17 of Point Beach Unit 1 and Region 16 of Point Beach Unit 2.
(5) Standardized fuel pellet-used in Region 18 and 19 of Point Beach Unit 1 and Region 17 and 18 of Point Beach Unit 2.

Figure 3.2-1 CONTROL ROD CLUSTER GROUPS

[redacted]

Figure 3.2-2 STANDARD FUEL NORMALIZED POWER DENSITY DISTRIBUTION (BOL) MAXIMUM POWER DENSITY = 1.364

[redacted]

Figure 3.2-3 NORMALIZED POWER DENSITY DISTRIBUTION (BOL) MAXIMUM POWER DENSITY = 1.505

[redacted]

Figure 3.2-4 STANDARD FUEL NORMALIZED POWER DENSITY DISTRIBUTION (BOL) IN A PLANE HAVING NO CONTROL RODS
MAXIMUM POWER DENSITY = 1.384

Figure 3.2-5 INITIAL BURNABLE ABSORBER ROD LOCATION

[redacted]

Figure 3.2-6 ARRANGEMENT OF BURNABLE ABSORBER RODS WITHIN AN ASSEMBLY

[redacted]

Figure 3.2-7 TYPICAL EQUILIBRIUM RELOAD LOADING PATTERN

[redacted]

Figure 3.2-8 UPGRADED CORE ASSEMBLY EQUILIBRIUM LOADING PATTERN AND IFBA PLACEMENT

[redacted]

Figure 3.2-9 OFA NORMALIZED POWER DENSITY DISTRIBUTION NEAR BEGINNING OF LIFE, UNRODDED CORE, HOT FULL POWER, EQUILIBRIUM XENON

[redacted]

Figure 3.2-10 OFA NORMALIZED POWER DENSITY DISTRIBUTION NEAR BEGINNING OF LIFE, GROUP D AT INSERTION LIMIT HOT FULL POWER, EQUILIBRIUM XENON

[redacted]

Figure 3.2-11 OFA NORMALIZED POWER DENSITY DISTRIBUTION NEAR END OF LIFE, UNRODDED CORE HOT FULL POWER, EQUILIBRIUM XENON

[redacted]

Figure 3.2-12 OFA NORMALIZED POWER DENSITY DISTRIBUTION NEAR END OF LIFE, GROUP D AT INSERTION LIMIT HOT FULL POWER, EQUILIBRIUM XENON

[redacted]

Figure 3.2-13 UPGRADED CORE NORMALIZED POWER DISTRIBUTION AT 150 MWD/MTU UNRODDED, HOT FULL POWER, EQUILIBRIUM XENON

PEAK $F_{\Delta H}$ = 1.555

[redacted]

Figure 3.2-13a UPGRADED CORE NORMALIZED POWER DISTRIBUTION AT 150 MWD/MTU D-BANK AT ROD INSERTION LIMIT, HOT FULL POWER, EQUILIBRIUM XENON PEAK $F_{\Delta H}$ = 1.574

[redacted]

Figure 3.2-14 UPGRADED CORE NORMALIZED POWER DISTRIBUTION AT 10600 MWD/MTU UNRODDED, HOT FULL POWER, EQUILIBRIUM XENON PEAK $F_{\Delta H}$ = 1.50

[redacted]

Figure 3.2-14a UPGRADED CORE NORMALIZED POWER DISTRIBUTION AT 10600 MWD/MTU D-BANK AT ROD INSERTION LIMIT, HOT FULL POWER, EQUILIBRIUM XENON PEAK $F_{\Delta H}$ = 1.515

[redacted]

Figure 3.2-15 EQUILIBRIUM CYCLE BOC, MOC AND EOC ASSEMBLY POWER DISTRIBUTIONS FOR 422V+ FUEL

[redacted]

Figure 3.2-16 EQUILIBRIUM CYCLE LOADING PATTERN WITH BOC AND EOC ASSEMBLY BURNUPS FOR 422V+ FUEL

[redacted]

Figure 3.2-17 MODERATOR TEMPERATURE COEFFICIENT vs. MODERATOR TEMPERATURE

[redacted]

Figure 3.2-18 DOPPLER COEFFICIENT vs. EFFECTIVE FUEL TEMPERATURE (BOL)

[redacted]

Figure 3.2-19 POWER COEFFICIENT

[redacted]

Figure 3.2-20 POWER COEFFICIENT

[redacted]

Figure 3.2-21 CALCULATED AND MEASURED DOPPLER DEFECT AND COEFFICIENTS AT BEGINNING OF LIFE, TWO-LOOP PLANT, 121 ASSEMBLIES, 12-FOOT CORE

[redacted]

Figure 3.2-22 COMPARISON OF CALCULATED AND MEASURED BORON CONCENTRATION FOR 2-LOOP PLANT, 121 ASSEMBLIES, 12-FOOT CORE

[redacted]

Figure 3.2-23 THERMAL CONDUCTIVITY OF UO_2 (DATA CORRECTED TO 95% THEORETICAL DENSITY)

[redacted]

Figure 3.2-24 HIGH POWER FUEL ROD EXPERIMENTAL PROGRAM

[redacted]

Figure 3.2-25 COMPARISON OF W-3 PREDICTION AND UNIFORM FLUX DATA

[redacted]

Figure 3.2-26 W-3 CORRELATION PROBABILITY DISTRIBUTION CURVE

[redacted]

Figure 3.2-27 COMPARISON OF W-3 CORRELATION WITH ROD BUNDLE DNB DATA (SIMPLE GRID WITHOUT MIXING VANE)

[redacted]

Figure 3.2-28 COMPARISON OF W-3 CORRELATION WITH ROD BUNDLE DNB DATA (SIMPLE GRID WITH MIXING VANE)

[redacted]

Figure 3.2-29 STABLE FILM BOILING HEAT TRANSFER DATA AND CORRELATION

[redacted]

Figure 3.2-30 COMPARISON OF W-3 PREDICTION AND NON UNIFORM FLUX DATA
($-0.15 \leq X_{DNB} \leq +0.15$)

[redacted]

Figure 3.2-31 COMPARISON OF W-3 PREDICTION WITH MEASURED DNB LOCATION

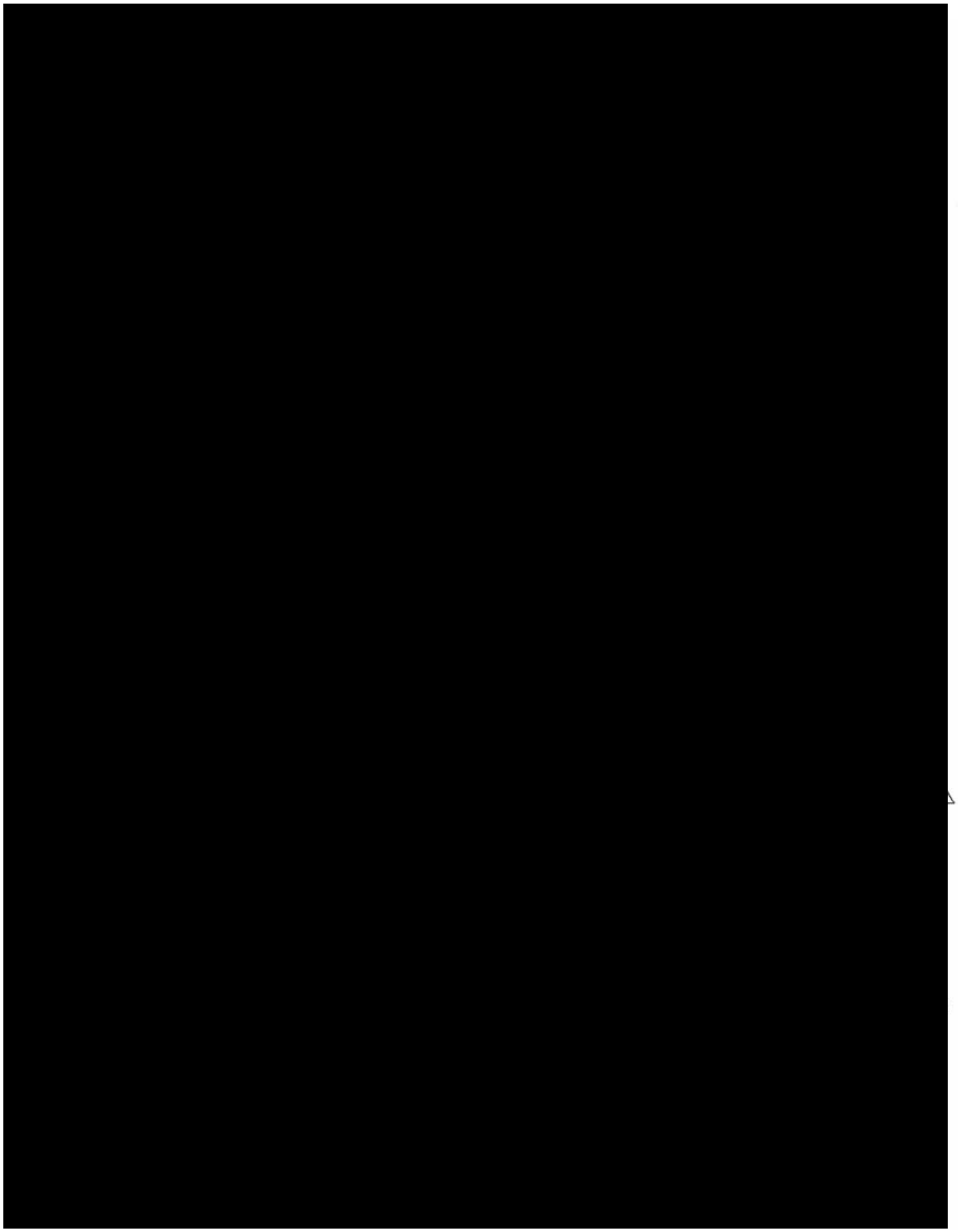

Figure 3.2-32 RADIAL POWER DISTRIBUTION

[redacted]

Figure 3.2-33 MEASURED vs. PREDICTED CRITICAL HEAT FLUX WRB-1 CORRELATION

[redacted]

Figure 3.2-34 REACTOR CORE CROSS SECTION

[redacted]

Figure 3.2-35 REACTOR VESSEL INTERNALS

[redacted]

Figure 3.2-36 BOTTOM NOZZLE FLOW HOLE COMPARISON

[redacted]

Figure 3.2-37 LOWER CORE SUPPORT STRUCTURE

[redacted]

Figure 3.2-38 UPPER CORE SUPPORT ASSEMBLY

[redacted]

Figure 3.2-39 GUIDE TUBE ASSEMBLY

[redacted]

Figure 3.2-40 FUEL ASSEMBLY AND CONTROL CLUSTER CROSS SECTION

[redacted]

Figure 3.2-41 FUEL ASSEMBLY OUTLINE

Sheet 1 of 7

[redacted]

Figure 3.2-41 FUEL ASSEMBLY OUTLINE

(Sheet 2 of 7)

[redacted]

Figure 3.2-41 FUEL ASSEMBLY OUTLINE

Sheet 3 of 7

[redacted]

Figure 3.2-41 FUEL ASSEMBLY OUTLINE

Sheet 4 of 7

[redacted]

Figure 3.2-41 FUEL ASSEMBLY OUTLINE

Sheet 5 of 7

[redacted]

Figure 3.2-41 14 X 14 422VANTAGE + (422V+) FUEL ASSEMBLY OUTLINE

Sheet 6 of 7

[redacted]

Figure 3.2-41 COMPARISON OF 14 X 14 OFA AND 422V + FUEL ROD DESIGNS

Sheet 7 of 7

[redacted]

Figure 3.2-42 SPRING CLIP GRID ASSEMBLY WITH SPLIT MIXING VANES

[redacted]

Figure 3.2-42a SPRING CLIP GRID ASSEMBLY WITH SPLIT MIXING VANES

[redacted]

Figure 3.2-43 REACTOR VESSEL STRESS CONCENTRATIONS

Sheet 1 of 3

[redacted]

Figure 3.2-43 REACTOR VESSEL STRESS CONCENTRATIONS

Sheet 2 of 3

[redacted]

Figure 3.2-43 REACTOR VESSEL STRESS CONCENTRATIONS

Sheet 3 of 3

[redacted]

3.3 RELOAD CORE DESIGN AND SAFETY ANALYSIS

At the reactivity end of life (end of cycle), the reactor is shut down for refueling. A portion of the fuel assemblies comprising the core are discharged, fresh fuel assemblies are added and, based on design calculations, a new core loading pattern is implemented. The core configuration following refueling operations comprises the reload core which will be operated until its respective end of reactivity life.

Nuclear design calculations are performed for each reload core to determine a proper core loading pattern which satisfies the cycle energy and safety analysis requirements. Particular attention is paid to peaking factors and core kinetics characteristics. If core characteristics fall outside of the range of values covered by the previous nuclear design or safety analysis, those core conditions or accidents so affected are reanalyzed (Reference 1).

As part of the reload safety evaluation, the mechanical, nuclear, and thermal-hydraulic characteristics of the reload core are assessed. Special conditions such as off-nominal operating conditions, LOCA limits, or special operational limitations are also addressed. Thus, each reload core is designed and provided with the same or better safety margins than the initial core analysis presented in this FSAR (Reference 2, Reference 3).

Fuel assemblies which comprise the reload region originally utilized Low-Parasitic (LOPAR) fuel also known as STD fuel assemblies. Starting with Unit 1, Region 15, Cycle 13, and Unit 2, Region 13, Cycle 11, the fuel assemblies are of the OFA design, described in Section 3.2. Upgraded OFAs have been inserted starting with the Point Beach Unit 1, Region 19, Cycle 17 core and the Unit 2, Region 18, Cycle 16 core. The upgraded OFA has a Reconstitutable Top Nozzle (RTN), a slightly longer fuel rod for a higher burnup capability, and a Debris Filter Bottom Nozzle (DFBN). The OFA design assemblies and the upgraded assemblies are fully compatible with previously irradiated OFA and STD assemblies in a reload core. Fuel enrichment and/or fuel rod internal pressurization may change to accommodate energy requirements of the respective duty cycles.

Commencing with Unit 1, Region 29, Cycle 27, and Unit 2, Region 27, Cycle 25, the Point Beach units were upgraded to the 14x14, 422VANTAGE+ (422V+) fuel design. This design uses the larger, 0.422" OD, fuel rod. Other major design features include the use of ZIRLO® or Optimized ZIRLO™ cladding, ZIRLO fabricated guide thimbles, instrumentation tubes, and mid-grids; mid-enriched annular pellets in axial blankets; and a pre-oxidized coating on the lower portion of the fuel rod. These reload cycles are based on an eighteen month operating cycle design.

The reactor core can consist of either OFA, and upgraded OFA assemblies, or any combination of previously burned OFA, previously burned upgraded OFA, and 422V+ assemblies. The use of previously-depleted STD fuel assemblies is no longer allowed (Reference 7).

In addition to incorporating upgraded OFA and 422V+ assemblies, the Point Beach core designs have deleted fuel assembly thimble plugs and deleted burnable absorber rods in RCC guide thimbles, and had previously used part length hafnium neutron absorber rods in core Peripheral Power Suppression Assemblies (PPSAs) to further reduce the fast neutron flux at the reactor pressure vessel welds.

A reduced neutron leakage fuel management scheme is presently used in the Point Beach cores. This scheme, defined as a low-low leakage loading pattern uses highly burned fuel in all assembly locations on the periphery. Use of highly burned fuel and absorber rods in core peripheral locations results in a reduced power in peripheral assemblies which is offset by power increases in the remaining fuel assemblies. This increased power has been accommodated by increasing the core peaking factor limits for $F_{\Delta H}$ and F_Q. The benefits of these loading patterns are:

1. Improved fuel utilization, and

2. Reduced fast neutron exposure of the reactor pressure vessel, with the corresponding reduction in the irradiation-induced embrittlement rate of the vessel material (Reference 4)

Peripheral Power Suppression Assemblies (PPSAs) were previously utilized to suppress fuel assembly powers at the flats of the core to shield key areas of the vessel from fast neutron flux. PPSAs are neutron-absorbing core component assemblies which locally suppress the power at the periphery of the fuel core near critical reactor vessel welds. The twelve assemblies on the core flats each contained a PPSA, with absorber in the lower six feet of the guide tubes. Each PPSA consisted of sixteen part-length hafnium absorber rods which were attached to a low profile spider. The mechanical design of the PPSAs was similar to the Westinghouse hafnium RCC assembly design, and would fit into the thimble tubes of the fuel assemblies. Two different assemblies exist, which were identical in design other than the length of hafnium contained within the rods. One assembly design utilized six foot long hafnium absorbers, which would protect both radial and vertical reactor vessel welds. The other assembly design utilized three foot long hafnium absorbers, which would protect only radial reactor vessel welds (Reference 3, Reference 4).

The PPSAs were removed from the Point Beach Unit 1 core beginning with Cycle 32 and were removed from the Point Beach Unit 2 core beginning with Cycle 35.

NOTE: Test fuel assemblies were used in some cores early in plant life. The following paragraph is retained for historical purposes.

When used, test assemblies may have removable fuel rods which can be examined to determine specialized fuel characteristics. If RTNs are not used for these assemblies, a special thimble plug device provides equivalent top nozzle flow characteristics while also providing the fuel rod holddown constraints of the top nozzle plate. Otherwise the test assemblies normally have the same mechanical and thermal-hydraulic characteristics as production fuel assemblies.

Reload fuel may also utilize axial blankets and Integral Fuel Burnable Absorbers (IFBA), as described below.

Axial blankets are sections of natural or mid-enriched uranium pellets at the top and bottom of the fuel stack of each fuel rod. Axial blankets may be annular to increase fuel rod plenum volume. Blanket length may also be increased to increase plenum volume.

The IFBA is a section of fuel pellets coated by a thin film of zirconium diboride (ZrB_2) burnable absorber material. The coated IFBA rod fuel stack was originally 96 inches in length and axially centered at the core midplane to obtain reasonable normal-operation, elevation dependent F_Q

values. The coating length may be reduced or increased, depending upon the number of IFBA rods and the core loading arrangement. Since Unit 1 Cycle 25 and Unit 2 Cycle 25 a nominal IFBA coating length of 120 inches has been used. In cycles in which full-length discrete burnable absorber rods are used, the fuel assemblies containing these absorbers will not have axial blankets. This will result in lower F_Q values and, therefore, the ΔI envelope will be conservative (Reference 5).

Where safety limits are not violated, limited substitutions of fuel rods by filler rods consisting of ZIRLO, Zircaloy-4 or stainless steel, or by vacancies, may be made to replace damaged fuel rods if justified by cycle-specific reload analysis. Replacement of leaking fuel rods will permit better utilization of the energy in the remaining non-leaking rods of fuel assemblies. In general, substitution of a limited number of fuel rods with filler rods or water holes has negligible effect on core physics parameters and consequently on the safety analysis. For each fuel cycle an analysis is conducted to ensure that, with each reload of fuel, all core design safety criteria are met (Reference 6).

REFERENCES

1. Bordelon, F. M., et. al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272-P-A (Proprietary), dated July 1985

2. WE letter to NRC, "Additional Information, Technical Specification Change Request No. 87, Safety Evaluation for Optimized Fuel, Point Beach Nuclear Plant, Units 1 And 2," dated September 6, 1983.

3. NRC letter to WE, "Approval to Allow Use of Westinghouse Optimized Fuel Assemblies At Point Beach Nuclear Plant For Units 1 And 2 Reloads," dated October 5, 1984.

4. NRC letter to WE, "Amendment Nos. 120 and 123 to Facility Operating License Nos. DPR-24 and DPR-27 (TACs 69349/69350)," dated May 8, 1989.

5. WE letter to NRC, "Additional Information for Technical Specification Change Request 127 - Increased Allowable Core Power Peaking Factors," dated October 28, 1988.

6. NRC letter to WE, "Point Beach Technical Specification Amendments No. 108 and 111," dated May 27, 1987.

7. NRC letter to WE, "Issuance of Amendments 193 and 198, Design and Operation of Fuel Cycles with Upgraded Westinghouse Fuel," dated February 8, 2000.

3.4 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

Rod Cluster Control Assemblies

The control rods or rod cluster control assemblies (RCCAs) each consist of a group of individual absorber rods fastened at the top end to a common hub or spider assembly. These assemblies, one of which is shown in Figure 3.4-1, are provided to control the reactivity of the core under operating conditions.

The absorber material used in the control rods is silver indium cadmium alloy which is essentially "black" to thermal neutrons and has sufficient additional resonance absorption to significantly increase its worth. The alloy is in the form of extruded single length rods which are sealed in chrome plated (EP-RCCA) stainless steel tubes to prevent the rods from coming in direct contact with the coolant.

The overall control rod length is such that when the assembly has been withdrawn through its full travel, the tip of the absorber rods remains engaged in the guide thimbles so that alignment between rods and thimbles is always maintained. Since the rods are long and slender, they are relatively free to conform to any small misalignments with the guide thimble. Prototype tests have shown that the RCCAs are very easily inserted and not subject to binding even under conditions of severe misalignment.

The spider assembly is in the form of a center hub with radial vanes containing cylindrical fingers from which the absorber rods are suspended. Handling detents and detents for connection to the drive shaft are machined into the upper end of the hub. A spring pack is assembled into a skirt integral to the bottom of the hub to stop the RCCA and absorb the impact energy at the end of a trip insertion. The radial vanes are joined to the hub and the fingers are joined to the vanes by furnace brazing. A center post which holds the spring pack and its retainer is threaded into the hub within the skirt and welded to prevent loosening in service. All components of the spider assembly are made from Type 304 stainless steel except for the springs which are Inconel X-750 alloy and the retainer which is of 17-4 PH material.

The absorber rods are secured to the spider so as to assure trouble free service. The rods are first threaded into the spider fingers and then pinned to maintain joint tightness, after which the pins are welded in place. The end plug below the pin position is designed with a reduced section to permit flexing of the rods to correct for small operating or assembly misalignments.

In construction, the silver indium cadmium rods are inserted into cold-worked stainless steel tubing which is then sealed at the bottom and the top by welded end plugs. The EP-RCCA tubes are then chrome plated. Sufficient diametral and end clearances are provided to accommodate relative thermal expansions and to limit the internal pressure to acceptable levels.

The bottom plugs are made bullet nosed to reduce the hydraulic drag during a reactor trip and to guide smoothly into the dashpot section of the fuel assembly guide thimbles. The upper plug is threaded for assembly to the spider and has a reduced end section to make the joint more flexible.

Stainless steel clad silver indium cadmium alloy absorber rods are resistant to radiation and thermal damage, thereby ensuring their effectiveness under all operating conditions. Rods of

similar design have been successfully used in the Saxton, Trino, Yankee Rowe, Indian Point 1, San Onofre, and Connecticut Yankee reactors.

Control Rod Drive Mechanism

Design Description

The magnetic latch control rod drive mechanisms (CRDMs) are used for withdrawal and insertion of the rod cluster control assemblies (RCCAs) into the reactor core and to provide sufficient holding power for stationary support.

Fast total insertion (reactor trip) is obtained by simply removing the electrical power, allowing the rods to fall by gravity.

The complete drive mechanism shown in Figure 3.4-2 consists of the internal latch assembly, the pressure vessel, the operating coil stack, the drive shaft assembly, and the position indicator coil stack.

Each assembly is an independent unit which can be dismantled or assembled separately. A full penetration weld attaches each drive to an adapter on top of the reactor pressure vessel and is connected to the control rod directly below by means of a grooved drive shaft. The upper section of the drive shaft is suspended from the working components of the drive mechanism. The drive shaft and control rod remain connected during reactor operation, including tripping of the rods.

Reactor coolant fills the pressure containing parts of the drive mechanism. All working components and the shaft are immersed in the coolant.

Three magnetic coils which form a removable electrical unit and surround the rod drive pressure housing induce magnetic flux through the housing wall to operate the working components. They move two sets of latches which lift or lower the grooved drive shaft. The three magnets are turned on and off in a fixed sequence by solid-state switches.

The sequencing of the magnets produces step motion over the 228 steps (144 inches) of normal control rod travel. The mechanism develops a lifting force approximately two times the static lifting load. Therefore, extra lift capacity is available for overcoming mechanical friction between the moving and the stationary parts. Gravity provides the drive force for rod insertion and the weight of the whole rod assembly is available to overcome any resistance.

The unit of steps is the preferred reference for control rod movement, as it corresponds to indications used in the Technical Specifications and by plant operators. One control rod step equals 5/8 inch of rod motion.

The mechanisms are designed to operate in water at 650°F and 2485 psig. The temperature at the mechanism head adapter will be much less than 650°F because it is located in a region where there is limited flow of water from the reactor core, while the pressure is the same as in the reactor pressure vessel.

A multi-conductor cable connects the mechanism operating coils to the 125 volt DC power supply. The power supply is described in Section 7.0.

Latch Assembly

The latch assembly contains the working components which withdraw and insert the drive shaft and attached control rod. It is located within the pressure housing and consists of the pole pieces for three electromagnets. They actuate two sets of latches which engage the grooved section of the drive shaft.

The upper set of latches moves up or down to raise or lower the drive rod by one step (5/8 inch). The lower set of latches has 1/16 inch axial movement to shift the weight of the control rod from the upper to the lower latches. In the de-energized condition, the latch assembly does not engage the drive shaft.

Pressure Vessel

The pressure vessel consists of the pressure housing and rod travel housing. The pressure housing is the lower portion of the vessel and contains the latch assembly. The rod travel housing is the upper portion of the vessel. It provides space for the drive shaft during its upward movement as the control rod is withdrawn from the core. The housings are designed in accordance with the requirements of the ASME Code, Section III, Class 1, 1998 Edition through 2000 Addenda.

Operating Coil Stack

The operating coil stack is an independent unit which is installed on the drive mechanism by sliding it over the outside of the pressure housing.

It rests on a pressure housing flange without any mechanical attachment and can be removed or installed while the reactor coolant system is pressurized.

The operator coils (A, B, and C) are made of round copper wire which is insulated with a double layer of filament type glass yarn. The design temperature limit of the coils is 200°C (392°F). Coil temperature can be determined by resistance measurement. Forced air cooling along the outside of the coil stack maintains the coil temperatures below 200°C (392°F).

Drive Shaft Assembly

The main function of the drive shaft is to connect the control rod to the mechanism latches. Grooves for engagement and lifting by the latches are located throughout the 228 steps of control rod travel. The grooves are spaced 5/8 inch apart to coincide with the mechanism step length and have 45° angle sides.

The drive shaft is attached to the control rod by the coupling. The coupling has two flexible arms which engage the grooves in the spider assembly hub.

A 1/4 inch diameter disconnect rod runs down the inside of the drive shaft. It utilizes a locking button at its lower end to lock the coupling and control rod. At its upper end, there is a disconnect assembly for remote disconnection of the drive shaft assembly from the control rod. During plant operation the drive shaft assembly remains connected to the control rod at all times.

Position Indicator Coil Stack

The position indicator coil stack slides over the rod travel housing section of the pressure vessel. It detects drive rod position by means of cylindrically wound differential transformers which span the normal 228 step length of the rod travel.

Drive Mechanism Materials

All parts exposed to reactor coolant, such as the pressure vessel, latch assembly, and drive rod, are made of metals which resist the corrosive action of the water.

Three types of metals are used exclusively: stainless steels, Inconel X-750, and cobalt based alloys. Wherever magnetic flux is carried by parts exposed to the reactor coolant, stainless steel is used. Cobalt based alloys are used for the pins and latch tips.

Inconel X-750 is used for the springs of both latch assemblies and 316 stainless steel is used for all pressure containment. Hard chrome plating provides wear surfaces on the sliding parts and prevents galling between mating parts during assembly.

Outside of the pressure vessel where the metals are exposed only to the reactor plant containment environment and cannot contaminate the main coolant, carbon and stainless steels are used. Carbon steel, because of its high permeability, is used for flux return paths around the operating coils. It is zinc plated 0.001 inch thick to prevent corrosion.

Principles of Operation

The drive mechanisms shown schematically in Figure 3.4-3 withdraw and insert their respective control rods as electrical pulses are received by the operator coils.

ON and OFF sequence, repeated by solid-state switches in the power programmer causes either withdrawal or insertion of the control rod. Position of the control rod is indicated by the differential transformer action of the position indicator coil stack surrounding the rod travel housing. The differential transformer output changes as the top of the ferromagnetic drive shaft assembly moves up the rod travel housing.

Generally, during plant operation, the drive mechanisms hold the control rods withdrawn from the core in a static position, and only the stationary gripper coil is energized on each mechanism.

Control Rod Withdrawal

The control rod is withdrawn by repeating the following sequence:

1. Movable Gripper Coil - ON

   The movable gripper armature rises and swings the movable gripper latches into the drive shaft groove.

2. Stationary Gripper Coil - OFF

   Gravity causes the stationary gripper latches and armature to move downward until the load of the drive shaft is transferred to the movable gripper latches. Simultaneously, the stationary gripper latches swing out of the shaft groove.

3. Lift Coil - ON

   The gap between the lift armature and the lift magnet pole closes and the drive rod rises one step length.

4. Stationary Gripper Coil - ON

   The stationary gripper armature rises and closes the gap below the stationary gripper armature and swings the stationary gripper latches into a drive shaft groove. The latches contact the shaft and lift it 1/32 inch. The load is so transferred from the movable to the stationary gripper latches.

5. Movable Gripper Coil - OFF

   The movable gripper armature separates from the lift armature under the force of three springs and gravity. Three links, pinned to the movable gripper armature, swing the three movable gripper latches out of the groove.

6. Lift Coil - OFF

   The gap between the lift armature and the lift magnet pole opens. The movable gripper latches drop one step length to a position adjacent to the next groove.

Control Rod Insertion

The sequence for control rod insertion is similar to that for control rod withdrawal:

1. Lift Coil - ON

   The movable gripper latches are raised to a position adjacent to a shaft groove.

2. Movable Gripper Coil - ON

   The movable gripper armature rises and swings the movable gripper latches into a groove.

3. Stationary Gripper Coil - OFF

   The stationary gripper armature moves downward and swings the stationary gripper latches out of the groove.

4. Lift Coil -OFF

   Gravity separates the lift armature from the lift magnet pole and the control rod drops down one step length.

5. Stationary Gripper Coil - ON

6. Movable Gripper Coil -OFF

The sequences described above are termed as one step and the control rod moves 5/8 inch for each step. Each sequence can be repeated at a rate of up to 72 steps per minute and the control rods can, therefore, be withdrawn or inserted at a rate of up to 45 inches per minute.

Control Rod Position Definitions

During any approach to criticality, except for physics tests, the control rod position resulting in reactor criticality shall not be lower than the insertion limit for zero power. That is, if the control rods were withdrawn in normal sequence with no other reactivity change, the reactor would not be critical until the control banks were above the insertion limit (Reference 1).

During power operation, the shutdown banks are fully withdrawn. Fully withdrawn is defined as a bank demand position equal to or greater than 225 steps. Evaluation has shown that positioning control rods at 225 steps, or greater, has a negligible effect on core power distributions and peaking factors. Due to the low reactivity worth in this region of the core and the fact that, at 225 steps, control rods are only inserted one step into the active fuel region of the core, positioning rods at this position or higher has minimal effect. This position is varied, based on a predetermined schedule, in order to minimize wear of the guide cards in the guide tubes of the RCCAs (Reference 2).

Control Rod Tripping

If power to the stationary gripper coil is cut off, as for tripping, the combined weight of the drive shaft and the RCCA is sufficient to move the latches out of the shaft groove. The control rod falls by gravity into the core. The tripping occurs as the magnetic field, holding the gripper armature against the lift magnet, collapses and the gripper armature is forced down by the weight acting upon the latches.

Part-Length Rod Drive Mechanisms

The part-length RCCAs have been removed as reactor operating experience has shown them to be unnecessary.

Fuel Assembly and RCC Mechanical Evaluation

To confirm the mechanical adequacy of the fuel assembly and RCCA, functional test programs have been conducted on a full scale San Onofre mock-up version of the fuel assembly and control rods. Additional tests were run on two full scale prototype assemblies for a 12 foot active core.

One of the 12 foot assemblies incorporated stainless steel guide tubes and the other incorporated Zircaloy 4 tubes.

Reactor Evaluation Center (REC) Tests

The prototype assemblies were tested under simulated reactor operating conditions (1875 psig, 575°F, and 17.8 fps flow velocity) in the Westinghouse Reactor Evaluation Center for a total of more than 6400 hours.

Each prototype assembly was subjected to scram cycling equivalent to one or more plant lifetimes. The test history for each prototype is summarized in Table 3.4-1.

Each of the three prototype fuel assemblies described in Table 3.4-1 remained in excellent mechanical condition. No measurable signs of wear on the fuel tubes or control rod guide tubes were found. The control rod was also found to be in excellent condition, having maximum wear measured on absorber cladding of approximately 0.001 inch.

Loading and Handling Tests

Tests simulating the loading of the prototype fuel assembly into a core location were also successfully conducted to determine that proper provisions had been made for guidance of the fuel assembly during refueling operation.

Axial and Lateral Bending Tests

In addition, axial and lateral bending tests were performed in order to simulate mechanical loading of the assembly during refueling operation.

Although the maximum column load expected to be experienced in service is approximately 1000 pounds, the fuel assembly can successfully be loaded to 2200 pounds axially with no damage resulting. This information is also used in the design of fuel handling equipment to establish the limits for inadvertent axial loads during refueling.

CRDM Housing Mechanical Failure Evaluation

An evaluation of the possibility of damage to adjacent CRDM housings in the event of a circumferential or longitudinal failure of a rod housing located on the vessel head is presented.

A CRDM schematic is shown in Figure 3.4-3. The operating coil stack assembly of this mechanism has a 10.8 inch by 10.8 inch cross section and a 39.875 inch length. The position indicator coil stack assembly (not shown in this figure) is located above the operating coil stack assembly. It surrounds the rod travel housing over nearly its entire length. The rod travel housing outside diameter is 3.8 inches and the position indicator coil stack assembly consists of a 1/8 inch thick stainless steel tube surrounded by a continuous stack of copper wire coils. This assembly is held together by two end plates (the top end plate is square), an outer sleeve, and four axial tie rods.

Effect of Rod Travel Housing Longitudinal Failures

Should a longitudinal failure of the rod travel housing occur, the region of the stainless steel tube opposite the break would be stressed by the reactor coolant pressure of 2250 psia. The most probable leakage path would be provided by the radial deformation of the position indicator coil assembly, resulting in the growth of axial flow passages between the rod travel housing and the stainless steel tube. A radial free water jet is not expected to occur because of the small clearance between the stainless steel tube and the rod travel housing and the considerable resistance of the combination of the stainless steel tube and the position indicator coils to internal pressure. Calculations based on the mechanical properties of stainless steel and copper at reactor operating temperature show that an internal pressure of at least 4000 psia would be necessary for the combination of the stainless steel tube and the coils to rupture.

Therefore, the combination of stainless steel tube and copper coils stack is more than adequate to prevent formation of a radial jet following a control rod housing split, which assures the integrity of the adjacent rod housings.

Effect of Rod Travel Housing Circumferential Failures

If circumferential failure of a rod travel housing should occur, the broken off section of the housing would be ejected vertically because the driving force is vertical and the position indicator coil stack assembly and the drive shaft would tend to guide the broken off piece upwards during its travel. Travel is limited to three feet by the missile shield, thereby limiting the projectile acceleration. When the projectile reaches the missile shield, it would partially penetrate the shield and dissipate its kinetic energy. The water jet from the break would push the broken off piece against the missile shield.

If the broken off piece were short enough to clear the break when fully ejected, it could rebound after impact with the missile shield. The top end plates of the position indicator coil stack assemblies would prevent the broken piece from directly hitting the rod travel housing of a second drive mechanism. Even if a direct hit by the rebounding piece were to occur, the low kinetic energy of the rebounding projectile would not be expected to cause significant damage. (Reference 5)

Based on the above, failure of a control rod housing due to either longitudinal or circumferential cracking would not cause damage to adjacent housings that would increase the severity of the initial accident.

Burnable Absorber Rods (No longer used)

The burnable absorber rods are statically suspended and positioned in RCC thimble tubes within the fuel assemblies at some nonrodded core locations. The absorber rods at each core location are grouped and attached together at the top end of the rods by a flat spider plate which fits within the fuel assembly top nozzle and rests on the top adapter plate. The plate (and the absorber rods) are held down and restrained against vertical motion through a spring pack which is attached to the plate and is compressed by the upper core plate when the reactor upper internals package is lowered into the reactor. This ensures that the absorber rods cannot be lifted out of the core by flow forces.

The absorber rods consist of pyrex glass tubes contained within Type 304 stainless steel tubular cladding which is plugged and seal welded at the ends to encapsulate the glass. The glass is also supported along the length of its inside diameter by a thin wall Type 304 stainless steel tubular inner liner. A typical burnable absorber rod is shown in longitudinal and transverse cross sections in Figure 3.4-4.

The rods are designed in accordance with the standard fuel rod design criteria; i.e., the cladding is free standing at reactor operating pressures and temperatures and sufficient cold void volume is provided within the rods to limit internal pressures to less than the reactor operating pressure assuming total release of all helium generated in the glass as a result of the B_{10} (n,α) reaction. The large void volume required for the helium is obtained through the use of glass in tubular form which provides a central void along the length of the rods. The resulting cladding stresses at temperature and pressure are given in Reference 3.

Based on available data on properties of pyrex glass and on nuclear and thermal calculations for the rods, gross swelling or cracking of the glass tubing is not expected during operation. Some minor creep of the glass at the hot spot on the inner surface of the tube is expected to occur but continues only until the glass comes into contact with the inner liner. The inner liner is provided to maintain the central void along the length of the glass and to prevent the glass from slumping or creeping into the void as a result of softening at the hot spot. The wall thickness of the inner liner is sized to provide adequate support in the event of slumping but to collapse locally before rupture of the exterior cladding if large volume changes due to swelling or cracking should possibly occur. The top end of the inner liner is open to receive the helium which diffuses out of the glass.

To ensure the integrity of the burnable absorber rods, the tubular cladding and end plugs are procured to the same specifications and standard of quality as are used for stainless steel fuel rod cladding and end plugs in other Westinghouse plants. In addition, the end plug seal welds are checked for integrity by visual inspection and x-ray. The finished rods are helium leak checked.

Water displacer rods may also be used for power distribution control. These rods consist of outer burnable absorber stainless steel tubes without any borosilicate glass or inner stainless tubes inserted in them. The rods are plugged and seal welded at each end and pressurized with helium.

Evaluations of Burnable Absorber Rods

The burnable absorber rods are positively positioned in the core inside RCCA guide thimbles and held down in place by attachment to a plate assembly compressed beneath the upper core plate and hence cannot be the source of any reactivity transient. Due to the low heat generation rate and the conservative design of the absorber rods, there is no possibility for release of the absorber as a result of helium pressure or cladding temperature during accident transients including loss-of-coolant.

In-pile testing of two of the rods in the Saxton reactor has been conducted to verify mechanical performance of the burnable absorber material and rod configuration in a power reactor environment.

A visual examination of the rods was made in early June 1968, and a visual and profilometer examination was made July 30, 1968 after an exposure of 1900 effective full power hours (~25%

B_{10} depletion). The rods were found to be in excellent condition and profilometry results showed no dimensional variation from the original new condition.

An experimental verification of the reactivity worth calculations for pyrex glass tubing is presented in Reference 4.

REFERENCES

1. WE letter to NRC, "Technical Specification Change Request 153 - Modification of Technical Specification 15.3.10.A.5, Zero Power Rod Insertion Limit," dated November 24, 1992.

2. WE letter to NRC, "Technical Specification Change Request 151 - Documentation of Rod Position in Steps Vice Inches," dated October 6, 1992.

3. WCAP-7113, "Use of Burnable Poison Rods in Westinghouse Pressurized Water Reactors," (Proprietary), October 1967.

4. WCAP-9000, "Nuclear Design of Westinghouse Pressurized Water Reactor with Burnable Poison Rods," (Proprietary), 1968.

5. Westinghouse Calculation CN-RVHP-04-10, "Point Beach Units 1 and 2 HAUP-Missile Impact Analysis," (Westinghouse Proprietary), Revision 5, dated May 18, 2006.

TABLE 3.4-1 PROTOTYPE FUEL ASSEMBLY AND RCC ASSEMBLY TESTS

Prototype	Test Time (Hours)	Number of SCRAMs	Total Rod Travel (Feet)	Total Driven Travel (Feet)	Total SCRAM Travel (Feet)
San Onofre, 10 foot assembly, stainless steel guide thimbles	4132	1461	38,927	27,217	11,710
12 foot assembly, stainless steel guide thimbles	1000	600	45,000	38,500	6,500
12 foot assembly, Zircaloy-4 guide thimbles	1277	600	124,200	117,700	6,500

Figure 3.4-1 TYPICAL ROD CLUSTER CONTROL ASSEMBLY

[redacted]

Figure 3.4-2 CONTROL ROD DRIVE MECHANISM ASSEMBLY

[redacted]

Figure 3.4-3 CONTROL ROD DRIVE MECHANISM SCHEMATIC

[redacted]

Figure 3.4-4 DETAIL OF BURNABLE POISON ROD

[redacted]