4 REACTOR

4.1 SUMMARY DESCRIPTION

4.1.1 REACTOR CORE

The reactor core is a multi-region core containing 121 fuel assemblies. The fuel rods are either cold worked ZIRLO® or partially recrystallized (pRXA) Optimized ZIRLO™ clad tubes containing slightly enriched uranium dioxide fuel.

The fuel assembly is a canless type with the basic assembly consisting of the rod cluster control guide thimbles fastened to the grids and the top and bottom nozzles. The fuel rods are held by the grids and grid springs which provide lateral and axial support for the fuel rods.

Full-length rod cluster control assemblies (also commonly referred to as control rods) are inserted into the guide thimbles of the fuel assemblies. The absorber sections of the control rods are fabricated of silver-indium-cadmium alloy sealed in stainless steel tubes.

There are 29 full-length control rods. The control rod drive mechanisms are of the magnetic latch type. The latches are controlled by three magnetic coils. They are so designed that upon a loss of power to the coils, the rod cluster control assembly is released and falls by gravity to shut down the reactor.

4.1.2 WESTINGHOUSE OPTIMIZED FUEL ASSEMBLIES/422 VANTAGE + FUEL ASSEMBLIES

The transition to an all-Westinghouse Optimized Fuel Assembly (OFA) core began in cycle 14 (region 16) in the spring of 1984 and was completed at the start of cycle 21 (region 23) in the spring of 1991. Ginna operated with all-OFA cores through the end of cycle 32 (region34) in the fall of 2006. OFA fuel assemblies consist of 179 0.400-inch diameter rodlets, of which typically 8 to 120 contain a burnable absorber consisting of a thin enriched zircdiboride coating on the surface of the fuel pellets. Solid natural uranium axial blankets were also introduced with OFA in cycle 14 (region 16).

The OFA top and bottom nozzles are fabricated from stainless steel. Both nozzles index the fuel assembly in the core and direct flow into and out of the assembly through perforated nozzle plates. The axial spacing between the top and bottom nozzle is established to accommodate the growth of the fuel rods due to irradiation effects on the zircaloy fuel tube. The optimized fuel assembly bottom nozzle can be removed to facilitate reconstitution. In cycle 20 (region 22) a removable top nozzle feature was also added to simplify reconstitution.

For cycle 21 (region 23) the debris filter bottom nozzle was introduced into the fuel assemblies to help reduce the possibility of fuel rod damage due to debris-induced fretting. The stainless steel debris filter bottom nozzle is similar to the conventional bottom nozzle design used previously. However, the debris filter bottom nozzle design incorporates a modified flow hole size and pattern. The relatively large flow holes in a conventional bottom nozzle are replaced with a new pattern of smaller flow holes in the debris filter bottom nozzle. The holes are sized to minimize passage of debris particles large enough to cause damage. The hole sizing was also designed to provide sufficient flow area, comparable pressure drop, and continued structural integrity of the nozzle. Significant testing to measure pressure drop and demonstrate structural integrity has been

performed to verify that the debris filter bottom nozzle is totally compatible with the previous design. An additional level of debris damage mitigation was added to the bottom portion of each rodlet in cycle 25 (region 27), with the introduction of a pre-oxidized protective coating on the lowermost portion of the fuel rod cladding, further guards against debris-induced damage at the bottom grid location. The oxide coating is applied to the outside diameter surface of the bottom of the fuel rod cladding using an induction heating process which is indistinguishable from in-reactor oxidation. The end result of the oxide coating process is to accelerate the oxidation process that naturally occurs in-core. Analyses explicitly account for the thermal effects of the oxide coating and confirm that even with the initial coating, the limiting naturally occurring oxide at the higher temperature elevations bounds the maximum expected oxide thickness in the coated segment. Fuel rod performance and the core safety considerations are not adversely affected because the oxide coating is a naturally occurring phenomenon accounted for in the fuel performance and thermal-hydraulic models.

Also introduced in cycle 25 (region 27) were mid-enriched (2.6% U235) axial blankets. A further enhancement to the axial blankets occurred in cycle 27 (region 29) when annular axial blankets were introduced. Annular blankets allow additional volume to reduce rod internal pressure concerns.

In cycle 28 (Region 30), VANTAGE + fuel product features were introduced into the Ginna OFA fuel assemblies. These features included ZIRLO® clad fuel rods, ZIRLO® fabricated guide thimble tubes and instrumentation tubes. ZIRLO® is a zirconium-based alloy that improves fuel assembly corrosion resistance and dimensional stability under irradiation. The chemical composition of the fuel rods and core components fabricated with ZIRLO® alloy is similar to the previous components fabricated from Zircaloy-4 except for a slight reduction in the content of Tin (Sn), Iron (Fe) and the elimination of Chromium (Cr). The ZIRLO® alloy also contains a nominal amount of Niobium (Nb). These composition changes, although small, are responsible for the improved corrosion resistance of ZIRLO® compared to Zircaloy-4.

The Region 30 ZIRLO® fuel rod is the same length as the Zircaloy-4 clad fuel rods used in the previous regions. Since the ZIRLO® clad fuel rods grow less than their Zircaloy-4 equivalents, the design is capable of accommodating extended lead rod average burnups beyond 60,000 MWD/MTU. Also, since ZIRLO® has improved corrosion resistance as compared to Zircaloy-4, the ZIRLO® clad fuel rods will have more margin to the rod internal pressure limit than their Zircaloy-4 equivalents.

Holddown of the Ginna fuel assemblies is provided by four sets of double-leaf springs. The Inconel 718 spring design permits both a high spring rate and large travel, which is required to accommodate the difference in thermal expansion between the zircaloy/ZIRLO® thimbles and the stainless steel reactor internals. This spring design also accommodates the growth of the zircaloy/ZIRLO® thimbles during service and prevents fuel assembly liftoff during MODES 1 and 2.

The fuel rod fretting evaluation performed on a Westinghouse 14 x 14 seven-grid OFA design has shown that even with no grid spring force acting on the fuel rod by the five zircaloy grids at end-of-life, the clad wear criterion is met. Since the Ginna OFA design contains nine grids, including seven zircaloy or ZIRLO® zircaloy grids, considerable additional wear margin exists for the fuel design to that for the seven-grid design.

The rod bow behavior of the optimized fuel assembly is expected to be better than that of the seven-grid Westinghouse fuel assembly. The optimized fuel assembly will have reduced grid spring forces due to the shorter span lengths and a higher fuel tube thickness-to-diameter ratio than the seven-grid fuel assembly. The zircaloy grid spring forces are lower during service than those typically used on Inconel grids. Therefore, lower friction forces are generated by the differential thermal expansion and irradiation growth of the fuel rods. This results in lower loads applied to the skeleton components than are present in the seven-grid Westinghouse assemblies. The skeleton components are conservatively designed to accept these loads with an adequate safety margin. The same conclusions apply to the OFA fuel with VANTAGE + features since the differential thermal expansion frictional forces are the same and the irradiation growth frictional forces are substantially less than those in the optimized fuel assembly.

A fuel transition to a Ginna specific version of the 422VANTAGE + (422V+) fuel assembly occurred in cycle 33 (region 35) to provide increased margins and uranium loading for the implementation of Extended Power Uprate (EPU) to 1775 MWt. This fuel assembly has a larger 0.422-inch diameter rod similar to the original Westinghouse fuel assemblies used at Ginna in cycles 1 through 7. The 422V+ assembly also has a reduced height standard top nozzle, allowing longer fuel rods and an increase in the fuel stack height from 141.4 to 143.25 inches.

All fuel resident in the Ginna core is now OFA with VANTAGE+ features (OFA/VANTAGE+) or 422V+.

The control rods used in the reactor core are compatible with the OFA and 422V+ fuel assemblies. Secondary sources were removed during the cycle 20/21 refueling since neutrons created by spontaneous fission in the burnt fuel provide sufficient source range detector response. Cycle 21 was operated with one thimble plug removed and a program for partial or full core thimble plug assembly removal was implemented starting in cycle 22. The resulting increase in core bypass flow is accounted for in the Chapter 15 safety analysis.

In Cycle 41, Optimized ZIRLO™ cladding material (Reference 4) was introduced into the 422V+ fuel assembly design to provide a benefit to the clad oxidation rate. The Optimized ZIRLO™ clad rods will have the same dimensions and features as the ZIRLO® clad rods, except Optimized ZIRLO™ clad material has lower tin content and a partially recrystallized (pRXA) structure.

4.1.3 RECONSTITUTED FUEL ASSEMBLIES

Ginna is licensed to use reconstituted fuel assemblies in reload cores. Each 14 x 14 fuel assembly includes 179 fuel rod locations, 16 guide tubes, and one instrument thimble. In accordance with cycle-specific reload analyses.fuel rods can be replaced with solid filler rods that are either zircaloy, ZIRLO®, or stainless steel as the result of reconstitution activities. This reconstitution process permits the continued use of these reconstituted fuel assemblies without increasing coolant activity (*Reference 2*).

4.1.4 STARTUP REPORT

A summary report of plant startup and power escalation testing shall be submitted following: (1) amendment to the operating license involving a planned increase in power level, (2) installation of fuel that has a different design or has been manufactured by a different fuel supplier, or (3) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests performed and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, or (2) 90 days following resumption of commercial power operation, whichever is earliest. If the Startup Report does not cover both events (i.e., completion of startup test program, and resumption of commercial power operation), supplementary reports shall be submitted at least every three months until both events have been completed.

REFERENCES FOR SECTION 4.1

1. Letter from J. E. Maier, RG&E, to H. R. Denton, NRC, Subject: Application for Amendment to OL, Westinghouse 14 x 14 Optimized Fuel for Cycle 14, dated December 20, 1983.

2. Westinghouse Electric Corporation, Westinghouse Fuel Assembly Reconstitution Evaluation Methodology, WCAP-13061-NP-A, July 1993.

3. S. L. Davidson, VANTAGE + Fuel Assembly Reference Core Report, WCAP-12610-P-A, April 1995, Westinghouse Proprietary.

4. Shah, H. H., "Optimized ZIRLO™," WCAP-12610-P-A & CENPD-404-P-A Addendum 1-A, July 2006, Westinghouse Proprietary.

4.2 FUEL SYSTEM DESIGN

4.2.1 DESIGN BASES

This section describes the fuel system design bases from the standpoint of performance objectives, principal design criteria, and safety limits.

4.2.1.1 Performance Objectives

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

The control rods, being long and slender, are relatively free to conform to any small misalignments. Tests show that the rods are very easily inserted and not subject to binding even under conditions of severe misalignments. In order to address issues of control rod binding as described in USNRC Information Notice 96-01, fuel assemblies are evaluated on a cycle basis for susceptibility to Incomplete Rod Insertion (IRI). Where warranted, detailed analysis is performed during the cycle design phase to ensure that IRI is precluded. The control rods provide sufficient control rod worth to shut the reactor down with sufficient shutdown margin in the hot condition at any time during the cycle life with the most reactive control rod stuck in the fully withdrawn position. Redundant equipment is provided to add soluble poison to the reactor coolant to ensure a similar shutdown capability when the reactor coolant is cooled to ambient temperatures.

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 each 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 clad temperatures during normal reactor operation and at power densities up to the design limit are conservatively evaluated and found to be consistent with safe operating limitations.

4.2.1.2 Principal Design Criteria

The design criteria cited in Sections 4.2.1.2.1 through 4.2.1.2.8 were used during the design and initial licensing of Ginna Station. They represent the Atomic Industrial Forum version of proposed criteria issued by the AEC for comment on July 10, 1967. Therefore, they are identified as AIF-GDCs. Conformance with the applicable General Design Criteria of 10 CFR 50, Appendix A, identified in Section 4.2.1.2.9, is discussed in Section 3.1.2.

4.2.1.2.1 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 MODES 1 and 2 with appropriate margins for uncertainties and for specified transient situations which can be anticipated (AIF-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 MODES 1 and 2 with appropriate margins for uncertainties and anticipated transient situations, including the effects of the loss of reactor coolant flow (Section 15.3), loss of electrical load (Section 15.2.2), loss of normal feedwater (Section 15.2.6), and loss of all offsite power (Section 15.2.5).

The reactor control protection system is designed to actuate a reactor trip for any anticipated combination of plant conditions, when necessary, to ensure departure from nucleate boiling (DNB) will not occur on the limiting fuel rods during MODES 1 and 2, operational transients, or transient conditions of moderate frequency, with a 95% probability and at a 95% confidence level. The integrity of the fuel cladding is ensured by preventing excessive fuel swelling, excessive cladding overheating, and excessive cladding stress and strain. This is achieved by designing the fuel rods so that the following conservative limits are not exceeded during MODES 1 and 2 or any anticipated transient condition:

A. DNB on limiting fuel rod does not occur.

B. Fuel center line temperature below melting point of uranium dioxide.

C. Internal gas pressure less than that required to increase the fuel clad diametral gap during MODES 1 and 2 and cause extensive DNB propagation to occur.

D. Clad stresses less than the yield strength of the cladding material.

E. Clad strain for MODES 1 and 2 is limited to 1% from the unirradiated condition. The transient limit is 1% from the pre-transient value.

The ability of fuel designed and operated to these criteria to withstand postulated normal and abnormal service conditions is shown by analyses described in Chapter 15 to satisfy the demands of plant operation well within applicable regulatory limits.

Should pellet/clad gap reopening be predicted to occur during the design cycle, analyses will be performed to show that applicable regulatory limits (17% oxidation) are still met.

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 transient results in a high pressurizer pressure, overtemperature delta T, or low steam generator water level trip and thereby prevents fuel damage for this transient. A loss of external electrical load at 50% of full power or less is normally controlled by rod cluster insertion together with a controlled steam dump to the condenser and atmosphere 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 could result in fuel centerline melting or DNB during the transient.

Neither the turbine trip nor the loss-of-flow events cause changes in coolant conditions to provoke a large 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.

4.2.1.2.2 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 (AIF-GDC 7).

The design of the reactor core and related protection systems ensures that power oscillations which could cause fuel damage in excess of acceptable limits are not possible or can be readily suppressed.

The potential for possible spatial oscillations of power distribution for this core has been reviewed. In summary it was concluded that the only potential spatial instability of a magnitude which could cause damage in excess of acceptable fuel damage limits was the xenon induced axial instability which may be a nearly free-running oscillation with little or no inherent damping. Part-length control rods were originally provided to suppress these oscillations, if they occurred. They have since been removed. Operating control strategies have been devised that do not require part-length rods and eliminate the potential for axial xenon instabilities.

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 information provided by the out-of-core instrumentation.

4.2.1.2.3 Redundancy of Reactivity Control

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

Two independent reactivity control systems are provided, one involving rod cluster control assemblies and the other involving chemical shimming.

4.2.1.2.4 Reactivity MODE 3 (Hot Shutdown) Capability

CRITERION: The reactivity control systems provided shall be capable of making and holding the core subcritical from any hot operating (MODES 1 and 2) condition (AIF-GDC 28).

The reactivity control systems provided are capable of making and holding the core subcritical from any operating or hot standby condition, including those resulting from power

changes. The maximum excess reactivity for the core originally occurred for the cold, clean condition at the beginning-of-life of the initial core.

With the Extended Power Uprate (EPU) and transition to 422V+ fuel, cores with more excess reactivity than the previous cores will be loaded.

The rod cluster control assemblies are divided into two categories comprising a control group and shutdown groups. The control group, used in combination with chemical shim control, provides control of the reactivity changes of the core throughout the life of the core at power conditions. This group of rod cluster control assemblies is used to compensate for short-term reactivity changes at power that might be produced due to variations in reactor power requirements or in coolant temperature. Chemical shim 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.

4.2.1.2.5 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 (AIF-GDC 29).

The control rod system provided is capable of making the core subcritical under any condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. The shutdown margin ensures subcriticality with the most reactive control rod fully withdrawn. The shutdown groups are provided to supplement the control group of rod cluster control assemblies to make the reactor subcritical with the required shutdown margin following trip from any credible operating condition to the hot zero-power condition assuming the most reactive rod cluster control assembly remains in the fully withdrawn position. Manually controlled boric acid addition is used to supplement the rod cluster control assemblies in maintaining the shutdown margin for the long-term conditions of xenon decay or plant cooldown. See Section 9.3.4 concerning details of the chemical and volume control system.

4.2.1.2.6 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 (AIF-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 two seconds following a trip signal by control rods. Boric acid injection is 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 or refueling water storage tank (RWST) and ready

for injection always exceeds that quantity required to reach MODE 5 (Cold Shutdown) while maintaining the required minimum shutdown margin. This quantity will also exceed the quantity of boric acid required to bring the reactor to MODE 3 (Hot Shutdown) and to compensate for subsequent xenon decay.

Boric acid is pumped from the boric acid storage tanks by one of two boric acid transfer pumps to the suction of one of three charging pumps which injects 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 operating at the nominal charging flow rate of 46 gpm and shut the reactor down with no rods inserted in approximately 81 minutes. Sufficient boric acid from the Boric Acid Storage Tanks (BAST) o RWST can be injected to compensate for xenon decay beyond the equilibrium level, with one charging pump operating at its minimum speed, and thereby delivering in excess of the required minimum of approximately 9 gpm into the reactor coolant system. If two charging pumps (or one pump at greater than minimum flow in the xenon decay case) and two boric acid transfer pumps are available, these time periods are reduced. Although the charging pumps are larger capacity than the boric acid transfer pumps, it is desirable to operate two charging pumps if two boric acid transfer pumps are operated. Additional boric acid injection is employed if it is desired to bring the reactor to MODE 5 (Cold Shutdown) conditions.

On the basis of the above, the injection of boric acid is shown to afford backup reactivity shutdown capability, independent of control rod clusters 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. Alternately, boric acid solution at lower concentration can be supplied from the refueling water storage tank (RWST). This solution can be transferred directly by the charging pumps or alternately by the safety injection pumps. The reduced boric acid concentration lengthens the time required to achieve equivalent shutdown.

4.2.1.2.7 Reactivity Control Systems Malfunction

CRITERION: The Reactor Trip System (RTS) shall be capable of protecting against any single malfunction of the reactivity control systems, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits (AIF-GDC 31).

The Reactor Trip System (RTS) is capable of protecting against any single anticipated malfunction of the reactivity control system, by limiting reactivity transients 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 rod mechanisms regardless of existing control signals.

Details of the effects of continuous withdrawal of a control rod are described in Section 15.4.1 and Section 15.4.2 and Section 15.4.4 describes the effects of continuous deboration.

4.2.1.2.8 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 (AIF-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 change of reactivity 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, greater than 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 reactivity insertion rate is analyzed in the detailed plant analysis assuming two of the highest worth groups to be accidentally withdrawn at maximum speed with 100% overlap, yielding reactivity insertion rates of the order of 9×10^{-4} Δk/sec which is 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 control rod to be withdrawn at a speed greater than 77 steps per minute.

4.2.1.2.9 Conformance With 1972 General Design Criteria

The adequacy of the Ginna Station reactor system design relative to the following General Design Criteria of 10 CFR 50, Appendix A, is discussed in Section 3.1.2.

GDC 1 Quality Standards and Records.

GDC 10 Reactor Design.

GDC 11 Reactor Inherent Protection.

GDC 12 Suppression of Reactor Power Oscillations.

GDC 13 Instrumentation and Control.

GDC 14 Reactor Coolant Pressure Boundary.

GDC 20 Protection System Functions.

GDC 25 Protection System Requirements for Reactivity Control Malfunctions.

GDC 26 Reactivity Control System Redundancy and Capability.

GDC 27 Combined Reactivity Control Systems Capability.

GDC 28 Reactivity Limits.

GDC 29 Protection Against Anticipated Operational Occurrences.

4.2.1.3 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 elements. Thus the release of unacceptable amounts of fission products to the coolant is prevented.

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

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

4.2.1.3.1 Nuclear Limits

At a full power level (license application power) the heat flux hot-channel factor, F_Q, specified in Table 4.2-1, 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 to their combined relationship so as not to exceed the F_Q or DNBR limits.

Part-length control rods were provided in the original design in order to control axial xenon oscillations to preclude adverse core conditions. However, constant axial offset operating strategies have been devised that ensure adequate control of axial xenon oscillations without the necessity of using the part-length rods; thus, they were removed. The protection system ensures that the nuclear core limits are not exceeded.

4.2.1.3.2 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:

A. Sufficient control is available to produce the required MODE 3 (Hot Shutdown) shutdown margin.

B. The shutdown margin is maintained with the most reactive rod cluster control assembly stuck in the fully withdrawn position.

C. The shutdown margin is maintained at cold shutdown by the use of soluble poison.

4.2.1.3.3 Thermal and Hydraulic Limits

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

A. There is at least a 95% probability that DNB will not occur on the limiting fuel rods during MODES 1 and 2, operational transients, or any condition 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.

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

The ratio of the heat flux causing DNB at a particular core location, as predicted by the W-3 correlation or the improved WRB-1 correlation, to the existing heat flux at the same core location is the DNBR. A Design Limit DNBR is defined in Section 4.4.3. Analytical assurance that DNB will not occur is provided by showing calculated DNBR to be higher for all conditions of normal operation, operational transients and transient conditions of moderate frequency. The Design Limit DNBR is chosen by using the Revised Thermal Design Procedure (RTDP) which includes appropriate margin to DNB for all operating conditions sufficient to assure compliance with the DNBR criteria above.

4.2.1.3.4 Mechanical Limits

4.2.1.3.4.1 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 is limited to prevent excessive movements resulting from seismic disturbances and thus prevent interference with rod cluster control assemblies. Core drop in the event of failure of the normal supports is limited so that the rod cluster control assemblies 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. Analysis performed for limited size breaks reported in WCAP-9748 (Proprietary) and WCAP-9749 (Non-Proprietary) June 1980, Westinghouse Owners Group Asymmetric LOCA Load Evaluation - Phase C, showed that the appropriate systems and components will maintain their functional capability to ensure a safe plant shutdown with a coolable core geometry. The systems and components examined were the reactor vessel assembly including internals, fuel, control rod drive mechanisms, vessel and component supports, reactor coolant loop piping, and attached emergency core cooling piping. Furthermore, in the resolution of Unresolved Safety Issue A-2, Asymmetric Loading, it was concluded that an acceptable basis has been provided so that asymmetric blowdown loads resulting from double-ended pipe breaks in main coolant loop piping need not be considered as a design basis for Ginna Station, provided that leakage detection systems exist to detect postulated flaws utilizing guidance from Regulatory Guide 1.45. Conformance with Regulatory Guide 1.45 is discussed in Section 5.2.5.5.

The structural integrity of various reactor internal critical components was evaluated in *Reference 16* for operation over vessel average temperatures ranging from 559.0°F to 581.2°F. The

evaluation concluded that the critical components maintained structural integrity over the temperature range with the revised design transients associated with the replacement steam generators.

In 1998 the Nuclear Regulatory Commission issued Information Notice (IN) 98-11, "Cracking of Reactor Vessel Internal Baffle Former Bolts in Foreign Plants". Rochester Gas and Electric participated with the Westinghouse Owners Group to respond to these concerns and evaluate the structural integrity of Ginna's baffle-former-bolts. In 1999 an ultrasonic inspection was performed on all accessible bolts. As a result of this examination 56 bolts (from a total population of 728) were replaced. The replacement bolts were manufactured from type 316 stainless steel rather than the type 347 stainless steel used in the original bolts. Type 316 steel is believed to be less susceptible to the flaw initiation mechanism.

In 2011, ultrasonic inspections performed on the 56 bolts replaced in 1999 showed no defects. At the same time, ultrasonic inspections performed on 99 of the remaining original bolts only showed one bolt to have a defect. An additional 25 original bolts were also replaced with the improved type 316 stainless steel bolts and ultrasonic inspections performed on 24 of those removed original bolts showed no defects (one was destroyed during removal process). At three locations, bolt holes were left empty because damage occurred during the bolt removal process which prevented insertion of new replacement bolts. The presence of these empty bolt holes was evaluated with respect to structural integrity of the baffle assembly and possible damage to adjacent fuel assemblies. The evaluations, performed in accordance with NRC-approved methods, showed no adverse impact on either area.

In addition to the evaluations described above, the Ginna reactor internals components were evaluated for plant license renewal. System and component materials of construction, operating history and programs used to manage aging effects are documented in NUREG-1786, Safety Evaluation Report (SER) Related to the License Renewal of R. E. Ginna Nuclear Power Plant, May 2004.

The impact of the Extended Power Uprate (EPU) on the conclusions reached in the Ginna License Renewal Application for the reactor internals and core supports were assessed. The NRC SER for the EPU found the proposed EPU acceptable with respect to the design of the reactor internal and core supports.

4.2.1.3.4.2 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.

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.

4.2.1.3.4.3 Control Rods

The criteria used for the design of the cladding on the individual absorber rods in the control rods are similar to those used for the fuel rod cladding. The cladding is designed to be free-standing under all operating conditions and maintain encapsulation of the absorber material throughout the absorber rod design life. Allowance for wear during operation is included in the rod cluster control assembly cladding thickness. 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 without 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.

4.2.1.3.4.4 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, Nuclear Vessels for Class 1 Vessel appurtenances.

The control rod drive assemblies provide rod cluster control assembly 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 rod cluster control assemblies 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 the rod cluster control assembly enters the dash-pot region.

4.2.2 FUEL SYSTEM DESIGN DESCRIPTION

The fuel assemblies are arranged in a roughly circular cross-sectional pattern. The assemblies are essentially identical in configuration, but contain fuel of different enrichments depending on the location of the assembly within the core.

The fuel is in the form of slightly enriched uranium dioxide ceramic pellets. The pellets are stacked to an active height of approximately 141 to 144 in. within ZIRLO® or Optimized ZIRLO™ tubular cladding, which is plugged and seal-welded at the ends to encapsulate the fuel. The typical enrichments of the fuel for the axial blanket and enriched regions are given in Table 4.2-2. The enrichments of the fuel for the various regions of the core are determined during the core design process for each reload to obtain the desired cycle energy. Heat generated by the fuel is removed by light water which flows upward through the fuel assemblies and acts as both moderator and coolant. Refueling takes place generally in accordance with a low leakage loading pattern with some fuel assemblies remaining in the core for up to four cycles.

The control rods, designated as rod cluster control 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 position, the absorber rods fit within hollow guide thimbles in the fuel assemblies. The guide thimbles are an integral part of the fuel assembly skeleton 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 4.2-1 shows a typical rod cluster control assembly.

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.

4.2.3 CORE COMPONENTS DESIGN DESCRIPTION

4.2.3.1 Fuel Assembly

The overall configuration of the fuel assemblies is shown in Figures 4.2-2 and 4.2-3. The assemblies are square in cross section, nominally 7.763 in. on a side, and have an overall height of approximately 160 in. 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 in. between rods. Of the total possible 196 rod locations per assembly, 16 are occupied by guide thimbles for the rod cluster control rods and one for in-core instrumentation. The remaining 179 locations contain fuel rods. In addition to fuel rods, a fuel assembly is composed of a top nozzle, a bottom nozzle, nine grid assemblies, 16 absorber rod guide thimbles, and one instrumentation thimble.

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 framework the fuel rods are contained and supported and the rod-to-rod centerline spacing is maintained along the assembly.

Defective fuel rods can be replaced with filler (dummy) rods fabricated from Zircaloy-4, ZIRLO®, or stainless steel to create a reconstituted fuel assembly. Reconstitution is accomplished by removing either the top or bottom nozzle, removing the defective rod(s), replacing the failed rod(s) with filler rod(s), and reattaching the nozzle. If a fuel assembly skeleton is damaged, serviceable fuel rods and dummy rods can be transferred to a new skeleton to form a reconstituted fuel assembly. The reconstituted fuel assemblies meet the same design requirements and satisfy the same design criteria as the original fuel assemblies.

4.2.3.1.1 Top Nozzle, Springs, and Clamps

The top nozzle, adapter plate, holddown springs, and clamps are shown in Figure 4.2-4. The perforated adapter plate directs the core flow through the nozzle enclosure into the upper internals. Two alignment holes are located in the top nozzle and mate with the pins in the upper core plate. The fuel assembly holddown provision consists of four sets of double-leaf springs that are clamped by screws at diagonally opposite corners of the top nozzle with the screws being secured in place by welded lock wires. During assembly a preload is normally applied to the springs, though it is possible to meet all required clearance tolerances with zero preload. The springs also prevent fuel-induced liftoff. The top nozzle, adapter plate, and clamp are stainless steel (type 304) whereas the springs are fabricated from Inconel.

4.2.3.1.2 Bottom Nozzle

The bottom nozzle is fabricated from type 304 stainless steel and consists of a perforated plate with four support legs (see Figure 4.2-5). The perforated plate directs the flow of the coolant upward toward the fuel rods. Indexing and positioning of the fuel assembly is controlled by alignment holes in two diagonally opposite legs which mate with two locating pins on the lower core plate.

Debris filter bottom nozzles (DFBN) were used commencing with the feed assemblies for cycle 21. The debris filter bottom nozzle is similar to the prior bottom nozzles except that it is designed to inhibit debris from entering the active fuel region of the core and minimize debris-related fuel failures. Composite debris filter bottom nozzles with reinforcing skirts have been used since the feed assemblies for cycle 29. The DFBN is a two-piece design incorporating a machined stainless steel adapter plate welded to a single low cobalt investment casting of the legs and skirts which enhances reliability during postulated adverse handling conditions while refueling. The composite DFBN is structurally and hydraulically equivalent to the previous DFBN design and meets all fuel assembly design criteria.

4.2.3.1.3 Guide Thimbles

The guide thimble tubes are fabricated from ZIRLO® tubing. The guide tubes are structural members which also provide channels for core components, e.g., control rods, sources, burnable absorbers, and thimble plugs.

For Optimized Fuel Assemblies (OFA) as shown in Figure 4.2-6, there are two sections with a large diameter and two with a smaller diameter. The larger diameter at the top permits rapid insertion of the control rods during a reactor trip. The lower short expanded diameter accommodates the mechanical fastening of the second grid from the bottom. Both reduced-diameter sections produce a dashpot action near the end of the control rod travel during a reactor trip which decelerates the control rod and reduces the impact forces between the control rod spider hub and fuel assembly adapter plate at the end of travel. Orifice holes are provided in the tube wall to allow water to exit, thus controlling the rod drop time. The thimble tube has an end plug welded to the bottom end.

The newer 422V+ fuel assembly incorporates the tube-in-tube internal dashpot design which consists of a constant diameter outer guide thimble assembly with a separate smaller diameter dashpot assembly which is inserted into the outer guide thimble assembly. Both the guide thimble diameters and the dashpot inside diameter are the same as the OFA. The dashpot assembly is retained by a press fit with the guide thimble plug at the bottom and to the guide thimble tube with two restraint bulges just above the bottom grid. The tube-in-tube design operates in the same manner as the OFA fuel currently in service but eliminates the swaged reduced diameter portion of the guide thimble tube. This provides increased structural integrity and eliminates a potential incomplete rod insertion contributor.

4.2.3.1.4 Instrumentation Tube

The instrumentation tube is fabricated from ZIRLO®. It is a constant diameter tube as shown in Figure 4.2-7 and is designed to accept the in-core instrumentation. The instrumentation tube is supported at the various grid elevations in the same manner as the fuel rods and is supported and positioned at the lower end by the bottom nozzle.

4.2.3.1.5 Grid Assemblies

The fuel rods are supported at intervals along their length by grid assemblies which maintain the lateral spacing between the rods throughout the design life of the assembly. Each fuel rod is given support at six contact points within each grid cell by a combination of support dimples and springs. The grid assembly consists of individual interlocking slotted straps that are joined either by brazing or welding depending on the material. The straps provide support springs and dimples along with hydraulic flow mixing vanes.

The top and bottom grid material is Alloy 718, which was chosen because of its corrosion resistance, high strength properties, and resistance to irradiation-induced stress relaxation. The middle seven grids are fabricated from Zircaloy-4 or ZIRLO® which was chosen because of its good neutron economy properties. The Inconel grids are furnace-brazed while the zircaloy grids are laser-welded. The magnitude of the grid restraining force on the fuel rod is set high enough to minimize potential fretting without overstressing the cladding at the

points of contact between the grids and fuel rods. The grid assemblies also allow axial thermal expansion and irradiation-induced growth of the fuel rods while imposing minimal axial restraining forces on the fuel rods.

The dimple contact area in the 422V+ ZIRLO® mid-grids have been increased to provide significantly reduced contact stresses and to reduce wear rates on the cladding. A balanced vane pattern has also been introduced to eliminate a known mechanism for fuel assembly vibration. The OFA Zircaloy-4 mid-grid design remains unchanged.

The outside straps on all grids contain guide vanes and guide tabs which, in addition to their mixing function, aid in guiding the fuel assemblies past projecting surfaces during handling or loading and unloading of the core.

4.2.3.1.6 Fuel Rods

The fuel rods consist of a stack of uranium dioxide ceramic pellets contained in slightly cold worked ZIRLO® tubing or in partially recrystallized (pRXA) Optimized ZIRLO™ tubing. The tubing is plugged and seal welded at the ends to encapsulate the fuel.

The top and bottom of the uranium stack (nominal 6.0 in.) contain the axial blanket pellets fabricated from slightly-enriched uranium. Commencing with Region 29 the blanket pellets are annular versus solid. In the center portion of the rod, the pellets are fabricated from slightly-enriched uranium. The tubing and end plug material is Zircaloy 4, ZIRLO®, or Optimized ZIRLO™ alloy. The function of the tube and end plugs is to contain the pellets and fission products. The bottom end plug has an internal pull grip feature for insertion and removal of the fuel rod to and from the assembly. Inside the top end of the fuel rod there is a stainless steel spring which is designed to prevent axial movement of the stack during shipment and handling of the fuel prior to irradiation.

4.2.3.1.7 Fuel Assembly Joints and Connections

All grids in the optimized fuel assembly design, with the exception of the bottom grid, are mechanically fastened to the thimble tubes by bulging into cylindrical sleeves which are attached to the grid straps. The sleeves used to attach the middle seven grids are fabricated from Zircaloy-4 and are welded to the grid straps. An illustration of a mid-grid joint is shown in Figure 4.2-8. The top grid sleeves are fabricated from stainless steel and is brazed to the Inconel grid straps. The top grid sleeve is then attached to the top nozzle adapter plate, as shown in Figure 4.2-9.

The bottom grid is fastened by clamping it between the thimble tube end plug and bottom nozzle via stainless steel inserts. The inserts are welded to the bottom grid straps. The bottom nozzle is fastened to the skeleton structure by means of stainless steel screws, which mate with internal threads in the thimble tube end plug. Reconstitutable design features have been incorporated into the bottom nozzle/thimble screw design.

As shown in Figure 4.2-9, the Ginna OFA/VANTAGE+ and 422V+ fuel assemblies are fitted with a top nozzle that is easily removable in the field to facilitate reconstruction of the defective fuel. It consists of a stainless steel nozzle insert, which is mechanically connected to the top nozzle adapter plate by means of a pre-formed circumferential bulge near the top of the

insert. The insert engages a mating groove in the wall of the adapter plate guide thimble through hole.

The insert has four equally spaced axial slots which will narrow to allow the insert to deflect inwardly at the location of the bulge, thus facilitating the installation or removal of the nozzle. The insert bulge is positively held in the adapter plate mating groove by placing a lock tube (same OD and ID as the guide thimble tube) into the insert. The lock tube is secured in place by local deformations that fit into the concave side of the insert's bulge and under a shoulder in the top nozzle adapter.

The bottom grid joint configuration is similar to the top grid design with stainless steel sleeves brazed in-place which allow the bottom grid to be mechanically attached to the guide thimble by two bulges located above the grid.

4.2.3.1.8 Fuel Assembly Identification

In addition to identifying the fuel assemblies in accordance with applicable NRC regulations, identification numbers are placed on assembly face three and top of the assembly. The numbers on face three are 2 in. in height and the numbers on the top are as large as practicable. The identification markings on the top of the fuel assembly are limited to a combined total of three digits, including numbers and letters.

4.2.3.2 Control Rods

The control rods or rod cluster control assemblies 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 4.2-1, 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 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 tips of the absorber rods remain 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.

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 rod cluster control assembly 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 centerpost which will hold 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 Alloy X-750 alloy and the retainer which is of 17-4

PH material. The absorber rods are secured to the spider so as to ensure 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. Sufficient diametral and end clearance 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. Ginna augmented the original rod cluster control assemblies with enhanced-performance rod cluster control assemblies. These assemblies have a thin chrome electroplate applied to a length of the stainless steel cladding in contact with the reactor internal guides to provide increased resistance to cladding wear. The enhanced-performance control rods also have reduced diameter absorber at the tips to allow more room within the clad for irradiation-induced swelling.

Withdrawal and insertion of the control rods into the core is accomplished by the control rod drive system described in Section 3.9.4.

4.2.3.3 Neutron Source Assemblies

The following is a historical discussion of neutron sources. Four neutron source assemblies were utilized initially in the core. These consisted of two assemblies with four secondary source rods each, and two assemblies with three secondary source rods and one primary source rod each. The rods in each assembly were fastened to a spider at the top end similar to the rod cluster control assembly spiders. The primary sources were discharged after the initial core. In subsequent cores, two or four secondary sources were used.

The secondary sources were removed at the cycle 20/21 refueling. The neutron emissions naturally occurring from the irradiated fuel provide a sufficient neutron source for startup.

4.2.3.4 Plugging Devices

The following is a historical discussion of thimble plugs. In order to limit bypass flow through the guide thimbles in fuel assemblies that do not contain either control rods or source assemblies, the fuel assemblies at those locations are fitted with plugging devices. The plugging devices consist of a flat spider plate with short rods suspended from the bottom surface and a spring pack assembly. 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 when the upper internals package is lowered into place. Similar short rods are also used on the source assemblies to fill the ends of all vacant fuel assembly guide thimbles.

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.

All thimble plugs had been removed from the core by the start of cycle 23.

4.2.3.5 Fuel Pellet and Cladding Design Considerations

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.

Perforation of fuel rod cladding which could release fission products or fuel material is directly related to cladding stress and strain under normal operating and overpower conditions. The stress limit during MODES 1 and 2 or conditions of moderate frequency is the 0.2% offset yield strength of the cladding. Cladding strain for MODES 1 and 2 is limited to 1% from the unirradiated condition. The transient clad strain limit is 1% from the pre-transient value.

For most of the fuel rod life the actual stresses and strains are considerably below the design limits so significant margins exist between actual operating conditions and the design limits.

Other parameters having an influence on cladding stress and strain and the design limits of these parameters are as follows:

A. Internal gas pressure. Rod internal gas pressure shall be limited to a value which would not cause (1) the fuel-clad diametral gap to increase due to outward cladding creep during steady-state operation and (2) extensive DNB propagation to occur.

B. Cladding temperature. The clad surface temperature (oxide to metal interface) shall not exceed that which is required to preclude a condition of accelerated clad oxidation.

C. Burnup. Fuel burnup affects both swelling of the fuel pellet and the release of fission gases during transient conditions. Also, the in-reactor time can affect the metallurgical properties of the cladding to varying degrees. Therefore, cladding stress and strain limits must be evaluated as a function of burnup to determine the limitations that exist.

D. Fuel temperature and kW/ft. During events of moderate frequency there shall be at least a 95% probability that fuel rods operating at the peak kW/ft will not exceed the uranium dioxide melting temperature. The melting temperature of unirradiated uranium dioxide is taken as 5080°F and decreases with burnup.

4.2.3.6 Reload Fuel Design

4.2.3.6.1 Reload Fuel Design - Westinghouse Optimized Fuel

Starting with cycle 14 Ginna Station began the transition to an all Westinghouse optimized fuel assembly fueled core. By cycle 21 the core consisted of all Westinghouse optimized fuel assemblies. Table 4.2-3 provides parameters associated with the optimized fuel assembly.

The 14 x 14 optimized fuel assembly is shown in Figure 4.2-3.

4.2.3.6.2 Reload Fuel Design - Westinghouse OFA/VANTAGE + Fuel

Starting with cycle 28, Ginna Station began the transition to an OFA fuel assembly with VANTAGE + fuel features. Table 4.2-3 provides parameters associated with the OFA/VANTAGE + fuel assembly.

4.2.3.6.3 Reload Fuel Design - Westinghouse 422V+ Fuel

Starting with cycle 33, Ginna Station began the transition to an all Westinghouse 422V+ fuel assembly core. Table 4.2-3 provides parameters associated with the 422V+ fuel assembly and the fuel assembly is shown in Figure 4.2-3.

4.2.3.7 Fuel Assembly and Rod Cluster Control Assembly Tests

To prove the mechanical adequacy of the original core fuel assembly and rod cluster control assembly, functional test programs were conducted on full scale San Onofre mock-up versions of the fuel assembly and control rods. (*Reference 1*).

4.2.3.7.1 Reactor Evaluation Center Tests

The prototype assemblies were tested under simulated reactor operating conditions (1900 psig, 575°F, 14 fps flow velocity) in the Westinghouse Reactor Evaluation Channel.

The components were subjected to a total environmental exposure of 4132 hours during which the rod cluster control assembly experienced a total travel of 38,927 linear feet. The travel was made up of 27,217 ft of normal driven travel and 11,710 ft of reactor trip travel, resulting from 1461 trips, which is equivalent to over two plant service lifetimes.

The fuel assembly 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 in.

4.2.3.7.2 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 MODE 6 (Refueling) operation.

The change to the short top nozzle 422V+ assembly design has necessitated changes to the manipulator crane gripper, the RCCA change fixture, the new fuel handling tool, the spent fuel handling tool, the portable RCCA change tool, and the RCCA stop on the fuel transfer cart. The new tooling is consistent with proven designs in service at the other plants.

4.2.3.7.3 Axial and Lateral Bending Tests

In addition, axial and lateral bending tests were performed in order to simulate mechanical loading of the assembly during MODE 6 (Refueling) operation.

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

4.2.4 DESIGN EVALUATION

4.2.4.1 Fuel and Cladding Evaluation - Original Core

The fission gas release and the associated buildup of internal gas pressure in the fuel rods was calculated by the FIGHT code based on experimentally determined rates. The increase of internal pressure in the fuel rod due to this phenomena was 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 clad creep was limited to less than 1% throughout core life. The associated stresses were below the yield strength of the material under all normal operating conditions.

To ensure that manufactured fuel rods met a high standard of excellence from the standpoint of functional requirements, many inspections and tests were performed both on the raw material and the finished product. These tests and inspections included chemical analysis, 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 uranium dioxide fuel pellets to attack by hot water protects against fuel deterioration or a 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 extensive 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.

4.2.4.2 Design Evaluation - Reload Optimized Fuel Assembly, OFA/VANTAGE+ Fuel Assembly, and 422 VANTAGE+ Fuel Assembly Designs

<u>4.2.4.2.1 Introduction</u>

The design and safety analysis of the optimized fuel assembly is discussed in WCAP-9500 (*Reference 2*) which the NRC has reviewed and found acceptable. However, the staff SER of WCAP-9500 requires that certain items be addressed on a plant specific basis. *Reference 3* includes the Ginna responses to staff questions related to plant specific items portions of which are discussed below. These sections have been selectively updated as required to reflect cores beyond cycle 14.

The design and safety analysis of the VANTAGE + fuel assembly is discussed in WCAP-12610-P-A (*Reference 20*) which the NRC has reviewed and found acceptable. However, the staff SER of WCAP-12610-P-A requires that certain plant specific analyses be completed

prior to the implementation of the VANTAGE + fuel product in a plant. *Reference 21* covers the plant specific analyses that are required to demonstrate acceptability of the VANTAGE + fuel at Ginna Station. *Reference 21* has been reviewed by the NRC staff and has been found to be acceptable.

4.2.4.2.2 Fuel Design

Table 4.2-3 presents fuel assembly, fuel rod, and fuel pellet design information for the OFA/ VANTAGE+ and 422V+ assemblies. Table 4.2-3 includes information on materials used and dimensions.

4.2.4.2.3 Design for Seismic and Loss-of-Coolant Accident Forces

Westinghouse has performed a structural integrity evaluation for the Ginna 422V+ 9-grid implementation. A homogeneous core of 422V+ and transition cores of OFA/VANTAGE+ and 422V+ fuel were analyzed for the combined seismic and LOCA loads and it was shown that the mid-grid impact forces for 422V+ and OFA are well below crush limits and a coolable core geometry is maintained. The stress analysis indicates that adequate margins for both fuel rods and thimble tubes for the 422V+ and OFA exist, so the fragmentation of the thimble tubes and fuel rods will not occur for combined seismic and LOCA loads.

Therefore, the 422V+ Ginna design is structurally adequate for Ginna seismic/LOCA loads in the homogeneous core and transition cores with OFA fuel. The OFA still satisfies all requirements in mixed core conditions during the 14x14 422V+ Ginna fuel transition.

4.2.4.2.4 Emergency Core Cooling System (ECCS) Calculation Loss-of-Coolant Accident Cladding Models

The WCOBRA/TRAC UPI Large Break evaluation model for Ginna includes NRC supplied loss-of-coolant accident cladding models as described in NUREG 0630, burst/blockage models. Additional information regarding the models are in *Reference 17*, *Reference 18*, *Reference 19*, *Reference 21*, and *Reference 20*.

4.2.4.2.5 Initial Fuel Conditions for Transient Analysis

The initial fuel temperatures used in the Ginna transient and accident analyses were calculated using the NRC approved Westinghouse fuel performance code, PAD-4.0 (*Reference 22*). In using PAD to generate fuel temperatures for input to safety analyses calculations, a conservative thermal safety model was used. Calculations of initial fuel stored energy used in safety analyses were also based on the results of conservative fuel average temperature calculations at the time of maximum densification. As a result, fuel temperatures at the end of one cycle are significantly less than those occurring at the time of maximum densification.

4.2.4.2.6 Predicted Clad Collapse Time

The Ginna evaluation was performed using *Reference 23*. Clad flattening or creep collapse depends on several design and physical events happening within a given time period. The first of these is fuel pellet hangup, then the formation of an axial gap in the fuel stack due to fuel densification below the hangup pellet, followed by cladding creep into the axial gap,

resulting in clad flattening. Fuel pellet hangup and cladding collapse are thought to be due to a combination of pellet cocking, fuel densification, and cladding creep. Therefore, those fuel design parameters that are important to clad flattening are fuel-to-cladding gap, initial fuel rod fill gas pressure, fuel densification, pellet cocking, and cladding creep rate.

Axial gaps greater than 0.5 inches can lead to cladding collapse and significant flux and power spiking. Axial gaps less than 0.5 inches have been shown to not result in cladding collapse. Current Westinghouse fuel data demonstrates that no large axial gaps (i.e. ≥ 0.3 inches) form in current generation fuel designs during in-reactor performance.

4.2.4.2.7 Nuclear Design

Nuclear design and analysis of Ginna cores are performed using the standard Westinghouse reload safety evaluation methodology. No changes in the nuclear design methodology or models were necessary due to the transition to OFA/ VANTAGE+ or 422V+ fuel assemblies. The most important nuclear design parameter change is the positive moderator temperature coefficient, for which the maximum value of +5.0 pcm/°F is expected to occur at the beginning-of-cycle condition. In particular, conservatively positive values of the moderator temperature coefficient were assumed in the accident evaluations. In general, the neutronic parameters used as input to the safety evaluation were chosen to bound the values obtained from the transition cycles. The required shutdown margin was computed using the negative temperature coefficient corresponding to the end-of-cycle condition and assuming all but the most reactive rod has inserted into the core. The required value of the shutdown margin was found to be 1.3% $\Delta\rho$. CENG will perform whole core power distribution measurements at startup (in addition to administrative procedures) to ensure against fuel misloading. Likewise, CENG will ensure that future cycles comply with the calculated values and bounds of this analysis. Table 4.2-4 is a listing of the neutronic parameters used in the safety analysis to provide bounding values against which cycle dependent parameters may be compared.

4.2.4.2.8 Fuel Assembly Hydraulic Lift-Off

From the precision flow calorimetric in cycle 13, the value for the reactor system flow obtained was approximately 195,000 gpm. The hold-down springs of the optimized fuel assembly and the VANTAGE + fuel assembly are designed to withstand lift-off of the assembly up to a flow rate of 100,000 gpm/loop or 200,000 gpm system flow and should therefore resist lift-off. Additional conservatism has also been built into the analysis to account for uncertainties in thermal and hydraulic parameters, fuel assembly hydraulic resistance, and worst case inlet flow maldistribution factors. The spring rate for the Ginna box nozzle which was implemented in cycle 27 (region 29) is lower than the previous nozzle design. This tends to make the fuel assembly less susceptible to guide thimble bow and distortion. The maximum and minimum contact force requirements are still met with this design.

A top nozzle holddown spring force analysis demonstrated that the functional requirement for fuel assembly holddown is met for both the transition core fuel assembly designs (OFA/VANTAGE+ and 422V+) as well as for a full-core application of 422V+ under extended power uprate conditions.

4.2.4.2.9 Thermal-Hydraulic Analysis

The thermal-hydraulic analysis of the 422V+ and OFA/VANTAGE+ mixed core was performed using the Revised Thermal Design Procedures (RTDP) (*Reference 6*) and the VIPRE code (*References 7*. The WRB-1 (*Reference 9*) critical heat flux correlations was used for both fuel assemblies. The RTDP and the VIPRE code used with the critical heat flux correlation have previously been approved by the NRC. Additional components of this application are noted below.

4.2.4.2.9.1 Sensitivity Factors

For Ginna, the VIPRE code and the WRB-1 DNB correlation have been used for the calculation of sensitivity factors for both the OFA/VANTAGE+ and 422V+ fuel. All parameter values are within the ranges of the codes and correlations used, and sensitivity factors have been determined specific to the fuel type over the range of Ginna parameters. Note that the parameter uncertainties used in the calculations conservatively bound actual Ginna parameters.

4.2.4.2.9.2 WRB-1 Correlation

WRB-1 correlation was approved for the 17 x 17 optimized fuel assemblies and 17 x 17 and 15 x 15 standard LOPAR fuel assemblies with a DNBR limit of 1.17 for the R-grid.

Ginna provided information to the NRC to justify the use of the WRB-1 critical heat flux correlation for the nine-grid 14 x 14 optimized fuel assemblies. The 14 x 14 optimized fuel assembly DNB test results were provided to the NRC in *Reference 12* which contains Supplement 1 to WCAP-8762 (*Reference 13*). These test results were used to demonstrate that the WRB-1 critical heat flux correlation correctly accounted for the geometry changes from the 0.422-in. R-grid design to the 14 x 14 optimized fuel assembly design. The DNB safety analyses for Ginna have been performed with the grid spacing term in the WRB-1 correlation set equal to 22 in., the longest grid spacing in the assembly. The WRB-1 correlation has been shown to accurately predict the 0.422 R-grid critical heat flux performance with grid spacings of 13 to 32 in. (*Reference 12*). The WRB-1 correlation is applicable to the Ginna 14 x 14 optimized fuel assembly and the 14 x 14 VANTAGE + fuel assembly fuel since the range of data covers the spacing for the nine-grid design for Ginna.

Based on the comparison to the FCEP parameters of the WRB-1 database of licensed fuel assembly designs, the Ginna 422V+ fuel assembly design was concluded to be licensable to the WRB-1 critical heat flux correlation (*Reference 24*). The geometric and fluid parameters that affect the applicability of the CHF correlation for the 422V+ nine-grid fuel assembly are bounded by, or have been shown by engineering evaluation to be "similar to or bracketed by" the ranges of the licensable database for the WRB-1 CHF correlation. Therefore the WRB-1 correlation with 1.17 limit DNBR and the associated ranges is acceptable for the 422V+ nine-grid fuel assembly design. Use of the WRB-1 correlation in DNB analyses for 422V+ is discussed in Section 4.4.3.1.

4.2.4.2.9.3 Rod Bow Penalties

Rod bow can occur between mid-grids, reducing the spacing between adjacent fuel rods and reducing the margin to DNB. Rod bow must be accounted for in the DNBR safety analysis of

Condition I and Condition II events. Westinghouse has conducted tests to determine the impact of rod bow on DNB performance; the testing and subsequent analyses were documented in *Reference 14*.

Currently, the maximum rod bow penalty for the OFA fuel assembly is 1.0% DNBR at an assembly average burnup of 24,000 MWD/MTU (*Reference 14* and 25). No additional rod bow penalty is required for burnups greater than 24,000 MWD/MTU since credit is taken for the effect of $F^N_{\Delta H}$ burndown due to the decrease in fissionable isotopes and the buildup of fission products (Reference 26). Based on the testing and analyses of various fuel array designs documented in *Reference 14*, including the 14x14 STANDARD assembly, the 14x14 OFA and the 14x14 422V+ fuel assemblies should have the same rod bow penalty applied to the analysis basis as that used for 14x14 STANDARD fuel assemblies.

For the OFA/VANTAGE+ and 422V+ fuel assemblies, sufficient margin (7.5% and 10.1% respectively) exists between the safety analysis limit DNBR and the design limit DNBR to accommodate this penalty as shown below.

	Westinghouse 14 x 14 OFA/ VANTAGE+ and 422V+Fuel Assembly	
	OFA/VANTAGE+	422V+
Correlation	WRB-1	WRB-1
Correlation limit DNBR (STDP)	1.17	1.17
Design limit DNBR (RTDP)	1.24	1.24
Safety analysis limit DNBR (RTDP)	1.34	1.38
DNBR Margin (RTDP)	7.5%	<10.1%

The DNBR margin is defined as:

Safety analysis DNBR value = Design DNBR value / (1 - Margin)

4.2.4.2.9.4 DNBR Design Limits

The core thermal-hydraulic analysis was performed using 1775 MWt core power, 2250 psia system pressure, a nominal T_{AVG} of 576°F, and 177,300 gpm primary system minimum measured flow. Use of a nominal T_{AVG} of 576°F bounds operation at lower nominal values of T_{AVG}. The DNBR design limits using RTDP are shown in the table above and are valid for both typical and thimble cells. For the OFA/VANTAGE+ 422V+ fuel assemblies the WRB-1 correlation was used with a design DNBR limit of 1.24 (RTDP). The safety analysis limit (SAL) DNBR calculated was 7.5% and 10.1% above the associated design limit for OFA/ VANTAGE+ and 422V+ fuel respectively. This margin is more than enough to account for the rod bow penalty, steam generator tube plugging, and thimble plug removal.

4.2.4.3 Design Evaluation of Reconstituted Fuel Assemblies

Filler rods were originally used in fuel assemblies to replace those fuel rods damaged by the baffle jetting problem in Westinghouse reactors. This concept was extended further to replace rods during reconstitution of fuel assemblies in other locations. In order to satisfy generic fuel design criteria, the dummy rods, which are now required to be solid filler rods, require thermal-hydraulic analyses to demonstrate that inclusion of these rods in a specific fuel cycle is acceptable with respect to the overall fuel performance and safety-significant conclusions. Such an analysis will follow the methodology described in *Reference 15*. Should more than 30 rods in the core, or 10 rods in any assembly, be replaced per refueling, a report describing the number of rods replaced and associated cycle-specific evaluation shall be submitted to the Nuclear Regulatory Commission prior to criticality.

4.2.5 CORE COMPONENTS TESTS AND INSPECTIONS

Fuel assemblies are manufactured and inspected in accordance with the Vendor's Quality Assurance Program.

Since cycle 14 was the first substantial application of 14 x 14 nine-grid Westinghouse optimized fuel assembly fuel (excluding lead test assemblies), a visual surveillance was performed. This was conducted in the containment area for a reasonable number of optimized fuel assemblies until they completed their fuel cycles and were put into the spent fuel pool (SFP).

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Table 4.2-1
NUCLEAR DESIGN DATA

STRUCTURAL CHARACTERISTICS[a]

1.	Fuel weight (UO_2), lb[b]	117,682-120,481
2.	ZIRLO®/ Optimized ZIRLO™ weight, lb	28,427
3.	Core diameter, in.	101.44
4.	Core height, in.[c]	143.25
	Reflector Thickness and Composition	
5.	Side-water plus steel (not counting baffle)	
	Corner	25.25 in.
	Flat	15.22 in.
6.	Number of fuel assemblies	121
7.	UO_2 rods per assembly	179

PERFORMANCE CHARACTERISTICS

8.	Heat output, MWt (as initially licensed)	1300
9.	Heat output, MWt (Previous Reactor Power Rating)	1520
10.	Heat output, MWt (current reactor and maximum calculated turbine rating)	1775
11.	Typical fuel burnup at BOL, MWd/MTU	20,400
12.	Typical average enrichment, wt % [a]	4.80
13.	Heat flux hot-channel factor (max), F_Q	2.60
14.	Nuclear enthalpy rise hot-channel factor (max), $F^N_{\Delta H}$	1.72 (422V + Ginna fuel 1.60 (OFA/VANTAGE+ fuel)

CONTROL CHARACTERISTICS

Rod Cluster Control Assemblies

15.	Material	5% cadmium; 15% indium; 80% silver
16.	Full-length, rod cluster control assemblies, number	29
17.	Number of absorber rods per rod cluster control assembly	16

a. For full core of Westinghouse 422V+ Ginna Fuel.

b. Fuel weight may vary depending on number of fuel assemblies with annular pellets in the blanket region and/or fabrication tolerances.

c. For full core of Westinghouse 422V+ Ginna fuel. The core height for full core of Westinghouse OFA/ VANTAGE+ is 414.4 inches.

Table 4.2-2
CORE MECHANICAL DESIGN PARAMETERS*

	OFA/VANTAGE+	422V+
ACTIVE PORTION OF THE CORE		
Equivalent diameter, in	101.44	
Active fuel height, in.	141.4	143.25
Length-to-diameter ratio	1.46	
Total cross section area, ft^2	50.6	
FUEL ASSEMBLIES		
Number	121	121
Rod array	14 x 14	14x14
Rods per assembly[a]	179	179
Rod pitch, in	0.556	0.556
Fuel weight (as UO_2), lb	103,748 - 105,996[b]	117,682-120,481
Number of grids per assembly	9	9
Number of guide thimbles	16	16
Diameter of guide thimbles (upper part), in	0.492 I.D. x 0.526 O.D.	0.492 I.D. x 0.526 O.D.
Diameter of guide thimbles (lower part), in	0.4465 I.D. x 0.4815 O.D.	0.492 I.D. x 0.526 O.D.
Diameter of dashpot	NA	0.4465 I.D. x 0.480 O.D.
FUEL RODS		
Number	21,659	21,659
Outside diameter, in.	0.400	0.422
Diametral gap, in	0.0070	0.0075
Clad thickness, in.	0.0243	0.0243
Clad material	Zircaloy-4/ZIRLO®	ZIRLO®/ Optimized ZIRLO™
Overall length, in	149.162	152.763

	OFA/VANTAGE+	422V+
FUEL PELLETS		
Material	UO_2 sintered	UO_2 sintered
Density (% of theoretical)	95	95
Fuel enrichments wt %, (typical)		
Axial blanket region	Natural or slightly enriched uranium, solid or annular	Slightly enriched uranium, annular
Enriched region	2.60, 3.30, 3.40, 3.60, 3.80, 3.90, 4.00, 4.2, 4.3, 4.6, 4.8	2.60, 3.30, 3.40, 3.60, 3.80, 3.90, 4.00, 4.2, 4.6, 4.8, 4.95
Diameter, in.	0.3444	0.3659
Length, in.		
Axial blanket region (natural uranium)	0.500	NA
Axial blanket region (slightly enriched)	0.500	NA
Enriched region	0.413	0.439
Annular blanket region (natural or slightly enriched)	0.500	0.500
ROD CLUSTER CONTROL ASSEMBLIES		
Neutron absorber	5% cadmium, 15% indium, 80% silver	5% cadmium, 15% indium, 80% silver
Cladding material	Type 304 SS - cold worked	Type 304 SS - cold worked
Clad thickness, in.	0.019	0.019
Number of clusters, full length	29	29
Number of control rods per cluster	16	16
Length of rod control	156.639 in. overall	156.639 in. overall
	148.759 in. insertion length[c]	148.759 in. insertion length
Length of absorber section	142.01 in. (full length)	142.01 in. (full length)

	OFA/VANTAGE+	422V+
CORE STRUCTURE		
Core barrel, in.		
I.D.	109.0	109.0
O.D.	112.5	112.5
Thermal shield, in.		
I.D.	115.3	115.3
O.D.	121.8	121.8

* All dimensions are for Westinghouse Optimized, VANTAGE + and 422V+ fuel assembly cold conditions.

a. Sixteen positions are occupied by guide thimbles to provide passage for control rods and one position contains an instrument thimble for in-core instrumentation.

b. Core fuel weight may vary depending on number of assemblies with annular blankets and/or fabrication tolerances.

c. From top of adaptor plate to bottom of end plug

Table 4.2-3
FUEL DESIGN

Basis	OFA/VANTAGE+	422V+
Fuel assemblies		
Number of fuel assemblies	121	121
UO_2 rods per assembly	179	179
Rod pitch, in.	0.556	0.556
Assembly pitch, in.	7.803	7.803
Number of grids per assembly	9	9
Material	7- Zircaloy 2- Inconel	7- ZIRLO® 2- Inconel
Guide tube material	Zircaloy/ZIRLO®	ZIRLO®
Fuel rods		
Number	21,659	21,659
Clad O.D., in	0.400	0.422
Diametral gap, in.	0.0070	0.0075
Clad thickness, in.	0.0243	0.0243
Clad material	Zircaloy/ZIRLO®	ZIRLO®/ Optimized ZIRLO™
Fuel pellets		
Material	UO_2	UO_2
Density, % theoretical	95	95
Diameter, in.	0.3444	0.3659
Length		
Axial blanket region (natural uranium)	0.500	NA
Axial blanket region (slightly enriched)	0.413	NA
Enriched region	0.313	0.439
Annular blanket region (natural or slightly enriched)	0.500	0.500

Table 4.2-4
KINETIC PARAMETERS USED IN TRANSIENT ANALYSIS (WESTINGHOUSE OFA/ VANTAGE+ AND 422V+ GINNA FUEL ASSEMBLY 14 x 14 FUEL)

Parameter	**Bounding Value**
Most positive moderator temperature coefficient, pcm/°F	$\leq$+5.0 (for power < 70%) $\leq$0.0 (for power $\geq$ 70%)
Most positive moderator density coefficient, Δk/gm/cc	$\leq$0.45
Doppler temperature coefficient, pcm/°F fuel	-0.91 to -2.90
Zero Power Doppler - only power coefficient, pcm/% power	-12.0 + 0.045Q to -24.0 + 0.100Q
B_{EFF} (fraction)	0.0043 to 0.0072
Normal operation $F_{\Delta H}$ (with uncertainties)	$\leq$ 1.72 (422V+ Ginna) $\leq$ 1.60 (VANTAGE+)
Maximum total peaking factor, F_Q	$\leq$ 2.60

4.3 RELOAD CORE NUCLEAR DESIGN

This section describes the nuclear design and evaluation of reload cores. The design bases for the nuclear design of the fuel and reactivity control system are described in Section 4.2.1. The design objectives and bases are reviewed and each of the design and evaluation phases of a reload core is discussed. 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. Relevant design procedures and methods are briefly described and design codes are referenced where appropriate.

The objective of the nuclear design process is to determine the number and enrichment of the feed assemblies and a preliminary loading pattern that meets the required energy output of the refueled core as defined in the design initialization. Constraints from the design initialization specify the approximate MODE 6 (Refueling) dates, the burnup window of the previous cycle, and sometimes an upper and/or lower bound on the number of feed assemblies (or alternatively on the feed enrichment).

Once the loading pattern is set, the nuclear evaluation phase begins. The primary objective of this phase is to determine whether all nuclear-related key safety parameters are within the bounding values used in the reference analysis. These parameters are used in the safety evaluation.

4.3.1 PRELIMINARY DESIGN PHASE

The detail and scope of the preliminary design process depends to a large extent on how similar the refueled core is to previous reload cores. When it differs significantly from previous reloads, detailed calculations are used, as outlined later in this section. When the reload is very similar to ones already designed, simpler calculational models may be used. These simpler calculational models are benchmarked to the more detailed models.

When a preliminary loading pattern that meets the required energy output is established, an evaluation is performed to ensure that the following criteria are satisfied.

1. The $F_{\Delta H}$ values with all-rods-out and D-bank-in to the insertion limit are below specified limits, with allowance for variation in the actual burnup of the previous cycle.
2. The moderator temperature coefficient satisfies Technical Specification requirements.
3. Sufficient rod worth is available to meet the N-1 rods shutdown margin criteria at all times.

During the preliminary design phase, operating history is used as much as possible and where this is not available, the best prediction of the operating history is used. Some of the parameters that comprise the operating history are power level, control rod position, average coolant temperature, and other parameters that may affect the nuclear models. Operating history is used to ensure that the nuclear model of the core represents the actual condition of the core.

With the completion of the preliminary design phase, the preliminary loading pattern including the number and enrichment of feed assemblies and the number of burnable absorber rods

if any, is fixed. Also, the three criteria specified above are met. The remaining effort consists of determining the nuclear related key safety parameters.

4.3.2 DETERMINATION OF NUCLEAR-RELATED KEY SAFETY PARAMETERS

A reload core can affect nuclear-related key safety parameters in three basic areas: core kinetic characteristics, control rod worths, and core power distributions. Key safety parameters can be determined by a comparison of the current reload core characteristics with the characteristics of previously analyzed reload cores, scoping studies that typically utilize efficient spatially dependent nuclear calculations, or explicit calculations using detailed techniques and models.

Each of the above methods is used in varying degrees for any particular reload evaluation. For example, if a reload core is identical to a previous reload (where plant operating parameters, fuel enrichment, cycle burnup, fuel arrangement, control rod pattern, etc., remain the same), a simple comparison would demonstrate that the previously evaluated parameters are applicable and that additional calculations are not required. This example, of course, is an ideal situation. Conversely, a reload core may possess characteristics unlike any previously evaluated core. For this example, comprehensive scoping calculations and explicit worst-case condition calculations would be required to evaluate limiting safety analysis parameters.

Most reload cores cannot be categorized by the above two examples. That is, reload cores possess varying degrees of similarity with previously evaluated reload cores and the evaluation methods recognize this fact.

The following discussion describes the methods for determining the nuclear related key safety parameters for the reload core. Three areas are addressed: control rod worth parameters, core reactivity parameters and coefficients, and other nuclear-related key safety parameters for specific events. Nuclear-related key safety parameters are identified and, where appropriate, a description of core conditions that are assumed in the evaluation of these parameters is discussed.

4.3.2.1 Reactivity Control Aspects

Reactivity control is provided by (1) a soluble chemical neutron absorber in the reactor coolant (boric acid, also called chemical shim), and (2) movable neutron absorbing control rods.

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 change in temperature of the reactor coolant from MODE 5 (Cold Shutdown) to the hot operating, zero power conditions, (2) changes in reactivity associated with changes in the fission product poisons, xenon and samarium, (3) reactivity losses associated with the depletion of fissile inventory and buildup of long-lived fission product poisons (other than xenon and samarium), and (4) changes in reactivity due to burnable absorber burnout.

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, and (4) reactivity changes associated with the power coefficient of reactivity.

The control 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 coolant temperature, power level, boron concentration, or xenon concentration, 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 control rods is specified to provide adequate shutdown at all operating and hot zero-power conditions with the most reactive rod stuck out of the core. The distribution of the various control group rods and shutdown rods within the core is shown in Figure 4.3-1.

A reload core can typically alter individual rod cluster control assembly worths and control and shutdown bank worths. These changes can be attributed to changes in the neutron flux distribution (and thus, reactivity importance) that are produced by the loading pattern of burned and fresh fuel assemblies and the fuel depletion which occurs during the reload fuel cycle. Changes in control rod worths may also affect rod insertion limits, trip reactivity, differential rod worths, and shutdown rod worth.

Prior to the evaluation of limiting control rod worth parameters, an initial evaluation of limiting control rod worth parameters is performed by rod worth calculations obtained using two-group three-dimensional models. These calculations are performed for the beginning and end of the reload fuel cycle at full and zero power conditions. The total worth of all the shutdown banks is also calculated at zero power conditions. In addition, the impact of the previous cycle burnup (burnup window) on the rod worth calculation is also evaluated for completeness. These calculations form the basis for the evaluation of the limiting control rod worth parameters.

4.3.2.1.1 Insertion Limits

Control rod insertion limits define the deepest individual control bank insertion that can be allowed, as a function of the reactor power level. One of the purposes for these limits is to physically restrict the value of the inserted integral rod worth in the core at any power level. This will ensure that the minimum shutdown margin requirement can be satisfied regardless of the core configuration during MODES 1 and 2. It should be recognized, however, that control rod insertion limits are not defined by reactivity constraints alone. The final determination of control rod insertion limits is dependent on peaking factor constraints that must be satisfied during MODES 1 and 2 and during certain accident conditions.

Insertion limits are calculated using two-group, one-dimensional axial models. The core is depleted using a three-dimensional model. The three-dimensional model is collapsed into an equivalent one-dimensional axial model. The calculations are performed at the beginning and end of the reload cycle. Subsequently, the axial model is used to compute power levels for various rod positions (with normal bank overlap) that would represent a pre-defined value of inserted integral rod worth (commonly referred to as the rod insertion allowance). Rod insertion limits are conservatively constructed by limiting the amount of rod insertion at any power level to a value that is less than the calculated amount.

4.3.2.1.2 Total Rod Worth

The total integral rod worth is evaluated by assuming that all the control and shutdown banks are inserted and that the most reactive individual rod cluster control assembly is fully withdrawn from the core. Calculations are performed at the beginning and end of the reload fuel cycle at hot zero power conditions. Two-group three-dimensional calculations are used to determine the worth of the most reactive stuck rod. Individual rods are withdrawn from an all-rods-in condition until the most reactive rod is identified. The stuck rod worth is subtracted from the total worth of all control and shutdown banks and the resultant quantity (called the N-1 rod worth) is further reduced for conservatism. This evaluation of the minimum N-1 rod worth is used to determine the shutdown margin that is available at both the beginning and end of the reload fuel cycle.

4.3.2.1.3 Trip Reactivity

The minimum trip reactivity at or near full power conditions and the trip reactivity shape (i.e., the inserted rod worth versus rod position) are control rod worth parameters evaluated for each reload core. The minimum trip reactivity is evaluated at the beginning and end of the reload fuel cycle to ensure that the previously established limit is valid for power levels near full power and for the entire cycle length.

The most limiting trip reactivity shape (accounting for the worst axial power distribution) is evaluated each reload fuel cycle to determine the minimum inserted rod worth versus rod position that would be produced by N-1 control rods entering the core at full power. This evaluation is performed with two-group one-dimensional axial calculations. The axial model is established by collapsing the three-dimensional model into an equivalent one-dimensional axial model. It is assumed that the control rods can be inserted as deep as the full power insertion limit and that the power distribution is within Technical Specifications limits. Using the most limiting axial power shape, a single shutdown bank, equal in worth to the minimum trip reactivity, is inserted into the core in a stepwise fashion. The results of these calculations are used to evaluate the minimum inserted rod worth versus rod position.

4.3.2.1.4 Differential Rod Worths

Maximum differential rod worths at full power and zero power conditions are evaluated for each reload core. These evaluations are performed at the beginning and end of the fuel cycle. Two-group, one-dimensional axial calculations are used to determine maximum differential rod worths.

The differential rod worths are obtained using the equivalent axial model, which has been obtained by collapsing the three-dimensional model with control bank cross sections that yield the total worth determined by the three-dimensional analyses for the bank fully inserted. Full power calculations are performed to determine the maximum differential worth of any control bank that could be moving during power operation. The control banks are assumed to move in normal sequence with programmed control bank overlap. At zero power conditions, the maximum differential rod worth of any two sequential control banks is determined by assuming that the banks are moving with 100% overlap. That is, both control banks are withdrawn simultaneously as in a postulated startup accident from a subcritical condition.

4.3.2.1.5 Summary

The control rod worth parameters are evaluated each reload fuel cycle. These key safety parameters are then factored into the reload safety evaluation.

4.3.2.2 Core Reactivity Parameters and Coefficients

The kinetic characteristics of the reactor core determine the response of the core to changing plant conditions or to operator adjustments made during MODES 1 and 2, as well as the core response during abnormal or accidental transients. These kinetic characteristics are quantified in terms of reactivity coefficients. The reactivity coefficients reflect the changes in the neutron multiplication due to varying plant conditions such as changes in power, moderator, or fuel temperatures. Since reactivity coefficients change during the life of the core, ranges of coefficients are employed in transient analysis to determine the response of the plant throughout life.

Reactivity coefficients are calculated on a core-wide basis using three-dimensional two-group calculations. For some accidents, power distributions during the transient do not change significantly from those occurring during normal operating conditions, ensuring negligible changes in the values of reactivity coefficients. However, for accidents leading to significant power distribution changes from those occurring during normal operating conditions (e.g., worst stuck rod configuration), reactivity coefficients are determined using the power distribution occurring during the accident. The exact values of the reactivity coefficient used in the safety analysis depend on whether the transient of interest is examined at beginning-of-life or end-of-life, whether the most negative or the most positive (least negative) coefficients produce conservative results, and whether spatial non-uniformity must be considered in the analysis. Conservative values of reactivity coefficients, considering various aspects of analysis, are used in the transient analysis. Table 4.2-4 illustrates the reactivity parameters and coefficients and the limiting values which are evaluated for each reload core.

Reactivity parameters and coefficients are evaluated by considering the following conditions.

A. Beginning, middle, and end of the reload fuel cycle.

B. Full power, part power, and zero power operation.

C. Rodded core configurations allowed by the Technical Specifications during power operation.

In addition to the above conditions, consideration is also given to the impact that the previous cycle burnup has on core reactivity parameters and coefficients. The evaluated reactivity parameters and coefficients are discussed below.

4.3.2.2.1 Moderator Temperature Coefficient

The moderator temperature (density) coefficient is defined as the change in reactivity per degree change in moderator temperature (density). The value of this coefficient is sensitive to changes in the moderator density, the moderator temperature (keeping the density constant), the soluble boron concentration, the fuel burnup, and the presence of control rods and/or burnable absorbers which reduce the required soluble boron concentration and increase the

"leakage" of the core. The moderator coefficient is calculated for the various plant conditions discussed above by performing two-group three-dimensional neutronic calculations, varying the moderator temperature (and density) by several degrees about each of the mean temperatures of interest.

4.3.2.2.2 Fuel Temperature Coefficient

The fuel temperature (doppler) coefficient is defined as the change in reactivity per degree change in effective fuel temperature. It is primarily a measure of the doppler broadening of Uranium-238 and Plutonium-240 resonance absorption peaks. The fuel temperature coefficient is calculated by performing two-group three-dimensional calculations. Moderator temperature is held constant and power level is varied. The spatial variation of fuel temperature is taken into account by calculating the effective fuel temperature as a function of local power density throughout the core. The doppler only contribution to the power coefficient is derived from the same calculations and is defined as the change in reactivity per percent change in power.

4.3.2.2.3 Boron Worth

The boron worth is defined as the change in reactivity per ppm change in the boron concentration. The value of this parameter depends on the boron concentration, on the moderator temperature (density), and on the presence of control rods and/or burnable absorbers. It is calculated for the various plant conditions discussed above by performing two-group neutronic calculations, varying the boron concentration about the reference values of interest.

4.3.2.2.4 Delayed Neutrons

Delayed neutrons play an important role in determining the dynamic response of the core. The delayed neutrons are emitted from fission products, called precursors, a short time after a fission event. The delayed neutron fraction in each of the precursor groups is, in general, different for different fissionable isotopes. The effective delayed neutron fraction for the entire core is obtained by weighting the delayed neutron fraction for different isotopes and precursor groups by the region-wise fraction of fissions in each isotope and the region-wise power sharing in the core. Region-wise power sharings for various core conditions described earlier are obtained from two-group three-dimensional neutronic calculations. The fraction of fissions in each isotope is obtained from region-wise macroscopic few-group cross-section calculations.

4.3.2.2.5 Prompt Neutron Lifetime

The prompt neutron lifetime value is obtained in a manner similar to the calculation of the effective delayed neutron fraction. Values of the prompt neutron lifetime are obtained from region-wise few-group cross-section calculations. These values are weighted by region-wise power sharings taken from two-group three-dimensional neutronic calculations for various core conditions to determine the core average prompt neutron lifetime.

4.3.2.2.6 Summary

Core reactivity parameters and coefficients evaluated in reload cores depend on the previous cycle burnup, the number and enrichment of fresh fuel assemblies, the loading pattern of burned and fresh fuel, the number and location of any burnable absorbers, etc. These coefficients and parameters do, however, exhibit predictable trends which are dependent on such core average parameters as burnup, boron concentration, moderator and fuel temperatures, and power level. As a result of these trends and past reload evaluation experience, reactivity parameters and coefficients can be evaluated using differing degrees of sophistication.

4.3.2.3 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, the heat flux hot-channel factor, F_Q, was established as specified in Table 4.2-1. For the hottest channel the nuclear enthalpy rise hot-channel factors, $F^N_{\Delta H}$, was established as specified in Table 4.2-1.

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

- The linear power density.
- The fuel cladding integrity.
- 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.

The presence of control rods, burnable absorbers, 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 ensure that the thermal design criteria are met.

4.3.3 EVALUATION OF RELOADS WITH OFA/VANTAGE+ AND 422V+ FUEL ASSEMBLIES

The key safety parameters evaluated for the transition to 422V+ fuel at extended power uprate conditions show that the expected ranges of variation for many of the parameters will lie within the normal cycle-to-cycle variations. The parameters which fall outside of these

ranges are those which are sensitive to fuel type, e.g., the moderator temperature coefficient. The accident evaluations, documented in Chapter 15, have considered ranges of parameters which are appropriate for the transition cycles and beyond.

The Advanced Nodal Code (ANC) (*Reference 1*) was implemented in the reload design analysis beginning with cycle 19. ANC is an advanced nodal analysis theory code capable of two- or three-dimensional calculations. Beginning with cycle 22, PHOENIX-P (*Reference 2*) computer code was implemented in the reload design analysis. PHOENIX-P is a two-dimensional transport theory based code that calculates lattice physics constants. These models supplement the "Standard Reload Safety Evaluation Methodology" (*Reference 3*). These are the same methods and models that have been used in other Westinghouse reload cycle designs.

A number of changes to the Technical Specifications were approved as part of the transition to 422V+ fuel assemblies and extended power uprate. These changes include (1) a reduction in the required shutdown margin to 1.3% Δp, (2) a reduction in the $F^N_{\Delta H}$ limit to 1.72 (422V+) and 1.60 (OFA/VANTAGE+), and (3) an increase in $F^N_Q(z)$ to 2.60.

Power distributions and peaking factors are primarily loading-pattern dependent. The usual methods, such as enrichment variation can be employed to ensure compliance with the peaking factor Technical Specifications.

4.3.4 TESTS FOR REACTIVITY ANOMALIES

Tests for reactivity anomalies or design errors are obtained during the reload startup tests. Review acceptance criteria are applied to the comparison of measured and predicted results at startup to identify reactivity anomalies.

Monitoring for reactivity anomalies over depletion of the fuel is accomplished by obtaining a measurement of the boron concentration, correcting the measurement to a set of reference plant operating conditions, and plotting the results versus fuel burnup. A reactivity anomaly can be identified by departure of the corrected measured boron concentration from the predicted boron value or path.

REFERENCES FOR SECTION 4.3

1. Y. S. Liu, et al., ANC: A Westinghouse Advanced Nodal Computer Code, WCAP-10966-NP-A (Non-Proprietary), September 1986.

2. T. Q. Nguyen, et al., Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores, WCAP-11597-A (Non-Proprietary), June 1988.

3. S. L. Davidson and W. R. Kramer (Ed.), Westinghouse Reload Safety Evaluation Methodology, WCAP-9273-A (Non-Proprietary), July 1985.

4.4 THERMAL AND HYDRAULIC DESIGN

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.

4.4.1 DESIGN BASIS

The design basis for the thermal and hydraulic design of the reactor is presented in Section 4.2.1.

4.4.2 DESCRIPTION AND EVALUATION OF THE THERMAL-HYDRAULIC DESIGN AND ANALYSIS OF RELOAD CORES

This section describes the thermal and hydraulic analysis of reload cores. The design objectives and bases are reviewed, and each of the design and evaluation phases of a reload core is discussed. Relevant design procedures and methods are briefly described and design codes are referenced where appropriate. Constraints from the design initialization specify the hydraulic conditions to be considered for the reload core analysis. The safety-related design bases for the thermal and hydraulic analysis are as specified in Section 4.2.

4.4.2.1 Hydraulic Evaluation

The hydraulic evaluation of the reload core requires a review of the fuel assembly design (nozzles, grids, fuel rods, etc.) that is to be inserted into the core. This design is compared with the design of the fuel assemblies that remain in the core. This comparison is made to ensure that the new fuel assemblies are hydraulically compatible with the fuel assemblies remaining in the core. In general the reload fuel assembly design is identical to the previous fuel assembly design. The best estimate flow rate and mechanical design flow rate are considered in evaluating the core pressure drop and fuel assembly hydraulic loads respectively. This evaluation is performed to verify the conservatism of the core pressure drop and the hydraulic loads upon which the fuel assembly holddown springs are designed.

4.4.2.2 Thermal and Hydraulic Key Safety Parameters

A list of thermal-hydraulic key safety parameters is given in Table 4.4-1. The core power, system pressure, inlet temperature, thermal design flow rate, and core bypass flow are defined during the design initialization phase of the reload design effort. The design radial power distribution for steady-state operation (a peak-to-average $F^N_{\Delta H}$ of 1.72 including measurement uncertainties) and design axial power shape are also defined during the initialization phase.

The values of these parameters are usually identical to the previous cycle design. The departure from nucleate boiling (DNB) correlation to be used in the DNB analyses is also defined during this initialization phase. Fuel density and sintering temperature are important to assess the effects of fuel densification. Changes in the above parameters are evaluated in the determination of the key safety parameters.

4.4.2.2.1 Engineering Hot-Channel Factors

Engineering hot-channel factors account for the influence of the variations of fuel pellet diameter, density, and enrichment. The heat flux engineering hot-channel factor, F^E_Q, which is applied in determining the peak kW/ft and in fuel pellet temperature evaluations, has been conservatively determined and generally will not vary for the reload case. F^E_Q does not need to be considered in DNB evaluation as stated in *Reference 1*. The enthalpy rise engineering hot-channel factor, $F^E_{\Delta H}$, is directly considered in RTDP by the convolution which sets the design limit DNBR.

4.4.2.2.2 Axial Fuel Stack Shrinkage

Axial fuel stack shrinkage due to fuel densification increases the linear kW/ft used for the fuel temperature calculations and heat flux used in DNB evaluations. The stack height factor is a multiplier on the linear kW/ft and heat flux which accounts for the fuel stack shrinkage. An acceptable model for determining the fuel stack shrinkage is given in *Reference 1*.

4.4.2.2.3 Fuel Temperatures

Fuel temperatures for safety analyses are computed for each first-core design. A summary of the computed quantities is given below:

- Fuel centerline temperature versus kW/ft.
- Fuel average temperature versus kW/ft.
- Fuel surface temperature versus kW/ft.

Temperatures are computed with the PAD 4.0 code (*26*).

Fuel parameters for reload fuel are evaluated to determine if the temperatures that were computed for the reference analysis are applicable to the current reload. Major fuel parameters of interest are pellet density, pellet sintering temperature, helium backfill pressure, and fuel pellet and rod dimensions. The dimensions are generally the same as the prior fuel design. If the reference analysis temperatures are not applicable, a new fuel temperature analysis is performed. If the reference analysis continues to apply, no further evaluation is required.

4.4.2.2.4 Rod Internal Pressure

The rod internal gas pressure of the lead rod in a reactor is limited to a value below that which could cause (1) the diametral gap to increase due to the outward cladding creep during steady-state operation and (2) extensive DNB propagation to occur. This precludes the outward clad creep rate from exceeding the fuel solid swelling rate and, therefore, ensures that the fuel-clad diametral gap will not reopen following contact or increase in size during steady-state operation. Restricting the fuel-clad gap from opening will prevent accelerated fission gas release at high burnup and preclude high burnup fuel from becoming limiting from a loss-of-coolant accident standpoint.

Fuel rod internal pressure is important in evaluating the possibility of clad flattening in pile (*Reference 6*) as well as assessing the degree of burst and blockage which may occur after a loss-of-coolant accident.

Pressures are computed with the PAD 4.0 code (*26*).

Fuel parameters for reload fuel are evaluated to determine if the pressures that were computed for the reference analysis are applicable to the current reload. Major fuel parameters of interest are pellet density, pellet sintering temperature, helium backfill pressure, power history, and fuel pellet and rod dimensions. The dimensions are generally the same as the prior fuel design. If first core or previous reload pressures are not applicable, a new fuel pressure analysis is performed. If the reference analysis continues to apply, no further evaluation is required.

4.4.2.2.5 Core Thermal Limits

Core thermal limits represent the locus of points of core thermal power, primary system pressure, and coolant inlet temperature which ensure that the DNB design basis (see Section 4.2) is satisfied. The design radial power distribution utilized is characterized by the enthalpy rise hot-channel factor $F^N_{\Delta H}$, which increases with decreasing power level. A typical value of

$F^N_{\Delta H}$ versus power is:

$$F^N_{\Delta H} = 1.72\,[1 + 0.3(1 - P)]$$

The design axial power distribution is a 1.75 chopped cosine. This power distribution is used in the VIPRE analysis to determine the core thermal limits. *Reference 7* further describes these criteria, assumptions, and methods.

The method for determining core thermal limits considers the variations in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and code uncertainties, coupled with the correlation uncertainties, to statistically obtain a DNB ratio (DNBR) uncertainty factor. Applying this factor leads to a limit DNBR value to be used for determining core thermal limits. These core thermal limits are used in accident analysis in conjunction with input parameters at their nominal or best estimate values. This method is described in detail in *Reference 23*.

The parameters listed in Table 4.4-1 are important in analyzing the core thermal limits. These parameters of the reload core are compared to those used in the reference analysis of the core thermal limits. A change in a parameter which results in the previous thermal limits being conservative (e.g., a decrease in the radial peaking factor, $F^N_{\Delta H}$) would not require a reevaluation. A change in a parameter which results in the previous thermal limits being nonconservative (e.g., an increase in the radial peaking factor, $F^N_{\Delta H}$) requires a reevaluation. For the reload core design which utilized the statistical method in the previous cycle, the variations in these parameters are reviewed to ensure that the DNBR uncertainty factor and limit DNBR remains applicable. If the variations in these parameters are changed or a new DNB correlation is used, the DNBR uncertainty factor, limit DNBR, and core thermal limits would be reevaluated. The reevaluation may be of two types: a simple quantitative evaluation performed to assess the effect of a change in a parameter on the core thermal limits, or a full reanalysis.

4.4.2.2.6 Key Safety Parameters for Specific Events

This section describes the DNB analysis for specific events. The methods used in reload evaluation of DNB are the same as those discussed in the previous section.

The DNB analysis of the loss of flow accident considers the plant parameters listed in Table 4.4-1, the heat flux and flow variations with time, and the design steady-state radial and axial power distribution in the VIPRE transient analysis to ensure that the DNB design basis is met.

An important parameter in the evaluation of the rod cluster control assemblies misalignment accidents is the radial power distribution. This radial power distribution (characterized by the enthalpy rise hot-channel factor, $F^N_{\Delta H}$) and the design axial power shape, as well as the other plant parameters listed in Table 4.4-1, are considered in the VIPRE analysis.

A DNB analysis of the single rod withdrawal at power accident is performed to determine the number of rods in DNB, as appropriate. The plant parameters listed in Table 4.4-1 which include the design steady-state radial and axial power distribution are considered in the VIPRE analysis. To determine the number of rods in DNB, the radial peaking factor $F^N_{\Delta H}$ is determined which satisfies the following conditions:

A. Minimum allowable DNBR is met.

B. Hot-channel exit quality is within the range of the DNB correlation.

A fuel census curve is then used to determine the percent of rods with powers greater than this $F^N_{\Delta H}$ and thus assumed to be in DNB.

In the DNB analysis of the hypothetical steam line break, the transient state points (combinations of reactor coolant pressure, inlet temperature, flow rate, and core power level) along with the radial power distribution, $F^N_{\Delta H}$ and axial power profile are included in the VIPRE analysis to ensure that the DNB design basis is met.

4.4.2.3 VIPRE Code

VIPRE is a subchannel code which has been developed to account for hydraulic and nuclear effects on the enthalpy rise in the core. The behavior of the hot assembly is determined by superimposing the power distribution among the assemblies upon the inlet flow distribution while allowing for flow mixing and flow distribution between assemblies. The average flow and enthalpy in the hottest assembly is obtained from the core-wide, assembly-by-assembly analysis. The local variations in power, fuel rod and pellet fabrication, and mixing (engineering hot-channel factors) within the hottest assembly are then superimposed on the average conditions of the hottest assembly in order to determine the conditions in the hot channel.

Further descriptions of this code and its applications are given in *Reference 9*.

4.4.2.3.1 **Steady-State Analysis**

The VIPRE computer program determines the coolant density, mass velocity, enthalpy, vapor void, static pressure, and DNBR distribution along parallel flow channels within a reactor core under all expected operating conditions. The core region being studied is considered to be made up of a number of contiguous elements extending the full length of the core. An element may represent any region of the core from several assemblies to a subchannel.

4.4.2.3.2 **Transient Analysis**

The VIPRE code is also used for transient DNB analysis (e.g., loss of flow and locked rotor transients).

The conservation equations needed for the transient analysis are included in VIPRE by including the necessary accumulation terms to the conservation equations used in the steady-state analyses. The input includes one or more of the following time dependent arrays: (1) inlet flow variation, (2) heat flux distribution, (3) system pressure history, and (4) inlet temperature variation.

4.4.3 THERMAL-HYDRAULIC METHODOLOGY FOR OFA/VANTAGE+ AND 422V+ FUEL ASSEMBLY DESIGN EVALUATION

4.4.3.1 General

The calculational methods used in the analysis of the OFA/VANTAGE+ and 422V+are (1) the VIPRE computer code, (2) the WRB-1 DNB correlation and (3) the revised thermal design procedure. In addition, the PAD 4.0 thermal model (*Reference 26*) is used to generate fuel temperatures for safety analysis.

The VIPRE code is used to perform steady-state thermal-hydraulic calculations. VIPRE calculates coolant density, mass velocity, enthalpy, void fractions, static pressure, and DNBR distributions along flow channels within a reactor core under all expected operating conditions. VIPRE is described in detail in *Reference 9*.

The WRB-1 DNB correlation (*Reference 13*) provides a significant improvement in critical heat flux predictions over previous DNB correlations.

The 17 x 17 optimized fuel assembly DNB tests showed that the WRB-1 correlation correctly accounted for the geometry changes in going from the 17 x 17 0.374-in. rod O.D. design to the 17 x 17 0.360-in. rod O.D. design, and that the correlation limit of 1.17 was still applicable (*Reference 14*). The 14 x 14 optimized fuel assembly design involved very similar geometry changes from the seven-grid 14 x 14 standard fuel design, namely, the reduction of the rod O.D. from 0.422-in. to 0.400-in. and the incorporation of a grid design with an increased height and strap thickness due to the change from Inconel to zircaloy. Confirmatory DNB tests performed on the 14 x 14 optimized fuel assembly typical cell geometry verified that the WRB-1 correlation accurately predicted critical heat flux values for this geometry type and that the correlation limit of 1.17 was still appropriate.

The WRB-1 correlation with a 95/95 correlation limit of 1.17 was also used in the DNB analyses for the Ginna 14 x 14 422V+ fuel. The use of the WRB-1 DNB correlation for this fuel design is based on the change notification which introduced the 14 x 14 422V+ mid-grid design (*Reference 27*). The basic change was reverting back to the larger O.D. fuel rod used with the original 14 x14 STANDARD fuel but with a new low pressure drop mid-grid design. The applicability of WRB-1 to this mid-grid was justified under the Westinghouse FCEP process (*Reference 28*).

The W-3 DNBR correlation (*References 15* and *16*) was 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. The W-3 correlation, which does not take credit for mixing vane grids, is used to calculate DNBR values in the heated region below 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 to 1000 psia, the W-3 correlation limit is 1.45 (*Reference 25*). For system pressures greater than 1000 psia, the W-3 correlation limit is 1.30.

The design method employed to meet the DNB design basis is the revised thermal design procedure (*Reference 23*). Uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, and computer codes are considered statistically. To this convolution of plant system and performance uncertainties is added the uncertainties of the correlation itself as defined from the test basis. These two major components of calculational uncertainty are combined to define an overall DNBR limit such that there is at least a 95% probability that the minimum DNBR of the limiting rod will be greater than or equal to this value to satisfy the DNB design criterion. This DNBR uncertainty establishes a design limit DNBR value that must be met in plant safety analyses. Since the parameter uncertainties are considered in determining the design limit DNBR value, the plant safety analyses are performed using values of input parameters without uncertainties. In addition, margin is allocated to the design limit DNBR values to set values designated as the safety analysis limit DNBR values. The plant allowance available between the safety analysis limit DNBR values and the design limit DNBR values will be used to offset DNBR penalties and for flexibility in the design and operation of this plant.

The table below indicates the relationship between the correlation limit DNBR, design limit DNBR, and the safety analysis limit DNBR values used for the Ginna fuel designs.

	Westinghouse 14 x 14 OFA/ VANTAGE+ and 422V+ Fuel Assembly	
	OFA/VANTAGE+	422V+
Correlation	WRB-1	WRB-1
Correlation limit DNBR (STDP)	1.17	1.17
Design limit DNBR (RTDP)	1.24	1.24

	Westinghouse 14 x 14 OFA/ VANTAGE+ and 422V+ Fuel Assembly	
Safety analysis limit DNBR (RTDP)	1.34	1.38
DNBR Margin (RTDP)	7.5%	10.1%

The margin between the design limit and the safety analysis limit DNBR is more than enough to offset the DNBR penalties associated with the Ginna core, which include rod bow, steam generator tube plugging, thimble plug removal, and transition core penalties.

4.4.3.2 Rod Bow

Rod bow can occur between mid-grids, reducing the spacing between adjacent fuel rods and reducing the margin to DNB. Rod bow must be accounted for in the DNBR safety analysis of Condition I and Condition II events. Westinghouse has conducted tests to determine the impact of rod bow on DNB performance; the testing and subsequent analyses were documented in *Reference 18*.

Currently, the maximum rod bow penalty for the OFA fuel assembly is 1.0% DNBR at an assembly average burnup of 24,000 MWD/MTU (*References 18 and 29*). No additional rod bow penalty is required for burnups greater than 24,000 MWD/MTU since credit is taken for the effect of $F^N_{\Delta H}$ burndown due to the decrease in fissionable isotopes and the buildup of fission products (*Reference 30*). Based on the testing and analyses of various fuel array designs documented in *Reference 18*, including the 14 x 14 STANDARD assembly, the 14 x 14 OFA and the 14 x 14 422V+ fuel assemblies should have the same rod bow penalty applied to the analysis basis as that used for 14 x 14 STANDARD fuel assemblies.

4.4.4 THERMAL AND HYDRAULIC TESTS AND INSPECTIONS

General hydraulic tests on models were initially used to confirm the design flow distributions and pressure drops (*References 19* and *20*). Fuel assemblies and control rod drive mechanisms were also tested. Onsite measurements were made to confirm the design flow rates.

Vessel and internals inspections were also reviewed to check such thermal and hydraulic design values as bypass flow. An extensive program of preoperational physics testing was performed using the in-core instrumentation system to verify that actual power distributions in the core were satisfactory.

4.4.5 REACTOR COOLANT FLOW MEASUREMENT

In the design of the reactor coolant system, design margin was applied to both the calculated system pressure drop and the pump design head as contrasted to "best estimate" calculations to ensure a system flow rate equal to or greater than design flow rate. Straightforward hydraulics techniques were employed together with detailed model tests using scaling techniques in accordance with Hydraulic Institute Standards. This design approach has been substantiated by measurements in all operating Westinghouse-designed plants.

Core safety limits are not particularly sensitive to the absolute value of reactor coolant system flow. In the course of plant startup, data pertinent to determining coolant flow both directly and indirectly were obtained to verify that flow is at or greater than design. A definite exact measurement of flow is not necessary for plant operation or for protection system purposes, as described later. Further there are no design provisions to vary flow, i.e., throttling devices; therefore, variations in absolute flow are not of concern during operation. Protection in the event of a loss of flow resulting from loss of power to one or both pumps is analyzed in Section 15.3.

In the original FSAR several methods were discussed that could be used to determine that actual coolant system flow was greater than the assumed design flow. The methods are described below and consist of a pump power measurement, a secondary heat balance coupled with coolant temperatures, elbow tap differential pressure measurements, core outlet thermocouple measurements, and a pump power-differential pressure iterative procedure. By operating each pump alone and both together, two different flow rates can be evaluated with the above methods as further confirmation of the pump flow characteristics.

4.4.5.1 Pump Power

Reactor coolant system flow rate can be determined from a measurement of reactor coolant pump input power by determining from the pump input power versus pump capacity characteristic curve the pump output in flow for the input power measured. Pump input power can be measured accurately. Pump power measurements are made on the actual pump motors prior to installation in the plant to determine motor characteristics.

4.4.5.2 Secondary Heat Balance

System flow rate is calculated by accurately measuring the secondary system power generation together with the corresponding measured hot- to cold-leg temperature differential in the reactor coolant system (loop delta T). Flow is equal to power divided by the reactor coolant enthalpy decrease. Further discussion of this method is in Section 4.4.5.8.

4.4.5.3 Elbow Tap Differential Pressure

Measurement of the elbow tap flow meter differential pressure provides a highly repeatable measure of system flow rate. The flow rate is determined from the measured 90-degree elbow differential pressure by documented (*Reference 21*) standard elbow characteristics.

4.4.5.4 Core Exit Thermocouple

The core differential temperature can be determined from the cold-leg temperature and a core exit thermocouple map. This is then compared with total generated secondary system power generation to determine total core flow. The core exit thermocouple system has 36 thermocouples positioned to measure fuel assembly coolant outlet temperatures at preselected core locations and three thermocouples to measure temperatures in the reactor vessel head area. The core exit thermocouple signals are converted to degrees Fahrenheit and input to the plant computer and a control room display. The core exit thermocouple system meets the requirements of NUREG 0737 and Regulatory Guide 1.97, Revision 3, for post-accident monitoring.

4.4.5.5 Pump Power-Differential Pressure

This procedure has been used experimentally in an existing plant. The results have produced calculated flow rates in close agreement with the analytically predicted most probable flow and consistent flow rates to within ±0.3% for a number of pumps. It is a refinement of the pump power method that utilizes a procedure to establish the actual pump input power performance curve more accurately. The actual operating curve is established from its known shape, determined from model tests, by interrelating pump input power and a relative change in system pressure drop under conditions of one and two pumps running. This procedure reduces the uncertainties associated with the absolute relation of the pump input power curve to flow. This procedure is described in more detail than the more familiar methods mentioned previously. Figure 4.4-1 is an example of a typical pump input power curve and is included to describe the procedure which is as follows:

A. With the reactor coolant system pressurized, all pumps are started. The flow within the loop to be measured is assumed to be equal to the design (represented by line 1 on Figure 4.4-1) and pump power (represented by line 2 on Figure 4.4-1) and a reference differential pressure measured. The intersection of lines 1 and 2 establishes a point on the assumed pump power input curve. This allows construction of the assumed curve by shifting the model test curve vertically until it intersects this point.

B. The other pump is stopped. The flow within the active loop increases because of the reduced flow through the reactor vessel. This increased flow above the assumed design flow is determined from the relative increase in the measured differential pressure.

C. This increased flow is then plotted on Figure 4.4-1 (line 3). Its intersection with the previously assumed pump curve will yield the amount of anticipated input power (line 4). If the anticipated input power equals the measured input power with one pump running, the originally assumed flow rate was correct.

The above procedure is all that is necessary to establish whether actual flow is less than, equal to, or greater than design flow. The sense of the difference between anticipated one loop operation input power and measured one loop input power will indicate this. If anticipated power is greater than measured power, the actual flow rate was greater than design. (This can be seen by following the construction of lines 5, 6, and 7 in Figure 4.4-1.) If it is desired to know the actual flow rate, the flow with all pumps operating must again be assumed and the construction of the lines repeated until anticipated one loop power equals measured one loop input power.

This procedure makes use of elbow tap (or steam generator) differential pressure readings. These readings are not used as absolute quantities but only in reference to each other in order to determine the magnitude of the change in flow from one point to another. Therefore, calibration or accurate knowledge of elbow characteristics and dimensions are not required.

The accuracy of this procedure is affected by the accuracy of measured input power, the accuracy of determining the relative change in flow, and the accuracy of the shape of the input power curve. From a review of data from full scale tests of smaller earlier model pumps and the accuracies associated with model tests and hydraulic scaling theory, it has been judged that an accuracy of ±0.5% is a conservative tolerance to apply to the accuracy of the shape of the curve. The relative change in flow between the two pump running condition

and the one pump running condition can be determined to an accuracy of ±0.5% by the use of pretest deadweight tester calibrated Foxboro differential pressure cells and a Hewlett-Packard digital voltmeter. Pump input power can be measured to an accuracy of ± 0.5% by use of procedures and instrumentation available from a test organization at the Westinghouse Large Rotating Apparatus Division. Typical instrumentation that would be used consists of a Weston Model 329 wattmeter and Westinghouse Model PA 151 volt and ammeters. These accuracies result in an expected total flow rate measurement accuracy of ± 2.5%.

4.4.5.6 Experience

Each of the above methods is employed to obtain an independent assessment of flow. Each is evaluated in terms of consistency with one another as well as between loops. Possible error allowances are established on the basis of various in-plant calibrations, i.e., loop temperature and in-core thermocouple isothermal calibrations. Experience has shown that all methods indicate greater than design flow with good agreement between loops and reasonable agreement between methods sufficient to validate greater than design flow. An example of measured data is listed below.

	Pump Power	Elbow Taps	Core Thermocouple
Loop A	113%	107%	
Loop B	110%	107%	105%

4.4.5.7 Low Flow Trip Setpoint

Elbow taps are used in the primary coolant system as an instrument device that indicates the status of the reactor coolant flow. The basic function of this device is to provide information as to whether or not a reduction in flow rate has occurred. The correlation between flow rate reduction and elbow tap read-out has been well established by the following equation (*Reference 21*):

$$\frac{deltaP}{deltaP_0} = \left(\frac{\omega}{\omega_0}\right)^2$$

(Equation 4.4-1)

where delta P_o is the referenced pressure differential, with the corresponding referenced flow rate ω_o and delta P is the pressure differential with the corresponding flow rate ω. The full-flow reference point is established during initial plant startup. The low-flow trip point is then established by extrapolating along the correlation curve. The technique has been used in providing core protection against low coolant flow in Westinghouse PWR plants. Field results have shown the repeatability of the setpoint to be within ± 1%. Transient analysis for a loss of flow assumes instrumentation error of ≤ 4%.

4.4.5.8 Precision Calorimetric Measurement for Reactor Coolant System Flow

The Improved Thermal Design Procedure, which was first used beginning with cycle 14, as well as the Revised Thermal Design Procedure, utilized beginning with cycle 26, require a reactor coolant system flow measurement with a high degree of accuracy such that flow measurement can be performed by determining the steam generator thermal output, corrected for the reactor coolant pump heat input and the loop's share of primary system heat losses, and the enthalpy rise (delta h) of the primary coolant. Assuming that the primary and secondary sides are in equilibrium, the reactor coolant system total vessel flow is the sum of the individual primary loop flows, i.e.

$$W_{RCS} = \Sigma W_L$$

The individual primary loop flows are determined by correcting the thermal output of the steam generator for steam generator blowdown (if not secured), subtracting the reactor coolant pump heat addition, adding the loop's share of the primary side system losses, dividing by the primary side enthalpy rise, and multiplying by the specific volume of the reactor coolant system cold leg. The equation for this calculation is

$$W_L = \frac{\gamma \cdot \left\{Q_{SG} - Q_p + \frac{Q_L}{N}\right\} V_C}{\left[h_H - h_C\right]}$$

(Equation 4.4-2)

where:	$W_L =$	loop flow (gpm)
	$\gamma =$	0.1247 gpm/(ft^3/hr)
	$Q_{SG} =$	steam generator thermal output (Btu/hr)
	$Q_p =$	reactor coolant pump heat adder (Btu/hr)
	$Q_L =$	primary system net heat losses (Btu/hr)
	$V_c =$	specific volume of the cold leg at T_C (ft^3/lb)
	$N =$	number of primary side loops
	$h_H =$	hot-leg enthalpy (Btu/lb)
	$h_c =$	cold-leg enthalpy (Btu/lb)

As an alternative to the individual loop methodology, it is also possible to obtain a calorimetric for both steam generators combined and calculate hot and cold leg average enthalpies to arrive at total reactor coolant system flow. The thermal output of the steam generator is determined by the same calorimetric measurement as for reactor power, which is defined as

$$Q_{SG} = (h_s - h_f)\, W_f$$

where

$h_s =$	steam enthalpy (Btu/lb)
$h_f =$	feedwater enthalpy (Btu/lb)
$W_f =$	feedwater flow (lb/hr)

The steam enthalpy is based on measurement of steam generator outlet steam pressure, assuming saturated conditions. The feedwater enthalpy is based on the measurement of feedwater temperature and steam pressure. The feedwater flow is determined by multiple measurements and the same calculation as used for reactor power measurements, which is based on the following:

$$W_f = (K)(F_a) \cdot \sqrt{\rho_f \cdot \Delta p}$$

(Equation 4.4-3)

where:	$K =$	feedwater venturi flow factor
	$F_a =$	feedwater venturi correction for thermal expansion
	$\rho_f =$	feedwater density (lb/ft^3)
	$\Delta p =$	feedwater venturi pressure drop (in. H_2O)

The feedwater venturi flow coefficient is the product of a number of constants including as-built dimensions of the venturi and calibration tests performed by the vendor. The thermal expansion correction is based on the coefficient of expansion of the venturi material and the difference between feedwater temperature and calibration temperature. Feedwater density is based on the measurement of feedwater temperature and feedwater pressure. The venturi

pressure drop is obtained from the output of the differential pressure cell connected to the venturi.

The reactor coolant pump heat adder is determined by calculation, based on the best estimates of coolant flow, pump head, and pump hydraulic efficiency.

The primary system net heat losses are determined by calculation, considering the following system heat inputs and heat losses:

- Charging flow
- Letdown flow
- Seal injection/seal return flow
- Reactor coolant pump thermal barrier cooler heat removal
- Pressurizer spray flow
- Pressurizer surge line flow
- Component insulation heat losses
- Component support heat losses
- Control rod drive mechanism heat losses

A single calculated sum for full power operation is used for these losses/heat inputs.

The hot-leg and cold-leg enthalpies are based on the measurement of the hot-leg temperature, cold-leg temperature, and the pressurizer pressure. The cold-leg specific volume is based on measurement of the cold-leg temperature and pressurizer pressure.

The reactor coolant system flow measurement is thus based on the following plant measurements.

- Steam line pressure (P_s)
- Feedwater temperature (T_f)
- Feedwater venturi differential pressure (delta P)
- Hot-leg temperature (T_H)
- Cold-leg temperature (T_C)
- Pressurizer pressure (P_p)
- Steam generator blowdown (if not secured)

and on the following calculated values.

- Feedwater venturi flow coefficients (K)
- Feedwater venturi thermal expansion correction (F_a)
- Feedwater density (ρ_f)

- Feedwater enthalpy (h_f)
- Steam enthalpy (h_s)
- Primary system net heat losses (Q_L)
- Reactor coolant pump heat adder (Q_p)
- Hot-leg enthalpy (h_H)
- Cold-leg enthalpy (h_c)

This measurement is performed for each cycle starting with cycle 13, verifying the conservatism of the design flow used in the safety analysis. The uncertainty of this measurement was calculated to be less than that assumed in determining the design limit for DNBR using the Revised Thermal Design Procedure.

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Table 4.4-1
THERMAL AND HYDRAULIC DESIGN PARAMETERS[a]

Total heat output, Mwt	1775
Total heat output, Btu/hr	6057×10^6
Heat generated in fuel, %	97.4
Maximum thermal overpower, %	18
Nominal system pressure, psia	2250
Hot-channel factors	
Heat flux	
Nuclear, F^N_Q	2.52
Engineering, F^E_Q	1.03
Total	2.60
Enthalpy rise	
Nuclear, $F^N_{\Delta H}$	1.72 (422V+) 1.60 (VANTAGE+)
Coolant flow	
Total flow rate, lb/hr (TDF)	60.5×10^6
Coolant temperature, °F	
Nominal inlet	540.2
Average rise in vessel	71.6
Average rise in core	76.0
Average in core	580.3
Average in vessel	576.0

Heat transfer	
Active heat transfer surface area, ft^2	98,507
Average heat flux, Btu/hr-ft^2	206,950
Maximum heat flux, Btu/hr-ft^2	538,070
Maximum thermal output, kW/ft	18.2
DNBR	
Minimum DNBR at nominal operating conditions	1.839 typical cell; 1.818 thimble cell
Pressure drop, psi	
Across core at flow of 193,600 gpm (B.E.)	24.7 (422V+) 26.6 (VANTAGE+)

a. These parameters correspond to zero percent steam generator tube plugging and high T_{AVG} (576°F). An evaluation has been performed (*Reference 17* of UFSAR Section 4.4) which shows that all safety limits are satisfied for 15% steam generator tube plugging.

4.5 REACTOR MATERIALS

Reactor vessel materials are discussed in Section 5.3.1. Control rod drive system structural materials and reactor internals materials are discussed below.

4.5.1 CONTROL ROD DRIVE SYSTEM STRUCTURAL 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, Alloy X-750, and cobalt-based alloys. Wherever magnetic flux is carried by parts exposed to the main coolant, stainless steel is used. Cobalt-based alloys are used for the pins and latch tips.

Alloy X-750 is used for the springs of both latch assemblies and type 304 stainless steel is used for all pressure containment. Hard chrome plating provides wear surfaces on the sliding parts and prevents galling between mating parts (such as threads) during assembly.

Outside of the pressure vessel, where the metals are exposed only to the 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-in. thick to prevent corrosion.

Additional information on the control rod drive system materials is presented in Section 3.9.4.

4.5.2 REACTOR INTERNALS MATERIALS

Information on reactor internals materials is presented in Section 3.9.5.

4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEM

Information on the functional design and evaluation of the control rod drive system is presented in Section 7.7. The mechanical design is discussed in Section 3.9.4. Information on the functional design and evaluation of the chemical and volume control system is presented in Section 9.3.4. Evaluation of the combined performance of reactivity control systems pertaining to the response of the plant to postulated process disturbances and to postulated malfunctions or failures of equipment are presented in Chapter 15 and Section 7.7.