CHAPTER 3

REACTOR

3.1 DESIGN BASIS

3.1.1 Performance Objectives

The reactor thermal power analyzed is 3216 MWt.

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

Rod Control Clusters are employed to provide sufficient reactivity control to terminate any credible power transient prior to reaching the design minimum departure from nucleate boiling ratio (DNBR) of the applicable limit. This is accomplished by ensuring sufficient control cluster worth to shut the reactor down by at least 1.3% in the hot condition with the most reactive control cluster stuck in the fully withdrawn position.

Redundant equipment is provided to add soluble poison to the reactor coolant in the form of boric acid to maintain shutdown margin when the reactor is cooled to ambient temperatures.

In addition, the control rod worth in conjunction with the boric acid injection from the refueling water storage tank (RWST) is sufficient to prevent an unacceptable return to power level as a result of the maximum credible steam line break (one safety valve stuck fully open) even assuming that the most reactive control rod is fully withdrawn.

With the Boron Injection Tank (BIT) functionally eliminated, the return to power following a credible steamline break accident has been evaluated showing that the event is bounded by the hypothetical steamline break. The departure from nucleate boiling (DNB) design basis is met with no consequential fuel failures predicted, and assuring that the return to power remains within the limits established for the protection of the health and safety of the public, with margin.

Plant specific analyses performed by Westinghouse for Indian Point Unit 3, have shown that the [Deleted] BIT may be bypassed, eliminated, or the concentration of its contents reduced, while continuing to meet applicable safety criteria.

The functional elimination of the BIT replaces the concentrated boric acid contained therein, with water from the [Deleted] RWST; this obviates the need to maintain the BIT and its associated piping at elevated temperatures.

The lowering of the minimum required boric acid concentration in the BIT:

1) reduces the potential for degradation of carbon steel components and supports as a result of leakage;
2) eliminates the need to maintain recirculation of boric acid through BIT;
3) eliminates the need to maintain the BIT heaters and heat tracing on the associated Safety Injection system piping and recirculation lines; and

4) eliminates the need for periodic checks of BIT concentration thereby reducing radiation exposure of plant personnel.
5) Eliminates the need to maintain closed the BIT inlet and outlet isolation valves.

Experimental measurements from critical experiments or operating reactors, or both, were used to validate the methods employed in the design. During design, nuclear parameters were calculated for every phase of operation of the first core and reload cycles and, where applicable, were compared with design limits to show that an adequate margin of safety existed. In the thermal hydraulic design of the core, the maximum fuel and clad temperatures during normal reactor operation and at overpower conditions were conservatively evaluated and found to be consistent with safe operating limitations.

3.1.2 Principal Design Criteria

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the bases for the Indian Point 3 design, were published by the Atomic Energy Commission in the Federal Register of July 11, 1967, and subsequently made a part of 10 CFR 50.

A study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission's Confirmatory Order of February 11, 1980 has been completed. The detailed results of the evaluation of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NRC) in 10 CFR 50 Appendix A, were submitted to the NRC on August 11, 1980, and approved by the Commission on January 19, 1982. These results are presented in Section 1.3.

Reactor Core Design

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

The reactor core, with its related control and protection system, was 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, provide for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations, including the effects of the loss of reactor coolant flow (Section 14.1.6), trip of the turbine generator (Section 14.1.8), loss of normal feedwater (Section 14.1.9) and loss of all offsite power (Section 14.1.12).

The Reactor Control and Protection System was designed to actuate a reactor trip for any anticipated combination of plant conditions, when necessary, to ensure a minimum Departure from Nucleate Boiling (DNB) ratio equal to or greater than the applicable limit.

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

1) Minimum DNB ratio equal to or greater than the applicable limit

2) Fuel center temperature below 4700° F

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

4) Clad stresses less than the Zircaloy or ZIRLO™ yield strength

5) Clad strain less than 1%

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

The reactor coolant pumps provided for the plant are supplied with sufficient rotational inertia to maintain an adequate flow coastdown and prevent core damage in the event of a simultaneous loss of power to all pumps.

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

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

Suppression of Reactor Power Oscillations

Criterion 7: 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.

The potential for possible spatial oscillations of power distribution for this core has been reviewed. It was concluded that low frequency xenon oscillations may occur in the axial dimension, and the control rods can suppress these oscillations. The core is expected to be stable to xenon oscillations in the X-Y dimension. Excore instrumentation is provided to obtain

necessary information concerning power distribution. 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 Excore instrumentation.) The analysis, detection and control of these oscillations is discussed in Reference 2 of Section 3.2.

Redundancy of Reactivity Control

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

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

Reactivity Hot Shutdown Capability

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

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

The Rod Cluster Control (RCC) assemblies are divided into two categories comprising control banks, and shutdown banks. The control banks used in combination with chemical shim control provide control of the reactivity changes of the core throughout the life of the core during power operation. These banks of RCC assemblies are used to compensate for short term reactivity changes at power that might be produced due to variations in reactor power level or in coolant temperature. The 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.

Reactivity Shutdown Capability

Criterion 29: 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.

The reactor core, together with the reactor control and protection system was designed so that the minimum allowable DNBR is at least the applicable limit and there is no fuel melting during normal operation including anticipated transients.

The shutdown groups are provided to supplement the control groups RCC assemblies to make the reactor at least 1.3% subcritical at the hot zero power condition following trip from any credible operating condition assuming the most reactive RCC assembly is in the fully withdrawn position.

Sufficient shutdown capability is also provided to prevent an unacceptable return to power level, assuming the most reactive rod to be in the fully withdrawn position for the most severe anticipated cooldown transient associated with a single active failure, e.g., accidental opening of a stream bypass, or relief valve, or safety valve stuck open. This is achieved by the combination of control rods and automatic boric acid addition via the Emergency Core Cooling System. With the BIT functionally eliminated, the return to power following a credible steamline break accident has been evaluated showing that the event is bounded by the hypothetical steamline break. The departure from nucleate boiling (DNB) design basis is met with no consequential fuel failures predicted, and assuring that the return to power remains within the limits established for the protection of the health and safety of the public, with margin.

The minimum shutdown margin was calculated to be at least 1.3% @EOL conditions assuming the maximum worth control rod in the fully withdrawn position allowing 10% uncertainty in the control rod calculations.

Manually controlled boric acid addition is used to maintain the shutdown margin for the long term conditions of xenon decay and plant cooldown. Redundant equipment is provided to guarantee the capability of adding boric acid to the Reactor Coolant System.

<u>Reactivity Holddown Capability</u>

Criterion 30: 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 that there will be no undue risk to the health and safety of the public.

Normal reactivity shutdown capability is provided within 2.7 seconds following a trip signal by control rods, with boric acid injection used to compensate for the long term xenon decay transient and for plant cooldown. As discussed in response to the previous criteria, the shutdown capability prevents return to critical as a result of the cooldown associated with a safety valve stuck fully open.

Any time that the reactor is at power, the quantity of boric acid retained in the boric acid tanks and ready for injection always exceeds that quantity required for the normal cold shutdown. This quantity always exceeds the quantity of boric acid required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay. Boric acid is pumped from the boric acid tanks by one of two boric acid transfer pumps to the suction of one of three charging pumps which inject boric acid into the reactor coolant. Any charging pump and either boric acid transfer pump can be operated from diesel generator power on loss of station power. Using either one of the two boric acid transfer pumps, in conjunction with any one of the three charging pumps, the RCS can be borated to hot shutdown even with the control rods fully withdrawn. Additional boration would be used to compensate for xenon decay. At a minimum CVCS design boration rate of 132 ppm/hr, the boron concentration required for cold shutdown can be reached well before xenon decays below its pre-shutdown level. The RWST is a suitable backup source for emergency boration. When two charging pumps are used to transfer borated water from the RWST to the reactor coolant, the boron concentration required for cold shutdown can be reached before xenon decays below its full power pre-shutdown level.

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, thus achieving the measure of reliability implied by the criterion.

Alternately, boric acid solution at lower concentration can be supplied from the refueling water tank. 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.

With the BIT functionally eliminated, the return to power following a credible steamline break accident has been evaluated showing that the event is bounded by the hypothetical steamline break. The departure from nucleate boiling (DNB) design basis is met with no consequential fuel failures predicted, and assuring that the return to power remains within the limits established for the protection of the health and safety of the public, with margin.

Reactivity Control System Malfunction

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

The reactor protection systems are capable of protecting against any single credible 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 and continuous deboration are described in Section 14.1 and Section 9.2, respectively.

Maximum Reactivity Worth of Control Rods

Criterion 32: 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.

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. A portion of these are designated shutdown rods and are fully withdrawn during power operation. The remaining rods comprise the control groups 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 yielding reactivity insertion rates no greater than 66 pcm/sec which is well within the capability of the overpower-overtemperature protection circuits to prevent core damage.

The rod drive design permits only two groups to be withdrawn at the same time. Should more than two groups try to move, insufficient power is available to activate the movable holding latches and the affected control rods would disengage and fall.

Specifically, a bank contains either one or two groups. Within a bank the group may be moved only sequentially one step at a time. Two banks may be moved simultaneously, e.g. banks C and D. However, group 1 in bank D may be moved together (one step), then group 2 in each bank simultaneously (one step). Therefore, no more than two groups can be moved together and this forms the basis of the assumption.

No single credible mechanical or electrical control system malfunction can cause a rod cluster to be withdrawn at a speed greater than 80 steps per minute (~50 inches per minute).

3.1.3 Safety Limits

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

The limiting conditions for operation established in the Technical Specifications specify the functional capacity of performance levels permitted to assure safe operation of the facility.

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

Nuclear Limits

At full power (license application power) the nuclear heat flux hot channel factor F_Q^N, is not exceeded. For any condition of power level, coolant temperature and pressure which is permitted by the control and protection system during normal operation and anticipated transients the hot channel power distribution is such that the minimum DNB ratio is greater than the applicable limit and the linear heat rate is less than 22.7 kW/ft. For any normal steady state operating condition, the maximum linear heat rate does not exceed 6.64 x F_Q kW ft, where F_Q is the maximum value dictated by the Core Operating Limits Report (COLR).

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

Fuel Enrichment Limits

Detailed nuclear analysis (refer to Reference 65 of Section 3.2) has been completed to demonstrate that the existing spent fuel storage racks can safely store fuel with initial enrichments up to 5.0 w/o U-235 provided they are done so as specified in Figures 9.5-2A, 9.5-2B, and 9.5-2C.

Control Bank Insertion Limits

The control bank insertion limits for D, C and B control banks were originally revised for Cycle 6 operation. The associated change in control bank insertion limits (refer to Reference 57 of Section 3.2) results in increased flexibility in core design, and a reduction in the calculated core peaking factor Fq at the bank insertion limit. The revised insertion limit curve for the current cycle is provided in the cycle-specific Core Operating Limits Report.

Reactivity Control Limits

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

1) A minimum hot shutdown margin as shown in the Technical Specifications is available assuming a 10% uncertainty in the control rod calculation (see Table 3.2-3).
2) This shutdown margin is maintained with the most reactive RCCA in the fully withdrawn position.
3) The shutdown margin is maintained at ambient temperature by the use of soluble poison.

Thermal and Hydraulic Limits

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

1) The minimum allowable DNBR during normal operation, including anticipated transients for the WRB-1 correlation is documented in Table 14.1-0. The minimum allowable DNBR for the W-3 correlation is 1.3 from 1000 to 2400 psia and 1.45 from 500 to 1000 psia.

2) Fuel temperature will not exceed 4700°F during any anticipated operating condition.

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 a departure from nucleate boiling (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 or WRB-1 correlation, to the existing heat flux at the same core location is the DNB ratio. The limiting DNB ratio corresponds to a 95% probability at a 95% confidence level that DNB does not occur and is chosen to maintain an appropriate margin to DNB for all operating conditions.

In connection with the functional elimination of the BIT, two hypothetical and one credible steamline break cases were either analyzed or evaluated:

Hypothetical steamline breaks:

1) Steampipe severance, downstream of the flow restrictor, with offsite power available;
2) Steampipe severance, downstream of the flow restrictor, without offsite power available;

Credible streamline break:

1) A failed secondary safety or relief valve, with offsite power available.

The DNB analyses show that the DNB design basis is met for the hypothetical steamline break with offsite power, and that no consequential fuel failures are anticipated. The hypothetical break without offsite power and the credible steamline break were evaluated and determined to be bounded by the results of the hypothetical steamline break with offsite power.

Mechanical Limits

Reactor Internals

The reactor internal components were 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 internals were further designed to maintain their functional integrity in the event of a major Loss-of-Coolant Accident. The dynamic loading resulting from the pressure oscillations because of a Loss-of-Coolant Accident does not cause sufficient deformation to prevent Rod Cluster Control assembly insertion.

Fuel Assemblies

The fuel assemblies were 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 was 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, subsequent handling during cooldown shipment and fuel reprocessing.

The fuel rods are supported at seven locations along their length within the fuel assemblies by brazed grid assemblies which were 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 and without imposing restraints of sufficient magnitude to result in buckling or distortion of the rods.

The fuel rod cladding was designed to withstand operating pressure loads without rupture and to maintain fuel integrity throughout design life.

Cycle 7 was the first cycle where Vantage 5 fuel was utilized. Features of this fuel include: debris filter bottom nozzle, reconstitutable top nozzle, six inch natural uranium axial blankets and the optimal use of integral fuel burnable absorber (IFBA) which incorporates a thin coating of ZrB2 on the outside cylindrical surface area of the fuel pellet.

Cycle 10 was the first cycle where Vantage + fuel was utilized. Features of this fuel included three intermediate flow mixers, enriched annular axial blankets, and low pressure drop midgrids, ZIRLO guide thimbles and midgrids. ZIRLO as a clad material was introduced in Cycle 9. Vantage + fuel also contains various high burnup features.

Beginning with Cycle 14, "15x15 upgrade" fuel was introduced to the core. See section 3.2.5.5 for details.

Rod Cluster Control Assemblies

The criteria used for the design of the cladding on the individual absorber rods in the rod cluster control assemblies (RCCA) are similar to those used for the fuel rod cladding. The cladding was designed to be free standing under all operating conditions and will maintain encapsulation of the absorber material throughout the absorber rod design life. Allowance for wear during operation is included for the RCCA cladding thickness. Cladding of RCCA's is stainless steel.

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.

Control Rod Drive Assembly

Each control rod drive assembly was designed as a hermetically sealed unit to prevent leakage of reactor coolant. All pressure containing components were designed to meet the requirements of the ASME Code Section III. Nuclear Vessels, for Class A vessels.

The control rod drive assemblies for the full length rods 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 for the full length rods provide a fast insertion rate during a "trip" of the RCCA which results in a rapid shutdown of the reactor for conditions that cannot be handled by the reactor control system.

The part length control rod drive mechanisms (Reference 46) were used to position the part length rod control cluster assemblies within the reactor core. Subsequent to initial plant operation (during the Cycle 1 to Cycle 2 refueling outage), the part length RCCAs were removed from the reactor. Their drive mechanisms and position indication system were retired in place.

3.2 REACTOR DESIGN

3.2.1 Nuclear Design and Evaluation

This section presents the nuclear characteristics of the core and an evaluation of the characteristics and design parameters which are significant to design objectives. Throughout this section, Cycle 1 and Current Cycle parameters have been utilized in as much as they represent examples and/or somewhat typical values for initial and subsequent cycles. The capability of the reactor to achieve these objectives while performing safely under normal operational modes, including both transient and steady state, is demonstrated.

3.2.1.1 Nuclear Characteristics of the Design

A summary of the reactor nuclear design characteristics for Cycle 1 and the current cycle is presented in Table 3.2-1. Figures 3.2-2 through 3.2-12 provide core configuration information for Cycle 1 and are included for historical purposes. For current cycle information refer to the cycle specific Nuclear Parameters and Operations Package and the cycle specific Core Operating Limits Report.

Reactivity Control Aspects

Reactivity control is provided by neutron absorbing control rods and by a soluble chemical neutron absorber (boric acid) in the reactor coolant. 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 cold shutdown to the hot operating, zero power condition; (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 poison burnup.

The control rods provide reactivity control for: (1) fast shutdown; (2) reactivity changes associated with changes in the average coolant temperature above hot zero power (core average coolant temperature is increased with power level); (3) reactivity associated with any void formation; (4) reactivity changes associated with the power coefficient of reactivity.

Chemical Shim Control

Control to render the reactor subcritical at temperatures below the operating range is provided by a chemical neutron absorber (boron). The boron concentration during refueling is the more restrictive of 2050 ppm or that required to be 5% subcritical. The boron concentration and the control rods together are required to provide at least 5 percent shutdown margin for these operations. Boron concentration requirements for refueling and Plant Startup (BOL) are found in Reference 72.

These boron concentrations are well within solubility limits at ambient temperature. Refueling concentration is also maintained in the spent fuel pit since it is directly connected with the refueling canal during refueling operations.

The initial Cycle 1, full power boron concentration without equilibrium xenon and samarium was 1228 ppm. As fission product poisons were built up, the boron concentration was reduced to 899 ppm.

This initial boron concentration was that which permitted the withdrawal of the control banks to their operational limits. The Cycle 1 xenon-free hot, zero power shutdown (k=0.99) with all but the highest worth rod inserted, could be maintained with the boron concentration of 669 ppm. This concentration is less than the full power operating value with equilibrium xenon.

Control Rod Requirements

Neutron-absorbing control rods provide reactivity control to compensate for more rapid variations in reactivity. The rods are divided into two categories according to their function. Some rods compensate for changes in reactivity due to variations in operating conditions of the reactor such as power or temperature. These rods comprise the control group of rods. The remaining rods, which provide shutdown reactivity, are termed shutdown rods. The total shutdown worth of all the rods was also specified to provide adequate shutdown with the most reactive rod stuck out of the core.

Control rod reactivity requirements at beginning- and end-of-life for Cycle 1 and the current cycle are summarized in Table 3.2-2. The installed worth of the control rods is shown in Table 3.2-3

The difference between required and installed worth is available for excess shutdown upon reactor trip. The control rod requirements are discussed below.

Total Power Reactivity Defect

Control rods must be available to compensate for the reactivity change incurred with a change in power level due to the Doppler and moderator temperature effects.

The average temperature of the reactor coolant is increased with power level in the reactor. Since this change is actually a part of the power dependent reactivity change, along with the Doppler effect and void formation, the associated reactivity change must be controlled by rods. The largest amount of reactivity that must be controlled is at the end of life when the moderator temperature coefficient has its most negative value. The moderator temperature coefficient range is given in Table 3.2-1, line 42, while the cumulative reactivity change is shown in the first line of Table 3.2-2. By the end of the fuel cycle, the nonuniform axial depletion causes a severe power peak at low power. The reactivity associated with this peak is part of the power defect.

Operational Maneuvering Band

The control group is operated at full power within a prescribed band of travel in the core to compensate for periodic changes in boron concentration, temperature, or xenon. The band has been defined as the operational maneuvering band. When the rods reach either limit of the

band, a change in boron concentration must be made to compensate for any additional change in reactivity, thus keeping the control group within the maneuvering band.

The fully withdrawn bank position can vary within a few steps from the reference fully withdrawn condition from cycle to cycle.

The greatest depth in the core to which the control banks may be inserted is known as the insertion limit. The maximum insertion limit on which the FSAR transient analyses are based is 23.5 percent D-Bank insertion, which corresponds to a D-Bank position of 176 steps at 100 percent power. The actual limit may be administratively decreased via a COLR change, to facilitate core design without having to reanalyze the FSAR transients.

Control Rod Bite

If sufficient boron is present in a chemically-shimmed core, the inherent operational control afforded by the negative moderator temperature coefficient is lessened to such a degree that the major control of transients resulting from load variations must be compensated for by control rods. The ability of the plant to accept major load variations is distinct from safety considerations, since the reactor would be tripped and the plant shut down safely if the rods could not follow the imposed load variations. In order to meet required reactivity ramp rates resulting from load changes, the control rods must be inserted a given distance into the core. The reactivity worth of this insertion has been defined as control rod bite.

The reactivity insertion rate must be sufficient to compensate for reactivity variation due to changes in power and temperature caused either by a ramp load change of five percent per minute, or by a step load change of ten percent. An insertion rate of $4 \times 10^{-5}\ \Delta p$ per second is determined by the transient analysis of the core and plant to be adequate for the most adverse combinations of power and moderator coefficients. To obtain this minimum ramp rate, one control bank of rods had to remain inserted at least 13 percent into the core at the Cycle 1 beginning-of-life. The reactivity associated with this bite was 0.03 percent.

Indian Point 3 is analyzed for continuous operation at the bite position, at no bite position (i.e., ARO) and for any rod position in between [(85)].

Xenon Stability Control

The control rods are capable of suppressing xenon induced power oscillations in the axial direction, should they occur. Out-of-core instrumentation was provided to obtain necessary information concerning power distribution. This instrumentation is adequate to enable the operator to monitor and control xenon induced power oscillations. Extensive analyses, with confirmation of methods by spatial transient experiments at Haddam Neck, have shown that any induced radial or diametral xenon transients would die away naturally.[(2)] A full discussion of axial xenon stability control can be found in Reference 3.

Excess Reactivity Insertion Upon Reactor Trip

The control requirements are nominally based on providing 1.3 percent shutdown at hot, zero power conditions with the highest worth rod stuck in its fully withdrawn position. The condition where excess reactivity insertion is most critical is at the end of a cycle when the steam line break accident is considered. The excess control available at the Cycle 1 end-of-cycle, hot zero

power condition with the highest worth rod stuck out, allowing a 10% margin for uncertainty in control rod worth is shown in Table 3.2-3.

Calculated Rod Worths

The complement of 53 full length control rods, arranged in the pattern shown in Figure 3.2-1 meets the shutdown requirements. Table 3.2-3 lists the calculated worths of this rod configuration for beginning-of-life, and end-of life, for Cycles 1 and the current cycle.

In order to be sure of maintaining a conservative margin between calculated and required rod worths, an additional amount has been added to account for uncertainties in the control rod worth calculations. The calculated reactivity worths listed were decreased in the design by 10 percent to account for any errors or uncertainties in the calculation. This worth was established for the condition that the highest worth rod is stuck in the fully withdrawn position in the core.

A comparison between calculated and measured rod worths in operating reactors shows the calculation to be well within the allowed uncertainty of 10%.

Reactor Core Power Distribution

The accuracy of power distribution calculations was initially confirmed through approximately one thousand flux maps during some twenty years of reactor operation under conditions very similar to those which were expected for Indian Point 3. Details of this confirmation are given in Reference 5.

Definitions

Power distributions are quantified in terms of hot channel factors. These factors are a measure of the peak pellet power within the reactor core and the total energy produced in a coolant channel, and are expressed in terms of quantities related to the nuclear or thermal design, namely:

Power density is the thermal power produced per unit volume of the core (kW/liter).

Linear power density is the thermal power produced per unit length of active fuel (kW/ft). Since fuel assembly geometry is standardized, this is the unit of power density most commonly used. For all practical purposes it differs from (kW/liter) by a constant factor which includes geometry and the fraction of the total thermal power which is generated in the fuel rod.

Average linear power density is the total thermal power produced in the fuel rods divided by the total active fuel length of all rods in the core.

Local heat flux is the heat flux at the surface of the cladding (Btu/ft^2/hr). For nominal rod parameters this differs from linear power density by a constant factor.

Rod power or rod integral power is the length integrated linear power density in one rod (kW).

Average rod power is the total thermal power produced in the fuel rods divided by the number of fuel rods (assuming all rods have equal length).

The hot channel factors used in the discussion of power distributions in this section are defined as follows:

F_Q, Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F_Q^N, Nuclear Heat Flux Hot Channel Factor, is defined as the maximum local rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod parameters.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod and eccentricity of the gap between pellet and clad.

Combined statistically, the net effect of the hot channel factors has a value of 1.03 to be applied to fuel rod surface heat flux.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

Manufacturing tolerances, hot channel power distribution and surrounding channel power distributions are treated explicitly in the calculation of the Departure from Nucleate Boiling Ratio.

It is convenient for the purposes of discussion to define subfactors of F_Q. However, design limits are set in terms of the total peaking factor.

F_Q = Total peaking factor or heat flux hot-channel factor
= (Maximum kW/ft) / (Average kW/ft)

without densification effects

$$
\begin{aligned}
F_Q &= F_Q^N \times F_Q^E \\
&= F_{xy}^N \times F_Z^N \times F_U^N \times F_Q^E
\end{aligned}
$$

where:

F_Q^N and F_Q^E are defined above.

F_U^N = factor for conservatism, assumed to be 1.05. (75% available thimbles)

F_{XY}^N = ratio of peak power density to average power density in the horizontal plane of peak local power.

F_Z^N = ratio of the power per unit core height in the horizontal plane of peak local power to the average value of power per unit core height. If the plane of peak local power coincides with the plane of maximum power per unit core height, then F_Z^N is the core average axial peaking factor.

To include the allowances made for densification effects, which are height dependent, the following quantities are defined:

$S(Z)$ = the allowance made for densification effects, at height Z in the core.

$P(Z)$ = ratio of the power per unit core height in the horizontal plane at height Z to the average value of power per unit core height.

Then:

F_Q = Total peaking factor

= (Maximum kW/ft) / (Average kW/ft)

Including densification allowance,

$$F_Q = \max_{\text{on } Z}\left\{\left[F_{XY}^N(Z)\right]\left[P(Z)\right]\left[S(Z)\right]\right\}\left(F_U^N\right)\left(F_Q^E\right)$$

Results reported in Reference 83 show that current fuel designs manufactured by Westinghouse are highly stable with respect to fuel densification and that the formation of small gaps in the axial pellet columns are very infrequent events. Reference 83 concludes that a densification power spike factor (S(Z)) of 1.0 is appropriate for current Westinghouse fuel, effectively eliminating this penalty from the safety analysis.

<u>Radial Power Distributions</u>

The power shape in horizontal sections of the core at full power is a function of the fuel and burnable poison loading patterns, and the presence or absence of a single bank of control rods. Thus, at any time in the cycle a horizontal section of the core can be characterized as: 1) unrodded, or 2) with group D control rods. These two situations, combined with burnup effects, determine the radial power shapes which can exist in the core at full power. The effect on radial power shapes of power level, xenon, samarium and moderator density are considered also, but they are quite small. The effect of non-uniform flow distribution is negligible. While radial power distributions in various planes of the core are often illustrated, the core radial enthalpy rise distribution, as determined by the integral of power up each channel, is of greater interest. Figure 3.2-2 through 3.2-4 show representative Cycle 1 radial power distributions for one eighth of the core for representative operating conditions. The conditions are:

1) Hot Full Power (HFP) – Beginning-of-Life (BOL) – unrodded – no xenon.
2) HFP – End-of-Life (EOL) – unrodded – equilibrium xenon
3) HFP – BOL – Bank D in – equilibrium xenon

Since the position of the hot channel varies from time to time, a single reference radial design power distribution is selected for DNB calculations. This reference power distribution is chosen conservatively to concentrate power in one area of the core, minimizing the benefits of flow redistribution. Assembly powers are normalized to core average power.

Axial Power Distributions

The shape of the power profile in the axial or vertical directions is largely under the control of the operator through either the manual operation of the control rods or automatic motion of the rods, responding to manual operation of the soluble boron system. Nuclear effects which cause variations in the axial power shape include: moderator density, Doppler effect on resonance absorption, spatial xenon and burnup. Automatically controlled variations in total power output and rod motion are also important in determining the axial power shape at any time. Signals are available to the operator from the excore ion chambers (which are long ion chambers outside the reactor vessel running parallel to the axis of the core). Separate signals are taken from the top and bottom halves of the chambers. The difference between top and bottom signals from each of four pairs of detectors is displayed on the control panel and called the flux difference, ΔI. Calculations of the core average peaking factor for many plants, and measurements from operating plants under many operating situations, are associated with either I or axial offset in such a way that an upper bound can be placed on the peaking factor. For these correlations, axial offset is defined as:

$$\text{Axial Offset} = (\Phi_t - \Phi_b)/(\Phi_t + \Phi_b)$$

where: Φ_t and Φ_b are the top and bottom detector readings.

Local Power Peaking

Fuel densification, which has been observed to occur under irradiation in several operating reactors, causes the fuel pellets to shrink both axially and radially. The pellet shrinkage combined with random hang-up of fuel pellets results in gaps in the fuel column when the pellets below the hung-up pellet settle in the fuel rod. The gaps vary in length and location in the fuel rod. Because of decreased neutron absorption in the vicinity of the gap, power peaking occurs in the adjacent fuel rods resulting in an increased power peaking factor. A quantitative measure of this local peaking is given by the power spike factor S(Z) where Z is the axial location in the core. The method used to compute the power spike factor is described in Reference 6.

Results reported in Reference 83 show that current fuel designs manufactured by Westinghouse are highly stable with respect to fuel densification and that the formation of small gaps in the axial pellet columns are very infrequent events. Reference 83 concludes that a densification power spike factor (S(Z)) of 1.0 is appropriate for current Westinghouse fuel, effectively eliminating this penalty from the safety analysis.

Limiting Power Distributions

According to the ANS classification of plant conditions, Condition I occurrences are those which are expected frequently or regularly in the course of power operation, maintenance, or maneuvering of the plant. As such, Condition I occurrences are accommodated with margin between any plant parameter and the value of that parameter which would require either

automatic or manual protective action, Inasmuch as Condition I occurrences occur frequently or regularly, they must be considered from the point of view of affecting the consequences of fault conditions (Conditions II, III and IV). In this regard, analysis of each fault condition described is generally based on a conservative set of initial conditions corresponding to the most adverse set of conditions which can occur during Condition I operation.

Implicit in the definition of normal operation is proper and timely action by the reactor operator. That is, the operator follows recommended procedures for maintaining appropriate power distributions and takes any necessary remedial actions when alerted to do so by the plant instrumentation. Thus, as stated above, the worst or limiting power distribution which can occur during normal operation is to be considered as the starting point for analysis of Condition II, III, and IV events.

Improper procedural actions or errors by the operator are assumed in the design as occurrences of moderate frequency (Condition II). Therefore, the limiting power shapes which result from such Condition II events are those power shapes which deviate from the normal operating condition at the recommended axial offset band, e.g., due to lack of proper action by the operator during a xenon transient following a change in power level brought about by control rod motion. Power shapes which fall in this category are used for determination of the reactor protection system setpoints so as to maintain margin to overpower or DNB limits.

The means for maintaining power distributions within the required hot channel factor limits are described in the Technical Specifications. A complete discussion of power distribution control in Westinghouse PWR's is included in Reference 8. Detailed background information on the design constraints on local power density in a Westinghouse PWR, on the defined operating procedures, and on the measures taken to preclude exceeding design limits is presented in the Westinghouse topical report on power distribution control and load following procedures.(9) The following paragraphs summarize these reports (8,9) and describe the calculations used to establish the upper bound on peaking factors.

The calculations used to establish the upper bound on peaking factors, F_Q and $F^N_{\Delta H}$, include all of the nuclear effects which influence the radial and/or axial power distributions throughout core life for various modes of operation, including load follow, reduced power operation, and axial xenon transients.

Radial power distributions are calculated for the full power condition, and fuel and moderator temperature feedback effects are included in these calculations. The steady state nuclear design calculations are done for normal flow with the same mass flow in each channel and flow redistribution effects neglected. The effect of flow redistribution is calculated explicitly where it is important in the DNB analysis of accidents. The effect of xenon on radial power distribution is small but is included as part of the normal design process. Radial power distributions are relatively fixed and easily bounded with upper limits.

The core average axial profile, however, can experience significant changes which can occur rapidly as a result of rod motion and load changes and more slowly due to xenon distribution. For the study of points of closest approach to axial power distribution limits, several thousand cases are examined. Since the properties of the nuclear design dictate what axial shapes can occur, boundaries on the limits of interest can be set in terms of the plant parameters which are readily observed. Specifically, the nuclear design parameters which are significant to the axial power distribution analysis are:

1) Core power level
2) Core height
3) Coolant temperature and flow
4) Coolant temperature program as a function of reactor power
5) Fuel cycle lifetimes
6) Rod bank worths
7) Rod bank overlaps

Normal operation of the plant assumes compliance with the following conditions:

1) Control rods in a single bank move together with no individual rod insertion differing by more than the rod group alignment limits specified in the Technical Specifications.
2) Control banks are sequenced with overlapping banks
3) The control bank insertion limits are not violated
4) Axial power distribution procedures, which are given in terms of flux difference control and control bank position, are observed.

The axial power distribution procedures referred to above are part of the required operating procedures which are followed in normal operation. Briefly, they require control of the axial offset (flux difference divided by fractional power) at all power levels within a permissible operating band of a target value corresponding to the equilibrium full power value. In the first cycle, the target value changed from about +10 to –3 percent linearly through the life of the cycle. This minimized xenon transient effects on the axial power distribution, since the procedures essentially kept the xenon distribution in phase with the power distribution.

Calculations are performed for normal operation of the reactor including load following maneuvers. Beginning, middle and end of cycle conditions are included in the calculations. Different histories of operation are assumed prior to calculating the effect of load follow transients on the axial power distribution. These different histories assume base loaded operation and extensive load following. For a given plant and fuel cycle, a finite number of maneuvers are studied to determine the general behavior of the local power density as a function of core elevation.

These cases represent many possible reactor states in the life of one fuel cycle, and they have been chosen as sufficiently definitive of the cycle by comparison with much more exhaustive studies performed on some 20 or 30 different, but typical, plant and fuel cycle combinations. The cases are described in detail in Reference 9 and they are considered to be necessary and sufficient to generate a local power density limit which, when increased by 5 percent for conservatism, will not be exceeded with a 95 percent confidence level. Many of the points do not approach the limiting envelope. However, they are part of the time histories which lead to the hundreds of shapes which do define the envelope. They also serve as a check that the reactor studied is typical of those more exhaustively studied.

Thus, it is not possible to single out any transient or steady-state condition which defines the most limiting case. It is not even possible to separate out a small number which form an adequate analysis. The process of generating a myriad of shapes is essential to the philosophy that leads to the required level of confidence. A maneuver which provides a limiting case for one reactor fuel cycle is not necessarily a limiting case for another reactor or fuel cycle with different control bank worths, enrichments, burnups, coefficients, etc. Each shape depends on the detailed history of operation up to that time and on the manner in which the operator

conditioned xenon in the days immediately prior to the time at which the power distribution is calculated.

The calculated points are synthesized from axial calculations combined with radial factors appropriate for rodded and unrodded planes in the first cycle. In these calculations, the effects on the unrodded radial peak of xenon redistribution that occurs following the withdrawal of a control bank (or banks) from a rodded region is obtained from two-dimensional X-Y calculations. A 1.03 factor to be applied on the unrodded radial peak was obtained from calculations in which xenon distribution was preconditioned by the presence of control rods and then allowed to redistribute for several hours. A detailed discussion of this effect may be found in Reference 9. The calculated values have been increased by a factor of 1.05 for conservatism (75% available thimbles), and by a factor of 1.03 for the engineering factor F_Q^E.

The envelope drawn over the calculated points in Figure 3.2-5 for Cycle 1 operation and in Figure 3 in the COLR for the current cycle represents an upper bound envelope on local power density versus elevation in the core. It should be emphasized that this envelope is a conservative representation of the bounding values of local power density. Expected values are considerably smaller and, in fact, less conservative bounding values may be justified with additional analysis or surveillance requirements. Additionally, the figures are based on a radial power distribution invariant with core elevation. Finally, as previously discussed, this upper bound envelope is based on procedures of load follow which require operation within an allowed deviation from a target equilibrium value of axial flux difference. These procedures are detailed in the Technical Specifications and are followed by relying only upon excore surveillance supplemented by the normal full core map requirement at every effective full power month, and by computer based alarms or manual logging on deviation and time of deviation from the allowed flux difference band.

To determine reactor protection system setpoints with respect to power distributions, three categories of events are considered, namely rod control equipment malfunctions, operator errors of commission and operator errors of omission. In evaluating these three categories of events, the core is assumed to be operating within the four constraints described below:

1) Control rods in a single bank move together with no individual rod insertion differing by more than the rod group alignment limits specified in the Technical Specifications.
2) Control banks are sequenced with overlapping banks
3) The control bank insertion limits are not violated
4) Axial power distribution procedures, which are given in terms of flux difference control and control bank position, are observed.

The first category comprises uncontrolled rod withdrawal (with rods moving in the normal bank sequence). Also included are motions of the banks below their insertion limits, which could be caused, for example, by uncontrolled dilution or primary coolant cooldown. Power distributions are calculated throughout these occurrences assuming short-term corrective action, that is, no transient xenon effects are considered to result from the malfunction. The event is assumed to occur from typical normal operating situations which include normal xenon transients. It is further assumed in determining the power distributions, that total core power level will be limited by reactor trip to below 120 percent. Since the study is to determine protection limits with respect to power and axial offset, no credit is taken for trip setpoint reduction due to flux difference. The peak power density which can occur in such events, assuming reactor trip at or

below 120 percent, is less than that required for centerline melt, including uncertainties and densification effects.

The second category assumes that the operator mispositions the rod bank in violation of the insertion limits and creates short term conditions not included in normal operating conditions.

The third category assumes that the operator fails to take action to correct a flux difference violation. The resulting F_Q is multiplied by an appropriate allowance for calorimetric error. It should be noted that a reactor overpower accident is not assumed to occur coincident with an independent operator error.

Analyses of possible operating power shapes show that the appropriate hot channel factors F_Q and $F^N_{\Delta H}$ for peak local power density and for DNB analysis at full power are the values addressed in the COLR.

F_Q can be increased with decreasing power as shown in the Technical Specifications. Increasing $F^N_{\Delta H}$ with decreasing power is permitted by the DNB protection setpoints and allows radial power shape changes with rod insertion to the insertion limits. It has been determined that, provided the above conditions 1) through 4) are observed, the COLR limits are met.

When a situation is possible in normal operation which could result in local power densities in excess of those assumed as the precondition for a subsequent hypothetical accident but which would not itself cause fuel failure, administrative controls and alarms are provided for returning the core to a safe condition.

Reactivity Coefficients

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

Moderator Temperature Coefficient

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

In order to reduce the dissolved poison requirement for control of excess reactivity, burnable poison rods have been incorporated in the core design. The result is that changes in the coolant density will have less effect on the density of poison, and the moderator temperature coefficient will be reduced.

The burnable poison for Cycle 1 was in the form of borated Pyrex glass rods clad in stainless steel. There were 1434 of these borated Pyrex glass rods in the form of clusters distributed throughout the initial core in vacant rod cluster control guide tubes, as illustrated in Figures, 3.2-6 through 3.2-7. Information regarding research, development and nuclear evaluation of the burnable poison rods can be found in Reference 1. These rods initially controlled 10% $\Delta\rho$ of the installed excess reactivity and their insertion into the core resulted in a reduction of the initial hot zero power boron concentration in the coolant to 1330 ppm. The moderator temperature coefficient is negative at operating conditions with burnable poison rods installed. Subsequent cycles utilized B_4C in AL_2O_3 in wet annular burnable absorbers (WABA) and ZrB_2 in Integral Fuel Burnable Absorbers (IFBA) and also the already mentioned Pyrex rods.

The effect of burnup on the moderator temperature coefficient is periodically calculated, and the coefficient becomes more negative with increasing burnup. This is due to the buildup of fission products with burnup, and dilution of the boric acid concentration with burnup. The reactivity loss due to equilibrium xenon is controlled by boron, and as xenon builds up boron is taken out. The calculated net effect and the predicted moderator temperature coefficient at equilibrium xenon for Cycle 1 and at full power BOL was $-0.84 \times 10^{-4}/^{\circ}F$. With core burnup, the coefficient became more negative as boron was removed because of a shift in the neutron energy spectrum due to the buildup of plutonium and fission products. At Cycle 1 EOL with no boron or rods in the core, the moderator coefficient was $-3.5 \times 10^{-4}/^{\circ}F$. Reference 72 provides the current cycle moderator temperature coefficient.

Moderator Pressure Coefficient

The moderator pressure coefficient has an opposite sign to the moderator temperature coefficient. Its effect on core reactivity and stability is small because of the small magnitude of the pressure coefficient, a change of 50 psi in pressure having no more effect on reactivity than half-degree change in moderator temperature. The calculated Cycle 1 BOL and EOL pressure coefficients are specified in Table 3.2-1, Line 43.

Moderator Density Coefficient

A uniform moderator density coefficient is defined as a change in the neutron multiplication* per unit change in moderator density. The range of the moderator density coefficient for Cycle 1 from BOL and EOL is specified in Table 3.2-1, Line 44.

Doppler and Power Coefficients

The Doppler coefficient is defined as the change in neutron multiplication* per degree change in fuel temperature. The coefficient was obtained in the past by calculating neutron multiplication as a function of effective fuel temperature by the code LEOPARD.[4] The results for Cycle 1 are shown in Figure 3.2-11. More recent calculations of Doppler and power coefficients are performed using the ANC[84] code.

> *NOTE: Neutron multiplication is defined at the ratio of the average number of neutrons produced by fission in each generation to the total number of corresponding neutrons absorbed.

In order to know the change in reactivity with power, it is necessary to know the change in the effective fuel temperature with power as well as the Doppler coefficient. It is very difficult to predict the effective temperature of the fuel using a conventional heat transfer model because of uncertainties in predicting the behavior of the fuel pellets. Therefore, an empirical approach is taken to calculate the power coefficient, based on operating experience of existing Westinghouse cores. Figure 3.2-12 shows the power coefficient as a function of power for Cycle 1 obtained by this method. The results presented do not include any moderator coefficient even though the moderator temperature changes with power level.

3.2.1.2 Nuclear Evaluation

The basis for confidence in the procedures and design methods comes from the comparison of these methods with many experimental results. These experiments include criticals performed at the Westinghouse Reactor Evaluation Center (WREC) and other facilities, and also measured data from operating power reactors. A summary of the results and discussion of the agreement between calculated and measured values is given in other Safety Analysis Reports such as the FSAR for Indian Point 2, Docket No. 50-247.

3.2.1.3 Enrichment Error

Any error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes that are more peaked than those calculated with the correct enrichments. There is an 8% uncertainty margin between the calculated worst value and the design value of power peaking assumed for the analysis of normal steady state operation and anticipated transients. The incore system of moveable flux detectors, used to verify power shapes at start of life, is capable of revealing any enrichment error or loading error which causes power shapes to be peaked in excess of the design value. These power shape measurements are taken at low power when extremely adverse power shapes can be tolerated.

An analysis of the effect of an inadvertent loading of an assembly with an enrichment increased by 20% over the nominal value, showed that the error was detectable at many of the detector locations in the core. In the case of a centrally placed assembly with this enrichment error, five flux detectors would show a signal more than 5% above the expected value. If the assembly bearing the enrichment error is placed off-center and as far from a flux detector as possible, the tilt caused by a 20% error in enrichment is detectable in more than half of the detector locations in the core, both as a flux increase over expected symmetric values and as a flux decrease on the opposite side of the core.

If the movable detector system fails to detect an enrichment error, then the power shapes are such that there is margin to the design conditions, and normal plant operation may be safely continued. It is incredible than any positive indication of power shape anomalies which are sufficiently large to cause a significant departure from design conditions, would be ignored.

These measurements are an integral part of the physics startup tests when considerable emphasis is placed on obtaining good power shape measurements.

These considerations, together with the fuel handling procedures described in Section 3.3, preclude power operation in the presence of any significant fuel enrichment error.

3.2.2 Thermal and Hydraulic Design and Evaluation

A large amount of material has been retained in this section as historical background. The thermal and hydraulic design parameters at the stretched power uprate conditions are provided in Table 3.2-4.

DNB Design Basis

There will be at least a 95 percent probability that departure from nucleate boiling (DNB) will not occur on the limiting fuel rods during normal operation and operational transients and any transient conditions arising from faults of moderate frequency (Condition I and II events), at a 95 percent confidence level. Historically, this has been conservatively met by adhering to the following thermal design basis: there must be at least a 95 percent probability that the minimum departure from nucleate boiling ration (DNBR) of the limiting power rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The DNBR limit for the correlation is established based on the variance of the correlation such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the calculated DNBR is at the DNBR limit.

DNB Analysis Method

The Westinghouse version of the VIPRE-01 (VIPRE) code was used to perform the thermal/hydraulic calculations for both the mini-uprate and stretched power uprate programs. The VIPRE code is equivalent to the THINC-IV (THINC) code and has been approved by the NRC for licensing applications to replace the THINC code. The use of VIPRE is in full compliance with the conditions specified in the NRC Safety Evaluation Report (SRE) on WCAP-14565-P-A(78).

The design method employed for both fuel types to meet the DNB design basis is the Revised Thermal Design Procedure (RTDP)(79). With the RTDP methodology, uncertainties in plant operating parameters, nuclear and thermal parameters, fuel fabrication parameters, computer codes and DNB correlation predictions are considered statistically to obtain DNB uncertainty factors. Based on the DNB uncertainty factors, RTDP design limit DNBR values are determined the most limiting fuel rod during normal operation and operational transients and during transient conditions arising from faults of moderate frequency (Condition I and II events as defined in ANSI N18.2)

To maintain DNBR margin to offset DNB penalties such as those due to fuel rod bow and potential transition core, the safety analyses were performed to DNBR limits higher than the design limit DNBR values. The difference between the design limit DNBRs and the safety analysis limit DNBRs result in available DNBR margin. The net DNBR margin, after consideration of all penalties, is available for operation and design flexibility. The Thermal Design Procedure (STDP) is used for those analyses where RTDP is not applicable. In the STDP method the parameters used in analysis are treated in a conservative way from a DNBR standpoint. The parameter uncertainties are applied directly to the plant safety input values to give the lowest minimum DNBR. The limit for STDP is appropriate DNB correlation limit increased by sufficient margin to offset the applicable DNBR penalties.

For this design, the WRB-1 correlation is used for analysis of the 15x15 Upgrade fuel assemblies with a correlation limit of 1.17 (both typical and thimble cells). When the core condition is outside the range of the WRB-1 correlation, the W-3 correlation is applied with a correlation limit of 1.30 (both cell types) with pressure greater than 1000 psia.

DNB With Physical Burnout

Westinghouse [(29)] has conducted DNB tests in a 25-rod bundle where physical burnout occurred with one rod. After this occurrence, the 25 rod test section was used for several days to obtain more DNB data from the other rods in the bundle. The burnout and deformation of the rod did not affect the performance of neighboring rods in the test section during the burnout or the validity of the subsequent DNB data points as predicted by the W-3 correlation. No occurrences of flow instability or other abnormal operation were observed.

DNB With Return to Nucleate Boiling

Additional DNB tests have been considered by Westinghouse[(30)] in 19 and 21 rod bundles. In these tests, DNB without physical burnout was experienced more than once on single rods in the bundle for short periods of time. Each time, a reduction in power of approximately 10% was sufficient to re-establish nucleate boiling on the surface of the rod. During these and subsequent tests, no adverse effects were observed on this rod or any other rod in the bundle as a consequence of operating in DNB.

Hydrodynamic and Flow Power Coupled Instability

The interaction of hydrodynamic and spatial effects has been considered and it is concluded that a large margin exists between the design conditions and those for which an instability is possible.

Heated channels in parallel can lead to flow instability. If substantial boiling takes place, periodic flow instabilities have been observed and, as long ago as 1938, Ledinegg[(24)] proposed a stability criterion on the basis of which the concept of inlet orificing has been developed to stabilize flow. More recent work[(25-27)] has demonstrated that periodic instabilities are possible which violate the Ledinegg criterion.

In normal flow channels with little or no boiling, the type of instability proposed by Ledinegg is not possible since it results primarily from the large changes in water density along the channel due to boiling. Moreover, the periodic instabilities examined by Quandt[(25-26)] and Meyer[(27)] are not exhibited in non-boiling channels of the type found in PWR cores. [(28)]

3.2.2.1 Thermal and Hydraulic Characteristics of the Design

Central Temperature of the Hot Pellet

The temperature distribution in the pellet is mainly a function of the uranium dioxide thermal conductivity and the local power density. The surface temperature of the pellet is affected by the cladding temperature and the thermal conductance of the gap between the pellet and the cladding.

The occurrence of nucleate boiling maintains maximum cladding surface temperature below about 657 °F at nominal system pressure. The contact conductance between the fuel pellet and cladding is a function of the contact pressure and the composition of the gas in the gap [11, 12], and may be calculated by the following equation:

$$h = 0.6P + k/ f(14.4 \times 10^{-6})$$

where

h = conductance in Btu/hr-ft^2-°F
P = contact pressure in psi
k = the thermal conductivity of the gas mixture in the rod
f = the correction factor for the accommodation coefficient

The thermal conductivity of uranium dioxide was evaluated from published results of work at ORNL [13] Chalk River [14], and WAPD. [15] The design curve for thermal conductivity is given in Figure 3.2-13. The section of the curve at temperatures between 0 °F and 3000 °F is based on the data of Godfrey, et al. [13]

The section of the curve between 3000 °F and 5000 °F was based on two factors:

1) Inpile observations of fuel melting dictate a positive temperature coefficient for conductivity above approximately 3000 °F, The temperature dependence in this range should conform to an exponential curve, since this reflects the most credible physical interpretation of the high temperature conductivity increase.

2) The area under the recommended curve is such that the integral is equal to approximately 97 w/cm as given by Robertson, et at [14] and Duncan. [15] This value is based upon the interpretation of fuel melt radius as determined at Hanford[16] and Chalk River. [14]

Thermal conductivity can be represented best by the following equation:

$$k = (11.8 + 0.0238T)^{-1} + 8.775x10^{-13}T^3$$
with k in w/cm- °C for 95 percent theoretical density and T in °C.

Based upon the above considerations, the maximum central temperature of the hot pellet at steady state nominal and overpower conditions are shown in Table 3.2-4. The temperature is well below the melting temperatures of the irradiated UO_2 (Refer to Criterion 6 in section 3.1.2), which is taken as the unirradiated fuel temperature of 5080°F[17] decreasing by 58°F per 10,000 MWd/metric ton uranium and covering the manufacturing/modeling uncertainties.

Fuel Thermal Performance

Ability to predict fuel performance at high burnup is based on both published data and proprietary data. Effects of irradiation on UO_2 melting point, fuel swelling, fission gas release, clad creep, clad yield strength, etc., have been incorporated into a computer program to enable the prediction of fuel performance. The proof of performance was determined by the Saxton and Zorita programs. Saxton fuel test rods operated to about 42,000 MWD/MTM peak burnup, and Zorita fuel test rods operated to about 32,000 MWD/MTM, have been used in verifying

performance models. In 1972, the Saxton lead rod accumulated additional burnup to about 53,000 MWD/MTM at the end of the Core III operation. Zorita test rods were exposed to about 45,000 MWD/MTM at the end of second cycle operation in mid 1972. In addition, special removable rod assemblies have been irradiated in the Zion, Surry Unit 1 and 2, and Trojan plants. These assemblies have provisions for removable rods which permit non-destructive examination of fuel rods during refueling shutdowns. Two Zion 15 x 15 high burnup test assemblies have been irradiated for five fuel cycles and have achieved assembly burnups of about 55,000 MWD/MTM. Profilometry measurements after cycle Nos. 1, 2, 3, and 4 have shown less fuel swelling and outward cladding strain than predicted by Westinghouse models. Examination of removable rod 17 x 17 assemblies irradiated in Surry Units 1 and 2 for up to four cycles have shown that cladding creep down is somewhat less than predicted by Westinghouse models.

Applying best estimate models to Indian Point 3 Region III peak burnup rod, the cladding strain damage limit was calculated to be reached at a power level greater than 21 kW/ft.

In the calculation of the steady-state performance of a nuclear fuel rod, the following interacting factors were considered:

1) Cladding creep and elastic deflection
2) Pellet swelling, thermal expansion, gas release, and thermal properties as a function of temperature
3) Internal pressure as a function of fission gas release, rod geometry, and temperature distribution.

These effects have been combined in a computer code which is considered to be Westinghouse proprietary information. With these interacting factors considered, the code determines the fuel rod performance characteristics for a given rod geometry, power history, and axial power shape. In particular, internal gas pressure, fuel and cladding temperatures, and cladding deflections are calculated. The fuel rod is divided lengthwise into several sections and radially into a number of annular zones. Fuel density changes, cladding stresses, strains and deformations, and fission gas releases are calculated separately for each segment. The effects are integrated to obtain the internal rod pressure.

The gap conductance between the pellet surface and the cladding inner diameter is calculated as a function of the composition and pressure of the gas mixture, and the contact pressure between clad and pellet. After computing the fuel temperature for each pellet's annular zone, the fractional fission gas release is assessed from the diffusion-trapping model described by Weisman, et al.[31] The total amount of gas released is based on the average fractional release within each axial increment and the gas generation rate which in turn is a function of burnup. Finally, the gas released is summed over all axial increments and the pressure is calculated.

The code shows good agreement in fit for a variety of published and proprietary data on fission gas release, fuel temperatures and clad deflections. Included in this spectrum are variations in power, time, fuel density, and geometry.

The worst strain conditions in the fuel rod occur at high burnup after the clad and fuel have reached mechanical equilibrium (cladding and fuel in intimate contact). At this time in life, the tolerances have negligible effect and therefore the linear heat rating is calculated using nominal dimensions.

The new PAD fuel rod performance code used to analyze VANTAGE + reloads contained an improved gap conductance model that calculates lower fuel temperatures. The new PAD code was approved by the NRC[80]. Fuel rod thermal evaluations are performed at rated power, maximum overpower, and during transients at various burnups for the stretched power uprate program. These analyses assure that design criteria for reactor core design (criterion 6) given in section 3.1.2 are met.

3.2.2.2 Westinghouse Experience With High Power Fuel Rods

A completed high power test program had the objective of defining failure limits for the combined effects of linear heat generation rate and burnup, providing increased assurance that plants have adequate performance and design margins to the fuel failure threshold, and verifying the adequacy of design methods and computer codes. Results from this program are given in Section 8 of Reference 50. Additional information on fuel rod experience is presented in Reference 51. A summary of the comprehensive experimental program to extend the operating experience to higher power and to higher exposures for fuel rods is provided in Figure 3.2-14.

The figure shows that thirty Saxton Plutonium Project non-pressurized fuel rods have operated at a design peak power level of up to 18.5 kW/ft to a peak exposure of approximately 30,000 MWD per MTM [Megawatt days per metric ton of metal (U + PU)]. No failures have occurred with this fuel. In the Saxton overpower test, two selected fuel rods from the Saxton Plutonium Project assemblies were removed after peak exposure of 18,000 MWD/MTM and inserted in a subassembly for short time irradiation at a design rating of 25 kW/ft. Results of this program indicate satisfactory performance of the fuel in every respect. The Saxton Plutonium Project was extended by irradiating approximately 250 rods to peak burnups of about 50,000 MWD/MTM at design linear power levels ranging from 9.5 to 23.6 kW/ft.

In the above tests (performed on non-pressurized rods), the strain fatigue experienced by the cladding was more severe than expected to occur for pressurized rods which would be placed under identical operating conditions.

Internally pressurized fuel rods have been under investigation[19] at Westinghouse for several years. These investigations include out-of-pile and in-pile experimental programs and analytical studies. Fuel rods internally pressurized with various gases have been irradiated in the Saxton reactor. Tests results show that initial pressurization is effective in substantially reducing the rate of cladding-creep on to the UO_2 fuel. The Saxton test results confirm the results of analyses which predict fuel-cladding mechanical interaction early in life for non-pressurized fuel rods and delayed interaction for initially pressurized fuel rods.

To verify the substantial design margin which exists in the fuel rods with regard to excessive internal pressures in a fuel rod, several highly pressurized Zircaloy-clad fuel rods were irradiated for several months in the Saxton reactor, then removed for examination. At an internal pressure of approximately 3500 psia, the fuel operated satisfactorily for the period of the test without any indication of failure. Two fuel rods, deliberately tested at unrealistically high internal pressures, experienced clad cracking but operated satisfactorily for the period of the test. Thus, even with excessive internal pressures that result in clad failures, the test results are favorable.

3.2.2.3 Sizing of Fuel Rod Plenum

The criterion for sizing the fuel rod plenum length was that the internal gas pressure is limited to a value below which could cause: 1) the diametral gap to increase due to outward cladding creep during steady state operation, and 2) extensive DNB propagation to occur. During operational transients, fuel rod clad rupture due to high internal gas pressure is precluded by meeting the above design basis. The end of life internal gas pressure depends on the initial gas pressure, void volumes (plenum, gap, dish, open porosity, etc.), the amount of fission gases released, and the amount of helium released from IFBA fuel. The estimated fraction of fission gases present in the gap and the plenum was about 20% for the maximum burnup rod (limiting case evaluation) at the end of three cycles of reactor operation.(31)

For the lead rod, the calculated internal pressure is less than the limit value; the clad stress is less than the yield strength of Zircaloy; the clad strain is less than 1% at end of life at normal operating conditions. The ability of fuel to withstand expected transients at the end of life has been evaluated. Such phenomena as high internal gas pressure have been studied under transient conditions and do not present any particular problem or significantly influence fuel behavior during transients.

The limiting criterion for evaluating mechanical performance for power transients is presently conservatively assumed to be yield stress. Both slow and rapid transients have been investigated. Slow transients, e.g., those caused by Xenon oscillations or local power shifts due to depletion, are generally of low magnitude and are of no concern since the resulting stresses will be small and will relax due to creep characteristics of the UO_2 and Zircaloy.

Rather rapid transients and transients of sufficient magnitude to cause high clad stresses could arise from some accidents such as major steam line break accident. Calculations indicate that the yield strength of the cladding could be exceeded in some fuel rods which would experience large power increase during this accident. The clad yield stress will also be exceeded during a loss of coolant accident because the yield strength of Zircaloy decreases to a very small value at high temperatures expected with this accident. It is not possible to preclude some rods from failure during severe transients. However, this is consistent with the plant design basis.

3.2.2.4 Heat Flux Ratio and DNB Correlation

WRB-1 Correlation

The DNB heat flux ratio (DNBR) as applied to typical cells (flow cells with walls heated) and thimble cells (flow cells with heated and unheated walls) is defined as:

$$DNBR = \frac{q''_{DNB,N}}{q''_{loc}}$$

Where:

$$q''_{DNB,N} = \frac{q''_{DNB,EU}}{F}$$

Where:

$q''_{DNB,EU}$ = the uniform heat flux as predicted by the WRB-1 DNB correlation (Ref. 58)

F = the flux shape factor to account for nonuniform axial heat flux distributions (Reference 18) with the "C" term modified as in Reference 32

q''_{loc} = the actual local heat flux.

The WRB-1 (Reference 58) correlation was developed based exclusively on the large bank of mixing vane grid rod bundle CHF data (over 1100 points) that Westinghouse has collected. The WRB-1 correlation, based on local fluid conditions, represents the rod bundle data which has better accuracy over a wide range of variables than the previous correlation used in design. This correlation accounts directly for both typical and thimble cold wall cell effects, uniform and nonuniform heat flux profiles, and variations in rod heated length and in grid spacing.

The applicable range of variables is:

Pressure: $1440 \leq P \leq 2490$ psia

Local Mass Velocity: $0.9 \leq \frac{G_{loc}}{10^6} \leq \frac{3.7 lb}{ft^2 - hr}$

Local Quality: $-0.2 \leq X_{loc} \leq 0.3$

Heated Length, inlet to CHF Location: $L_h \leq 14$ feet

Grid Spacing: $13 \leq g_{sp} \leq 32$ inches

Equivalent Hydraulic Diameter: $0.37 < d_e < 0.60$ inches

Equivalent Heated Hydraulic Diameter: $0.46 < d_h < 0.58$ inches

Figure 3.2-15A shows measured critical heat flux plotted against predicted critical heat flux using the WRB-1 correlation.

As documented in References 81 and 85, a 95/95 limit DNBR of 1.17 is appropriate for 15x15 VANTAGE+ and Upgrade fuel assemblies.

The W-3 DNB Correlation

The W-3 DNB correlation [18 and 32] is used where the primary DNB correlation is not applicable. The WRB-1 correlation is 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 primary correlation. For system pressure in the range of 500 to 1000 psia, the W-3 correlation is 1.45 [59]. For system pressure

greater than 1000 psia, the W-3 correlation is limited to 1.30. A cold wall factor [33] is applied to the W-3 DNB correlation to account for the pressure of the unheated thermal surfaces.

Historical Information of W-3 Correlation

Departure from Nucleate Boiling (DNB) is predicated upon a combination of hydrodynamic and heat transfer phenomena and is affected by the local and upstream conditions, including the flux distribution. In reactor design, the heat flux associated with DNB and the location of DNB are both important. The magnitude of the local fuel rod temperature after DNB depends upon the axial location where DNB occurs.

The W-3 DNB correlation[18] was developed to predict the DNB flux and the location of DNB equally well for a uniform and an axially non-uniform heat flux distribution. This correlation replaced the preceding WAPD q" and H DNB correlations published in Nucleonics[20], May 1963, in order to eliminate the discontinuity of the latter at the saturation temperature and to provide a single unambiguous criterion for the design margin.

The W-3 correlation, and several modifications of it, have been used in Westinghouse critical heat flux (CHF) calculations. The W-3 correlation was originally developed from single tube data[32], but was subsequently modified to apply to the "L" –grid[33] rod bundle data. These modifications to the W-3 correlation have been demonstrated to be adequate for reactor rod bundle design. The W-3 DNB correlation[18] incorporates both local and system parameters in predicting the local DNB heat flux. This correlation includes the nonuniform flux effect and the upstream effect which includes inlet enthalpy or length. The local DNB heat flux ratio (defined as the ratio of the DNB heat flux to the local heat flux) is indicative of the contingency available in the local heat flux without reaching DNB. The sources of the data used in developing this correlation included:

WAPD-188	1958	CU-TR-NO. 1 (NW-208)	1964
ASME PAPER 62-WA-297	1962	CISE-R-90	1964
CISE-R-63	1962	DP-895	1964
ANL-6675	1962	AEEW-R-356	1964
GEAP-3766	1962	BAW-3238-7	1965
AEEW-R213 AND 309	1963	AE-RTL-778	1965
CISE-R-74	1963	AEEW-355	1965
CU-MPR-XIII	1963	EUR-2490.e	1965

The comparison of the measured to predicted DNB flux of this correlation is given in Figure 3.2-15. The local flux DNB ratio versus the probability of not reaching DNB is plotted in Figure 3.2-16. This plot indicates that with a DNBR of 1.3 the probability of not reaching DNB is 95% at a 95% confidence level.

Rod bundle data without mixing vanes agreed very well with the predicted DNB flux as shown in Figure 3.2-17 and rod bundle data with mixing vanes (Figure 3.2-18 show, on the average, an 8% higher value of DNB heat flux than predicted by the W-3 DNB correlation.

It should be emphasized that the inlet subcooling effect of the W-3 correlation was obtained from both uniform and non-uniform data. The existence of an inlet subcooling effect has been demonstrated to be real and hence the actual subcooling should be used in the calculations. The W-3 correlation was developed from tests with flow in tubes and rectangular channels. Good agreement is obtained when the correlation is applied to test data for rod bundles.

Departure from Nucleate Boiling Ratio

The DNB heat flux ratio (DNBR) as applied to the design when all flow cell walls are heated is:

$$\mathrm{DNBR} = \left(q''_{\mathrm{DNB,N}}\right)\left(F'_s\right)(0.986) \Big/ q''_{\mathrm{loc}}$$

where:

$$q''_{\mathrm{DNB,N}} = \left(q''_{\mathrm{DNB,EU}}\right) \Big/ F$$

and:

$q''_{\mathrm{DNB,EU}}$ is the uniform DNB heat flux as predicted by the W-3 DNB correlation (18) when all flow cell walls are heated.

F is the flux shape factor to account for non-uniform axial heat flux distributions [18] with the "C" term modified as in Reference 32.

F'_s is the modified spacer factor which uses an axial grid spacing coefficient, KS = 0.046, and a thermal diffusion coefficient, TDC = 0.019, based on the 26-inch grid spacing data.

q''_{loc} is the actual local heat flux.

The DNBR as applied to this design when a cold wall is present is:

$$\mathrm{DNBR} = \left(q''_{\mathrm{DBN,N,CW}}\right)\left(F'_S\right)\left(0.986\right) \Big/ q_{\mathrm{loc}}$$

where:

$$q''_{\mathrm{DNB,N,CW}} = \left(q''_{\mathrm{DBN,EU,Dh}}\right)\left(\mathrm{CWF}\right) \Big/ F$$

$q''_{\mathrm{DBN,EU,Dh}}$ is the uniform heat flux as predicted by the W-3 cold wall DNB correlation[33] when not all flow cell walls are heated (thimble cold wall cell).

CWF = Cold Wall Factor

<u>Local Non-Uniform DNB Flux</u>

The local non-uniform $q''_{DNB,N}$ is calculated as follows

$$q''_{DNB,N} = \frac{q''_{DNB,EU}}{F}$$

Where:

$$F = \frac{C}{q''_{local\_at\_\ell_{DNB}} * (1 - e^{-C\ell_{DNB}})} \int_0^{\ell_{DNB}} q''(z) e^{-C(\ell_{DNB} - z)} dz \qquad (5)$$

ℓ_{DNB} = distance from the inception of local boiling to the point of DNB, in inches.

Z = distance from the inception of local boiling measured in the direction of flow, in inches.

The empirical constant, C, as presented in Reference 18 has been updated through the use of more recent non-uniform DNB data. However, the revised expression does not significantly influence (less than once percent deviation from that of Reference 18) the value of the F-factor and the DNBR. It does provide a better prediction of the location of DNB. The new expression is:

$$C = 0.15 \exp\left[\frac{(1 - X_{DNB})^{4.31}}{(G/10^6)^{0.478}}\right] \text{ (in)}^{-1} \qquad (6)$$

Where

G = mass velocity, $\frac{lb}{hr - ft^2}$

X_{DNB} = quality of the coolant at the location where DNB flux is calculated.

In determining the F-factor, the value of q'' local at DNB in equation (5) was measured as $Z = \ell_{DNB}$, the location where the DNB flux is calculated. For a uniform flux, F becomes unity so that $q''_{DNB,N}$ reduces to $q''_{DNB,EU}$ as expected. The comparisons of predictions by using W-3 correlations and the non-uniform DNB data obtained by B&W [21], Winfrith [22 & 23] and Fiat are given in Figure 3.2-19 and 3.2-20. The criterion for determining the predicted location of DNB is to evaluate the ratio of the predicted DNB flux to the local heat flux along the length of the channel. The location of the minimum DNB ratio is considered to be the location of DNB.

Procedure for Using W-3 Correlation

In predicting the local DNB flux in a non-uniform heat flux channel, the following two steps are required:

1) The uniform DNB heat flux, $q''_{DNB,EU}$, is computed with the W-3 correlations using the specified local reactor conditions.

2) This equivalent uniform heat flux is converted into corresponding non-uniform DNB heat flux, $q''_{DNB,N}$, for the non-uniform flux distribution in the reactor. This is accomplished by dividing the uniform DNB flux by the F-factor. [8] Since F is generally greater than unity $q''_{DNB,N}$ will be smaller than $q''_{DNB,EU}$.

To calculate the DNBR of a reactor channel, the values of $\left(q''_{DNB,N}/q''_{loc}\right)$ along the channel are evaluated and the minimum value is selected as the minimum DNBR incurred in that channel.

The W-3 correlation depends on both local and inlet enthalpies of the actual system fluid, and the upstream conditions are accommodated by the F-factor. Hence, the correlation provides a realistic evaluation of the safety margin of heat flux.

Application of the W-3 Correlation in Design

During steady state operation at the nominal design conditions, the DNB ratios are determined. Under other operating conditions, particularly overpower transients, more limiting conditions develop than those existing during steady state operation. The DNB correlations are sensitive to several parameters. In addition, thermal flux general under transient conditions is also sensitive to many parameters. Therefore, for each case studied, a conservative combination of the significant parameters is used as an initial condition. These parameters include:

a) Reactor coolant system pressure
b) Reactor coolant system temperature
c) Reactor power (determined from secondary plant calorimetrics)
d) Core power distribution (hot channel factors)

For transient accident conditions where the power level, system pressure and core temperature may increase, the DNBR is limited to a minimum value of 1.30. The Reactor Control and Protection System is designed to prevent any credible combination of conditions from occurring which would result in a lower DNB ratio.

For the W-3 correlation, the 95/95 limit DNBR is 1.30 at system pressure greater than or equal to 1000 psi. For lower pressure application (500-1000 psi), the 95/95 limit DNBR is 1.45 (Reference 59).

3.2.2.5 Film Boiling Heat Transfer Coefficient

Heat transfer after departure from nucleate boiling was conservatively assumed to be limited by film boiling immediately, and the period of transition boiling neglected.

The correlation used to evaluate these film boiling heat transfer coefficients was developed by Tong, Sandberg and Bishop [34] and is shown in Figure 3.2-21.

$$(hD/k)_f = 0.0193(DG/\mu)_f^{0.80}(C_p\mu/k)_f^{1.23}(p_g/p_b)^{0.68}(p_g/p_\ell)^{0.068-}$$

where: $p_b = p_g a + p_\ell(1-a)$
and

Cp = heat capacity at constant pressure, Btu/lb-F

D = equivalent diameter of flow channel, feet

H = heat transfer coefficient. Btu/hr-ft^2-F

G = mass flow rate, lb/hr-ft^2

k = thermal conductivity, Btu/hr-ft-F

a = void fraction

p = density, lbm/ft^3

μ = viscosity, lbm/ft-hr

Subscripts:

g = evaluation of the property at the saturated vapor condition

ℓ = evaluation of the property at the saturated liquid condition

f = evaluation of the property at the average film temperature

w = evaluation of the property at the wall temperature

b = evaluation of the property at the average bulk fluid condition.

The heat transfer correlation was developed for flow rates equal or greater than 0.8 x 10^6 lb/hr/sq ft over a pressure range of 580 to 3190 psia, for qualities as high as 100 percent, and heat flux from 0.1 to 0.65 x 10^6 Btu/hr/sq ft.

3.2.2.6 Hot Channel Factors

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

Definition of Engineering Hot Channel Factor

Each of the total hot channel factors is the product of a nuclear hot channel factor describing the neutron flux distribution and an engineering hot channel factor to allow for variations from design conditions. The engineering hot channel factors account for the effects of flow conditions and fabrication tolerances and are made up of subfactors accounting for the influence of the variations of fuel pellet diameter, density and enrichment, inlet flow distribution, flow redistribution, and flow mixing.

Heat Flux Engineering Subfactor, F_q^E

The heat flux engineering hot channel factor is used to evaluate the maximum linear heat generation rate in the core. This subfactor is determined by statistically combining the fabrication variations for fuel pellet diameter, density and enrichment, and has a value of 1.03 at the 95 percent probability level with 95 percent confidence. No DNB penalty need be taken for the short relatively low intensity heat flux spikes caused by variations in the above parameters, as well as fuel pellet eccentricity and fuel rod diameter variation.[35]

Enthalpy Rise Engineering Subfactor, $F_{\Delta H}^E$

The effect of variations in flow conditions and fabrication tolerances on the hot channel enthalpy rise in reload analysis is directly considered in the Westinghouse version of VIPRE-01 code (VIPRE)[77, 78] thermal subchannel analysis under any reactor operating condition. The items presently considered contributing to the enthalpy rise engineering hot channel factor are discussed below:

Pellet diameter, density and enrichment.

Design values employed in the VIPRE analysis related to the above fabrication variations are based on applicable limiting tolerances such that these design values are met for 95 percent of the limiting channels at a 95 percent confidence level. Measured manufacturing data on Westinghouse fuel show that the tolerances used in this evaluation are conservative. In addition, each fuel assembly is checked to assure the channel spacing design criteria are met. The effect of variations in pellet diameter, enrichment and density is considered in establishing RTDP design limit.

Inlet Flow Maldistribution

Data have been considered from several 1/7 scale hydraulic reactor model tests [36, 37, 38] in arriving at the core inlet flow maldistribution criteria to be used in the subchannel analyses. THINC-1 analyses using these data have indicated that a conservative design basis is to

consider a five percent reduction in the flow to the hot assembly.[39] The design basis of 5% flow reduction to the hot assembly is also used in the VIPRE analysis for the stretch power uprate.

Flow Redistribution

The flow redistribution accounts for the reduction in flow in the hot channel resulting from the high flow resistance in the channel due to the local or bulk boiling. The effect of the flow redistribution is inherently considered in the VIPRE analysis

Flow Mixing

Mixing vanes have been incorporated into the spacer grid design. These vanes induce flow mixing between the various flow channels in a fuel assembly and also between adjacent assemblies. This mixing reduces the enthalpy rise in the hot channel resulting from local power peaking or unfavorable mechanical tolerances. The subchannel mixing model now incorporated in the THINC or VIPRE code and used in reload reactor design is based on experimental data.[40]

3.2.2.7 Core Pressure Drop and Hydraulic Loads

Core and vessel pressure losses are calculated by equations of the form:

$$\Delta P_L = (K + F\frac{L}{D_e})\frac{\rho V^2}{2g_c(144)}$$

where:

ΔP_L = unrecoverable pressure drop, lbf/in^2

ρ = fluid density, lbm/ft lbf/in^3

L = length, feet

D_E = equivalent diameter, feet

V = fluid velocity, ft/sec

g_c = 32.174 lbm-ft/lbf-sec^2

K = form loss coefficient, dimensionless

F = friction loss coefficient, dimensionless

Fluid density is assumed to be constant at the appropriate value for each component in the core and vessel. Because of the complex core and vessel flow geometry, precise analytical values for the form and friction loss coefficients are not available. Therefore, experimental values for these coefficients are obtained from geometrically similar models.

The results of full scale tests of core components and fuel assemblies are utilized in developing the core pressure loss characteristic in reload reactor design. The pressure drop for the vessel has been obtained by combining the core loss with correlation of 1/7 scale model hydraulic test data on a number of vessels [(36)(37)] and form loss relationships. [(41)] Moody [(42)] curves have been used to obtain the single phase friction factors.

The fuel assembly holddown springs were designed to keep the fuel assemblies in contact with the lower core plate under all Condition I and II events, with the exception of the turbine overspeed transient associated with a loss of external load. The holddown springs were designed to tolerate the possibility of an overdeflection associated with fuel assembly liftoff for this case and provide contract between the fuel assembly and the lower core plate following this transient. More adverse flow conditions occur during a Loss-of-Coolant Accident.

Hydraulic loads at normal operating conditions are calculated considering the best estimate flow and accounting for the minimum core bypass flow based on manufacturing tolerances. Core hydraulic loads at cold plant startup conditions are based on the best estimate flow, but are adjusted to account for the coolant density difference. Conservative core hydraulic loads for a pump overspeed transient, which could possibly create flow rates 20 percent greater than the best estimate flow, are evaluated to be approximately twice the fuel assembly weight. The hydraulic forces are not sufficient to lift a rod control cluster during normal operation even if the rod cluster is detached from its coupling.

3.2.3 Mechanical Design and Evaluation

The reactor core and reactor vessel internals are shown in cross-section in Figure 3.2-22 and in elevation in Figure 3.2-23. The core, consisting of the fuel assemblies, control rods, source rods, burnable poison rods, and guide thimble plugging devices, provides and controls the heat source for the reactor operation. The internals, consisting of the upper and lower core support structure, were designed to support, align, and guide the core components, direct the coolant flow to and from the core components, and to support and guide the in-core instrumentation. A listing of the core mechanical design parameters is given in Table 3.2-5.

The fuel assemblies are arranged in a roughly checkered circular and/or zoned pattern. The assemblies are all 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 144 inches within Zircaloy-4 or ZIRLO™ tubular cladding which is plugged and seal welded at the ends to encapsulate the fuel. The fuel rods are internally pressurized with helium during fabrication. The enrichments of the fuel for the first three regions and current cycle in the core are given in Table 3.2-5. Heat generated by the fuel is removed by demineralized light water which flows upward through the fuel assemblies and acts as both moderator and coolant.

The core is divided into fuel assembly regions of different enrichments. The loading arrangement for the initial cycle is indicated on Figure 3.2-24. Refueling originally took place generally in accordance with an inward loading schedule, but now a modified loading schedule for low - low leakage core design is used.

The control rods, designated as Rod Cluster Control Assemblies (RCCA), 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 assemblies and occupy locations within the regular fuel rod pattern where fuel rods have been deleted. In the withdrawn position, the absorber rods are guided and supported laterally by guide tubes which form an integral part of the upper core support structure. Figure 3.2-25 shows a typical rod cluster control assembly.

As shown in Figure 3.2-23 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. During the refueling outage for Cycle 8/9, the upper core plate was modified such that locations A5 and A11 were each missing one pin and location B13 was missing both pins. A special analysis was performed by Westinghouse showing that this configuration was acceptable for normal and design basis event operation. This analysis conservatively assumed absence of both alignment pins in these three locations, and also for location A6, which has two intact pins, one of which has been straightened. [71]

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.

Reactor Internals

Design Description

The reactor internals were designed to support and orient the reactor core fuel assemblies and control rod assemblies, absorb the control rod dynamic loads and transmit these and other loads to the reactor vessel flange, provide a passageway for the reactor coolant, and support incore instrumentation. The reactor internals are shown in Figure 3.2-23.

The internals were designed to withstand the forces due to weight, preload of fuel assemblies, control rod dynamic loading, vibration, and earthquake acceleration. The internals were analyzed in a manner similar to that employed for Connecticut Yankee, San Onofre, Zorita, Saxton and Yankee. Under the loading conditions specified, which included conservative effects of design earthquake loading, the structure satisfied stress values prescribed in Section III, ASME Nuclear Vessel Code.

The reactor internals were fabricated primarily from type 304 stainless steel.

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

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

Lower Core Support Structure

The major containment and support member of the reactor internals is the lower core support structure, shown in Figure 3.2-23. This support structure assembly consists of the core barrel, the core baffle, and lower core plate and support columns, the thermal shield, the intermediate diffuser plate and the bottom support plate which is welded to the core barrel. All the major material for this structure is type 304 stainless steel. The core support structure is supported at its upper flange from a ledge in the reactor vessel head flange and its lower end is restrained in its transverse movement by a radial support system attached to the vessel wall. Within the core barrel are axial baffle and former plates which are attached to the core barrel wall and form the enclosure periphery of the assembled core. The lower core plate is positioned at the bottom level of the core below the baffle plates and provided support and orientation for the fuel assemblies.

The lower core plate is a 2-inch thick member through which the necessary flow distributor holes for each fuel assembly were machined. Fuel assembly locating pins (two for each assembly) are also inserted into this plate. Columns were placed between this plate and the bottom support plate of the core barrel in order to provide stiffness and to transmit the core load to the bottom support plate. Intermediate between the support plate and lower core support plate was positioned a perforated plate to diffuse uniformly the coolant flowing into the core.

The one-piece thermal shield is fixed to the core barrel at the top with rigid bolted connections. The bottom of the thermal shield is connected to the core barrel by means of axial flexures. This bottom support allows for differential axial growth of the shield/core barrel but restricts radial or horizontal movement of the bottom of the shield. Rectangular tubing in which material samples can be inserted and irradiated during reactor operation are welded to the thermal shield and extend to the top of the thermal shield. These samples are held in the rectangular tubing by a preloaded spring device at the top and bottom.

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

A small amount of water also flows between the baffle plates and core barrel to provide additional cooling of the barrel. Similarly, a small amount of the entering flow is directed into the vessel head plenum to provide cooling of the head. Both these flows eventually are directed into the upper support structure plenum and exit through the vessel outlet nozzles.

Vertically downward loads from weight, fuel assembly preload, control rod dynamic loading and earthquake acceleration are carried by the lower core plate partially into the lower core plate support flange on the core barrel shell, and partially through the lower support columns to the lower core support and thence through the core barrel shell to the core barrel flange supported by the vessel head flange. Transverse loads from earthquake acceleration, coolant cross flow, and vibration are carried by the core barrel shell to be distributed to the lower radial support to the vessel wall, and to the core barrel flange. Transverse acceleration of the fuel assemblies is transmitted to the core barrel shell by direct connection of the lower core plate to the barrel wall and by a radial support type connection of the upper core plate to slab sided pins pressed into the core barrel.

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

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

In the event of downward vertical displacement of the internals, energy absorbing devices limit the displacement by contacting the vessel bottom head. The load is transferred through the energy devices of the internals.

The energy absorbers, which are cylindrical in shape, are contoured on their bottom surface to the reactor vessel bottom head geometry. Their number and design were determined so as to limit the forces imposed to less than yield. Assuming a downward vertical displacement the potential energy of the system is absorbed mostly by the strain energy of the energy absorbing devices.

The free fall in the hot condition is on the order of ½ inch, and there is an additional strain displacement in the energy absorbing devices of approximately ¾ inch. Alignment features in the internals prevent cocking of the internals structure during this postulated drop. The control rods are designed to provide assurance of control rod insertion capabilities under this assumed drop of internals condition. The drop distance of about 1 ¼ inch is not enough to cause the tips of the shutdown group of RCC assemblies to come out of the guide tubes in the fuel assemblies.

<u>Upper Core Support Assembly</u>

The upper core support assembly, shown in Figure 3.2-28, consists of the top support plate, deep beam sections, and upper core plate between which are contained 48 support columns and 61 guide tube assemblies. The support columns establish the spacing between the top support plate, deep beam sections, and the upper core plate and are fastened at top and bottom to these plates and beams. The support columns transmit the mechanical loadings between the two plates and serve the supplementary function of supporting thermocouple guide tubes. The

guide tube assemblies, shown on Figure 3.2-29, sheath and guide the control rod drive shafts and control rods and provide no other mechanical functions. They are fastened to the top support plate and are guided by pins in the upper core plate for proper orientation and support. Additional guidance for the control rod drive shafts is provided by the control rod shroud tube which is attached to the upper support plate and guide tube.

The upper core support assembly, which is removed as a unit during refueling operation, is positioned in its proper orientation with respect to the lower support structure by flat-sided pins pressed into the core barrel which in turn engage in slots in the upper core plate. At an elevation in the core barrel where the upper core plate is positioned, the flat-sided pins are located at angular positions of 0°, 90°, 180°, and 270°. Four slots are milled into the core plate at the same positions. As the upper support structure is lowered into the main internals, the slots in the plate engage the flat-sided pins in the axial direction. Lateral displacement of the plate and of the upper support assembly is restricted by this design. Fuel assembly location pins protrude from the bottom of the upper core plate and engage the fuel assemblies as the upper assembly is lowered into place. Proper alignment of the lower core support structure, the upper core support assembly, the fuel assemblies and control rods is thereby assured by this system of locating pins and guidance arrangement. The upper core support assembly is restrained from any axial movements by a large circumferential spring which rests between the upper barrel flange and the upper core support assembly and is compressed by the reactor vessel head flange.

Vertical loads from weight, earthquake acceleration, hydraulic loads and fuel assembly preload are transmitted through the upper core plate via the support columns to the deep beams and top support plate and then the reactor vessel head. Transverse loads form coolant cross flow, earthquake acceleration, and possible vibrations are distributed by the support columns to the top support plate and upper core plate. The top support plate is particularly stiff to minimize deflection.

Incore Instrumentation Support Structures

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

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

In addition to the upper in-core instrumentation, there are reactor vessel bottom port columns which carry the retractable, cold worked stainless steel flux thimbles that are pushed upward into the reactor core. Conduits extend from the bottom of the reactor vessel down through the concrete shield area and up to a thimble seal line. The minimum bend radii are about 144 inches and the trailing ends of the thimbles (at the seal line) are extracted approximately 15 feet during refueling of the reactor in order to avoid interference within the core. The thimbles are closed at the leading ends and serve as the pressure barrier between the pressurized water and the containment atmosphere.

Mechanical seals between the retractable thimbles and the conduits are provided at the seal line. During normal operation, the retractable thimbles are stationary and move only during refueling or for maintenance, at which time a space of approximately 15 feet above the seal line is cleared for the retraction operation. Section 7.4 contains more information on the layout of the incore instrumentation system.

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

Evaluation of Core Barrel and Thermal Shield

The internals design was based on analysis, test and operational information. Troubles in previous Westinghouse PWR's were evaluated and information derived was considered in this design. For example, the Westinghouse design uses a one-piece thermal shield which is attached rigidly to the core barrel at one end and flexured at the other. The early designs that malfunctioned were multi-piece thermal shields that rested on vessel lugs and were not rigidly attached at the top.

Early core barrel designs that have malfunctioned in service, now abandoned, employed threaded connections such as tie rods, joining the bottom support to the bottom of the core barrel, and a bolted connection that tied the core barrel to the upper barrel. The malfunctioning of core barrel designs in earlier service was believed to have been caused by the thermal shield which was oscillating, thus creating forces on the core barrel. Other forces were induced by unbalanced flow in the lower plenum of the reactor. In the Indian Point 3 RCC design there are no fuel followers to necessitate a large bottom plenum in the reactor. The elimination of these fuel followers enabled Westinghouse to build a shorter core barrel.

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

Substantial scale model testing was performed at WAPD. This included tests which involved a complete full scale fuel assembly which was operated at reactor flow, temperature and pressure conditions. Tests were run on a 1/7 scale model of the Indian Point 3 reactor. Measurements taken from those tests indicated very little shield movement, on the order of a few mils when scaled up to Indian Point. Strain gauge measurements taken on the core barrel also indicated very low stresses. Testing to determine thermal shield excitation due to inlet flow disturbances was included. Information gathered from these tests was used in the design of the thermal shield and core barrel. It was concluded from the testing program and the analyses with the

experience gained that the design as employed on the Indian Point Plant is adequate. Further confirmation of the internals design was made on Indian Point 2. Deflection gauges were mounted on the thermal shield top and bottom for the hot-functional test. Six such gauges were mounted in the top of the thermal shield equidistant between the fixed supports and eight located at the bottom, equidistant between the six flexures, and two next to flexure supports. The internals inspection, just before the hot-functional test, included looking at mating bearing surfaces, main welds, and welds that are used on bolt locking devices. At the conclusion of the hot-functional test, measurement readings were taken from the deflectometers on the shield and the internals were re-examined at all key areas for any evidence of malfunction.

The final report of the Indian Pont 2 vibrational test, Reference 52, was transmitted to the Deputy Director for Reactor Projects in August, 1972. This report supports the use of Indian Point 2 internals as the prototype for Indian Point 3 internals.

<u>Core Components</u>

<u>Design Description</u>

Fuel Assembly

The original fuel design for Indian Point 3 was the Westinghouse Low Parasitic (LOPAR) fuel assembly. For the Cycle 5 reload, a new design fuel, the Westinghouse Optimized Fuel Assembly (OFA) was introduced (and LOPAR was phased out by Cycle 7). The major design difference between the two designs is the use of the five middle Zircaloy grids for the new design versus five middle Inconel grids for the old design. (Reference 54)

For the Cycle 7 reload, another new design, VANTAGE-5, was introduced (and by Cycle 11, the only OFA assembly was the central assembly).

For the Cycle 9 reload, ZIRLO™ clad was introduced and continues to be used.

For the Cycle 10 reload, a third design, VANTAGE+ was introduced.

For the Cycle 11 reload, the VANTAGE+ design was enhanced to include “PERFORMANCE+” features. This enhanced design has all the features of the VANTAGE+ design and also includes the Protective Bottom Grid (Reference 76).

Beginning with the Cycle 14 core, the design known as 15x15 Upgrade was introduced. This design includes an enhanced grid and IFMs to reduce grid-rod fretting. [See Section 3.2.5.5]

The overall configuration of the fuel assemblies is shown in Figures 3.2-30 and 3.2-31. The assemblies are square in cross-section, nominally 8.426 inches on a side and have an overall height of 13 feet 4 inches.

<u>VANTAGE 5 Fuel</u>

The Vantage 5 Fuel assembly has been designed to be compatible with the OFAs, reactor internals interfaces, the fuel handling equipment, and refueling equipment. The VANTAGE 5 design dimensions are essentially equivalent to the IP3 OFA assembly design from an exterior assembly envelope and reactor internals interface standpoint. (Reference 73)

The significant new mechanical features of the VANTAGE 5 design relative to the OFA design in operation include the following:

- Integral Fuel Burnable Absorber (IFBA)
- Reconstitutable Top Nozzle
- Slightly longer fuel rod and assembly for extended burnup capability
- Axial Blankets
- Redesigned fuel rod bottom end plug to facilitate reconstitution capability

Other different mechanical features are the use of a standardized chamfer pellet design and the Debris Filter Bottom Nozzle (DFBN).

<u>VANTAGE + Fuel</u>

Vantage + uses the following V5 features

- Reconstitutable Top Nozzle (RTN)
- Extended Burnup Fuel Assembly Design
- Extreme Low Leakage Loading Pattern
- Enriched Integral Fuel Burnable Absorbers (IFBAs)
- [Deleted] DFBN
- Axial Blankets

In addition V+ incorporates the following features as described in Reference 74

- ZIRLO™ Fuel Cladding
- Low Pressure Drop (LPD) Mid-grids
- Integral Flow Mixer grids (IFMs)
- ZIRLO™ guide thimbles and instrumentation tubes
- Variable Pitch Fuel Rod Plenum Spring
- Mid-Enriched Annular Fuel Pellets in Axial Blanket
- Fuel Assembly and Fuel Rod Dimensional Modifications

The fuel rods in a fuel assembly are arranged in a square array with 15 rod locations per side and a nominal centerline-to-centerline pitch of 0.563 inch between rods. Of the total possible 225 rod locations per assembly, 20 are occupied by guide thimbles for the RCCA rods and one for in-core instrumentation. The remaining 204 locations contain fuel rods. In addition to fuel rods, a fuel assembly is composed of a top nozzle, a bottom nozzle, seven grid assemblies, twenty guide thimbles, and one instrumentation thimble. Three IFM grids and one protective bottom grid (P-Grid/RPG) may also be present. Occasionally, stainless steel, zirconium alloy filler rods, or slightly enriched uranium fuel rods will replace failed fuel rods in reconstituted fuel assemblies. These rods are identical in shape and size to fuel rods. Special analyses are performed prior to any new fuel cycle utilizing reconstituted fuel.

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 secured 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.

Bottom Nozzle

The bottom nozzle is a square box-like structure which controls the coolant flow distribution to the fuel assembly and functions as the bottom structural element of the fuel assembly. The nozzle, which is square in cross-section, was fabricated from 304 stainless steel parts consisting of a perforated plate, four angle legs, and four pads or feet. The angle legs are welded to the plate forming a plenum space for coolant inlet to the fuel assembly. The perforated plate serves as the bottom limit for radiation and thermally induced growth for the fuel rods. The bottom support surface for the fuel assembly is formed under the plenum space by the four pads which are welded to the corner angles. The bottom nozzle now has more but smaller flow holes to mitigate possible fuel damage from debris.

Coolant flow to the fuel assembly is directed from the plenum in the bottom nozzle upward to the interior of the fuel assembly and to the channel between assemblies through the holes in the plate. The flow holes in the plate were sized and positioned beneath the fuel rods so that the rods cannot pass through the plate.

The RCC guide thimbles, which carry axial loads imposed on the assembly, are fastened to the bottom nozzle. These loads as well as the weight of the assembly are distributed through the nozzle to the lower core support plate. Indexing and positioning of the fuel assembly in the core is controlled through two holes in diagonally opposite pads which mate with locating pins in the lower core plate. Lateral loads imposed on the fuel assembly are also transferred to the core support structures through the locating pins. For Vantage 5 fuel the bottom nozzle has been made thinner to allow for increased fuel rod growth and higher burnup. This feature was maintained for V+ fuel. The bottom nozzle was modified starting with the Cycle 19 15x15 Upgrade fuel assembly design to eliminate the side communication flow holes in order to reduce the potential for debris intrusion. This design change is referred to as the modified Debris Filter Bottom Nozzle or mDFBN.

Top Nozzle

The top nozzle is a box-like structure, which functions as the fuel assembly upper structural element and forms a plenum space where the heated fuel assembly discharge coolant is mixed and directed toward the flow holes in the upper core plate. The nozzle is comprised of an adapter plate enclosure, top plate, two clamps, four 2-leaf holdown springs and assorted hardware. All parts with the exception of the springs and their hold down bolts were constructed of type 304 stainless steel. The springs were made from age hardenable Inconel 718 and the bolts from Inconel 600. The assemblies of the OFA design have about 4.5% increase in hydraulic resistance to flow compared to the LOPAR design. This results in an increased lift force to the OFA and requires 3-leaf holdown springs in the top nozzle instead of 2-leaf springs used for LOPAR assemblies.

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

The nozzle enclosure is actually a square tubular structure which forms the plenum section of the top nozzle. The bottom end of the enclosure is welded to the periphery of the adapter plate, and top end is welded to the periphery of the top plate.

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

Hold-down forces of sufficient magnitude to oppose the hydraulic lifting forces on the fuel assembly are obtained by means of the leaf springs which are mounted on the top plate. The springs are fastened to the top plate at the two corners where alignment holes are not used and radiate out from the corners parallel to the sides of the plate. Fastening of each pair of springs was accomplished with a clamp which fits over the ends of the springs and two bolts (one per spring) which pass through the clamp and springs, and thread into the top plate. At assembly, the spring mounting bolts were torqued sufficiently to reload against the maximum spring load and then lockwelded to the clamp which is counter-bored to receive the bold head.

The spring load is obtained through deflection of the spring pack by the upper core plate. The spring pack form is such that it projects above the fuel assembly and is depressed by the core plate when the internals are loaded into the reactor. The free end of the spring pack is bent downward and captured in a key slot in the top plate to guard against loose parts in the reactor in the event (however remote) of spring fracture. In addition, the fit between the spring and key slot and between the spring and its mating slot in the clamp are sized to prevent rotation.

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

Westinghouse has developed a top nozzle design to eliminate the potential of generating a loose part from fracture of the fuel assembly holddown spring screws. This design is called the Westinghouse Integral Nozzle (WIN). The WIN uses the same basic two piece nozzle as the standard design except that the top nozzle casting has been modified to include an integral pad in place of the previously separate clamp. As the name implies, these pads are cast as integral parts of the top nozzle casting. The WIN springs includes manufacturing process modifications for added margin against primary water stress corrosion cracking. Unlike previous bolted designs, the WIN design provides a wedged rather than a clamped joint. The tails of the spring packs are a slip-fit interface in the respective clamp pad cavities. Once preloaded, each spring pack is effectively locked in place during normal operation by the reaction forces generated at its tail. The flow holes, thermal characteristics, and method of attachment are all unchanged from the previous top nozzle design. The design was first introduced with the Cycle 19 Region

21 15x15 Upgrade fuel assemblies. The current top nozzle design is such as to allow easy reconstitution.

Guide Thimbles

The control rod guide thimbles in the fuel assembly provide guided channels for the control rods during insertion and withdrawal. They were fabricated from a single piece of Zircaloy-4 tubing, which is drawn to two different diameters. Starting with Cycle 10, the guide thimble material is ZIRLO™. The larger inside diameter at the top provides a relatively large annular area for rapid insertion during a reactor trip and to accommodate a small amount of upward cooling flow during normal operations. The bottom portion of the guide thimble is a reduced diameter to produce a dashpot action when the absorber rods near the end of travel in the guide thimbles during a reactor trip. The transition zone at the dashpot section is conical in shape so that there are no rapid changes in diameter in the tube.

The OFA design guide thimbles are the same as the guide thimbles for LOPAR design except for a 13 mil inner diameter and water diameter reduction. The OFA guide thimble diameter provides adequate diametral clearance for control rods, source rods, burnable absorber rods, and thimble plugs.

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

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

The top ends of the guide thimbles are expanded into stainless steel sleeves which are welded to the top grid. The sleeves are fitted through individual holes in the adaptor plate and welded around the circumference of the holes.

The 15x15 Upgrade incorporates the new tube-in-tube guide thimble design which utilizes a separate dashpot tube assembly that is inserted into the guide thimble assembly, pulled to a press fit over the thimble end plug and bulged into place. As the dashpot in this design can provide additional lateral support in that bottom thimble span, it is expected that there will be additional resistance to lateral deformation and incomplete rod insertions as a result of this design modification. The thimble screw for the tube-in-tube design is slightly longer than in the previous guide thimble tube design so that it can properly engage with the threads on the new guide thimble end plug and extend through the end plug of the dashpot tube assembly.

Grids

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

Different types of grid assemblies are used in the fuel assembly. One type having mixing vanes which project from the edges of the straps into the coolant stream, is used in the high heat region of the fuel assemblies for mixing of the coolant. A grid of this type is shown in Figure 3.2-

32. Grids of another type, located at the bottom and top ends of the assembly, are of the non-mixing type. They are similar to the mixing type with the exception that mixing vanes are not used on the internal straps. They are made of Inconel.

The spacing between grids is shown on Figure 3.2-31. The variation in span lengths is the result of optimization of the thermal-hydraulic and structural parameters. The grids are fastened securely to each guide thimble.

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

Inconel 718 was chosen for the grid material for the top and bottom grid because of its good corrosion resistance and high strength properties. After the combined brazing and solution annealing temperature cycle, the grid material was age hardened to obtain the material strength necessary to develop the required grid spring forces.

The OFA design features five middle Zircaloy-4 grids which have thicker and wider straps than the LOPAR Inconel grids to compensate for the difference in natural strength properties. Zircaloy grids maintain their integrity during the most severe load conditions of a combined seismic/LOCA event.

The Vantage+ fuel assemblies have five Low Pressure Drop (LPD) mid grids and three intermediate flow mixing (IFM) grids for increased DNB margin. These are all made of ZIRLO™.

The I-Spring design, introduced in Cycle 14 with the 15x15 Upgrade fuel, uses a revised spring, dimple and strap design to reduce the probability of grid-rod fretting.

Fuel Rods

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

At assembly, the pellets are stacked in the cladding to the required fuel height. The compression spring is then inserted into the top end of the fuel and the end plugs pressed into the ends of the tube and welded. All fuel rods are internally pressurized with helium during the welding process. [(47)] A hold-down force of approximately six times the weight of the fuel is obtained by compression of the spring between the top end plug and the top of the fuel pellet stack.

The fuel pellets are right circular cylinders consisting of slightly enriched uranium-dioxide powder which has been compacted by cold pressing and then sintered to the required density. The ends of each pellet are dished slightly to allow the greater axial expansion at the center of the pellets to be taken up within the pellets themselves and not in the overall fuel length.

A different fuel enrichment, as listed in Table 3.2-5, was used for each of the three regions in the first core loading. (Checker Board Pattern, see Figure 3.2-24.) Subsequent regions were uniquely designed for their enrichments. Current cycle core design parameters and the core loading pattern are provided in Table 3.2-5, Figure 3.2-24A and Figure 3.2-24B.

Each fuel assembly is identified by means of a serial number engraved on the upper nozzle. The fuel pellets are fabricated by a batch process so that only one enrichment region is processed at any given time. The serial numbers of the assemblies and corresponding enrichment are documented by the manufacturer and verified prior to shipment.

Each assembly is assigned a specific core loading position prior to insertion. A record is then made of the core loading position, serial number, and enrichment.

During initial core loading and subsequent refueling operations, detailed written handling and check-off procedures were utilized throughout the sequence. The initial core was loaded in accordance with the core loading diagram similar to Figure 3.2-24 which shows the location for each of the three enrichment types of fuel assemblies used in the loading together with the serial number of the assemblies in the region.

Extensive administrative controls (as discussed in Sections 3.2.3 and 3.3.3) render the possibility of loading fuel assemblies with incorrect enrichments or without their burnable poison rods extremely unlikely. Independent checks are made, prior to fuel loading, of each fuel assembly matching the contents of the assembly with its position in the core. Further checks are provided during core loading utilizing detailed written handling and check-off procedures.

Achieving criticality during core loading is prohibited in any case as the subcritical neutron flux is continuously monitored [Deleted] to detect any unexpected rise in the subcritical neutron flux. Core loading is stopped should the subcritical count rate increase unexpectedly or behave in an unstable manner. Procedures require the fuel handlers to receive permission from Reactor Engineering prior to unlatching a fuel assembly that has been inserted into the core.

Any such loading error, not significant enough to be detected during initial core loading is of no consequence from a criticality standpoint and would be detected by the power distribution map. During subsequent refueling operations the flux profile is flatter and loading errors can be detected by the power distribution map. (See the Technical Specifications)

<u>Rod Cluster Control Assemblies</u>

The control rods or rod cluster control assemblies (RCCA) each consist of a group of individual absorber rods fastened at the top end to a common hub or spider assembly. These assemblies, one of which is shown in Figure 3.2-25 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 stainless steel tubes to prevent the rods from coming in direct contact with the coolant. All control rods in the core have chrome plating on the stainless steel tube to reduce wear.

The overall control rod length is such that when the assembly has been withdrawn through its full travel, the tip 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. Prototype tests have shown that the RCC assemblies are very easily inserted and not subject to binding even under conditions of severe misalignment.

The spider assembly is in the form of a center hub with radial vanes supporting 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 RCC 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 holds the spring pack and its retainer is threaded into the hub within the skirt and welded to prevent loosening in service. All components of the spider assembly are made from type 304 stainless steel except for the springs which are Inconel X-750 alloy and the retainer which is of 17-4 PH material.

The absorber rods are secured to the spider so as to assure trouble free service. The rods were first threaded into the spider fingers and then pinned to maintain joint tightness, after which the pins were welded in place. The end plug below the pin position was 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 were 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 were provided to accommodate relative thermal expansions and to limit the internal pressure to acceptable levels.

The bottom plugs were 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.

Silver-indium-cadmium alloy absorber rods are resistant to radiation and thermal damage thereby ensuring their effectiveness under all operating conditions.

<u>Neutron Source Assemblies</u>

Neutron source assemblies are utilized in the core. A Neutron Source may be defined as a Primary Source, Secondary Source, or a recently Irradiated Fuel Assembly. In Cycle 1, these consisted of two assemblies with four secondary source rods each and two assemblies with one primary source rod each. In subsequent cycles, secondary sources were utilized through Cycle 18 and then removed for Cycle 19. The secondary source rods are fastened to a spider at the top end of the assembly. The initial core primary source rods were attached to a burnable poison assembly. Beginning with Cycle 14, secondary source assemblies using baseplate mounts began to be introduced into the core.

In the core, the neutron source assemblies are inserted into the RCC guide thimbles in fuel assemblies at unrodded locations. The location and orientation of the assemblies in the initial core is shown in Figure 3.2-33.

The primary and secondary source rods of the initial core utilized the same type of cladding material as the absorber rods (cold-worked type 304 stainless steel tubing) in which the sources are inserted. The source rods contain Sb-Be pellets. The primary source rods each contained capsules of Pu 238-Be source material at a neutron strength of approximately 2 x 10^8 neutrons/sec. Design criteria for the source rods are: cladding is free standing, internal pressures are always less than reactor operating pressure, and internal gaps and clearances are provided to allow for differential expansions between the source material and clad. For reload cores, the primary source rods have been removed and only the secondary source rods or recently irradiated fuel assemblies remained in core locations determined by the reload design. The secondary source assemblies were removed beginning with Cycle 19.

Plugging Devices

In order to limit bypass flow through the RCC guide thimbles in fuel assemblies which do not contain either control rods, source assemblies, or burnable poison rods, the fuel assemblies at those locations were fitted with plugging devices. The plugging devices consist of a flat base plate with short rods suspended from the bottom surface and a spring pack assembly and mixing device attached to the top surface. At installation in the core, the plugging devices fitted with the fuel assembly top nozzles and rested on the adapter plate. The short rods project the upper ends of the thimble tubes to reduce the bypass flow area. The spring pack was compressed by the upper core plate when the upper internals package was lowered into place. Similar short rods are also used on the source assemblies to fill the ends of all vacant fuel assembly guide thimbles and on burnable poison rod assemblies in these guide thimble positions where there are no burnable poison rodlets.

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

As of Cycle 17, thimble plugging devices were removed from the core.

Burnable Poison Rods

The burnable poison rods are statically suspended and positioned in vacant assembly guide thimble tubes within the fuel assemblies at non-rodded RCC core locations. The poison rods in each fuel assembly are grouped and attached together at the top end of the rods by a flat base plate which fits with the fuel assembly top nozzle and rests on the top adapter plate.

The base plate and the poison rods are held down and restrained against vertical motion through a spring pack which is attached to the plate and is compressed by the upper core plate when the reactor upper internals package is lowered into the reactor. This ensures that the poison rods cannot be lifted out of the core by flow forces.

The old poison rods consist of borosilicate glass tubes contained within type 304 stainless steel tubular cladding which is plugged and seal welded at the ends to encapsulate the glass. The glass is also supported along the length of its inside diameter by a thin wall type 304 stainless steel tubular inner liner. A typical old burnable poison rod is shown in longitudinal and transverse cross-sections in Figure 3.2-34. These rods can still be used for fluence reduction on the reactor vessel, as well as for burnable poison. The new poison rod design is referred to as Wet Annular Burnable Absorber (WABA). The WABA design has an annular aluminum oxide-boron carbide (Al_2OB_4C) absorber pellets contained within two concentric zircaloy tubings with water flowing through the center tubes as well as around the outer tubes. Eight demonstration poison rods of a new design, consisting of annular pellets of Al_2OB_4C contained within two concentric zircaloy tubings were also employed in the Cycle 3 and Cycle 4 cores. Cycles 5 and beyond have used large complements of WABA. Cycle 7 also utilized the Integral Fuel Burnable Absorber (IFBA). This consists of a thin coating of zirconium diboride on the cylindrical surface of the pellet. Cycles 9 and beyond have used IFBA and WABA.

The rods were designed in accordance with the standard fuel rod design criteria; i.e.., the cladding is free standing at reactor operating pressures and temperatures and sufficient cold void volume is provided within the rods to limit internal pressures to less than the reactor operating pressure assuming total release of all helium generated in the glass or annular pellets as a result of the $B_{10}(n,\gamma)$ reaction. A more detailed discussion of the old burnable poison rod design is found in WCAP-9000. [(45)] A more detailed discussion of the new burnable poison rods is found in WCAP-10021 (Revision 1) [(55)]. The new burnable poison rod is shown in longitudinal and transverse cross-sections in Figure 3.2-34a.

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

The top end of the inner liner is open to receive the helium which diffuses out of the glass.

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

Starting in Cycle 11, Hafnium Flux Suppressors have been placed in the eight core locations closest to the reactor vessel to suppress neutron fluence on the reactor vessel, thereby prolonging vessel life. These hafnium flux suppressors replace eight borosilicate-glass (Pyrex) burnable absorbers that were used up to Cycle 10. The hafnium flux suppressors are constructed with the same upper fixture as the Pyrex and WABA burnable absorbers currently available for use, and are handled in the same manner during refueling. The flux suppressors use unclad, hafnium rods and may be used in future operating cycles. (Reference 76)

Protective Bottom Grid

Starting with Cycle 11, the fuel assemblies (known as PERFORMANCE+) have a Protective Bottom Grid, which is an extra grid strap located at the bottom of the fuel pins, between the Bottom Nozzle and the bottom grid. The purpose of this protective bottom grid is to capture debris entering the fuel assembly and trap it at an elevation below the top of the fuel pin end plug. Therefore, any debris caught by the protective bottom grid cannot fret through or otherwise damage the fuel pin cladding in such a manner that the fuel pellets or rod plenum is exposed, thus making the fuel assembly more resistant to debris-related fuel failures.

The PERFORMANCE+ design also has the fuel pins mounted lower in the grid cage than fuel assembly designs before Cycle 11, almost touching the bottom nozzle. During operation, it is expected that pin growth and grid relaxation will allow the pin to rest on the bottom nozzle. The resultant transfer of weight from the grid cage to the bottom nozzle is expected, and may result in less bowing of the irradiated fuel assembly. (Reference 76)

The Protective Bottom Grid (P-Grid) has been redesigned starting with the Cycle 19 15x15 Upgrade fuel assembly design to reduce stresses that caused dimple cracking. The new design referred to as the Robust Protective Grid (RPG) implements design changes such as increasing the maximum nominal height of the grid, increasing the ligament length and the radii of the ligament cutouts, and the use of four additional spacers or inserts to help strengthen the grid. The nominal height of the grid was increased to allow “V-notch” window to help minimize flow-induced vibration caused by vortex shedding at the trailing edge of the inner grid straps. These design changes incorporated into the RPG design help address the issues of fatigue failures and failures due to stress corrosion cracking.

Evaluation of Core Components

Fuel Evaluation

The fission gas release and the associated buildup of internal gas pressure in the fuel rods were calculated by an overall fuel rod design model [(48)] which incorporates the time-dependent fuel densification. [(49)] 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 fission product gap inventory is a maximum. IFBA fuel has an additional contribution to rod internal pressure in the form of helium which is assumed to release at a rate of 100 percent.

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

To assure that manufactured fuel rods have 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, elevated temperature, tensile testing of fuel tubes, dimensional inspection, X-ray of both end plug welds, ultrasonic testing and helium leak tests.

In the event of cladding defect, the high resistance of uranium dioxide fuel pellets to attack by hot water protects against fuel deterioration. 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.

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

A survey of high burnup uranium dioxide [43] fuel element behavior indicates that for an initial uranium dioxide void volume, which is a function of the fuel density, it is possible to conservatively define the fuel swelling as a function of burnup. The fuel swelling model considered the effect of burnup, temperature distribution, and internal voids. It was incorporated in the overall fuel rod design model. [48]

Actual damage limits depend upon neutron exposure and normal variation of material properties and are greater than the design limits. For most of the fuel rod life the actual stresses and strains are considerably below the design limits. Thus, significant margins exist between actual operating conditions and the damage limits.

The other parameters having an influence on cladding stress and stain are as follows:

1) Internal gas pressure
The maximum rod internal pressure under nominal conditions is substantially less than the calculated pressure at the design limits. The end-of-life internal gas pressure depends upon the initial pressure, void volume, and fuel rod power history and IFBA loading. However, it does not exceed the design limit defined in Section 3.1.2.

2) Cladding temperature
The strength of the fuel cladding is temperature dependent. The minimum ultimate strength reduces to the design yield strength at an average cladding temperature of approximately 850°F. The maximum average cladding temperature during normal operating conditions (715°F) is given in Table 3.2-4, along with many other thermal and hydraulic design parameters.

3) Burnup
Fuel burnup results in fuel swelling which, along with thermal expansion, causes tensile cladding strain. Since rod power levels, and hence fuel temperature, decreases with burnup, the fuel pellet diameter increase with burnup is somewhat mitigated by the reduced thermal expansion. The strain design limits and stress design limits are met throughout the burnup lifetime of the fuel. These strain and stress design limits are below the cladding damage limits.

4) Fuel temperature and kW/ft

The fuel is designed so that the maximum fuel temperature will not exceed 4700°F during normal operation or malfunction transients. The peak linear heat generation rate (expressed in kW/ft) should not exceed a value which would cause fuel centerline melting.

Evaluation of Burnable Poison Rods

The burnable poison rods are positioned in the core inside RCC assembly guide thimbles and held down in place by attachment to a baseplate assembly compressed beneath the upper core plate and hence cannot be ejected and cause a reactivity transient. Due to the low heat generation rate, and the conservative design of the poison rods, there is no possibility for release of the poison as a result of helium pressure or clad heating during accident transients, including loss of coolant.

Many nuclear plants are using or have used burnable poison rods of the old design. Reference 51 describes the test operational experience of the old poison rod design. Eight demonstration poison rods of the new design were used in Cycles 3 and 4. Post irradiation examination following Cycles 3 and 4 indicated rodlets performed as expected and these were no anomalies. Further details of examinations performed can be referenced in WCAP-10021 (Revision 1). [55] The design was retained for Cycles 5 and beyond.

Effects of Vibration and Thermal Cycling on Fuel Assemblies

Analyses of the effect of cyclic deflection of the fuel rods, grid spring fingers, RCC control rods, and burnable poison rods due to hydraulically induced vibrations and thermal cycling show that the design of the components is good for the nominal design lifetime of 12 Effective Full-Power Years.

In the case of the fuel rod grid spring support, the amplitude of a hydraulically induced motion of the fuel rod is extremely small (approximately 0.001), and the stress associated with the motion is significantly small (less than 100 psi). Likewise, the reactions at the grid spring due to the motion is much less than the preload spring force and contact is maintained between the fuel clad and the grid spring and dimples. Fatigue of the clad and fretting between the clad and the grid support is not anticipated.

The effect of thermal cycling on the grid-clad support is merely a slight relative movement between the grid contact surfaces and the clad, which is gradual in nature during heat-up and cool-down. Since the number of cycles of the occurrence is small over the life of a fuel assembly, negligible wear of the mating parts is expected.

These conclusions have been verified by fuel operating experience in a number of nuclear plants as described in Reference 51.

The dynamic deflection of the full length control rods and the burnable poison rods is limited by their fit with the inside diameter of either the upper portion of the guide thimble or the dashpot. With this limitation, the occurrence of truly cyclic motion is questionable. However, an assumed cyclic deflection through the available clearance gap results in an insignificantly low stress in

either the clad tubing or in the flexture joint at the spider or retainer plate. The above consideration assumes the rods are supported as cantilevers from the spider, or the retainer plate in the case of the burnable poison rods.

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

Control Rod Drive Mechanism

Full Length Rods

Design Description

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

Fast total insertion (reactor trip) is obtained by simply removing the electrical power allowing the rods to fall by gravity. Typical total insertion time is less than 1.8 seconds and always less than 2.7 seconds.

The complete drive mechanism, shown in Figure 3.2-36, consists of the internal (latch) assembly, the pressure vessel, the operating coil stack, the drive shaft assembly, and the rod position indictor coil stack.

Each assembly is an independent unit which can be dismantled or assembled separately. Each mechanism pressure housing is threaded onto an adapter on top of the reactor pressure vessel and seal welded. The operating drive assembly is connected to the control rod (directly below) by means of a grooved drive shaft. The upper section of the drive shaft is suspended from the working components of the drive mechanism. The drive shaft and control rod remain connected during reactor operation, including tripping of the rods.

Main coolant fills the pressure containing parts of the control rod drive mechanism. All working components and the shaft are immersed in the main coolant and depend on it for component damping.

Three magnetic coils, which form a removable electrical unit and surround the rod drive pressure housing, induce magnetic flux through the housing wall to operate the working components. They move into sets of latches which lift, lower, and hold the grooved drive shaft.

The three magnets are turned on and off in a fixed sequence by solid-state switches for the full length rod assemblies.

The sequencing of the magnets produces step motion over the 144 inches of normal control rod travel.

The mechanism develops a lifting force approximately two times the static lifting load. Therefore, extra lift capacity is available for overcoming mechanical friction between the moving and the stationary parts. Gravity provides the drive force for rod insertion and the weight of the whole rod assembly is available to overcome any resistance.

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

A multi-conductor cable connects the mechanism operating coils to the 125 volt DC power supply. The power supply is described in Section 7.3.2.

Latch Assembly

The latch assembly contains the working components which withdraw and insert the drive shaft and attached control rod. It is located within the pressure housing and consists of the pole pieces for three electromagnets.

They actuate two sets of latches which engage the grooved section of the drive shaft.

The upper set of latches move up or down to raise or lower the drive rod by 5/8 inch. The lower set of latches have a maximum 1/16 inch axial movement to shift the weight of the control rod from the upper to the lower latches.

Pressure Vessel

The pressure vessel consists of the pressure housing and rod travel housing. The pressure housing is the lower portion of the vessel and contains the latch assembly. The rod travel housing is the upper portion of the vessel. It provides space for the drive shaft during its upward movement as the control rod is withdrawn from the core.

Operating Coil Stack

The operating coil stack is an independent unit which is installed on the drive mechanism by sliding it over the outside of the pressure housing. It rests on a pressure housing flange without any mechanical attachment and can be removed and installed while the reactor is pressurized.

The three operator coils are made of round copper wire which is insulated with a double layer of filament type glass yarn.

The design operating temperature of the coils is 200^{o}C. Average coil temperature can be determined by resistance measurement. Forced air cooling along the outside of the coil stack maintains a coil casing temperature of approximately 120^{o}C or lower.

Drive Shaft Assembly

The main function of the drive shaft is to connect the control rod to the mechanism latches. Grooves for engagement and lifting by the latches are located throughout the 144 inches of control rod travel. The grooves are spaced 5/8 inch apart to coincide with the mechanism step length and have 45° angle sides.

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

A ¼ inch diameter disconnect rod runs down the inside of the drive shaft. It utilizes a locking button at its lower end to lock the coupling and control rod. At its lower end, there is a disconnect assembly. For remote disconnection of the drive shaft assembly from the control rod, a button at the top of the drive rod actuates the connect/disconnect assembly.

During plant operation, the drive shaft assembly remains connected to the control rod at all times. It can be attached and removed from the control rod only when the reactor vessel head is removed.

Position Indicator Coil Stack

The position indicator coil stack slides over the rod travel housing section of the pressure vessel. It detects drive rod position by means of a cylindrically wound differential transformer which spans the normal length of the rod travel (144 inches).

Drive Mechanism Materials

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

Three types of metals are used exclusively: stainless steel, Inconel X, and cobalt based alloys. Wherever magnetic flux is carried by parts exposed to the main coolant, 400 series stainless steel is used. Cobalt based alloys are used for the pins, latch tips, and bearing surfaces.

Inconel X is used for the springs of both latch assemblies and 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 Reactor 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 to approximately 0.001 inch thickness to prevent corrosion.

Principles of Operation

The drive mechanisms, shown schematically in Figures 3.2-35 and 3.2-36 withdraw and insert their respective control rods as electrical pulses are received by the operator coils.

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

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

Control Rod Withdrawal:
The control rod is withdrawn by repeating the following sequence:

1) Movable Gripper Coil – ON
   The moveable gripper armature raises and swings the movable gripper latches into the drive shaft groove.

2) Stationary Gripper Coil – OFF
   Gravity causes the stationary gripper latches and armature to move downward until the load of the drive shaft is transferred to the movable gripper latches. Simultaneously, the stationary gripper latches then swing out of the shaft groove.

3) Lift Coil – ON
   The 5/8 inch gap between the lift armature and the lift magnet pole closes and the drive rod raises one step length.

4) Stationary Gripper Coil – ON
   The stationary gripper armature raises and closes the gap below the stationary gripper magnetic pole, swings the stationary gripper latches into a drive shaft groove. The latches contact the shaft and lift it 1/16 inch. The load is so transferred from the movable to the stationary gripper latches.

5) Movable Gripper Coil –OFF
   The movable gripper armature separates from the lift armature under the force of the spring and gravity. Three links, pinned to the movable gripper armature, swing the three movable gripper latches out of the groove.

6) Lift coil – OFF
   The gap between the lift armature and the lift magnet pole opens. The moveable gripper latches drop 5/8 inch to a position adjacent to the next groove.

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

1) Lift Coil – ON
   The movable gripper latches are raised to a position adjacent to a shaft groove.

2) Movable Gripper Coil – ON
   The movable gripper armature raises and swings the movable gripper latches into a groove.

3) Stationary Gripper Coil – OFF
   The stationary gripper armature moves downward and swings the stationary gripper latches out of the groove.

4) Lift Coil – OFF
   Gravity and spring force separate the lift armature from the lift magnetic pole and the control rod drops down 5/8 inch.

5) Stationary Gripper Coil – ON

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

Control Rod Tripping:

The holding or static mode is with the stationary gripper coil. If power to the stationary gripper coil is cut off, as for tripping, the combined weight of the drive shaft and the rod cluster control assembly is sufficient to move the latches out of the shaft groove. The control rod falls by gravity into the core. The tripping occurs as the magnetic field, holding the stationary gripper plunger half against the stationary gripper pole, collapses and the stationary gripper plunger half is forced down by the weight acting upon the latches.

Fuel Assembly and RCCA mechanical Evaluation

To confirm the mechanical adequacy of the fuel assembly and full length RCC assembly, functional test programs were conducted on a full scale Indian Point 2 prototype 12 ft can less fuel assembly and control rod. The prototype assembly was tested under simulated conditions of reactor temperature, pressure, and flow for approximately 1000 hours. The prototype mechanism accumulated 2,260,892 steps and 600 scrams. At the end of the test the CRDM was still operating satisfactorily. A correlation was developed to predict the amplitude of flow excited vibration of individual fuel rods and fuel assemblies. Inspection of the fuel assembly and drive line components did not reveal significant fretting. The wear of the absorber rods, fuel assembly guide thimbles, and upper guide tubes was minimal. The control rod free fall time against 125% of nominal flow was less than 1.5 seconds to the dashpot (10 ft of travel).

Additional tests had previously been made on a full scale San Onofre mock up version of the fuel assembly and control rods. [44]

Mockup Tests at 1/7 Scale for Indian Point 2

A 1/7 scale model of the Indian Point 2 internals was designed and built for hydraulic and mechanical testing. The tests provided information on stresses and displacements at selected locations on the structure due to static loads, flow induced loads, and electromagnetic shaker loads. Flow distribution and pressure drop information were obtained. Results of the static tests indicated that mean strains in the upper core support plate and upper support columns are below design limits. Strains and displacements measured in the model during flow tests verified that no damaging vibration levels were present. Additional information gained from the tests were the natural frequency and damping of the thermal shield and other components in air and water. Model response can be related to the full scale plant for most of the expected exciting phenomena, but across the board scaling is not possible. Specifically exciting phenomena which are strongly dependent on Reynolds number cannot be scaled. In areas where Reynolds number may be important, either: (1) the measured vibration amplitudes were many times lower than a level that would be damaging, or (2) full scale vibration data were obtained.

Loading and Handling Tests

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

Axial and Lateral Bending Tests

In addition, axial and lateral bending tests were performed in order to simulate mechanical loading of the assembly during refueling operation. Although the maximum column load expected to be experienced in service is approximately 1000 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.

3.2.4 ZIRLO Clad Fuel

Fuel assemblies containing fuel rods fabricated with the advanced zirconium alloy cladding material ZIRLO™ are being used in the Indian Point Unit 3 Cycle 9 core and beyond. These fuel assemblies will have fuel rods fabricated with ZIRLO™ cladding to obtain additional operational benefit from the cladding's improved corrosion resistance. The transition to VANTAGE 5 fuel was made at Indian Point 3 for Cycle 7 as described in the submittal to the NRC dated January 20, 1989 (IPN-89-007). Cycle 10 began the use of V+ fuel which utilizes ZIRLO™ for guide thimbles and intermediate spacers as well as clad.

This report shows, based on both evaluations and analyses, that no unreviewed safety questions exist as a result of inserting ZIRLO™ clad fuel rods into the Indian Point 3 reactor core. This report shows that the subsequent proposed changes to the Indian Point Unit 3 Technical Specifications will not involve significant hazard considerations.

3.2.4.1 Background

Westinghouse has developed a new zirconium based fuel rod clad alloy, known as ZIRLO™, to enhance fuel reliability and achieve extended burnup. This alloy provides significant improvement in fuel rod clad corrosion resistance and dimensional stability under irradiation. ZIRLO™ cladding corrosion resistance has been evaluated in long-term, out-of-pile tests over a wide range of temperatures (up to 600°F in water tests, up to 932°F in steam tests). Additional tests have also been conducted in lithiated water environments. The improved corrosion resistance of ZIRLO™ cladding has also been demonstrated to very high burnups in the BR-3 reactor.

A conditional licensing approval for the use of this advanced alloy cladding in two demonstration fuel assemblies for the North Anna Unit 1 reactor core was given in a USNRC letter dated May 13, 1987. The USNRC granted an exemption (Reference 66) from the provision of 10CFR50.46, 10 CFR50.44 and 10CFR51.52 with respect to the use of the North Anna demonstration fuel assemblies with the advanced cladding material, ZIRLO™. The Information required to support the licensing basis for the implementation of the ZIRLO™ clad fuel rods in Indian Point 3 is given in References 67 and 68. The fuel assemblies will be utilized in Indian Point 3, beginning with Cycle 9.

3.2.4.2 Areas Assessed

The following areas have been assessed during the safety evaluation process: chemical/mechanical properties, neutronic performance, thermal and hydraulic performance, cladding performance under non-LOCA conditions, and cladding performance under LOCA conditions.

Reference 69 addresses the VANTAGE 5 design and its application to a 17 x 17 fuel assembly. The VANTAGE 5 design may be applied to other fuel assembly arrays (14 x 14, 15 x 15) where such applications are evaluated on a plant specific basis and licensed in accordance with NRC requirements. Indian Point Unit 3 has been licensed for 15 x 15 VANTAGE 5 as already noted. Subsequently, the applicable models and methods employed to address the 15 x 15 VANTAGE 5 design have been licensed for Indian Point Unit 3. The principal difference between the Indian Point Unit 3 Region 11 fuel and the licensed 15 x 15 VANTAGE 5 fuel is the use of ZIRLO™ cladding. The use of ZIRLO™ cladding does not alter the previously licensed models and methods of Reference 69 with the exception of the LOCA model and methodology as noted in Chapter 14. The revised LOCA model and methodology were used as the basis to evaluate the effects of the change in cladding material. These evaluations have shown that the present LOCA related design bases and limits remain valid. Where the models and methods of Reference 69 are not affected by ZIRLO™ cladding, Indian Point 3 plant specific evaluations and analyses have also shown that the current design bases and limits remain valid.

3.2.4.3 Previous Irradiation Experience

Fuel rods fabricated with ZIRLO™ cladding have been previously irradiated in a foreign reactor (BR-3 reactor) at linear power levels up to 17 kw/ft, and burnups significantly greater than those planned for the Indian Point 3 fuel assemblies. Corrosion and hydriding data obtained on the ZIRLO™ were compared with the reference Zircaloy-4 cladding of fuel rods irradiated as controls in the same test assemblies. Based on the irradiation results of the test assemblies in the foreign reactor, the Indian Point Unit 3 ZIRLO™ cladding waterside corrosion and hydriding

will be significantly less than that expected for the Zircaloy-4 clad fuel rods. The irradiation test results substantiate a lower clad irradiation growth $(\Delta L / L)$ and creepdown for the ZIRLO™ cladding compared to Zircaloy-4 cladding.

Two demonstration fuel assemblies, containing ZIRLO™ clad fuel rods, began irradiation in the North Anna Unit 1 reactor during June 1987. The ZIRLO™ clad fuel rods achieved over 21,000 MWD/MTU burnup in their first cycle (complete during February 1989). Visual inspection during refueling showed no abnormalities. One demonstration assembly with ZIRLO™ TM clad fuel rods underwent a second cycle of irradiation and achieved over 37,000 MWD/MTU burnup (completed January 1991). Visual inspection of the two cycle ZIRLO™ clad fuel rods during refueling showed no abnormalities. Cladding corrosion measurements showed that the reduced corrosion obtained with the ZIRLO™ clad rods was significantly better than that anticipated on the basis of licensing basis evaluations. The present and future irradiation results are and will be considered in the design of the fuel rods with ZIRLO™ cladding to assure that all fuel rod design bases are satisfied for the planned irradiation life of the Indian Point Unit 3 fuel assemblies.

3.2.4.4 Chemical/Mechanical Properties

This chemical composition (see Table 3.2-6) of the ZIRLO™ clad fuel rods in the Indian Point 3 fuel assemblies is similar to Zircaloy-4 except for slight reductions in the content of tin (Sn), iron (FE), and zirconium (Zr) and the elimination of chromium (Cr). ZIRLO™ cladding also contains a nominal amount of niobium (Nb). These small composition changes are responsible for the improved corrosion resistance compared to Zircaloy-4. The physical and mechanical properties are very similar to Zircaloy-4 while in the same metallurgical phase. However, the temperatures at which the metallurgical phase changes occur are different for Zircaloy-4 and ZIRLO™ cladding (Appendix A of Reference 67). These differences are considered in the evaluations discussed below for cladding behavior under non-LOCA and LOCA conditions. Further aspects of the ZIRLO™ cladding performance under LOCA conditions are given in Reference 67). Evaluations have been performed using the NRC approved fuel rod performance code (Reference 70) to verify that the fuel rod design bases and design criteria are met for assemblies containing ZIRLO™ clad fuel rods. The fuel rod design bases, criteria and models, which are affected by the use of ZIRLO™ cladding are described in Reference 67.

3.2.4.5 Neutronic Performance

The design and predicted nuclear characteristics of fuel rods with ZIRLO™ cladding are similar to those of VANTAGE 5 design (Reference 69). The evaluations have shown (Reference 67) that the nuclear design bases are satisfied for fuel rods with ZIRLO™ cladding and that the use of ZIRLO™ cladding will not affect the standard nuclear design analytical models and methods to accurately describe the neutronic behavior of fuel rods with ZIRLO™ cladding. The safety limit characteristics of the VANTAGE 5 fuel design (Reference 69) are not affected.

3.2.4.6 Thermal and Hydraulic Performance

The thermal and hydraulic design bases for fuel rods with ZIRLO™ cladding are identical to those of the VANTAGE 5 design (reference 69). Since the use of the ZIRLO™ clad fuel does not cause changes affecting the parameters which are major contributors in this area (i.e., DNB, core flow, and rod bow), the design bases of the VANTAGE 5 design (Reference 69) remain valid.

3.2.5 VANTAGE + Design Features

The design features of the V+ fuel design include the following:

- Current Vantage 5 (w/o IFMs) Fuel Features
  - Reconstitutable Top Nozzle (RTN)
  - Extended Burnup Fuel Assembly Design
  - Extreme Low Leakage Loading Pattern
  - Enriched Integral Fuel Burnable Absorbers (IFBAs)
  - Debris Filter Bottom Nozzle (DFBN)
  - Axial Blankets
- ZIRLO™ fuel cladding
- Low Pressure Drop (LPD) Mid grids
- Integral Flow Mixing grids (IFMs)
- ZIRLO™ guide thimble and instrument tubes
- Variable Pitch Fuel Rod Plenum Spring
- Mid-enriched Annual Fuel Pellets in Axial Blanket
- Fuel Assembly and Fuel Rod Dimensional Modifications
- Low Cobalt Top and Bottom Nozzles
- Coated Cladding (Pre-oxidized ZIRLO™ cladding)
- Gripable Top End Plug

3.2.5.1 VANTAGE 5 Fuel Features

The use of ZIRLO™ fuel cladding was implemented in the VANTAGE 5 fuel design beginning in Cycle 9 after issuance of the V5 SER.

With respect to the Extended Burnup Fuel Assembly Design, the V+ fuel design includes an increase in the region average discharge burnup from 40,000 + to 50,000 + MWD/MTU. The effects of the increase in extended burnup on the performance of the fuel are included in the safety analysis via the reload safety analysis parameters which are taken into account in the reload design process.

3.2.5.2 Mid-enriched Annular Fuel Pellets in Axial Blankets and Enriched IFBAs

Axial blankets reduce power at the ends of the rod which increases axial peaking toward the middle of the rod. Used alone, axial blankets reduce DNB margin, but the effect may be offset by the presence of IFBAs which flatten the power distribution. The net effect on the axial shape is a function of the number and configuration of IFBAs in the core and the time in core life. The effects on the reload safety analysis parameters due to axial blankets, including annular fuel pellets in the axial blanket, and IFBAs, including enriched IFBAs, are taken into account in the reload design process. The axial and radial power distribution assumptions used in the safety analysis kinetics calculations have been determined to be applicable for evaluating the axial blankets and IFBAs in the IP3 V+ fuel design.

3.2.5.3 LPD Mid Grids, IFM ZIRLO Guide Thimbles and Instrument Tubes, Low Cobalt Top and Bottom Nozzles and Fuel Assembly and Fuel Rod Dimensional Modifications.

The IP3 V+ fuel design incorporates the use of ZIRLO™ LPD mid grids, IFMs, guide thimbles and instrument tubes, and includes minor fuel assembly and fuel rod dimensional modifications to accommodate the Extended Burnup Fuel Assembly design.

With respect to the non-LOCA accident analysis, the use of ZIRLO™ guide thimbles and instrument tubes, low cobalt top and bottom nozzles and the minor fuel assembly and fuel rod dimensional modifications have no direct effect on the analysis results since the characteristics of these features are not specifically modeled in the transient analysis. Any effects of these items on the performance of the fuel are included in the safety analysis via the reload safety analysis parameters (e.g., fuel temperatures, flow rates, pressure drops) which are taken into account in the reload design process.

The use of the LPD mid grids in the V+ fuel design are primarily for the purpose of offsetting the increase in flow resistance associated with the introduction of the IFMs. Therefore, with respect to the parameters which affected the non LOCA accident analysis, the V+ fuel design is hydraulically compatible with the resident V5 fuel design and has no direct effect on the non LOCA safety analysis. Any localized assembly to assembly hydraulic differences which may affect the performance of the fuel are included in the safety analysis via the reload safety analysis parameters which are taken into account in the reload design process and documented in the Reload Safety Evaluation.

With respect to the implementation of IFMs V+ fuel design, it should be noted that although the use of IFMs provides a DNB benefit caused by enhanced flow mixing, the benefit is not fully realized in the safety analysis since the resident V5 fuel design does not include IFMs and therefore is more limiting with respect to DNB. As a result, the Core Thermal Limits and resulting OTΔT and OPΔT reactor trip setpoints applicable for the transition to VANTAGE + Fuel are currently based on the DNB performance of the more limiting VANTAGE 5 (w/o IFMs) fuel. With the transition to a full core V+ fuel complete, the DNB benefit associated with the IFMs is fully realized.

3.2.5.4 Variable Pitch Fuel Rod Plenum Spring

The optimized fuel rod plenum spring feature of the V+ fuel design is primarily to provide increased fuel rod plenum volume which benefits rod internal pressure concerns. Any effects of this spring design on the performance of the fuel is included in the safety analysis via the reload safety analysis parameters which are taken into account in the reload design process.

3.2.5.5 15x15 Upgrade Design

Beginning with Cycle 14, a modified fuel design, designated “15x15 Upgrade”, was introduced to the Indian Point Unit 3 Core (IP3).

Fuel cladding is ZIRLO on all 15 x 15 Upgrade fuel assemblies. ZIRLO is also used on intermediate grid straps, IFMs and thimble tubes. The Westinghouse model designation for this modified design fuel is 15x15 Upgrade Fuel with I-Spring (hereafter called 15x15 Upgrade). The IP3 core was fully loaded with 15x15 Upgrade fuel in Cycle 15, with the exception of one

VANTAGE+ assembly in the center core location, which is a location not considered susceptible to grid-rod fretting. Cycle 16 and subsequent cycle cores consist only of 15 x 15 Upgrade fuel.

The 15x15 Upgrade fuel design includes a number of new features:

1) I-Spring mid-grid design to reduce grid-to-rod fretting
2) Enhanced Intermediate Flow Mixers (IFMs) to reduce grid-to-rod fretting
3) Balanced vane pattern on all grids to provide more balanced flow for vibration reduction.
4) Tube-in-tube guide thimble design, replacing the Vantage+ dashpot design. The new guide thimbles are designed to improve margin to Incomplete Rod Insertion (IRI).

In addition to these features, the 15x15 Upgrade Fuel includes fuel performance and reliability features that have been added to IP3 fuel in previous cycles. These include: debris resistant bottom nozzles (DFBN, mDFBN), fuel rod oxide coatings, protective bottom grid (P-Grid, RPG) and bead-blasted top nozzle spring screws which were later eliminated in the Westinghouse Integral Nozzle (WIN) design. The grade of Inconel used in the spring screws is Type 718, which is stronger than the previously-used Type 600.

Beginning with Cycle 15, mixed axial blankets were introduced to the IP3 core. Axial blankets for previous Unit 3 reload regions have consisted of annular pellets. Starting with the Cycle 15 feed fuel, a combination of annular and solid pellets are used. Annular blankets continue to be utilized in IFBA fuel rods, and solid blanket pellets are used in non-IFBA fuel rods. This approach maximizes the assembly uranium loading while continuing to provide the necessary free volumes in IFBA rods to accommodate helium gas release.

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TABLE 3.2-1

NUCLEAR DESIGN DATA
(For Cycle 1 and Cycle 21

		Cycle 1	Cycle 21
STRUCTURAL CHARACTERISTICS			
1.	Fuel Weight [Deleted], MTU	88.60	87.55
2a.	Zircaloy Weight, lbs	44,450	1,350
2b.	ZIRLO™ Weight, lbs	N/A[2]	51,300
3.	Core Diameter, inches	132.7	Same as Cycle 1
4.	Core Height, inches	144	Same as Cycle 1
	Reflector Thickness and Composition		
5.	Top – Water Plus Steel	~10 inches	Same as Cycle 1
6.	Bottom – Water Plus Steel	~10 inches	Same as Cycle 1
7.	Side – Water Plus Steel	~15 inches	Same as Cycle 1
8.	H_2O/U, (cold) Core (volume)	4.03	N/C[1]
9.	Number of Fuel Assemblies	193	Same as Cycle 1
10.	UO_2 Rods per Assembly	[Deleted] 204	Same as Cycle 1
PERFORMANCE CHARACTERISTICS			
11.	Heat Output, MWt (initial rating)	3025	3216
12	Heat Output, MWt (maximum calculated turbine rating)	3216	Same as Cycle 1
13.	Fuel Burnup, MWD/MTU	17,346	27,060 (maximum)

[1] Not Calculated for Cycle 21
[2] Not Applicable to Cycle 1

TABLE 3.2-1
(Cont.)

NUCLEAR DESIGN DATA
(For Cycle 1 and Cycle 21)

		Cycle 1	Cycle 21
14.	Region 1 enrichment, w/o[Deleted]	2.28	4.95[3] (Region 21)
15.	Region 2 enrichment, w/o[Deleted]	2.80	4.60[3] (Region 22A) 4.95[3] (Region 22B)
16.	Region 3 enrichment, w/o[Deleted]	3.30	4.60[3] (Region 23A) 4.95[3] (Region 23B)
17.	Equilibrium Enrichment, w/o	3.20	4.80 (approx.)
18.	Nuclear Heat Flux Hot Channel Factor, F_Q^N	2.49[4]	2.23[9]
19.	Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^N$	1.55	1.65[9]
CONTROL CHARACTERISTICS			
	Effective Multiplication (Beginning of Life) With Rods in; No Boron		
20.	Cold, No Power, Clean	1.197	N/C[5]
21.	Hot, No Power, Clean	1.144	N/C[5]
22.	Hot, Full Power, Clean	1.131	N/C[5]
23.	Hot, Full Power, Xe & Sm Equilibrium	1.091	N/C[5]
24.	Material	5% Cd; 15% In; 80% Ag	Same as Cycle 1
25.	Full Length RCC Assemblies	53	Same as Cycle 1

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[3] [Deleted] Region has 4.0 w/o top and 3.6 w/o bottom 8 inch mixed blankets. [Deleted]
[4] Nuclear peaking factor limits were revised in response to ACRS concerns by a generic peaking factor envelope for Cycle 1 operation (see Figure 3.2-5).
[5] Not Calculated for Cycle 21

TABLE 3.2-1
(Cont.)

NUCLEAR DESIGN DATA
(For Cycle 1 and Cycle 21)

		Cycle 1	Cycle 21
26.	Partial Length RCC Assemblies	8	Removed
27.	Number of Absorber Rods per RCC Assembly	20	Same as Cycle 1
28.	Total Rod Worth, BOL, %	See Table 3.2-2	See Table 3.2-2
29.	Fuel Loading Shutdown; Rods in (k = 0.85)	2000 ppm	N/C[6]
	(k = 0.90)	1810 ppm	N/C[6]
	(k = 0.95)	N/A[7]	1807 ppm[8]
30.	Shutdown [Deleted] with Rods Inserted, Clean, Cold, BOL	1000 ppm (k=0.99)	N/C[6]
31.	Shutdown [Deleted] with Rods Inserted, Clean, Hot, BOL	548 ppm (k=0.99)	941 ppm (k=0.987)
32.	Shutdown [Deleted] with No Rods Inserted, Clean, Cold, BOL	1500 ppm (k=0.99)	N/C[6]
33.	Shutdown with No Rods Inserted, Clean, Hot, BOL	1476 ppm (k=0.99)	2014 ppm (k=0.987)
	To Maintain k=1 at Hot Full Power, No Rods Inserted (34, 35, 36):		
34.	Clean (0 MWD/MTU)	1228 ppm	1397 ppm
35.	Xenon Equilibrium (150 MWD/MTU)	934 ppm	1019 ppm [Deleted]
36.	Xe and Sm Equilibrium (1000 MWD/MTU) [Deleted]	899 ppm	1078 ppm
37.	Shutdown, All But One Rod Inserted, Clean, Cold, BOL	1099 ppm (k=0.99)	1391 ppm (k = 0.987 [Deleted])
38.	Shutdown, All But One Rod Inserted, Clean, Hot, BOL	669 ppm (k=0.99)	1051 ppm (k=0.987 [Deleted])

[6] Not Calculated for Cycle 21
[7] Not Applicable to Cycle 1
[8] Calculated Value – Actual value used is 2050 ppm per the COLR.
[9] Fuel pellet thermal conductivity degradation evaluations resulted in a reduction of Fq from 2.5 to 2.3 and Fdh from 1.7 to 1.65.

TABLE 3.2-1
(Cont.)

NUCLEAR DESIGN DATA
(For Cycle 1 and Cycle 21)

		Cycle 1	Cycle 21
BURNABLE POISON RODS			
39.	Number and Material	1434 Borosilicate Glass	[Deleted] 1136 WABA 12144 IFBA
40.	Worth Hot, Δk/k	10.0%	N/C[1]
41.	Worth Cold, Δk/k	8.0%	N/C[1]
KINETIC CHARACTERISTICS			
42.	Moderator Temperature Coefficient at Hot Full Power, pcm/°F	-4 to -31	-10.77 to -34.11
43.	Moderator Pressure Coefficient, Δk/psi	0.3×10^{-6} to 4.0×10^{-6}	N/C[1]
44.	Moderator Density Coefficient, $\Delta k \cdot cm^3/gm$	-0.1 to 0.47	<0.54
45.	Doppler Coefficient, pcm/°F	-1.0 to -2.0	-0.9 to -3.2
46.	Delayed Neutron Fraction, %	0.51 to 0.72	0.51 to 0.62
47.	Prompt Neutron Lifetime, seconds	[Deleted] 1.8×10^{-5}	1.16×10^{-5} to 1.45×10^{-5}

[1] Not Calculated for Cycle 21

TABLE 3.2-2

REACTIVITY REQUIREMENTS FOR CONTROL RODS
(For Cycle 1 and Cycle 21)

Requirement	Percent $\Delta\rho$ Beginning of Life[10]	End of Life
CYCLE 1		
Control Banks		
Total Control	2.31	3.43
Shutdown Banks		
Shutdown	1.00	1.72
Total Requirement	3.31	5.15
CYCLE 21		
Control Groups		
Total Control Bank Requirement (1)	2.00	2.84
Control Rod Worth (HZP)		
All Rods Inserted Less Most Reactive Stuck Rod[11]	5.76	6.06
(2) Less 10%	5.18	5.45
Shutdown Margin		
Calculated Margin [(2)-(1)]	3.18	2.61
Required Shutdown Margin	1.3	1.3

Cycle 21 Note:
A) Rod worths are calculated at ARO, HFP boron concentration.
B) T_{mod} at 547.0°F at HZP and 570.0°F at HFP. (Nominal vessel Tavg)

[10] 150 MWD/MTU.
[11] K-10 is the most reactive stuck rod at BOL (0.78% $\Delta\rho$) and K-10 is the most reactive stuck rod at EOL (0.81% $\Delta\rho$).

TABLE 3.2-3

CALCULATED ROD WORTHS, $\Delta\rho$
(For Cycle 1 and Cycle 21)

CYCLE 1

Core Condition	Rod Configuration	Worth*	Less 10%**	Design Reactivity Requirements	Shutdown Margin
BOL, HFP	53 Rods in	9.76%			
BOL, HZP	52 Rods in; Highest Worth Rod Stuck Out	7.75%	6.97%	2.31%	4.66%
EOL, HFP	53 Rods in	9.45%			
EOL, HZP	52 Rods in; Highest Worth Rod Stuck Out	7.57%	6.81%	3.43%	3.38%

CYCLE 21

Core Condition	Rod Configuration	Worth*	Less 10%**	Reactivity Requirements	Shutdown Margin
BOL, HFP	53 Rods in	7.47%			
BOL, HZP	52 Rods in; Highest Worth Rod Stuck Out	5.76%	5.18%	1.30%	3.18%
EOL, HFP	53 Rods in	8.45%			
EOL, HZP	52 Rods in; Highest Worth Rod Stuck Out	6.06%	5.45%	1.30%	2.61%

Legend

BOL = Beginning-of-Life
EOL = End-of-Life
HFP = Hot Full Power
HZP = Hot Zero Power

* The values for worth are for rods at the insertion limit.
** Calculated rod worth is reduced by 10% to allow for uncertainties.

TABLE 3.2-4

THERMAL AND HYDRAULIC DESIGN PARAMETERS
(For Cycle 1 and Cycle 21)

	Cycle 1	**Cycle 21**
Total Heat Output, MWt	3025	3216
Total Heat Output, Btu/hr	10,324x10^6	10,973x10^6
Heat Generated in Fuel, %	97.4	Same as Cycle 1
Nominal System Pressure, psia	2250	Same as Cycle 1
Hot Channel Factors		
Heat Flux		
Engineering, F_q^E	1.03	Same as Cycle 1
Total, F_q^{T*}	2.32	2.30**
Enthalpy Rise – Nuclear $F_{\Delta H}^N$	1.55	1.65**
Coolant Flow		
Total Flow Rate, lbm/hr	136.3x10^6	134.8x10^6
Average Velocity Along Fuel Rods, ft/sec	15.6	14.9
Average Mass Velocity, lbm/hr•ft^2	2.54x10^6	2.42x10^6

*The total heat flux hot channel factor shown is a generic limit. The actual value is axially dependent and is documented in Figure 3.2-5 for Cycle 1 and in the COLR for the current cycle.

** Fuel pellet thermal conductivity degradation evaluations resulted in a reduction of Fq from 2.5 to 2.3 and Fdh from 1.7 to 1.65.

TABLE 3.2-4
(Cont.)

THERMAL AND HYDRAULIC DESIGN PARAMETERS
(For Cycle 1 and Cycle 21)

	Cycle 1	Cycle 21
Coolant Temperature, °F		
Nominal Inlet	542.6	541.9
Average Rise in Vessel	57.8	56.2
Average Rise in Core	60.3	60.0
Average in Core	573.0	573.1
Average in Vessel	571.5	570.0
Nominal Outlet of Hot Channel	633.8	N/C*
Heat Transfer**		
Active Heat Transfer Surface Area, ft^2	52,200	52,100
Average Heat Flux $Btu/hr{\bullet}ft^2$	193,000	205,200
Maximum Heat Flux, $Btu/hr{\bullet}ft^2$	448,000	472,000
Maximum Thermal Output, $^{kw}/_{ft}$	14.5	15.3
Maximum Clad Surface Temperature BOL at Nominal Pressure, °F	657	N/C*
Maximum Average Clad Temperature BOL at Rated Power, °F	715	N/C*

*Not Calculated for Cycle 21
**Cycle 1 values do not include fuel densification effects described in WCAP-8146

TABLE 3.2-4
(Cont.)

THERMAL AND HYDRAULIC DESIGN PARAMETERS
(For Cycle 1 and Cycle 21)

	Cycle 1	Cycle 21
Fuel Central Temperatures for nominal fuel rod dimensions, °F		
Maximum at 100% Power	4100	3670*
Maximum at Overpower Power	4350 (112% power)	4250* (120% power)
DNB Ratio		
Minimum DNB Ratio at nominal operating conditions	1.80	2.50
Pressure Drop, psi		
Across Core	21	26

*The fuel central temperatures were calculated using the PAD 4.0 model (Ref. 80)

TABLE 3.2-5

CORE MECHANICAL DESIGN PARAMETERS[Deleted]
(For Cycle 1 and Cycle 21)

	Cycle 1	Cycle 21
Active Portion of the Core		
Equivalent Diameter, inches	132.7	Same as Cycle 1
Active Fuel Height, inches	144.0	Same as Cycle 1
Length-to-Diameter Ratio	1.09	Same as Cycle 1
Total Cross-Section Area, ft^2	96.06	Same as Cycle 1
Fuel Assemblies		
Number	193	Same as Cycle 1
Rod Array	15 x 15	Same as Cycle 1
Rods per Assembly	204*	Same as Cycle 1
Rod Pitch, inches	0.563	Same as Cycle 1
Overall Dimensions, inches	8.426 x 8.426	Same as Cycle 1
Fuel Weight, MTU	88.60	87.55
Number of Grids per Assembly	7	11
Number of Guide Thimbles	20	Same as Cycle 1
Diameter of Guide Thimbles (upper part), inches	0.545 O.D. x 0.515 I.D.	0.533 O.D. x 0.499 I.D.
Diameter of Guide Thimbles (lower part), inches	0.484 O.D. x 0.454 I.D.	0.533 O.D. x 0.455 I.D.
Fuel Rods		
Number	39,372	Same as Cycle 1
Outside Diameter, inches	0.422	Same as Cycle 1
Diametral Gap, inches	0.0075	0.0075**
Clad Thickness, inches	0.0243	Same as Cycle 1
Clad Material	Zircaloy-4	ZIRLO™
Overall Length	151.8	152.880
Length of End Cap, overall, inches	0.688	0.350 (top), 0.810 (bottom)
Length of End Cap, inserted in rod, inches	0.250	0.13

NOTE: All dimensions are for cold conditions.
*Twenty-one rods are omitted: Twenty provide passage for control rods and one to contain in-core instrumentation.
**Nominal gap.

TABLE 3.2-5
(Cont.)

CORE MECHANICAL DESIGN PARAMETERS[Deleted]
(For Cycle 1 and Cycle 21)

	Cycle 1	Cycle 21
Fuel Pellets		
Material	UO_2 sintered	Same as Cycle 1
Density (% of Theoretical)		
Region 1, 2, & 3	95 (10.4 g/cc)	
[Deleted]		
Region 21, 22 & 23		95.5 (10.47 g/cc) (nominal)
Feed Enrichments w/o		
Region 1	2.28	4.95 (Region 21)
Region 2	2.80	4.60 Region 22A)
Region 3	3.30	4.95 (Region 22B)
		4.60 (Region 23A)
		4.95 (Region 23B)
Diameter, inches		
Region 1, 2, & 3	0.3659	Same as Cycle 1
Length, inches	0.600	0.439 (Enriched) 0.500 (Blanket)
Rod Cluster Control Assemblies		
Neutron Absorber	5% Cd, 15% In, 80% Ag	Same as Cycle 1
Cladding Material	Type 304 SS-Cold Worked	Chrome Plated SS
Clad Thickness, inches	0.019	Same as Cycle 1
Number of Clusters		
Full Length	53	Same as Cycle 1
Number of Control Rods per Cluster	20	Same as Cycle 1
Length of Control Rod, inches	158.459 (overall)	Same as Cycle 1
	150.579 (insertion length)	Same as Cycle 1
Length of Absorber Section, inches	142.000	Same as Cycle 1

NOTE: All dimensions are for cold conditions.

TABLE 3.2-5
(Cont.)

CORE MECHANICAL DESIGN PARAMETERS[Deleted]
(For Cycle 1 and Cycle 21)

	Cycle 1	Cycle 21
Core Structure		
Core Barrel, inches		
I.D.	148.0	Same as Cycle 1
O.D.	152.5	Same as Cycle 1
Thermal Shield, inches		
I.D.	158.5	Same as Cycle 1
O.D.	164.0	Same as Cycle 1
Burnable Poison Rods and Flux Suppressors		
Material	Borosilicate Glass	IFBA WABA Hafnium
Number	1434 (first cycle) [Deleted]	12144 (IFBA) 1136 (WABA) 0 (Hafnium) [Deleted]
Outer Tube, inches (outer diameter)	0.4390	N/A (IFBA) 0.381 (WABA) 0.381 (Hafnium)
Inner Tube, inches (outer diameter)	0.2365	N/A (IFBA) 0.267 (WABA) N/A (Hafnium)
Clad Material	Type 304 SS	N/A (IFBA) Zircaloy-4 (WABA) N/A (Hafnium)
Inner Tube Material	Type 304 SS [Deleted]	N/A (IFBA) Zircaloy-4 (WABA) N/A (Hafnium)
Boron loading, [Deleted] gm/cm*	0.0576	0.000871 (IFBA) 0.00603 (WABA) N/A (Hafnium)

NOTE: All dimensions are for cold conditions.
* Natural B for Cycle 1 and B-10 for current cycle.

TABLE 3.2-6

NOMINAL COMPOSITION OF ZIRLO™ AND ZIRCALOY-4 CLADDING

Element	Zircaloy-4 (wt %)	ZIRLO™ (wt %)
Sn	1.6	1.0
Fe	0.21	0.1
Cr	0.1	0.0
Nb	0.0	1.0
Zr	>97.0	>97.0

3.3 TESTS AND INSPECTIONS

3.3.1 Physics Tests

1. Tests to Confirm Reactor Core Characteristics

   A detailed series of startup physics tests were performed from zero power up to and including 100% power. As part of these tests, a series of core power distribution measurements were made by means of the core movable detector system. These measurements were analyzed and the results compared with the analytical predictions upon which safety analyses were based.

2. Tests Performed During Operation

   To detect and eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration, necessary to maintain adequate control characteristics, is normalized to accurately reflect actual core conditions. When full power is reached initially, and with the control groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation continues, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted and corrected if necessary. This normalization is completed before a cycle burnup of 60 Effective Full Power Days (EFPD) is reached. Thereafter, actual boron concentration can be compared with the predicted concentration, and the reactivity prediction of the core can be continuously evaluated and adjusted.

   Any reactivity anomaly greater than one percent would be unexpected, and its occurrence would be thoroughly investigated and evaluated.

3.3.2 Thermal and Hydraulic Tests and Inspections

General hydraulic tests on models have been used to confirm the design flow distributions and pressure drops (1,2). Fuel assemblies and control and drive mechanisms are also tested in this manner. Appropriate on-site measurements are made to confirm the design flow rates.

Vessel and vessel internals inspections were also reviewed to confirm such thermal and hydraulic design values as bypass flow.

A summary report of appropriate plant testing shall be submitted to the NRC following (1), an amendment to the license involving a planned increase in power level, (2) installation of fuel that has a different design and (3) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the testing and comparison of these values with acceptance criteria.

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, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup programs, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

3.3.3 Core Component Tests and Inspections

To ensure that all materials, components and assemblies conformed to the design requirements, a release point program was established with the manufacturer. This required surveillance of raw materials, special processes, i.e., welding, heat treating, nondestructive testing, etc. and those characteristics of parts which directly affect the assembly and alignment of the reactor internals. The surveillance was accomplished by the issuance of Inspection Reports by the quality control organization after conformance had been verified.

A resident quality control representative performed a surveillance/audit program at the manufacturer's facility and witnessed the required tests and inspections and issued the inspection reports.

Components and materials supplied by Westinghouse to the assembly manufacturer were subjected to a similar program. Quality Control engineers developed inspection plans for all raw materials, components, and assemblies. Each level of manufacturing was evaluated by a qualified inspector for conformance, i.e., witnessing the ultrasonic testing of core plate raw material. Upon completion of specified events, all documentation was audited prior to releasing the material or component for further manufacturing. All documentation and inspection releases are maintained in the quality control central records section. All materials are traceable to the mill heat number.

In conclusion, a set of "as built" dimensions were taken to verify conformance to the design requirements and assure proper fitup between the reactor internals and the reactor pressure vessel.

3.3.3.1 Fuel Quality Assurance

Surveillance audits of the fuel fabrications and the Westinghouse Quality Assurance Program are performed by Entergy.

Quality Assurance Program

The Westinghouse Nuclear Fuel Division's quality assurance program plan is included in Reference 3 that is summarized below.

The program provides for control over all activities affecting product quality, commencing with design and development and continuing through procurement, materials handling, fabrication, testing and inspection, storage and transportation. The program also provides for the indoctrination and training of personnel and for the auditing of activities affecting product quality through a formal auditing program.

Westinghouse drawing and product, process, and material specifications identify the inspections to be performed.

Quality Control

Quality Control (QC) philosophy is based on the following inspections being performed to a 95 percent confidence that at least 95 percent of the product meets specification, unless otherwise noted:

1) Fuel system components and parts.
   The characteristics inspected depend upon the component parts; the QC program includes dimensional and visual examinations, check audits of test reports, material certification, and nondestructive examination, such as X-ray and ultrasonic.

   All material used in this core is accepted and released by QC.

2) Pellets.
   Inspection is performed for dimensional characteristics such as diameter, density, length, and squareness of ends. Additional visual inspections are performed for cracks, chips and surface conditions according to approved standards.

   Density is determined in terms of weight per unit length and is plotted on zone charts used in controlling the process. Chemical analyses are taken on a specified sample basis throughout pellet production.

3) Rod inspection.
   The fuel rod inspection consists of the following nondestructive examination techniques and methods, as applicable:

   a) Each rod is leak tested using a calibrated mass spectrometer, with helium being the detectable gas.

   b) All weld enclosures are X-rayed or Ultrasonically tested. X-rays are taken in accordance with Westinghouse specifications meeting the requirements of ASTM-E-142.

   c) All rods are dimensionally inspected prior to final release. The requirements include such items as length, camber, and visual appearance.

   d) All of the fuel rods are inspected by X-ray or other approved methods to ensure proper plenum dimensions.

e) All of the fuel rods are inspected by gamma scanning, or other approved methods to ensure that no significant gaps exist between pellets.

f) All fuel rods are active gamma scanned to verify enrichment control prior to acceptance for assembly loading.

g) Traceability of rods and associated rod components is established by QC.

4) Assemblies:

Each fuel assembly is inspected for compliance with drawing and/or specification requirements. Other incore control component inspection and specification requirements are given in Reference 3.

5) Other inspections:

The following inspections are performed as part of the routine inspection operation:

a) Tool and gage inspection and control, including standardization to primary and/or secondary working standards. Tool inspection is performed at prescribed intervals on all serialized tools. Complete records are kept of calibration and conditions of tools.

b) Audits are performed of inspection activities and records to ensure that prescribed methods are followed and that records are correct and properly maintained.

c) Surveillance inspection, where appropriate, and audits of outside contractors are performed to ensure conformance with specified requirements.

6) Process control:

To prevent the possibility of mixing enrichments during fuel manufacture and assembly, strict enrichment segregation and other process controls are exercised.

The UO_2 powder is kept in sealed containers. The contents are fully identified by labels completely describing the contents. Isotopic content is confirmed by analysis.

Powder withdrawal from storage can be made by only one authorized group, which directs the powder to the correct pellet production line. All pellet production lines are physically separated from each other and pellets of only a single nominal enrichment and density are produced in a given production line at any given time.

Finished pellets are placed on trays and transferred to segregated storage racks within the confines of the pelleting area. Samples from each pellet lot are tested for isotopic content and impurity levels prior to acceptance by QC. Physical barriers prevent mixing of pellets of different nominal densities and enrichments in this storage area. Substandard pellets are reprocessed and utilized for recycle material.

A serialized traceability code label is electroetched into the tube, to provide unique identification. This identification is maintained and used throughout assembly fabrication and inspection.

Loading of pellets into the clad, which previously had bottom end plug welded in place, is performed in isolated production lines, and again only one density and enrichment are loaded on a line at a time except that the axial blanket pellets are usually a lower enrichment.

The "annular" axial blanket pellets are sintered with a hole in the center to increase plenum volume. The axial blanket pellets are carefully controlled on separate trays located in separate cabinets and have the added characteristic of the centerline hole to minimize improper placement in the pellets stacks.

The loading of IFBA burnable poison rods containing ZrB_2 coated pellets is performed in a separate production line physically separated from standard UO_2 production line. All other operations are similar with the same close control to prevent the misloading of pellets with different enrichment.

The top end plugs are inserted and then welded to seal the tube. At the time of installation into an assembly, a matrix is generated to identify each rod's position within a given assembly. The top nozzle is inscribed with a permanent identification number providing traceability to the fuel contained in the assembly.

The preceding discussion and the references stated describe the efforts in the application of quality control and the use of reliability techniques during the development, design, fabrication, and shipment of fuel assemblies.

Upon delivery of the fuel to the site, an inspection is performed for evidence of shipping damage, loose parts and debris. The shipping container internals, seals, container bolts, clamping fixture, and shock overload probe indicators are inspected. After removal from the container, the fuel itself is inspected for evidence of shipping damage to approved Inspection Procedures. The above inspections, as a minimum, are accomplished for all fuel assemblies and are appropriately documented.

3.3.3.2 Burnable Poison Rod Tests and Inspections

The end plug seal welds are checked for integrity by visual inspection, and X-ray or Ultrasonic testing. The finished rods are helium leak checked.

REFERENCES

1. Hetsroni, G., "Hydraulic Tests of the San Onofre Reactor Model," WCAP-3269-8, 1964.

2. Hetsroni, G., "Studies of the Connecticut-Yankee Hydraulic Model," WCAP-2761, 1965.

3. Moore, J., "Nuclear Fuel Division Quality Assurance Program Plan," WCAP-7800-Revision 5, November 1979.