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4.1 SUMMARY DESCRIPTION

This Chapter describes and evaluates those systems most pertinent to the fuel barrier and the control of core reactivity. Section 4.2, Fuel System Design, describes the mechanical aspects of the fuel material (uranium dioxide), the fuel cladding, the fuel rods, and the arrangement of fuel rods in bundles. Of particular interest is the ability of the fuel to serve as the initial barrier to the release of radioactive material. The mechanical design of the fuel is sufficient to prevent the escape of significant amounts of radioactive material during normal modes of reactor operation.

Section 4.3, Nuclear Design, describes the nuclear aspects of the reactor core. The design of the boiling water reactor core and fuel is based on a proper combination of design variables, such as moderator to fuel volume ratio, core power density, thermal hydraulic characteristics, fuel exposure level, nuclear characteristics of the core and fuel, heat transfer, flow distribution, void content, heat flux, and operating pressure. All of these conditions are dynamic functions during reactor operations. Design analyses and calculations are performed for specific steady state, transient, and accident conditions. Included in this section are summaries of results of the steady state analyses for the fuel cycle, reactivity control, and control rod worths. Also included are discussions of the reactivity coefficients and spatial xenon characteristics of the core. Transients and accident analysis results are presented in Chapter 15.

Section 4.4, Thermal and Hydraulic Design, describes the thermal and hydraulic characteristics of the core. The low coolant saturation temperature, high heat transfer coefficient, and neutral water chemistry of the boiling water reactor are significant advantages in minimizing Zircaloy temperatures and associated temperature dependent hydride pickup. This results in improved cladding performance at long exposures. The relatively uniform fuel cladding temperatures throughout the boiling water reactor core minimize migration of the hydrides to cold cladding zones and reduce the thermal stresses. A discussion of fuel failure mechanisms and the parameters associated with fuel damage is included in this section.

Section 4.5, Reactor Materials, describes the materials used in the Control Rod Drive System and the reactor vessel internal structures.

Section 4.6, Functional Design of Reactivity Control Systems, establishes that the Control Rod Drive System, including the control rod drive mechanisms and hydraulic control system, are designed and installed to provide the required functional performance and are properly isolated from other equipment.

The system is designed so that sufficient energy is available to drive the control rods into the core under conditions associated with abnormal operational transients and accidents. Control rod insertion speed is sufficient to prevent fuel damage as a result of any abnormal operational transient.

The Control Rod Drive System mechanical design is described in detail in Subsection 3.9.4.

A Standby Liquid Control System (Liquid Poison System) provides an independent and diverse method, other than control rods, to make the reactor subcritical, even in the cold condition. The insertion of control rods is expected always to ensure prompt shutdown of the reactor; hot shutdown can be achieved by insertion of only a few of the many independent control rods. The Standby Liquid Control System is described in detail in Section 9.3.5.

A summary of important reactor design and performance characteristics is provided in Table 4.1-1.

TABLE 4.1-1
(Sheet 1 of 1)

REACTOR DESIGN DATA

REACTOR	
Design thermal output	1930 MWt
Operating pressure (steam dome)	1035 psia
Steam flow rate	7.259×10^6 lb/hr
Recirculation flow rate	61.0×10^6 lb/hr
Operating temperature	546°F

CORE	
	GE8x8NB & GE11 Fuel
Number of fuel assemblies	560
Fuel assembly rod geometry	8x8 & 9x9
Number of control rods	137
Shape	Cruciform
Core inlet enthalpy	517.5 Btu/lb
Core average void fraction, active coolant	39.4% (variable)
Fuel	UO_2
Cladding material	Zircaloy-2

4.2 FUEL SYSTEM DESIGN

4.2.1 Design Bases

The OCNGS core consists of 560 fuel assemblies. The core composition includes Exxon fuel assemblies and GE pressurized fuel assemblies. The design bases comprise limits which are consistent with proven practice, as substantiated in the references cited in Subsection 4.2.5.

4.2.2 Description

4.2.2.1 EXXON Type VB Fuel Assemblies

The Exxon Type VB fuel assembly is made up of an 8x8 square array of rods, with an active fuel length of 144 inches. The four rods at the center of the assembly are inert rods filled with solid Zircaloy-2. One of the center rods is the spacer capture rod. The four center rods constitute a passive heat sink within the bundle, which serves to lower the peak clad temperature for the calculated Loss-of-Coolant Accident.

The spacing between rods is maintained by seven Zircaloy spacers containing Inconel springs equally spaced along the length of the fuel rods, and by tie plates at the upper and lower ends of the rods. The spacer capture rod maintains the axial location of the spacers.

The ends of all rods have extensions which fit into holes in the upper and lower tie plates. Eight of the peripheral fuel rods, called tie rods, are threaded into the lower tie plate and pass through the upper tie plates, where they are secured by quick disassembly locking lugs. The remaining rods are restrained within holes in the upper and lower tie plates. Coil springs on each rod, captured by the upper fuel rod extensions act together to force the upper tie plate upward against the tie rod nuts. Each spring also acts individually to assure that the fuel rod remains seated in the lower tie plate.

The level of U-235 enrichment is identified on the end surface of each fuel rod upper end cap by a letter code. During and after assembly, the correct position of each rod is verified by comparison with a template, which specifies the location of each enriched uranium rod. In addition, notches are used to identify individual fuel rods for enrichment and poison loading. A listing of mechanical characteristics of the assemblies is presented in Table 4.2-1.

4.2.2.1.1 Spacers

The spacers are designed to maintain the correct rod to rod spacing, but to allow for differential axial expansion. The spacer uses an egg crate type design of criss crossed narrow zircaloy strips interlocked and welded together with a peripheral band to form a cell for each fuel rod. Each of the fuel rods is centered in its cell by an Inconel spring, which holds the rod against support dimples with sufficient force to minimize flow induced vibrations. Guide tabs are provided on the upper edge of the spacers to avoid possible hang up during channeling operations.

4.2.2.1.2 Upper and Lower Tie Plates

The upper tie plate is a flat perforated plate of cast and machined stainless steel. It is designed to maintain the correct position of the fuel rods, provide flow passages and maintain the position of the fuel assembly within the reactor top grid assembly. A handle is attached to the plate for

loading, unloading and general handling of the fuel assembly. A boss on the lifting bail points to the nearest control rod when the fuel assembly is correctly oriented in the reactor core.

The lower tie plate consists of a flat perforated plate and an inlet box section, all cast and machined as a single unit from stainless steel. The perforated plate is designed to support, and maintain the correct position of the rods and to permit coolant flow through the fuel assembly. The inlet box section distributes coolant from the assembly support section to the fuel rods.

4.2.2.1.3 Fuel Pellets and Rods

The fuel consists of compacted and sintered uranium dioxide powder formed into cylindrical pellets. The nominal density of the pellets is 93.5 percent of the theoretical density (TD) of UO_2. All pellets are dished to remove a nominal one percent of the fuel volume. This dished volume is provided to accommodate fuel axial expansion and irradiation induced swelling.

The fuel pellets are stacked in 0.5015-inch OD Zircaloy-2 cladding rods, sealed by welding Zircaloy-2 caps to each end. The atmosphere within the rods is helium. The requirement that the cladding be free standing is met by specifying a minimum wall thickness of 0.0336 inches (0.0360 inches thickness is provided). The thickness provides sufficient margin against collapse of the cladding onto the pellets due to primary coolant pressure.

The design of the fuel pellets and rods has considered fuel swelling behavior, thermal expansion, densification, distortion of the pellets and fission gas release. Other design considerations include mechanical, nuclear and thermal hydraulic factors. Consideration of these factors at the maximum design peak burnup has resulted in the selection of appropriate pellet densities, dishing requirements, diametral gap and fuel rod gas plenum dimensions.

4.2.2.1.4 Nuclear Characteristics of the Fuel Assembly

The fuel assemblies are designed and fabricated to have a nominal average enrichment of 2.50 w/o U-235. Four of the fuel rods contain gadolinium as a burnable poison. The UO_2 to Gd_2O_3 poison distribution is 2.69/1.00 w/o for those rods. All of the pellets in the burnable poison rods contain gadolinium.

Calculated infinite multiplication factors and other neutronic parameters are provided with the reload submittals for each cycle of the core. Although the values for the uncontrolled infinite multiplication factor generally indicate a positive coolant temperature coefficient, changes of reactivity with coolant temperature during normal startup and operation with a normal control rod pattern are negative. Since the heatup transient during reactor startup occurs while the control rod pattern is at about half density, the temperature defect for this fuel is definitely negative.

4.2.2.2 GE Pressurized Fuel Assemblies

The General Electric fuel assembly consists of a fuel bundle and a channel that surrounds it. Fuel assembly parameters are given in Table 4.2-1. The rods of all bundle types are spaced and supported in a square (8x8, 9x9 and 10x10 fuel) array by the upper and lower tie plates and seven spacers. The lower tie plate has a nose piece which has the function of supporting the fuel assembly in the reactor. The upper tie plate has a handle for transferring the fuel bundle from one location to another. The identifying assembly serial number is engraved on the top of the handle. No two assemblies bear the same serial number. A boss projects from one side of the handle to aid in ensuring proper fuel assembly orientation. Both upper and lower tieplates

are fabricated from Type 304 stainless steel castings. Zircaloy-4 fuel rod spacers with Inconel-X springs are employed to maintain rod-to-rod spacing. Finger springs located between the lower tie plate and the channel are utilized on some fuel assemblies to control bypass flow through that assembly. For GNF2 fuel, the spacers are made entirely from Alloy X750 and no finger springs are used.

4.2.2.2.1 Fuel Rods

Each fuel rod consists of high density ceramic uranium dioxide fuel pellets stacked within Zircaloy-2 cladding which is evacuated, backfilled with helium and sealed with zircaloy end plugs welded in each end. The helium backfill pressure is 3 atm for the GE fuel design. The fuel pellets are manufactured by compacting and sintering uranium dioxide powder into right cylindrical pellets with flat ends and chamfered edges. The average pellet immersion density is approximately 95 to 97 percent of the theoretical density of UO_2. Ceramic uranium dioxide is chemically inert to the cladding at operating temperatures and is resistant to attack by water. Several U-235 enrichments are used in the fuel assemblies to reduce the local peak to average fuel rod power ratios. Selected fuel rods within each reload bundle also incorporate small amounts of gadolinium as burnable poison. Gd_2O_3 is uniformly distributed in the UO_2 pellet and forms a solid solution. Details of UO_2-Gd_2O_3 fuel are given in NEDE-20943-P (proprietary) and NEDO-20943 (January 1977) (Reference 1).

The fuel rod cladding thickness is adequate to be essentially free standing in the 1000 psi range. Adequate free volume is provided within each fuel rod in the form of a pellet to cladding gap and a plenum region at the top of the fuel rod to accommodate thermal and irradiation expansion of the UO_2 and the internal pressures resulting from the helium fill gas, impurities, and gaseous fission products liberated over the design life of the fuel. A plenum spring, or retainer, is provided in the plenum space to minimize movement of the fuel column inside the fuel rod during fuel shipping and handling. A hydrogen getter is also provided in the plenum space as assurance against chemical attack from the inadvertent admission of moisture or hydrogenous impurities into a fuel rod during manufacturing in some GE fuel designs. The content and subsequent reaction of the getter is described in NEDE-24011-P-A (Reference 2).

Three types of fuel rods are used in a fuel bundle: tie rods, standard rods and part length rods. The tie rods in each bundle have lower end plugs that thread into the lower tie plate casting and threaded upper end plugs that extend through the upper tie plate casting. A stainless steel hexagonal nut and locking tab are installed on the upper end plug to hold the fuel bundle together. These tie rods support the weight of the bundle only during fuel handling operations when the assembly hangs by the handle. During operation, the fuel assembly is supported by the lower tie plate. Additional expansion spring design information is given in NEDO-20377 (Reference 3). In the advanced GE fuel design, a third type of fuel rod, called a part length rod, is used. Part length rods provide the ability to better utilize axial power shaping resulting in more efficient fuel design.

4.2.2.2.2 Water Rods

The GE 8x8 fuel assembly can be designed to contain one centrally located water rod, or two or more water rods. The GE 9x9 and 10x10 fuel designs have two centrally located water rods.

4.2.2.2.3 Other Fuel Assembly Components

The primary function of the fuel spacer is to provide lateral support and spacing of the fuel rods, with consideration of thermal-hydraulic performance, fretting wear, strength, neutron economy, and producibility. The spacer represents an optimization of all these considerations.

Finger springs (Reference 2) may be employed to control the bypass flow through the channel to lower tie plate flow path for some fuel assemblies. These finger spring seals, located between the lower tie plate and the channel, provide control over the flow through this path due to channel wall deflections by maintaining a nearly constant flow area as the channel wall deforms.

The upper and lower tie plates serve the functions of supporting the weight of the fuel and position the rod ends during all phases of operation and handling. Reload fuel bundles with alternate path bypass holes in the lower tie plate may be used at Oyster Creek. These holes are drilled to augment flow in the bypass region. The method by which this bypass flow is taken into account is given in Reference 2. The advanced GE fuel designs incorporate a debris filter lower tie plate. This debris filter lower tie plate is very similar to the regular tie plate except for the upper portion, or the grid plate. Additionally, a debris filter cartridge may be used, internal to the lower tie plate that acts as an additional barrier for debris to enter the fuel assembly.

A licensing topical report (Reference 5) provides a complete description and analytical results for channels supplied by the General Electric Company and used in conjunction with the reload fuel described herein. However, the following functional description is included for completeness. The description and evaluation (Reference 5) and the information given below applies only to channels supplied by General Electric.

The General Electric BWR Zircaloy fuel channel performs the following functions:

a. Forms the fuel bundle flow path outer periphery for bundle coolant flow.

b. Provides surfaces for control rod guidance in the reactor core.

c. Provides structural stiffness to the fuel bundle during lateral loadings applied from fuel rods through the fuel spacers.

d. Minimizes, in conjunction with the finger springs and bundle lower tie plate, bypass flow at the channel/lower tie plate interface.

e. Transmits fuel assembly seismic loadings to the top guide and fuel support of the core internal structures.

f. Provides a heat sink during a Loss-of-Coolant Accident (LOCA).

g. Provides a stagnation envelope for in-core fuel sipping.

The channel is open at the bottom. The upper end of the fuel assemblies in a four bundle cell is positioned in the corners of the cell against the top guide beams by the channel fastener springs. At the top of the channel, two diagonally opposite corners have welded tabs, one of which supports the weight of the channel from a threaded raised post on the upper tie plate. One of these raised posts has a threaded hole. The channel is attached using the threaded

channel fastener assembly, which also includes the fuel assembly positioning spring. Channel to channel spacing is provided for by means of spacer buttons located on the upper portion of the channel adjacent to the control rod passage area.

For the P8x8R fuel design, the channels have a uniform thickness of 80 mils for Oyster Creek. The channels for the GE 9x9 and 10x10 fuel designs may have thinner sides and thicker corners.

4.2.2.3 Core Cell and Core Configuration

A core cell is defined as a control rod and four fuel assemblies that immediately surround it. Each core cell is associated with a four lobed fuel support piece. Around the outer edge of the core, certain fuel assemblies are not immediately adjacent to a control rod and are supported by individual peripheral fuel support pieces. The 560 fuel assemblies are arranged in a configuration that approximates a right cylinder, with 548 assemblies arranged in core cell configurations with 12 peripheral assemblies.

The Control Cell Core (CCC) design concept was first introduced at OCNGS during Cycle 11. The basic CCC configuration is a concept developed and patented by General Electric. In this design, control rods are inserted adjacent only to the low power "Control Cell" fuel assemblies while operating at power. The advantages in using the CCC design are: a reduction in fuel duty, the elimination of conventional rod pattern exchanges and an increase in operating margins to thermal limits. The major disadvantage to CCC loading is the reduction of available locations for fresh fuel. This requires a higher enrichment to meet energy needs. Control Cell Core operation may or may not be used in any given cycle.

4.2.3 Design Evaluation

The evaluation of the fuel system design is contained in the supplemental reload licensing submittal for each core cycle.

4.2.4 Testing and Inspection Plan

Testing and inspection programs are conducted by the fuel manufacturer and submitted to NRC as topical reports. Major characteristics of the OCNGS reactor fuel assemblies are provided in Table 4.2-1.

4.2.5 References

(1) Potts, G.A., "Urania - Gadolinia Nuclear Fuel Physical and Irradiation Characteristics and Material Properties," NEDE-20943-P (Proprietary) and NEDO-20943, January 1977.

(2) "General Electric Standard Application for Reactor Fuel (GESTAR II)," NEDE-24011-P-A, as revised at the time of the specified reload cycle.

(3) "8x8 Fuel Bundle Development Support," NEDO-20377, February 1975.

(4) “GE Fuel Bundle Designs,” NEDE–31152P, as revised at the time of the specified reload cycle.

(5) "BWR Fuel Channel Mechanical Design and Deflection," NEDE-21354-2–P (Proprietary) and NEDO-21354–2, July 1977.

TABLE 4.2-1
(Sheet 1 of 1)

MECHANICAL CHARACTERISTICS OF FUEL ASSEMBLIES

	GE8x8EB	GE8x8NB	GE11
Fuel Assembly			
Rod array	8x8	8x8	9x9
Average enrichment (w/o U-235)	3.38	3.38/3.48	3.69/3.74
Rod pitch, inches	0.640	0.640	0.566
Number of full length fuel rods	60	60	66
Number of part length fuel rods	0	0	8
Number of spacers	7	7	7
Fuel weight (kg U/kg UO_2)	171.4/194.7	171.5/194.6	170.6/193.6
Fuel Rods			
Fill gas	Helium	Helium	Helium
Active fuel length, inches	145.24	145.24	145.24
Initial gap, inches	0.008	0.008	0.008
Water Rods			
Material	Zr-2	Zr-2	Zr-2
Outside diameter, inches	0.591/.483	1.34	0.98
Thickness, inches	0.030/0.026	0.040	0.030
Number of Water Rods	2/2	1	2
Cladding			
Material	Zr-2	Zr-2	Zr-2
Outside diameter, inches	0.483	0.483	0.440
Inside diameter, inches	0.419	0.419	0.384
Fuel Pellets			
Diameter, inches	0.411	0.411	0.376
Length, inches	0.410	0.410	0.380
Density (percent TD)	96.5	96.5	97.0

4.3 NUCLEAR DESIGN

4.3.1 Design Bases

The nuclear performance characteristics of the core are designed to provide a nuclear dynamic response which:

a. Has a strong negative reactivity feedback under severe transient conditions.

b. Contributes negative reactivity feedback consistent with the requirements of overall plant nuclear hydrodynamic stability.

c. Has a reactivity response which regulates or damps changes in power level and spatial distribution of power production in the core to a level consistent with safe and efficient operation.

The first characteristic is a major factor, through the Doppler and moderator coefficients, in providing shutdown mechanisms in the event of a reactivity excursion. The latter two characteristics assure, along with other parameters discussed in Subsection 4.3.2.7, that there will be no inherent tendency for undamped oscillations.

Excess reactivity provides the driving force for the dynamic response of the reactor core. The core must be designed to have a dynamic response which contributes substantially to reactor control and ease of operation. The dynamic behavior of the core is characterized in terms of several reactivity coefficients. These are:

a. Fuel Temperature or Doppler coefficient

b. Moderator void coefficient

c. Moderator temperature coefficient.

It is also convenient in some cases to characterize the composite, simultaneous effect of all three coefficients above into a single term called the power coefficient.

In UO_2 fuel the Doppler coefficient provides potential for a large instantaneous negative reactivity feedback to any power rise, either gross or locally, of the core. The magnitude of the Doppler coefficient is inherent in the fuel design and does not vary significantly among various light water moderated UO_2 fuel designs having low enrichment. This coefficient provides negative feedback to terminate large core transients. The design provides limitations on the combination of rate and magnitude of potential reactivity additions, as discussed in Subsection 4.3.2.5.

The moderator void coefficient contributes to nuclear-hydrodynamic stability. The considerations to assure stability are discussed in Subsection 4.3.2.7. Since a number of plant parameters, including the void coefficient, contribute to stability requirements, no specific coefficient value can be used as a design basis. In general, to assure stability, the void coefficient during power operation must not become too negative. The water to fuel ratio provides a void coefficient consistent with other plant parameters in meeting stability limits.

A second requirement is imposed on the moderator void coefficient, or more precisely on the moderator density coefficient interior to the fuel assemblies. In Doppler terminated or controlled transients, this moderator coefficient is relatively slow acting due to the long heat transfer time constant of the fuel. Nevertheless, it should not result in a significant positive reactivity contribution to the core as heat transfers from fuel to coolant. This condition is satisfied if the moderator density or void coefficient interior to the fuel assemblies is designed to be zero or slightly negative in the cold core condition.

The moderator temperature coefficient includes temperature effects interior to fuel assemblies and in the moderator in the gaps between flow channels. This coefficient is even slower acting than the moderator void coefficient as it takes on the order of minutes for the water gaps to reach temperature equilibrium with the circulating coolant. Because of the relative slowness of the water gap temperature to respond in the time domain of transients, the reactivity feedback due to temperature coefficient is negligible for any transient. However, the requirements on the magnitude and sign of the void coefficient restrict the temperature coefficient to become and remain negative prior to attaining the normal reactor operating temperature.

Finally, perturbations of reactor power level result in shifts of power distribution in the core due to the effects of xenon variations. The inherent nuclear characteristics of the core lead to strong damping of such oscillations. Such damping is provided by the operating power coefficient of the core. Operating experience and analysis has indicated that in large boiling water reactors, a power coefficient more negative than about -0.01 Delta k/k Delta P/P provides damping of xenon induced power shifts to the point that they can be maintained within normal operating limits by minor control rod adjustments. (Reference 1)

4.3.2 Description

4.3.2.1 Nuclear Design Description

The reactor is a light water moderated reactor, fueled with slightly enriched uranium dioxide. At operating conditions, the moderator is permitted to boil, producing a spatially variable density of steam voids within the core. The use of water moderator produces a neutron energy spectrum such that the fissions are produced principally by thermal neutrons.

The presence of U-238 in uranium dioxide fuel leads to the production of significant quantities of plutonium during core operation. This plutonium contributes both to fuel reactivity and power production of the reactor. In addition, direct fissioning of U-238 by fast neutrons yields approximately seven percent of the total power and contributes to an increased fraction of delayed neutrons in the core. The U-238 contributes a strong negative Doppler coefficient of reactivity, which improves the inherent response of the reactor and limits the peak power in excursions. The strong negative void reactivity effect contributes to the overall plant stability and to the damping of xenon oscillation effects.

The Oyster Creek reactor was designed to achieve a first core average discharge exposure of 15,000 MWD/T. In reactivity level and reactivity coefficients, the fuel is approximately the same as that used in other operating General Electric reactors. Table 4.3-1 shows historical fuel types used from Cycles 1 through 19. The original Oyster Creek core contained 560 (7x7) fuel assemblies, designated Type I, manufactured by General Electric Company. These assemblies contained no gadolinia. Poison curtains were used for supplementary reactivity control. In the fall of 1971, a partial reload was performed and twenty four (24) fuel assemblies containing gadolinia, manufactured by General Electric Company and designated Type II, were loaded.

The poison curtains were also removed at this time. The Type II assemblies were the subject of Facility Change Request No. 1.

The Oyster Creek Cycle 5 reload (Reference 2) consisted of Exxon Nuclear Company (Exxon) Type III F (7x7), Type VB fuel (8x8) and Type V (8x8) fuel bundles. Type V fuel characteristics were presented in Facility Change Request No. 6. This was the last Facility Change Request. The Type VB fuel described in the Cycle 5 reload submittal is the same as the Type V fuel except for: (1) a decrease in fuel enrichment and burnable poison content, and (2) a decrease in fuel pellet density. The smaller diameter 8x8 rods have a lower maximum linear heat generation rate, and a larger cladding thickness to diameter ratio which results in increased safety margins when compared to the 7x7 fuel assemblies. In particular, the maximum design linear power and maximum fuel temperature are substantially reduced with the 8x8 fuel design.

In Cycle 10, GE fuel was reintroduced into Oyster Creek with the P8x8R design. Barrier fuel was first loaded during Cycle 12 with the GE 8x8EB fuel design.

In Cycle 15 the GE 8x8NB fuel design was introduced into Oyster Creek. This fuel design includes a large central water rod, higher fuel enrichments, axial zoning of both fuel enrichment and Gd concentration, and ferruled spacers. GE 8x8NB is also a barrier design.

In Cycle 19 the GE11 fuel design was introduced into Oyster Creek. This fuel design includes a 9x9 fuel rod array, two large central water rods, higher fuel enrichments, axial zoning of both fuel enrichment and Gd concentration, ferruled spacers and a thick corner, thin wall channel design.

In Cycle 23, the GNF2 fuel design was introduced into Oyster Creek. This fuel design includes a 10x10 fuel rod array, two large central water rods, higher fuel enrichments, axial zoning of both fuel enrichment and Gd concentration, Inconel Alloy X-750 spacers and a thick corner, thin wall channel design. The fuel design incorporates a short and a long part length fuel rod design and an option to use a debris filter cartridge in the lower tie plate.

Information relative to the General Electric reload fuel is presented in NEDE-31152 (Reference 20). The information contained in the report represents information that is independent of a cycle by cycle reload application. Wherever possible, results of bounding analyses are given.

4.3.2.2 Power Distribution

During fuel burnup, control rods are used in part to counteract the power distribution effect of steam voids as indicated by the in-core flux monitors. Taken together, the control rod and void distributions may be used to flatten gross power beyond that possible in the non boiling core. The design provides considerable flexibility to control gross power distribution. This permits control of fuel burnup and isotopic composition throughout the core to the extent necessary to counteract the effect of voids on the axial distribution at the end of a fuel cycle, when few control rods remain in the core.

Core power distribution can be described in terms of various peaking factors. The local radial, axial, gross and total peaking factors are design parameters related to reload core analysis. Their respective definitions are given below.

Local Peaking Factor

The ratio of the heat flux in the highest powered rod at a given plane in a particular fuel assembly to that of the average rod in the plane.

Radial Peaking Factor (RPF)

The ratio of the fuel assembly power in a particular assembly to the power of the average fuel assembly.

Axial Peaking Factor (APF)

The ratio of the heat flux at the axial plane of interest to the heat flux averaged over the active length of the fuel (assembly or rod) of interest.

Gross Peaking Factor (GPF)

The product of the radial peaking factor for a fuel assembly and the maximum axial peaking factor for the same fuel assembly.

Total Peaking Factor (TPF)

The total peaking factor is that peaking factor which, when multiplied by the average linear heat generation rate of a specific bundle (fuel assembly) type, yields the Technical Specification limit on MLHGR for that bundle type.

These peaking factors determine, directly or indirectly, the thermal performance parameters of the fuel such as maximum linear heat generation rate (MLHGR), maximum average planar linear heat generation rate (MAPLHGR) and minimum critical power ratio (MCPR). The relations between the various peaking factors and the core thermal performance parameters are as follows:

a Maximum Linear Heat Generation Rate (MLHGR)

The MLHGR is the maximum linear heat generation rate expressed in kW/ft) in any fuel rod allowed by the Technical Specifications for a given fuel type. The MLHGR is attained when the product of the local, radial and axial peaking factors in an axial segment of a fuel bundle equals the total peaking factor for that fuel type.

b. Maximum Average Planar Linear Heat Generation Rate MAPLHGR)

The MAPLHGR is the maximum average linear heat generation rate (expressed in kW/ft) in any plane of a fuel bundle allowed by the Technical Specifications for that fuel type. This parameter is obtained by averaging the LHGR over each fuel rod in the plane of interest and its limiting value is selected such that:

1. The peak clad temperature during the design basis Loss-of-Coolant Accident will not exceed 2200°F.

2. The peak oxidation fraction during the design basis Loss-of-Coolant Accident will not exceed 17%.

The MPLHGR is attained when the gross peaking factor equals the Technical Specifications value.

c. Minimum Critical Power Ratio (MCPR)

The MCPR operating limit is the minimum CPR allowed by the Technical Specifications for a given bundle type. The CPR is a function of several parameters, the most important of which are bundle power, bundle flow and bundle R-factor. The R-factor is dependent upon the local power distribution, but is only indirectly related to the local peaking factor. The limiting value of CPR is selected for each bundle type such that 2during the most limiting event of moderate frequency, the CPR in that bundle will not be less than the safety limit CPR. The MCPR is attained when the bundle power, R-factor, flow, and other relevant parameters combine to yield the Technical Specifications value. Therefore, MCPR is not directly related to any of the peaking factors described above.

The peaking factor, by itself, does not constitute a limiting condition. The thermal performance parameters such as MLHGR, MAPLHGR and MCPR do limit unacceptable combinations of these peaking factors.

For a given lattice at a given void fraction, the maximum local peaking factor will occur at different fuel rods as the exposure increases. This is due to the different depletion and generation rate of the various fissile nuclides in each fuel rod.

Figures showing the local pin power distribution as a function of void fraction and exposure for typical General Electric fuel lattices are provided in Reference 4.

The axial power shapes and axial peaking factors are dependent on the fuel bundle types and exposures, the in-core locations, the control rod pattern, and the specific reload cycle. These axial power shapes and axial peaking factors are calculated by the three dimensional BWR simulator, which takes all of the effects into account. Therefore, while it is possible to provide curves showing the variation of the local peaking factor as a function of exposure, it is not possible to do so for the axial peaking factor.

4.3.2.3 Reactivity Coefficients

Reactivity coefficients (the differential changes in reactivity produced by differential changes in core conditions) are useful in calculating relative stability and evaluating response of the core to external disturbances. The base initial condition of the system and the postulated initiating event determine which of the several defined coefficients are significant in evaluating the response of the reactor.

The coefficients of interest, relative to BWR systems, are discussed herein individually with references to the types of events in which they significantly affect the response.

There are two primary reactivity coefficients that characterize the dynamic behavior of boiling water reactors during power operation: (1) The Doppler coefficient and (2) the moderator void coefficient. Also associated with the BWR is a moderator temperature coefficient and a power reactivity coefficient. However, the power coefficient is just a combination of the other coefficients and the moderator temperature coefficient is of little importance in the power

operating range since the moderator temperature remains essentially constant. As a result, these two quantities are no longer calculated for the reload cores. The Doppler and moderator void coefficients are unique for each core and reload. Representative reactivity coefficients and other nuclear parameters are given in Table 4.3-2.

4.3.2.3.1 Doppler Coefficient

The Doppler coefficient is of prime importance in reactor safety. The Doppler coefficient is a measure of the reactivity change associated with an increase in the absorption of resonance energy neutrons caused by a change in the temperature of the fuel. The Doppler reactivity coefficient provides instantaneous negative reactivity feedback to any rise in fuel temperature, on either a gross or local basis. The magnitude of the Doppler coefficient is inherent in the fuel design and does not vary significantly among BWR designs having low fuel enrichment. For most structural and moderator materials, this effect is not significant; but in U-238 and Pu-240 an increase in temperature produces a comparatively large increase in the absorption cross section. The resulting increased absorption of neutrons causes a significant loss in reactivity. In BWR fuel, in which approximately 98 percent of the uranium in UO_2 is U-238, the Doppler coefficient provides an immediate reactivity response that opposes fuel fission rate changes.

Although the reactivity change caused by the Doppler effect is small compared to other power related reactivity changes during normal operation, it becomes very important during postulated rapid power excursions in which large fuel temperature changes occur. The most severe power excursions are those associated with rod drop accidents. A local Doppler feedback associated with a 3000°F to 5000°F temperature rise is available for terminating the initial excursion.

The Doppler reactivity decrement is derived directly from the lattice calculations by performing calculations at different fuel temperatures as described in detail in References 5 and 16.

Maximum and minimum calculated Doppler coefficients for P8x8R fuel central lattices as a function of fuel temperature and exposure are shown in Figures 4.3-1 and 4.3-2. Uncertainty in the nominal Doppler coefficient application of the point model to three dimensional analyses and effect of exposure on the Doppler coefficient are discussed in detail in Reference 5, in response to NRC questions in Reference 6 and for currently approved methods in Reference 16.

The Doppler coefficient for a core made up of a number of different bundle types (lattice designs) at characteristic exposures can be determined analytically.

Figure 4.3-3 shows the Doppler coefficient as a function of fuel temperature and steam voids for unirradiated fuel in the initial OCNGS core. The behavior with exposure is assumed constant for accident analyses although in fact contributions from plutonium, particularly PU-240, will increase the magnitude of the coefficient by 10 to 15 percent at the end of cycle (EOC).

Figure 4.3-4 illustrates the total Doppler reactivity defect (the negative integrated Doppler reactivity coefficient) for the initial core under normal steady state operating conditions up to an average fuel temperature of about 1000°F. This curve includes the effects on the Doppler reactivity defect of both fuel temperature and steam voids characteristic of normal operation.

Figure 4.3-5 shows the total Doppler reactivity available under abnormal conditions for the initial core. Doppler defects are shown for adiabatic fuel heating transients starting from cold, hot standby and rated power fuel temperatures. Fuel temperatures on the abscissa represent

effective average fuel temperatures in the core. This shows that substantially more Doppler negative reactivity is available than is required to terminate an excursion caused by removal of any single control rod from a normal pattern.

4.3.2.3.2 Moderator Void Coefficient

One of the most important considerations for reactor safety is void reactivity. The moderator void coefficient must be large enough to prevent power oscillations due to spatial xenon changes, yet small enough that pressurization transients do not unduly limit plant operation. In addition, the void coefficient in a BWR has the ability to flatten the radial power distribution and to provide ease of reactor control due to the void feedback mechanism. The overall void coefficient is always negative over the complete operating range, since the BWR design is undermoderated. The reactivity change due to the formation of voids results from the reduction in neutron slowing down due to the decrease in the water to fuel ratio.

A detailed discussion of the methods used to calculate void reactivity coefficients, their accuracy and their application to plant transient analysis is presented in References 5 and 16.

The maximum and minimum calculated reactivity effect of a change in void for P8x8R central lattices is shown in Figure 4.3-6. The void coefficient as a function of voids can be calculated using a point model approach. Figure 4.3-7 shows the moderator void coefficient of reactivity for beginning of life and at 10,000 MWD/T fuel exposure.

4.3.2.3.3 Moderator Temperature Coefficient

The moderator temperature coefficient is not a significant reactivity coefficient because its effect is limited to primarily the reactor startup range. Once the reactor reaches the power producing range, boiling begins and the moderator temperature remains essentially constant. As with the moderator void coefficient, the moderator temperature coefficient is associated with a change in the moderating power of the water. The temperature coefficient is negative during power operation.

The range of values of moderator temperature coefficients encountered in current BWR lattices does not include any that are significant from the safety point of view. Typically, the temperature coefficient may range from $+4x10^{-5}$ k/k°F to $-14x10^{-5}$ k/k°F, depending on base temperature and core exposure. The small magnitude of this coefficient (relative to that associated with steam voids) and combined with the long time-constant associated with transfer of heat from the fuel to the coolant, makes the reactivity contribution of moderator temperature of small importance.

Because of its relative insignificance, current core design criteria do not impose limits on the value of the temperature coefficient. A measure of design control over the temperature coefficient is exercised by applying a design limit to the void coefficient. This constraint implies control over the water-to-fuel ratio of the lattice; this, in turn, controls the temperature coefficient. Thus, imposing a quantitative limit on the void coefficient effectively limits the temperature coefficient.

Figure 4.3-8 shows the moderator temperature coefficient of reactivity for beginning of life and 10,000 MWD/T fuel exposure.

4.3.2.4 Control Requirements

The primary purpose of a reactivity control system is to compensate for and regulate the excess reactivity designed into the core. This is accomplished in the Oyster Creek reactor using control rods. For the initial core loading, the control rods were supplemented by temporary control curtains which were removed during the first refueling outage.

The control rods are designed to provide adequate control of the maximum excess reactivity anticipated during the equilibrium fuel cycle operation. The initial core loading had an excess reactivity somewhat higher than that of the equilibrium core. Thus, the basis for design of the temporary control curtains was that they shall compensate the reactivity difference between the initial and equilibrium cores.

There are several types of control rods in the OC core, including All B_4C – Conventional Control Rod, Duralife-230 (D-230), Marathon, and Marathon Ultra MD (also referred to as Marathon-5S). These control rod types provide the same reactivity control. The D-230 was designed to increase core residence time. D-230 control rods are primarily used in the control cells of the CCC core. The first GE Marathon control rod was loaded in the OC core during the 12R refueling outage (November 1988). Since the OC Marathon control rod is the lead test rod, GE inspected the Marathon control rod during EOC-12 and EOC-13 refueling outages. The inspection revealed no abnormalities. OC is evaluating the Marathon control rod performance for expanded use in future cycles. Marathon Ultra MD controls rods were introduced in the 23R refueling outage (November 2010).

4.3.2.4.1 All B_4C - Conventional Control Rod

The cruciform-shaped control rods contain a number of vertical stainless steel tubes filled with boron carbide powder, compacted to approximately 70 percent of theoretical density. The control rod is shown in Figure 4.6-7. Plugs are welded into the ends of the tubes to seal them. The boron carbide powder is separated longitudinally into independent compartments by stainless steel balls at approximately 18 inch intervals, held in place by a slight crimping of the tube. This feature tends to spread uniformly any compaction of the powder during control rod life. A free volume of approximately 30 percent is provided in each tube as a plenum for helium from the B-Li (n, alpha) reaction.

The tubes are held in cruciform array by a stainless steel sheath extending the full length of the poison section. A cruciform shaped top casting and handle aligns the tubes and provides structural rigidity at the top of the control rod. Rollers attached to the top casting maintain the spacing between the control rod and the fuel assembly channels. A similar connector casting is located at the bottom of the control rod and contains a velocity limiter section and rollers to position the lower part of the control rod in the control rod guide tube, located below the core. These bottom rollers always remain in the guide tube during operation. A coupling at the bottom of the control rod is connected and locked to the control rod drive index tube by an expandable ball and socket joint. The control rods are capable of being positioned in six inch steps to control neutron flux distribution in the reactor core. They are moved individually at an average rate of approximately three inches per second.

Fuel reactivity is a maximum at ambient temperatures with no fission product poisons, and control strength is a minimum at this condition, therefore shutdown capability is evaluated assuming a beginning of cycle, cold, xenon free core.

The basic criterion is that the core in its maximum reactivity condition be subcritical (effective multiplication factor, k_{eff}, less than 1.0) with the control rod of highest worth fully withdrawn and all operable rods fully inserted. At most times in core life more than one control rod drive could fail mechanically and this criterion would still be met. The criterion has been established in order to provide a substantial shutdown margin with all rods in and also facilitate maintenance and testing of control rod drive.

In order to assure that the basic criterion will be satisfied an additional design margin was adopted: that the k_{eff} will be less than 0.99 with the rod of highest worth fully withdrawn. Thus the design requirement is k_{eff} less than 0.99, whereas the minimum condition for operation is k_{eff} less than 1.0 with the rod of highest worth fully withdrawn. This limit allows control rod testing at any time in core life and assures that the plant can be shut down by control rods alone.

This limit applies any time in plant life. In addition to the control rod shutdown requirements, the Standby Liquid Control System (Liquid Poison System), discussed in Subsection 9.3.5, can shut the plant down at any time in plant life.

The reactivity control system is designed such that, under conditions of normal operation: (1) sufficient reactivity compensation is always available to make the reactor adequately subcritical from its most reactive condition, and (2) means are provided for continuous regulation of the core excess reactivity and reactivity distribution.

For gross or local reactor power disturbances resulting from operator error or equipment malfunctions the control rods will respond upon signal of the Reactor Protection System (see Section 7.2) to prevent fuel damage.

The inherent safety features of the reactor design as described in Section 4.6, in combination with reactivity control system devices are such that the consequences of a potential nuclear excursion accident, caused by any single component failure within the reactivity control system itself, cannot result in damage either by motion or rupture to the reactor primary coolant system.

Reactivity balances have not been used in describing BWR behavior because of the strong dependence of, for example, rod worth on temperature and void fraction. Therefore, the design process does not produce components of a reactivity balance at the conditions of interest but gives the effective multiplication factor representing all effects combined. Further, any listing of components of a reactivity balance is quite ambiguous unless the sequence of the changes is clearly defined.

Reactivity status of the BWR core under conditions of interest is calculated using accurate representations of the fuel and control conditions. In the cold condition a calculation in two dimensions is performed for the entire core with the control rod of maximum worth withdrawn. For final design purposes, a k_{eff} value of 0.99 or less is acceptable. For the operating condition at full power, the fuel Doppler effect, xenon concentrations and space dependent void fractions are used as input, and the result is the control rod pattern required for a critical reactor, i.e. k_{eff} = 1.0. Temperature and void coefficients effective at some base condition are calculated as differential quantities in detail as described in Subsection 4.3.2.3.

Although reactivity balances are not useful in BWR design, it is possible to perform additional calculations to obtain the reactivity changes associated with identifiable changes in core

components or conditions taken either singly or in specified combinations. For each case, the base conditions, proposed change and desired result must be specified in detail.

The change in the core equilibrium xenon reactivity as a function of exposure is shown in Figure 4.3-9. At the beginning of life (BOL) the equilibrium xenon reactivity is -0.0288 Delta k/k. Assuming that the average core exposure at the end of life (EOL) is 10,000 MWD/T, the equilibrium xenon reactivity is -0.02625 Delta k/k.

The maximum xenon reactivity buildup upon shutdown from full power and the rate of reactivity burnout on return to full power from maximum shutdown xenon buildup were calculated for both the BOL and EOL reactor conditions. It was assumed that the average core exposure at EOL was 10,000 MWD/T for this analysis.

In the analysis, the maximum rate of change of reactivity is obtained by assuming an instantaneous return to full power. The results of these calculations are shown in Figure 4.3-10 for the EOL and BOL.

From this analysis, it was determined that the maximum reactivity addition due to the burnup of xenon was +0.00010 (Delta k/k)/min. Assuming a control rod worth of 0.001 Delta k/k with an insertion rate of 3 in/sec, the reactivity addition due to the control blade insertion is -0.00125 (Delta k/k)/min. Therefore, it is seen that even a very weak control rod can more than compensate for the reactivity addition due to xenon burnout.

If the control rod system fails to respond to an insertion signal, the Standby Liquid Control System is more than adequate to compensate for the reactivity addition due to xenon burnout. The Standby Liquid Control System (SLCS) is capable of injecting sufficient boron concentration to provide a negative reactivity worth equal to the combined effects of rated coolant voids, fuel Doppler, xenon, samarium and temperature changes plus shutdown margin to bring the reactor from full power to cold shutdown. The negative reactivity insertion for the SLCS is significantly greater than the reactivity addition due to xenon burnout. Refer to section 9.3.5 for further discussion of the SLCS.

It must be noted that normal procedure following failure of the control rods to respond to an insertion signal would be a manual scram which would insert all rods simultaneously.

4.3.2.4.2 <u>DURALIFE-230 Control Rod:</u>

The D-230 control rod has the following changes over the conventional design:

- A hafnium absorber plate six inches long is added to the tip of each wing to increase rod life time.
- The three full length B_4C absorber rods at the edge of each wing have been replaced with a hafnium strip.8
- The new lighter weight velocity limiter offsets the weight of the hafnium absorber plates.
- Improved absorber rod tube material to eliminate cracking during control rod lifetime.88

Figure 4.6-8 shows the D-230 control rod assembly. Typical parameters for control rods are described in Table 4.3-5. The mechanical and nuclear properties of the D-230 do not differ from the original all-B_4C control rod assembly.

The end of life for the conventional control blade is reached when a control blade is predicted to have greater than 34 percent B^{10} depletion averaged over the upper one fourth (upper 3 feet) of the blade. The end of life for the D-230 control blade is 62% B^{10} depletion. The B^{10} depletion is tracked for each control rod in the core. The core residence time of a D-230 control rod is approximately 230% over that of a conventional control rod.

4.3.2.4.3 Marathon Control Rod

The Marathon control rod design has the following major changes over the conventional design:

1. The function of the sheath and the absorber rods are performed by a single square tube structure.

2. The full length tie rod is replaced with a segmented tie rod.

3. The B_4C absorber is contained in separate capsules.

4. The absorber tube structure is fabricated from a special high purity stabilized 304 stainless steel.

5. Each absorber tube contains a combination of B_4C, hafnium, and empty space.

The typical parameters of Marathon and existing control rods are described in Table 4.3.5.

Figure 4.6-9 shows the Marathon control rod assembly. The Marathon control rod is fabricated by laser welding multiple square tubes together to form the cruciform configuration. The region between each pair of square tubes contains helium and is sealed on the top and on the bottom by welding.

All interfacing dimensions between the velocity limiter and the guide tube, fuel support casting and control drives are consistent with the conventional design. The mechanical and nuclear properties of the Marathon control rod do not differ from the original all B_4C control rods. The end of life for the Marathon control rod is provided by Reference 19.

4.3.2.4.4 Marathon Ultra MD Control Rod

The Marathon Ultra MD is a derivative version of the Marathon control rod design. The outer absorber tubes are edge welded together to form the cruciform CRB shape and they are filled with capsules containing boron carbide (B_4C) powder. The design changes made to the Marathon control rod result in a more producible, medium duty version design. The design differences are in absorber tube geometry, capsule geometry, capsule length, potential use of a new velocity limiter design, and full length tie rod.

A nuclear evaluation of the Marathon Ultra MD control rod shows that the initial cold and hot reactivity worths are with $\pm$ 5% of the original equipment control rod. Therefore, the Marathon Ultra MD is a direct nuclear replacement for previous control rod designs. End of life criteria are specific in Reference 19.

The structure of the Marathon Ultra MD control rod has been evaluated during all normal and upset conditions, and has been found to be mechanically acceptable. The fatigue usage of the control rod has been found to be well below lifetime limits.

4.3.2.5 Control Rod Patterns and Reactivity Worths

In an operating reactor there is a spectrum of possible control rod worths, depending on the reactor state and on control rod patterns chosen for operation. Control rod withdrawal sequences are selected prior to operation to result in patterns which achieve optimum core performance and, simultaneously, low individual rod worths.

The first barrier or line of defense in preventing the establishment of high worth control rods is the reactor operator and the control rod withdrawal sequence. By procedures the reactor operator follows the rod by rod withdrawal sequence provided by the nuclear engineer. This sequence, in addition to providing for an efficient startup, minimizes the reactivity worth of the control rods to be withdrawn. The Control Rod Worth Minimizer (RWM) has been designed as a backup to the operator so that if procedures are violated, the RWM will block rod motion. See Section 7.7 for a detailed discussion of the Control Rod Worth Minimizer.

From a control rod worth standpoint and resulting reactivity excursions, the RWM is not required above approximately 10 percent power, thus operating without the RWM above this power level is of no consequence. In addition, experience has shown that power reactors do operate at power (above 10 percent rated power) a large fraction of the time. This time can be used to perform maintenance and repairs on the RWM which gives a higher availability during those times when the RWM is needed.

Given, however, that the RWM is out of service during operation at less than approximately 10 percent power, less favorable control rod patterns which contain high worth rods in the withdrawal sequence could be set up, but only if the reactor operator violates an operating procedure.

The RWM prevents withdrawal errors and in addition prevents more than two insertion errors. With the RWM operating as above, there are no control rods in the withdrawal sequence with reactivity worths greater than approximately 1 percent Delta k. If the reactor operator operates within the bounds established by procedures, whether the RWM is operational or not, the maximum control rod worths of in-sequence control rods are the same. Therefore equivalent operator errors would result in the same rod worths for operations with or without the backup control of the RWM. There is nothing inherent in the RWM which, because it is operational, gives lower rod worths than when the operator is running the plant by the same rules without it.

It should be made clear, however, that there is no connection between the stuck rod Technical Specifications and the RWM, i.e., the RWM provides no interlock action which ensures that the stuck rod specification is met. If following this specification should require changes in the control rod pattern, these can be accommodated to some extent by the RWM or new control rod patterns which do accommodate the situation can be programmed into the RWM.

A second concern is whether a single failure in the reactivity control system can accidentally lead to a reactivity addition rate to the core such that the resulting nuclear excursion has potential for primary system damage. As discussed in numerous prior BWR submittals, the potential for such damage can be correlated against peak fuel energy density (cal/gm) resulting

from the excursion. Having defined a limit on peak fuel enthalpy, a locus of control rod worth and rod withdrawal velocity (reactivity rate determining parameters) can be found for which the limit is satisfied. Such a locus is shown in Figure 4.3-11.

If the control system is designed such that the combination of maximum rod worth and rod velocity under postulated failure conditions is such that these parameters fall below the locus, then there is no concern. If such is not the case, then additional design features must be added to accomplish this. These added design features can take two forms. First, control rod worths can be limited so that for any possible rod velocity, fuel enthalpies are constrained below the limit. Conversely, rod velocities can be restricted to a range such that for all credible control rod worths, the peak fuel enthalpy is again held within limits. Limit lines corresponding to such constraints are shown in Figure 4.3-11. Obviously, combinations of such devices also may satisfy the limiting enthalpy.

For nuclear excursions hypothesized to yield peak fuel enthalpies in excess of 425 cal/gm, there exists the potential of thermal to mechanical energy conversion efficiencies of a few percent. Prompt fuel dispersal and heat transfer time constants of several milliseconds are anticipated in this range. For excursions of this severity, large pressure gradients would be generated which would almost certainly damage vessel internals. If sufficient energy were contained in fuel with enthalpies above 425 cal/gm, the integrity of the primary system would be threatened.

Recognizing uncertainty in this damage threshold and adopting an additional margin, a conservative value of 280 cal/gm, corresponding to just molten fuel, was chosen as a reactivity control system design limit. That is, with all features of the control system operable, no single failure was permitted to lead to an excursion yielding more than 280 cal/gm maximum fuel enthalpy.

To meet the prescribed enthalpy limit, both the rod worth minimizer and control rod velocity limiter were simultaneously studied for the Oyster Creek and similar BWR plants. These protective devices, identified as engineered safeguards in early NRC reviews, were conceived, designed, and tested to limit both rod worths and velocities. The Rod Worth Minimizer (RWM) was developed as a constraint against control rod patterns leading to high worth rods. Control rod drive thimble supports were designed to preclude the high rod velocities associated with potential thimble failure and rod ejection. Rod velocity limiting devices were conceived to restrict the rate of rod drop associated with other types of possible control drive failure.

Early work favored the RWM as the primary protective device due to uncertainty in the feasibility of development of reliable velocity limiting devices. However, subsequent design and prototype testing of the latter devices showed that they were capable of meeting the maximum velocity limit of 3.11 ft/sec. Since these devices, as finally designed, are fixed and passive with no moving parts, they are extremely reliable. This excellent performance and high reliability led to consideration of the velocity limiting devices as the primary protective equipment instead of the RWM. With rod velocities reliably constrained to 3.11 ft/sec or less, analysis indicates that:

1. rod drop accidents involving in-sequence control rods (no operator errors) will always result in peak fuel enthalpies less than 280 cal/gm;

2. above 2% power, rod drop accidents involving maximum worth rods developed due to the worst single operator error will always result in peak fuel enthalpies less than 280 cal/gm; and

3. above 10% power, even multiple operator errors will not produce rod worths large enough to exceed fuel enthalpies of 280 cal/gm.

The control rod drop accident (RDA) results for plants using banked position withdrawal sequences (BPWS) show that in all cases the peak fuel enthalpy in an RDA would be much less than the 280 cal/gm design limit even with the maximum incremental rod worth. The BPWS is developed prior to initial operation of the unit following any refueling outage and the requirement that the operator follow the BPWS is supervised by the RWM or a second licensed operator. If it is necessary to deviate slightly from the BPWS sequence (i.e., due to an inoperable control rod) no further analysis is needed if the maximum incremental rod worth in the modified sequence is equal to or less than 1.0% delta k.

The RWM also interlocks against rod insertion. In regards to insertion errors, it must be understood that an insertion error is in no way equivalent to a withdrawal error. This is because the inserted rod cannot fall out of the core and the nuclear importance is shifted away from the inserted rod and its nearby neighbors. For an erroneously withdrawn control rod, the control rod can fall out of the core and the nuclear importance shifts toward the withdrawn control rod and its neighbors.

Multiple insertion errors of course could lead to a high worth control rod being set up, but many errors are required. Thus, the RWM allows only two insertion errors which is well below any point at which high worth control rods could be set up.

The two-insertion error permissive allows the operator some freedom to correct offnormal conditions which may arise, such as adjusting a slightly excessive period, or to allow passing over a stuck rod which could be included in a withdrawal sequence. Analysis has shown that a rod left inserted in the latter case will never assume a worth higher than it would have had at the time of normal withdrawal. This is because continued withdrawal of other rods diminishes the statistical importance of any such rod left in position.

If the RWM is out of service, normal startup procedure would still be followed but would not be automatically monitored. Additional personnel monitoring would be used instead, as discussed in paragraph 3.2.B.2.b of the Technical Specifications (only one reactor startup without an operable RWM per calendar year is allowed, as specified in the Technical Specifications). If no operational errors were committed, rod worths and accident potential would be exactly the same as if the RWM were in operation. Rod grouping in startup sequences utilized in the RWM are exactly those that are the basis for the detailed sequence employed in a normal startup whether monitored by the RWM or not. The question then becomes one of evaluation of the probability of significant operational errors occurring in a startup unmonitored by the RWM.

A typical startup withdrawal sequence may require pulling one rod of every eight in the reactor in a scattered pattern. All such rods would constitute a RWM group. The specific sequence of withdrawal within such a group does not significantly affect rod worth so no safety problem can be created by variations in the order in which rods are withdrawn within the given group. Only a withdrawal outside a specified group need be considered.

Subsequent normal rod group withdrawals would generally result in successively a one fourth, a three eighths, and one half rod out patterns. Any rod passed by in a normal sequence does not increase in worth as additional rods are pulled.

The more likely withdrawal errors, if not prevented by the RWM, therefore, would not lead to rod worths above those that can be tolerated in an excursion. Excessive rod worths may arise only by approaching the classical pattern consisting of an isolated rod surrounded by a region of withdrawn rods. Such a pattern is unique, obvious, and difficult to achieve. It is inconsistent with any normal startup procedure and would lead to difficulties in arriving at power operating patterns consistent with core thermal limits and may produce scram signals from the Intermediate Range Monitor System. Stated simply, high worth patterns are not of the type that can be set up inadvertently by one or two operator errors in the course of normal operation. They generally would require a deliberate and obvious deviation from normal procedures. If such patterns were required by special test conditions, they would be subjected to special surveillance. Further, should a mechanical failure occur on a rod in such a pattern, it would be immediately obvious to any trained operator by the lack of in-core instrument response as withdrawal commenced.

Consideration of the spectrum of potential rod worths; the relatively few unique rod configurations and reactor states that yield worths of concern; the probability of inadvertently setting up these configurations; and the further conditions of required failure and operator action to cause an accident lead to the conclusion that no compromise of plant safety would result if the RWM were out of service and the reactor were started up using normal carefully planned and monitored startup procedures.

To limit the worth of the rod which would be dropped, the RWM is used below 10 percent power to enforce the rod withdrawal sequence. The RWM is programmed to follow the BPWS control rod sequences shown in Figure 4.3-13. The rod drop accident design limit restricts peak enthalpies in excess of 280 cal/gm for any possible plant operation or core exposure.

4.3.2.6 Criticality of Reactor While Shutdown

The Technical Specifications require that the refueled core must be capable of being made subcritical with margin in the most reactive condition throughout the subsequent operating cycle and during refueling with the most reactive control rod in its full-out position and all other rods fully inserted.

As exposure accumulates and burnable poison depletes in the least burned fuel bundles, an increase in core reactivity may occur. The nature of this increase depends on specifics of fuel loading and control state. For example, if one control cell is loaded with two fresh bundles, then the core reactivity calculated with that single control rod removed will probably increase as exposure accumulates. However, the core reactivity calculated with a different rod out may decrease with increased exposure.

Cold k_{eff} is calculated with the most reactive control rod out at various exposures through the cycle. At each exposure point, a search is made for the single most reactive rod, the location of which may change with exposure. The value R, is defined as the difference between the most reactive rod out k_{eff} at BOC and the maximum calculated most reactive rod out k_{eff} at any exposure point. The most reactive rod out k_{eff} at any exposure point is equal to or less than:

$$(\text{Fully Controlled } k_{eff})\text{BOC} + (\text{Most Reactive Rod Worth})\text{BOC} + R$$

The shutdown margin for the core is obtained by subtracting the maximum calculated most reactive control rod out k_{eff} from the critical k_{eff} of 1.0.

These values are unique to each reload. They are calculated using the BWR simulator code to determine the core reactivity with all rods withdrawn and with all rods inserted. The core is assumed to be in a xenon free condition. The value of R includes equilibrium Samarium. An example tabulation of the values for reload Cycle 18 is provided in Table 4.3-3.

During core alterations, fuel assemblies are allowed to be placed in intermediate positions prior to being placed in their final core loading position. A significant amount of reactivity can be added to a subcritical configuration by the addition of a highly reactive assembly. The shutdown margin requirements, which are evaluated from the final core configuration, may not be met for all intermediate fuel assembly positions. Therefore, fuel shuffles must be conducted in a manner to control the intermediate position of highly reactive fuel assemblies in order to meet shutdown margin requirements.

4.3.2.7 Stability

In order to insure that the Oyster Creek reactor will exhibit acceptable stability, the General Electric Company has established criteria and design guides which the final design must meet (Reference 3). Compliance with these criteria have been achieved, thus stable dynamic performance is expected.

The stability of the plant, including the basic process, associated equipment and control systems, has been evaluated by an extensive plant analytical simulation model. The same perturbations introduced in the model to determine stability were introduced into the plant during startup testing to determine actual stability margins.

Whenever there is negative feedback, whether it be inherent self regulation in the process or added to the process by a control system, the potential for instability must be considered. There are many definitions of stability, but for feedback processes and control systems, the following definitions may be used: a system is stable if, following a disturbance, it settles to a steady, non cyclic state. A system may also be defined as being stable even if oscillatory, provided the oscillations are less than a prescribed magnitude. Instability then, is a continuous departure from a final steady state value, or it may be a greater than prescribed oscillations about the final steady state value.

It is possible for an unstable process to be stabilized by the addition of a control system. For example, it would be possible to design an automatic control rod system which uses the in-core neutron monitors to stabilize a nuclear reactor which exhibited spatial xenon instability. In general, however, it is preferable that a process with inherent feedback be designed to be stable by itself before it is combined with other processes and control systems.

For the Oyster Creek reactor, four types of stability have been considered.

a. Process control system stability

b. Power-flow (nuclear-hydraulic) stability

c. Interchannel hydraulic stability

d. Xenon spatial stability

4.3.2.7.1 Process Control System Stability

Process control system stability is concerned with control system dynamics and basic process dynamics respectively. The dynamics of the control systems and basic process determine the dynamics of the reactor system. The most meaningful way to check the system dynamic response and thus system stability is to perturb the system and measure the response of the important variables such as steam flow, pressure, and neutron flux.

To this end, a startup test program was established and criteria formulated on the minimum acceptable response. To determine that the final design was acceptable prior to testing, the plant analytical model was used. Using final design parameters the model was perturbed in the same manner as the plant and the response of the important variables (steam flow, pressure, and neutron flux) showed conformance to the stability criteria.

The plant model considers the entire reactor system, neutronics, heat transfer, hydraulics, and the basic processes, as well as associated control systems such as the flow controller, pressure regulator, and feedwater controller. Although the control systems may be stable when analyzed individually, final control system settings must be made in conjunction with the operating reactor so that the entire system is stable. The model solves the dynamic equations which represent the reactor system in the time domain. The various variables are represented as a function of time. The model is constantly being refined as new experimental or reactor operating data are obtained to improve the capability and accuracy of the model.

The time response of each of the important variables of the reactor system (neutron flux, pressure, and steam flow) to small step, or near step disturbances can be underdamped but must show a decay ratio of less than 1-to-4. Each of the following disturbances are imposed, one at a time.

a. A pressure set point change of at least ±5 psi.

b. A control rod position change equivalent to a local power change of at least five percent of the magnitude of power at the time of the disturbance.

c. A recirculation flow change equivalent to a power change of at least five percent of the magnitude of flow at the time of the disturbance.

d. A reactor water level set point change of at least 6 inches.

The above perturbations were imposed on the detailed analytical model of the entire reactor system and were imposed on the plant during the startup test period. A decay ratio of 1-to-4 is equivalent to an open loop gain and phase margin of 6 db and 30°. A decay ratio of 1-to-4 is an accepted standard (Reference 7) in the control industry for many process systems for good dynamic performance.

The responses of steam flow, pressure, and neutron flux to the various perturbations in all cases exhibited a decay ratio less than 1-to-4, thus good dynamic response is indicated.

4.3.2.7.2 Power Flow and Inter-Channel Hydraulic Stability

Power-Flow and Inter-Channel hydraulic stability are addressed in Reference 3. Inter-Channel stability is evaluated as a fundamental aspect of fuel design. The fuel designs in operation at Oyster Creek meet all requirements associated with Inter-Channel hydraulic stability.

Under certain conditions, BWRs may be susceptible to coupled neutron/thermal-hydraulic instabilities. These instabilities are characterized by periodic power and flow oscillations and are the result of density waves (i.e., regions of highly voided coolant periodically sweeping through the core). If the flow and power oscillations become large enough, and the density wave contains a sufficiently high void fraction, then the fuel cladding integrity safety limit could be challenged. Analytical studies have demonstrated that, for some plants, existing reactor protection systems may not assure automatic protection for these events. Reference 3 identifies several stability long-term solutions (options), approved by the NRC, that provide protection against violation of the fuel cladding integrity safety limit for anticipated oscillations.

Oyster Creek has implemented stability long-term solution Option II (Reference 21). The Option II solution provides fuel cladding integrity safety limit protection by automatically detecting and suppressing reactor instability. This is accomplished through a modification to the neutron monitoring system. Specifically, the original analog average power range monitor (APRM) trip bias units have been replaced with digital flow control trip reference (FCTR) cards (see Section 7.5). The Option II solution implements modified flow biased APRM Scram and Rod Block setpoints, as well as an operational exclusion region, in the low flow region (below approximately 45% core flow) of the Power Operation Curve (see Section 4.4.3.1).

4.3.2.7.3 Xenon Spatial Stability

The above discussions were concerned with process control system, power-flow, and inter-channel hydraulic stability. Attention is also given to xenon induced disturbances and especially the effects of these disturbances on the flux distribution.

For the Oyster Creek core size, lattice design, and power density it has been found that for a power coefficient which is more negative than -0.01, (Delta k/k)/(Delta P/P), the xenon oscillations are well damped. The power coefficient is the damping mechanism for xenon stability considerations.

The power coefficients for the plant at the beginning and end of life are approximately as follows:

1. Beginning of Life -0.06 Delta k/k/Delta P/P

2. End of Life -0.04 Delta k/k/Delta P/P

Notice that even at the end of life the power coefficient is more negative than that -0.01 (Delta k/k)/(Delta P/P), necessary for strong damping of xenon oscillations.

The model which is used to predict stability against xenon induced power oscillations is based on the Randall and St. John model expansion method. (Reference 8). This basic model of Randall and St. John was extended to include the effects of temperature and space dependence of xenon oscillations. The effects of temperature and space are given in reports by Chernick, Lellouche, and Wallman. (References 9 and 10)

The important assumptions in the development of this model are:

1. One group diffusion theory is adequate for analyzing xenon oscillations.

Since xenon oscillations occur primarily in large thermal reactors, the assumption of monoenergetic diffusion theory can probably be judged reasonably good.

2. Delayed neutrons are neglected.

   Since the period of the xenon oscillations is of the order of hours, the effect of neglecting delayed neutrons is small.

3. Linearized first order perturbation theory is adequate.

   Any use of linearized equations coupled with first order perturbations implies small perturbations for consistency and determines the response of the system to small changes. Near the oscillation threshold the assumption of small perturbations is valid to determine the response of the reactor; however, neither the assumed linearity of the determining equations nor the "smallness" of the perturbation can be guaranteed.

4. Mode coupling is negligible.

   Mode coupling arises from the expansion of the perturbation amplitudes in orthonormal eigenfunctions of the unperturbed boundary value problem and the multiplication by the orthogonal eigenfunctions followed by integration over the volume. The cross mode terms are small because of the orthogonality of the g_v's. They are zero if the flux is flat; furthermore, Canosa, Reference 11 has shown that their neglect has little effect in the flux ranges of interest for BWR design.

5. Neutron absorption in iodine is neglected.

   The assumption that the absorptions in iodine are negligible closely corresponds to the physical situation; however, it affects the normalization differential equations.

   The estimated accuracy in determining threshold fluxes, oscillation periods, and damping ratios is of the order of 10 percent.

The important parameters used to predict stability against xenon-induced power oscillations are core size (length/diameter ratio), flux or power level power coefficient, percent of power flattening, and migration area or void transportation in axial direction. The relative stability of any given reactor with respect to power distributional effects, normally called "harmonics," depends on the length to diameter ratio of the core. Figure 4.3-14 shows the relative stability of the radial, azimuthal and axial power distributions as a function of L/D ratio. From the figure it is seen that axial and azimuthal oscillations have a much higher probability of occurrence than radial oscillations for the Oyster Creek reactor. This plot only gives information regarding the relative stability of the various power distributions for a given core.

Figure 4.3-15 shows the effect of the flux or power level on the stability of the axial power distribution. This data which was adopted from Lellouche (Reference 10) includes the effect of void transport in the axial direction.

From this figure it is seen that the Oyster Creek reactor is well damped, and that there is a factor of 7 between the threshold power coefficient at which xenon oscillations would be induced and the calculated power coefficient at the end of the power cycle.

Flux flattening tends to reduce the stability of the various geometrical modes. This is due to the fact that the buckling differences and consequently the criticality differences between the harmonics are reduced as the flux shape is flattened. This flux flattening may cause the radial power distribution to be subject to azimuthal oscillations (a rotating edge to edge tilt) which are less well damped than the axial oscillations. Figure 4.3-16 which is based on work by Randall and St. John (Reference 8) shows the critical power coefficients, where the onset of neutral oscillation occurs, as a function of flux flattening. From this plot it is seen that the reactor is well damped even when the flat zone is 100 percent of the radius.

Operating experience with the Dresden I and KRB reactor indicates observable but highly damped xenon response to both power level and power distribution changes, indicating that the results given above are reliable.

4.3.2.8 Vessel Irradiation

Any material that is exposed to a neutron flux suffers some property change; for example, irradiation of steels increases the ductile to brittle transition temperature. Data is available relating fluence to property changes, thus it is important to know how much irradiation the reactor vessel has received so that the vessel is pressurized at a safe limit above the transition temperature.

Three iron and copper wire neutron flux dosimeters in a capsule were installed near the wall of the Oyster Creek reactor vessel at construction and were removed at the Poison Curtain Removal Outage in the Fall of 1971. An irradiation activation technique was used to determine the activity of the iron and copper isotopes of interest. A computer code (ANISN) was used to obtain the neutron flux spectrum at the vessel wall. The activation cross section for the iron dosimeters was determined using the calculated neutron spectrum and cross section data from ASTM E-263. Since the cross-section for copper in a neutron flux spectrum is not accurately known, the results for the copper dosimeter wires are not included in this discussion.

Each of the three iron wires (commercially pure iron thermocouple wire, 24-gauge) and three copper wires (99.999-percent pure) were formed into loops with the ends twisted together and enclosed in a cylindrical capsule containing helium atmosphere. The capsule was inserted in a special slot in the material surveillance specimen holder at 30 plant azimuth. The monitor holder was mounted on the vessel wall, 106.5 inches from the core center, at the midplane of the core. The flux dosimeter capsule was situated adjacent to the vessel inner wall and was assumed to have received the same neutron flux as the vessel wall. This arrangement is described in Reference 12.

The irradiation of the dosimeters ended on September 18, 1971, at an integrated power output of 908,036.165 megawatt days thermal. Battelle-Columbus Laboratories of Columbus, Ohio, was chosen to determine the fast neutron fluence experienced by the dosimeters. The reactions of interest in the flux dosimeters are $Fe^{54}(n,p)\ Mn^{54}$ for the iron wires, and $Cu^{63}(n,d)\ Co^{60}$ for the copper wires. Radioactivation techniques following ASTM methods were used in evaluating the activity of the manganese. To obtain the neutron spectrum at the reactor vessel wall, a computer code, ASISN, was employed with the results given in Table 4.3-4. The activation cross section for iron is known for all neutron energies and is presented in Reference

13. Therefore, the iron activation cross-section at the vessel wall was calculated to be 0.200 barns.

As a result of the above, the fast neutron exposures at the reactor vessel wall for the three iron dosimeters were determined to be 1.175 x 10^{71} nvt, 1.15 x 10^{17} nvt and 1.20 x 10^{17} nvt.

Using the average value of 1.175 x 10^{17} nvt for the fast neutron fluence and a total power output of 908,036.165 megawatt days thermal (MWDt) at the end of Fuel Cycle 1A, the fast neutron fluence received by the reactor vessel wall for each megawatt day of power was calculated to be 1.294 x 10^{11} nvt.

However, the 0.200 barn cross section for the FE^{54} (n,p) Mn^{54} has an inherent uncertainty of approximately 25 percent. Applying this uncertainty in the conservative direction (i.e., a cross-section of 0.150 barns) the fast neutron fluence for each MWDt was adjusted to 1.725 x 10^{11} nvt.

Based upon the more conservative 0.150 barn FE^{54} cross-section, the predicted end of life fast neutron fluence for the reactor vessel was calculated to be 3.32 x 10^{18} nvt. This figure is approximately one third the design value of 1 x 10^{19} nvt and, therefore, presents no safety problems.

In order to verify Battelle's counting techniques, portions of the same flux wires were sent to the Vallecitos Nuclear Center Laboratory of the General Electric Company. The GE results were essentially identical to those of Battelle.

The neutron fluence at the inner surface of the reactor vessel as of March 31, 1977 was estimated to have been 5.352 x 10^{17}. Based on 1930 MWt, 5.15 EFPY were accumulated as of that date.

In Reference 22, as part of the transition to the BWR Vessel and Internals Project Integrated Surveillance Program, Oyster Creek committed to perform a reactor vessel fluence evaluation using a method in accordance with Regulatory Guide (RG) 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” dated March 1, 2001. All future evaluations of reactor vessel fluence will be performed using a method in accordance with RG 1.190.

4.3.3 Analytical Methods

A detailed description of the analytical methods used in the nuclear design analysis is provided in Reference 3.

4.3.4 Changes

The evaluation of the nuclear design is continued in the periodic reload submittals for each core cycle.

4.3.5 References

(1) Nuclear Characteristics of Large Advanced BWR's, R.L. Crowther and D.L. Fischer, TID-7672, ANS Topical Meeting, September 26-27, 1963, San Francisco, California.

(2) Oyster Creek 1975 Reload for Cycle 5 Licensing Data Submittal, XN-74-52, Revision 3, January 22, 1975.

(3) General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A, as revised at the time of the specific reload cycle.

(4) BWR/4 and BWR/5 Fuel Design, NEDE-20944-P (Proprietary) and NEDO-20944, October 1976.

(5) Generation of Void and Doppler Reactivity Feedback for Application to BWR Design, R.C. Stirn, General Electric Co., NEDO-20964-A, December 1, 1986.

(6) Generation of Void and Doppler Reactivity Feedback for Application to BWR Design (Amendment No. 1), NEDO-20964-1, July 1977.

(7) Process Instruments and Controls Handbook, Considine, McGraw Hill, 1957.

(8) "Xenon Spatial Oscillations," D. Randall and D.S. St. John, Nucleonics, March 1958.

(9) "The Effect of Temperature on Xenon Stability," J. Chernick, G. Lellouche, and W. Wollman, NSE: 10, 120-131 (1961).

(10) "Space Dependent Xenon Oscillations," G. Lellouche,S. Gerald, NSE: 12, 482-489 (1962).

(11) "Xenon Induced Oscillations," J. Canosa and H. Brooks, NSE: 26, 237-253 (1966).

(12) "Mechanical Property Surveillance of General Electric Boiling Water Reactor Vessels," NEDO 10115, J.P. Higgins and F.A. Brandt, July 1969.

(13) "Standard Method for Measuring Fast Neutron Flux by Radioactivation of Iron," ASTM Designation E263-70, 1971 Annual Book of ASTM Standards, pp.781-786.

(14) Deleted.

(15) Deleted.

(16) Deleted.

(17) Deleted.

(18) Deleted.

(19) General Electric Hitachi BWR Control Rod Lifetime, NEDE-30931-P.

(20) General Electric Fuel Bundle Designs, NEDE-31152-P, as revised at the time of the specific reload cycle.

(21) Application of Stability Long-Term Solution Option II to Oyster Creek, NEDC-33065P, Rev. 0, April 2002.

(22) Letter from M.P. Gallagher (AmerGen Energy Company, LLC) to U.S. NRC, “Additional Information Supporting the Request for License Amendment Regarding Reactor Vessel Specimen Removal Schedule,” dated September 10, 2003.

(23) General Electric Hitachi Licensing Topical Report, Marathon-5S Control Rod Assembly, NEDE-33284P, Revision 1, November 2007.

TABLE 4.3-1
(Sheet 1 of 3)

HISTORICAL FUEL TYPES UTILIZED DURING CYCLES 1 THROUGH 19

Fuel Cycle	Fuel Type GE I (7x7)	GE II (7x7)	ENC III (7x7)	ENC III-E (7x7)	ENC III-F (7x7)	ENC V (8x8)	ENC V-B (8x8)
Cycle 1A 5/3/69 - 9/18/71	560	--	--	--	--	--	--
Refueling Change	(-24)	(+24)					
Cycle 1B 11/11/71 - 5/1/72	536	24	--	--	--	--	--
Refueling Change	(-136)	(+132)	(+4)				
Cycle 2 6/20/72 - 4/13/73	400	156	4	--	--	--	--
Refueling Change	(-148)			(+148)			
Cycle 3 6/4/73 - 4/13/74	252	156	4	148	--	--	--
Refueling Change	(-68)	(-4)			(+72)		
Cycle 4 7/1/74 - 3/29/75	184	152	4	148	72	--	--
Refueling Change	(-112)				(+36)		
Cycle 5 5/25/75 - 12/27/75	72	152	4	148	108	4	72
Refueling Change	(-56)						(+56)
Cycle 6 3/10/76 - 4/23/77	16	152	4	148	108	4	128
Refueling Change	(-16)	(-108)	(-4)				(+128)
Cycle 7 8/1/77 - 9/16/78	--	44	--	148	108	4	256
Refueling Change		(-44)		(-84)	(-40)		(+168)
Cycle 8 12/5/78 - 1/5/80	--	--	--	64	68	4	424
Refueling Change				(-64)	(-44)		(-152) (+160)

TABLE 4.3-1
(Sheet 2 of 3)

HISTORICAL FUEL TYPES UTILIZED DURING CYCLES 1 THROUGH 19

Fuel Type

Fuel Cycle	ENC III-F (7x7)	ENC V (8x8)	ENC V-B (8x8)	GE P8DRB239 (8x8)	GE P8DR B265 H(8x8)	GE P8DRB299ZA (8x8)	GE P8DR299 Z (8x8)
Cycle 9 7/15/80 - 2/12/83	24	4	532	--	--	--	--
Refueling Change	(-24)	(-4)	(-144)	(+112)	(+60)		
Cycle 10	--	--	388	112	60	--	--
Refueling Change			(-188)	--	(+4)	(+48)	(+136)

Fuel Cycle	ENC V-B (8x8)	GE P8DRB239 (8x8)	GE P8DRB265H (8x8)	GE P8DRB299ZA (8x8)	GE P8DR299Z (8x8)	GE P8DRB321 (EB) (8x8)	GE P8DQB338-12Gz (8x8)	GE P8DQB338-11GZ (8x8)
Cycle 11	200	112	64	48	136	--	--	--
Refueling Change	(-171)	(-1)	--	--	(+20)	(+152)	--	--
Cycle 12	29	111	64	48	156	152	--	--
Refueling Change	(17)	(-87)	(-36)	--	(-4)	(+68)	(+60)	(+16)
2								
Cycle 13	12	24	28	48	152	220	60	16
Refueling Change	(-12)	(-24)	(-28)	(-48)	(-56)	--	(+132)	(+36)
Cycle 14	--	--	--	--	96	220	192	52
Refueling Change					(-96)	(076)	(+4)	(+8)

Fuel Cycle	GE P8DRB321(EB) (8x8)	GE P8DQB338-12GZ (8x8)	GE P8DQB338-11GZ (8x8)	GE P8DWB338-11GZ (8x8)	GE P8DWB348-12GZ (8x8)
Cycle 15	144	196	60	48	112
Refueling Change	(-144)	(-36)	(-8)	(+40)	(+148)
Cycle 16	--	160	52	88	260
Refueling Change		(-140)	(-44)	(+40)	(+144)

TABLE 4.3-1
(Sheet 3 of 3)

HISTORICAL FUEL TYPES UTILIZED DURING CYCLES 1 THROUGH 19

Fuel Cycle	GE P8DQB338-12GZ (8x8)	GE P8DQB338-11GZ (8x8)	GE P8DWB338-11GZ (8x8)	GE P8DWB348-12GZ (8x8)
Cycle 17	20	8	128	404
Refueling Change	(-20)	(-8)	(-48)	(-108)
			(+48)	(+136)
Cycle 18	--	--	128	432

Fuel Cycle	GE P8DWB338-11GZ (8x8)	GE P8DWB348-12GZ (8x8)	GE11 P9HUB369-12GZ (9x9)	GE11 P9HUB274-13GZ (9x9)
Cycle 18	128	432		
Refueling Change	(–40)	(–156) (+6)	(+144)	(+46)
Cycle 19	88	282	144	46

TABLE 4.3-2
(Sheet 1 of 1)

REPRESENTATIVE REACTIVITY COEFFICIENTS AND NUCLEAR PARAMETERS

Coefficient		Reload Cycle 18
Void Coefficient at Core Average Voids, (Delta k/k)/(percent void)		
	BOC	-1.15×10^{-3}
	EOC	-1.26×10^{-3}
Fuel Temperature (Doppler) Coefficient (Delta k/k)/F		
	BOC	-1.84×10^{-5}
	EOC	-2.25×10^{-5}
Parameter		
Delayed Neutron Fraction		
	BOC	0.00633
	EOC	0.00540
*, Neutron Lifetime, Microseconds		
	BOC	35.4
	EOC	38.1

TABLE 4.3-3
(Sheet 1 of 1)

CALCULATED CORE EFFECTIVE MULTIPLICATION FACTOR, CONTROL SYSTEM WORTH AND SHUTDOWN MARGIN

	Cycle 18
$k_{critical}$ (with uncertainty)	0.9910
k_{eff} (cold-68 F) Beginning of Cycle	
Uncontrolled*	1.1071
Fully Controlled	0.9578
Strongest Control Rod Withdrawn	0.9792
R, Maximum Increase in Cold Core Reactivity With Exposure Into Cycle, Delta k	0.0000
Minimum Shutdown Margin, Delta k	0.0118

* All controls rods out of the core.

TABLE 4.3-4
(Sheet 1 of 1)

FLUX SPECTRUM AT OYSTER CREEK PRESSURE VESSEL WALL

Group	Energy Interval (Mev)	Flux (m/cm^2-sec)
1	12.2 - 14.9	0.0368
2	10.0 - 2.2	0.194
3	8.19 - 10.0	0.609
4	6.70 - 8.19	1.54
5	5.49 - 6.70	2.25
6	4.49 - 5.49	2.47
7	3.68 - 4.49	1.90
8	3.01 - 3.68	1.94
9	2.47 - 3.01	2.52
10	2.02 - 2.47	2.64
11	1.65 - 2.02	2.22
12	1.35 - 1.65	2.19
13	1.11 - 1.35	2.02
14	9.06 x 10-1 – 1.11	1.64
15	6.08 x 10-1 - 9.07 x 10-1	3.42
16	4.08 x 10-1 - 6.08 x 10-1	2.58
17	1.11 x 10-1 x 4.08 x 10-1	5.45
18	1.50 x 10-2 x 1.11 x 10-1	4.57
19	3.35 x 10-3 x 1.50 x 10-2	2.84
20	5.83 x 10-4 - 3.35 x 10-3	3.02
21	1.01 x 10-4 x 5.83 x 10-4	3.00
22	2.90 x 10-5 - 1.01 x 10-4	2.13
23	1.07 x 10-5 - 2.90 x 10-5	1.68
24	3.06 x 10-6 - 1.07 x 10-5	2.04
25	1.13 x 10-6 - 3.06 x 10-6	1.57
26	4.14 x 10-7 - 1.13 x 10-6	1.48
27	0 - 4.14 x 10-7	40.8

TABLE 4.3-5
(Sheet 1 of 1)

Typical Parameters for GE Control Rod Assemblies. "D" Lattice.

	Current All-B_4C	Duralife-230	Marathon
Control Rod Weight, lb.	218	215	176
Absorber Rod - B_4C			
Number Per Control Rod	84	60	56
Length, in.	143	137	143.7
Density, grams/cm^3	1.76 (Nominal 70% Theoretical)		
Absorber Tube (B_4C)			Rad Resist
Cladding Material	Commercial 304-SS	High Purity	304-SS
Outside Diameter, in.	0.188	0.220	.250(Inside)
Wall Thickness, in.	0.025	0.020	0.024
Absorber Strip - Hafnium			
Number Per Control Rod	0	4	44 / 12
Length, in.	--	143	6" / 108.6"
Thickness, in.	--	0.220	0.220
Density, Grams/cm^3	--	13.3	13.0
Absorber Plate -- Hafnium			
Number Per Control Rod	--	4	--
Length, in.	--	6	--
Width, in.	--	3.32	--
Thickness, in.	--	0.220	--
Density, Grams/cm^3		-- 13.3	--
Absorber Capsule			
Length, in.	--	--	11.4
Inside Diameter, in.	--	--	0.235
Wall thickness, in.	--	--	0.003
Material	--	--	304-SS
Sheath Thickness, in.	0.056	0.045	N/A
Pin Material	Haynes Alloy 25	PH13 - 8 MO	PH 13-8 MO
Roller Material	Stellite	Inconel x 750	Inconel x 750

4.4 THERMAL AND HYDRAULIC DESIGN

4.4.1 Design Basis

The design basis for the thermal and hydraulic characteristics incorporated in the core design, in conjunction with the plant equipment characteristics, nuclear instrumentation, and the Reactor Protection System, is to ensure that no fuel damage will occur in normal operation or operational transients caused by reasonable expected single operator error or equipment malfunction. Limits on plant operation are established to assure that the plant can be safely operated and not pose any undue risk to the health and safety of the public. The criteria used to establish these limits are not intended to predict failure of the fuel if they are exceeded, but are rather considered prudent design practice.

4.4.2 Description of Thermal and Hydraulic Design of the Reactor Core

Fuel rod damage is defined as a perforation of the cladding which would permit the release of fission products to the reactor coolant. The mechanisms which could cause fuel rod damage are: (1) rupture of the fuel rod cladding due to strain caused by relative expansion of the $U0_2$ pellet, and (2) severe overheating of the fuel rod cladding caused by inadequate cooling.

4.4.2.1 Linear Heat Generation Rate

The peak linear heat generation rate (LHGR) calculated for normal and abnormal operating transients must be less than or equal to the LHGR at which 1 percent strain is calculated to occur.

A value of 1 percent strain of the Zircaloy cladding has been established as the safety limit below which fuel damage due to overstraining of the fuel cladding is not expected to occur. The model used in the evaluation of the 1 percent strain limit is described in detail in References 4 and 25. Dimensions used in conjunction with the model for this evaluation are the most limiting combination of tolerances.

The 1 percent strain safety limit was established based on General Electric data on the strain capability of irradiated Zircaloy cladding segments from fuel rods operated in several BWRs (Reference 2). None of the data obtained fall below the 1 percent strain value; however, a statistical distribution fit to the available data indicates the 1 percent strain value to be approximately the 95 percent point in the total population. This distribution implies, therefore, a small (less than 5 percent) probability that some cladding segments may have plastic elongation less than 1 percent at failure. Based on this data, and the NRC regulatory guidance provided in NUREG-0800, the 1 percent strain safety limit was established.

The metallurgical state of the Zircaloy-2 cladding for fuel rods, tie rods and water rods is given in Reference 3. Values of thermal conductivity, coefficient of thermal expansion, and plastic strain capability for Zircaloy-2 cladding are given in Table 4.4-1.

The effect of temperature on the ultimate and yield strengths of unirradiated and irradiated Zircaloy-2 tubing is given in Reference 4.

The liner heat generation rate (LHGR) corresponding to 1 percent strain has been calculated and results are presented in Table 4.4-2 for the P8x8R fuel design. Because gadolinia urania fuel has a lower thermal conductivity and melting temperature than UO_2 fuel, there is a reduction

in the LHGR calculated to cause 1 percent diametral strain for gadolinia-urania fuel rods. However, to compensate for this, the gadolinia-urania fuel rods are designed to provide margins similar to standard UO_2 rods.

The values calculated as resulting in 1 percent strain in the cladding are used during specific plant evaluations of transients due to single operator error or equipment malfunctions to ensure that the safety limit is not exceeded.

Based on these evaluations it has been determined that for all P8x8R fuel, the power required to produce 1 percent plastic strain in the cladding is greater than 160 percent of the design maximum steady state power, throughout life, for all rod types in the assembly. This ratio considers the presence of a calculated power spiking penalty being added to the maximum linear heat generation rate. (Subsection 4.4.2.1.1).

The linear heat generation rate and associated evaluations for the GE11 and GNF2 fuel designs are contained in References 3 and 25, respectively.

4.4.2.1.1 Maximum Linear Heat Generation Rate (MLHGR)

The MLHGR is the maximum linear heat generation rate (expressed in kW/ft) in any fuel rod allowed by the Technical Specifications for a given fuel type. The MLHGR is attained when the product of the local, radial and axial peaking factors in an axial segment of a fuel bundle equals the total peaking factor for that fuel type.

The Technical Specifications MLHGR limit for all P8x8R fuel assemblies is 13.4 kW/ft. A power spike allowance due to fuel densification is added to the 13.4 kW/ft LHGR for analyses sensitive to localized power increases (i.e., cladding temperatures and stress and strain evaluations). This assures with 95 percent confidence that less than 1 fuel rod in the core will exceed the MLHGR for which the fuel has been designed and precludes Technical Specification requirements to reduce the 13.4 kW/ft LHGR by a power spike peaking penalty as a function of axial core location. The limiting thermal-mechanical power-exposure envelope for the fuel designs applicable to BWR/2 type plants, including the power spike allowance for fuel densification, is provided in References 25 and 26. For the Oyster Creek specific application, more limiting LHGR limits have been conservatively applied as compared to these BWR/2 LHGR limits. A discussion of the effects of fuel densification is given below.

Evaluations to determine the effects of fuel densification are made using the models described in References 21 and 22. Possible effects due to fuel densification are: (1) power spikes due to axial gap formation, (2) increase in linear heat generation rate (LHGR) because of pellet length shortening, (3) creep collapse of the cladding due to axial gap formation, and (4) changes in stored energy due to decreased pellet cladding thermal conductance resulting from increased radial gap size.

The power spiking allowance, calculated using the referenced methods, resulted in a power spiking penalty at the top of the core which is less than or equal to 2.2 percent for the P8x8R design. The power spiking penalty as a function of elevation from the bottom of the core is conservatively expressed by:

$$\frac{\Delta P}{P} x = \frac{\Delta P}{PL} \frac{x}{L}$$

where:

$\frac{\Delta P}{P}$ x = power spiking penalty at elevation x from bottom of core;

$\frac{\Delta P}{PL}$ power spiking penalty at top of core;

x = elevation from bottom of core; and

L = fuel column length.

No power increase is calculated due to densification. A fuel pellet expands 1.2 percent in going from the cold to hot condition at 13.4 kW/ft. While this increase in length from the cold to hot condition is not taken credit for either in design calculations or in the process of core performance analysis during reactor operation, the expansion more than offsets the decrease in pellet length due to densification.

The calculated pellet decrease in length due to densification is less than the increase in length due to thermal expansion of the pellet in going from cold to hot condition. Therefore, no power increase is calculated due to densification.

Cladding creep collapse is not predicted to occur in the P8x8R fuel design. Details of this analysis are given in Reference 4.

The effects of densification on stored energy are considered in the Loss-of-Coolant Accident evaluation. Stored energy in the fuel pellet at the initiation of the LOCA is calculated using the model and assumptions described in Reference 5. Analysis of the LOCA is presented in References 23, 24, and 26.

4.4.2.1.2 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

The MAPLHGR is the maximum average linear heat generation rate (expressed in kW/ft) in any plane of a fuel bundle allowed by the Technical Specifications for that fuel type. This parameter is obtained by averaging the LHGR over each fuel rod in the plane and its limiting value is selected such that:

1. The peak clad temperature during the design basis Loss-of-Coolant Accident will not exceed 2200°F, and

2. The peak oxidation fraction during the design basis Loss-of-Coolant Accident will not exceed 17%.

The MAPLHGR is attained when the product of the gross peaking factor (radial peaking factor times axial peaking factor) and the average rod peaking factor (1.0) equals the Technical Specification value.

The Technical Specification limits for average planar LHGR for the various fuel types as a function of exposure assure that the peak cladding temperature following the postulated design

basis Loss-of-Coolant Accident will not exceed the 2200°F limit specified in 10CFR50.46 (January 4, 1974) considering the postulated effects of fuel pellet densification.

The peak cladding temperature following a postulated Loss-of-Coolant Accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ±20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are below the limits specified in 10CFR50.46.

The cladding surface temperature is calculated using the cladding surface heat flux at a given axial position on a fuel element, in conjunction with the overall cladding-to-coolant film coefficient representing the combined effects of crud and oxide resistances and the liquid film resistance based on the Jens-Lottes (Reference 6) wall superheat equation. The impact of high cladding temperature, such as decreased yield strength and reduced cladding thickness due to oxidation, is considered in the design evaluation.

Details of considerations when determining cladding surface temperature are given in Reference 4. The model used to calculate the fuel cladding temperature is documented in Reference 1. Fuel cladding temperature as a function of heat flux for the P8x8R design is shown in Figure 4.4-1 for the beginning of life conditions and in Figure 4.4-2 for late in life conditions.

4.4.2.2 Critical Power Ratio

There are three different types of boiling heat transfer for water in a forced convection system: nucleate boiling, transition boiling, and film boiling. Nucleate boiling, at lower heat fluxes, is an extremely efficient mode of heat transfer, allowing large quantities of heat to be transferred with a very small temperature rise at the heated wall. As power is increased, the boiling heat transfer surface alternates between film and nucleate boiling, leading to fluctuations in heated wall temperatures. The bundle power at the point of departure from the nucleate boiling region into the transition boiling region anywhere in the bundle is called the critical power. Transition boiling begins at the critical power, and is characterized by fluctuations in cladding surface temperature. Film boiling occurs at the highest heat fluxes; it begins as transition boiling comes to an end. Film boiling heat transfer is characterized by stable wall temperatures which are higher than those experienced during nucleate boiling.

The objective for normal operation and transient events is to maintain nucleate boiling and thus avoid a transition to film boiling. Operating limits are specified to maintain adequate margin to the onset of the boiling transition. The parameter used for core design and monitoring is the critical power ratio. This is defined as the ratio of the critical power (bundle power at which some point within the assembly experiences onset of boiling transition) to the operating bundle power. The critical power is determined at the same mass flux, inlet temperature and pressure which exists at the specified reactor condition. Thermal margin is stated in terms of the minimum value of the critical power ratio (MCPR), which corresponds to the most limiting fuel assembly in the core. To ensure that adequate margin is maintained, a design requirement based on a statistical analysis was selected. This requirement states that moderate frequency transients caused by a single operator error or equipment malfunction shall be limited such that, considering uncertainties in manufacturing and monitoring the core operating state, more than 99.9 percent of the fuel rods would be expected to avoid boiling transition (Reference 12). Both

the transient (safety) thermal limit and normal operating thermal limit for the fuel in terms of MCPR are derived from this basis.

The MCPR operating limit is the minimum CPR allowed by the Technical Specifications for a given bundle type and the limit is a function of power and flow. The CPR is a function of several parameters, the most important of which are bundle power, bundle flow and local power distribution. The operating limit of CPR is selected for each bundle type such that during the most limiting event of moderate frequency, the CPR in that bundle will not be less than the safety limit CPR. The MCPR is attained when the bundle power, bundle flow and other relevant parameters combine to yield the Technical Specifications value.

For historical perspective, it is to be noted that the design criteria for evaluating the thermal design margin was revised from one using critical heat flux (CHF) to one using critical power (CP) at the time of the OCNGS cycle 5 reload. This revision in criteria was established in Supplement 4 of Reference 7 which provides an introduction to the evaluation of critical power and a critical power criteria to determine the operating margin to boiling transition.

A fuel cladding integrity safety limit of 1.07 was established for General Electric (GE) P8x8R fuel for the Oyster Creek Cycle 10 reload. The safety limit has been conservatively applied to the GE 8x8EB and GE 8x8NB fuel designs based on an NRC approved generic safety limit of 1.06 for these fuel designs (Reference 3).

During Oyster Creek's Cycle 15 operation, GE issued a 10 CFR Part 21 notification stating that the generic safety limit may be non-conservative for GE fuel designs. GE performed a cycle specific safety limit evaluation and determined that a 1.07 safety limit is acceptable for the GE fuel designs in the Cycle 15 core loading. Since Oyster Creek was using a 1.07 safety limit, no action was necessary. However, a cycle specific safety limit must be determined for each reload.

The safety limit MCPR is now calculated on a cycle–by–cycle basis and is reported in the technical specifications**.** Separate safety limit MCPR values are calculated based on the number of operating recirculation loops. One value is calculated for operation with four or five recirculation loops in service and another value is calculated for operation with only three recirculation loops in service. Each new safety limit is calculated using NRC approved methods (Reference 3) and includes procedures for application to plant and cycle specific evaluations that were reviewed by the NRC. These procedures have been revised to incorporate cycle specific parameters that include: 1) actual core loading; 2) conservative variations of projected control blade patterns; 3) the actual bundle parameters (e.g., local peaking); and 4) the fuel cycle exposure range.

The GEXL plus (Reference 3) critical power correlation is used to calculate transient delta CPR values for any design basis transient or accident where CPR is the limiting parameter. The minimum CPR operating limit ensures that the fuel cladding integrity safety limit is not violated for these transients or accidents. Representative delta CPR values for different transients are given in Table 4.4-3.

The resulting operating MCPR limits combined with the transient analysis results provide assurance that the fuel cladding integrity safety limit will not be violated during anticipated operating transients.

4.4.2.3 Hydraulic Model

Core steady state thermal hydraulic analyses are performed using a model of the reactor core. This model includes hydraulic descriptions orifices, lower tieplates and fuel rods, fuel rod spacers, upper tieplates, the fuel channel and core bypass flow paths. The orifice, lower tieplate fuel rod spacers upper tieplate and, where applicable, holes in the lower tieplate are hydraulically represented as being separate, distinct local losses of zero thickness. The fuel channel cross section is represented by a square section with enclosed area equal to the unrodded cross sectional area of the actual fuel channel.

The flow distribution to the fuel assemblies and bypass flow paths is calculated on the assumption that the pressure drop across all fuel assemblies and bypass flow paths is the same. This assumption has been confirmed by measuring the flow distribution in boiling water reactors (References 13, 14 and 15). The components of bundle pressure drop considered are friction, local, elevation and acceleration, and are calculated as described in detail in Reference 3. Pressure drop measurements made in operating reactors confirm that the total measured core pressure drop and calculated core pressure drop are in good agreement. There is reasonable assurance, therefore, that the calculated flow distribution throughout the core is in close agreement with the actual flow distribution of an operating reactor.

An iteration is performed on flow through each flow path (fuel assemblies and bypass paths), which equates the total differential pressure (plenum to plenum) across each path and matches the sum of the flows through each path to the total core flow. The total core flow less the control rod cooling flow enters the lower plenum. A fraction of this passes through various bypass paths documented in Subsection 4.4.2.3.2. The remainder passes through the orifice in the fuel support (experiencing a pressure loss), where more flow is lost through the fit-up between the fuel support and the lower tieplate and through the lower tieplate holes and into the bypass region. The majority of the flow continues through the lower tieplate (experiencing a pressure loss), where some flow is lost through the flow path defined by the fuel channel and lower tieplate into the bypass region. This bypass flow is restricted on those fuel assemblies with finger springs.

Within the fuel assembly, heat balances on the active coolant are performed nodally. Fluid properties are expressed as the bundle average at the particular node of interest and are based on 1967 International Standard Steam-Water Properties. In evaluating fluid properties, a constant pressure model is used.

The relative axial power and radial power distributions are used with the bundle flow to determine the axial coolant property distribution, which gives sufficient information to calculate the pressure drop components within each fuel assembly type. When the equal pressure drop criterion described above is satisfied, the flow distributions are established.

4.4.2.3.1 Pressure Drop Calculations

Friction pressure drop is calculated with a basic model incorporating a friction factor and two phase friction multiplier similar to that used throughout the nuclear power industry. The formulation for the two-phase friction multiplier used in the model is based on data that compares closely to that found in Reference 16. Significant amounts of friction pressure drop data in multirod geometries representative of modern BWR plant fuel bundles have been taken. Both the friction factor and two-phase multipliers were correlated on a best-fit basis. Checks are

made on a continuing basis to ensure the best models are used over the full range of interest to boiling water reactors.

The local pressure drop is defined as the irreversible pressure loss associated with an area change, such as the orifice, lower tieplate, and spacers of a fuel assembly. The addition of orificing to the fuel channel adds single phase pressure drop to the channels which has been proven experimentally to stabilize the channel hydrodynamically. The general local pressure drop model is similar to the friction pressure drop and is also similar to that used throughout the nuclear power industry. The formulation for the two phase local multiplier is similar to that reported in Reference 17. Empirical constants were added to fit the results to data taken for the specific designs of the boiling water reactor fuel assembly. These data were obtained from tests performed in single phase water flow to calibrate the orifice, the lower tieplate, and the holes in the lower tie plate, and in both single and two phase flow, to derive the best fit design values for spacer and upper tieplate pressure drop. The range of test variables was specified to include the range of interest for boiling water reactors. New test data are obtained whenever there is a significant design change to ensure the most applicable methods are used.

The elevation pressure drop is based on the basic relation involving void fraction and saturated water and vapor density. The void fraction model used is an extension of the Zuber-Findlay (Reference 18) model and uses an empirical fit constant to predict a large block of steam void fraction data. Checks against new data are made on a continuing basis to ensure the best models are used over the full range of interest to boiling water reactors.

A reversible pressure change occurs when an area change is encountered, and an irreversible loss occurs when the fluid is accelerated through the boiling process. The basis formulations for the acceleration pressure changes for two-phase flow are provided in Reference 4. The total acceleration pressure drop in boiling water reactors is on the order of a few percent of the total pressure drop.

4.4.2.3.2 Bypass Flow

The bypass flow paths considered in the analyses are described in Table 4.4-4 and are shown schematically in Figure 4.4-3. Typical values of the fraction of bypass flow through each flow path are given in Reference 19.

Due to the low flow velocity, the pressure drop in the bypass region above the core support plate is essentially all elevation head. Thus, the sum of the core support plate differential pressure and the bypass region elevation head is equal to the core differential pressure.

Full scale tests, documented in Reference 20, have been performed to establish the flow coefficients for the expression used to calculate bypass flow through the major flow paths. These tests simulate actual plant configurations, which have several parallel flow paths and, therefore, the flow coefficients for the individual paths could not be separated. However, analytical models of the individual flow paths were developed as an independent check of the tests. The models were derived for actual BWR design dimensions and considered the effects of dimensional variations. These models predicted the test results when the as-built dimensions were applied. When using these models for hydraulic design calculations, nominal drawing dimensions are used. This is done to yield the most accurate prediction of the expected bypass flow. With the large number of components in a typical BWR core, deviations from the nominal dimensions will tend to statistically cancel, resulting in a total bypass flow best represented by that calculated using nominal dimensions.

Increases in channel wall permanent deflection at the lower tieplate result in increased bypass flow through the channel to lower tieplate flow path. For plants without finger springs, this flow path contributes a significant portion (approximately 80 percent) of the total core bypass flow. Changes in the flow through this path affect the total core bypass flow, which in turn, affects the active coolant flow, void coefficient and operational transients. To provide control over this flow path, optional finger spring seals, which control the flow rate through the channel to lower tieplate flow path over a wide range of channel wall deflections, may be added to most reload fuel assemblies. Finger springs are located between the lower tieplate and the channel for the purpose of controlling the flow through that path. The finger springs provide control of the flow through this path by maintaining a nearly constant flow area as the channel wall deforms.

Full scale tests have been performed on finger spring seals to determine the flow characteristics through the lower tieplate and fuel channel flow path as a function of pressure drop and channel deformation. The pressure drop and channel deflections used in the tests covered the range expected during reactor service. The fuel bundle components used for the tests were standard lower tieplates, finger springs and a standard channel section. The flow characteristics as determined by these tests are used in design calculations (Reference 3).

In cases where finger spring seals are installed on some reload fuel bundles, the hydraulic analysis takes into account the differences between finger spring and nonfinger spring fuel.

Because finger spring seals reduce the bypass flow through the lower tieplate channel path, the active coolant flow in a bundle with finger spring seals will normally be larger than the flow in a bundle without the seals. This is true, assuming that both bundles are operating at the same power and have identical channel to lower tieplate deflections. To accommodate this variation, for otherwise identical fuel assemblies, the bypass flow characteristics are individually modeled in the analysis. The model yields lower tieplate and channel bypass flow as a function of pressure drop for defined channel deflection at the lower tieplate, and the existence (or absence) of finger spring seals.

Variations in bundle flow rates for flow assemblies with and without finger spring seals are handled by supplying separate hydraulic constants for the finger spring seal and nonfinger spring seal bundles. This ensures that the core flow distribution is properly calculated.

4.4.3 Description of Thermal and Hydraulic Design of the Reactor Coolant System

Coolant yet to flow through the reactor core flows from the discharge of the reactor recirculation pumps through the bottom plenum region of the reactor to the fuel assembly inlet orifices at the bottom of the core. Differential inlet orificing diverts flow to the inner, higher powered zone of the core. It also decreases the dependence of fuel assembly flow on assembly power level, and increases flow stability margins. Except for a small fraction which bypasses the fuel assemblies and cools the core components between the fuel channels, the recirculation flow travels vertically upward within the fuel assemblies to cool the fuel rods. Fission energy is transferred primarily as thermal energy from the fuel to the coolant. This produces a two phase steam-water mixture which increases in quality toward the top of the fuel assemblies. This mixture passes out of the fuel assemblies into a mixing plenum, thence upward through the steam separators.

Table 4.4-5 summarizes the thermal and hydraulic characteristics of the Reactor Coolant System. Refer to Chapter 5 for a detailed description of the system including figures illustrating its configuration and approximate dimensions.

4.4.3.1 Power-Flow Operation Map

The general control capability and characteristics of the reactor are depicted in Figure 4.4-4. Power can be adjusted with recirculation flow control along constant rod lines, which is shown in Figure 4.4-4. Power adjustments with control rods can also be made along any constant recirculation flow line. However, routine operation is restricted from entering the Buffer and Exclusion Regions because of concerns involving thermal-hydraulic instability.

4.4.4 Evaluation

The evaluation of the thermal and hydraulic design is continued in the supplemental reload licensing report for each core cycle.

4.4.5 Testing and Verification

Principal thermal-hydraulic characteristics were evaluated by core performance tests in the engineering startup program during the step-wise approach to full power. The evaluation procedure included the use of in-core nuclear instrumentation and normal plant process instrumentation, combined with the use of detailed precalculations to establish actual core performance characteristics including core flow, power, power distribution, minimum critical heat flux ratio (minimum critical power ratio), steam quality, and void distribution.

In addition, plant instrumentation allows evaluation of these principal core thermal-hydraulic characteristics at any time during operation.

The detailed core power and MCPR distribution is calculated periodically. The plant is operated as necessary to maintain MCPR and the linear heat generation rates within the applicable Technical Specification values.

4.4.6 Instrumentation Requirements

Instrumentation employed in monitoring and measuring those thermal-hydraulic parameters important to safety is discussed in detail in Chapter 7.

4.4.7 References

(1) "General Electric Standard Safety Analysis Report," Proprietary Supplement to Amendment 14, Docket No. STN-50-447, May 1974.

(2) H. E. Williamson and D. C. Ditmore, "Experience with BWR Fuel Through September 1971," NEDO-10505, May 1972.

(3) "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A, as revised at the time of the specific reload cycle.

(4) General Electric Standard Application for Reactor Fuel (Supplement For United States), NEDE-24011–P–A–US, as revised at the time of the specific reload cycle.

(5) "GEGAP-III, A Model for the Prediction of Pellet-Cladding Thermal Conductance in BWR Fuel Rods," NEDC-20181-A, Revision 1, September 1986.

(6) Jens, W. R., and Lottes, P. A., "Analysis of Heat Transfer, Burnout, Pressure Drop, and Density Data for High Pressure Water, USAEC Report-4627, May 1951.

(7) Oyster Creek Nuclear Generating Station, Docket No. 50-219, FDSAR Amendment 76 (January 31, 1975).

(8) "The XN-2 Critical Power Correlation," XN-75-34(A), Revision 1, September 9, 1976.

(9) M. A. Alammer, et al, "BWR-2 Transient Analysis Model Using the RETRAN Code", September 3, 1987.

(10)
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(11)

(12) "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application," January 1977 (NEDE-10958-PA and NEDO-10958-A).

(13) "Core Flow Distribution in a Modern Boiling Water Reactor as Measured in Monticello," NEDO-10299A, October 1976.

(14) H. T. Kim and H. S. Smith, "Core Flow Distribution in a General Electric Boiling Water Reactor as Measured in Quad-Cities Unit 1," NEDO-10722A, August 1976.

(15) "Brunswick Steam Electric Plant Unit 1 Safety Analysis Report for Plant Modifications to Eliminate Significant In-Core Vibrations," NEDO-21215, March 1976.

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(18) N. Zuber and J. A. Findlay, "Average Volumetric Concentration in Two-Phase Flow Systems," Transactions of the ASME Journal of Heat Transfer, November 1965.

(19) "BWR Fuel Channel Mechanical Design and Deflection," NEDE-21354-2-P (Proprietary) and NEDO-21354-2, July 1977.

(20) "Supplemental Information for Plant Modification to Eliminate Significant In-Core Vibration," NEDE-21156 (Proprietary), January 1976.

(21) "Supplement 1 to the Technical Report on Densification of General Electric Reactor Fuels," Regulatory Staff (U.S. Atomic Energy Commission), December, 1973.

(22) "Modified GE Model for Fuel Densification," V. A. Moore, Letter to I. S. Mitchell, Docket 50-321, March 22, 1974.

(23) NEDC-31462P Oyster Creek Nuclear Generating Station SAFER/CORECOOL/GESTR-LOCA Loss of Coolant Accident Analysis.

(24) GE-NE–0000–0001–7486–01P, “Oyster Creek Generating Station Loss of Coolant Accident Analysis for GE11,” June 2002.

(25) NEDC-33270P, "GNF2 Advantage Generic Compliance with NEDE-24011-P-A (GESTAR II)," as revised at the time of the specific reload cycle.

(26) 0000-0098-3503-R2, "Oyster Creek Generating Station GNF2 ECCS-LOCA Evaluation" November 2010.

TABLE 4.4-1
(Sheet 1 of 1)

ZIRCALOY-2 MATERIAL PROPERTIES

Parameter	Value
Thermal Conductivity, Btu/hr - ft-°F (600 to 800°F)	9-10
Coefficient of Thermal Expansion in/in-°F	
Radial	3.2×10^{-5}
Longitudinal	2.9×10^{-6}
Total Plastic Strain Capability, percent	1.0

TABLE 4.4-2
(Sheet 1 of 1)

LINEAR HEAT GENERATION RATE
CALCULATED TO CAUSE 1 PERCENT PLASTIC DIAMETRAL STRAIN
FOR P8x8R FUEL

Fuel Exposure Megawatt-days per (Metric ton uranium MWd/MTu)	LHGR at Calculated 1% Plastic Strain(kw/ft)*	
(MWd/t)	UO_2	Gd**
0	24.7	21.9
20,000	23.0	20.4
40,000	19.6	17.3

* The values reported have been reduced by an amount equal to the calculated power spiking penalty (percent).

** Results for gadolinia are applicable for maximum concentration used in reload fuel design.

TABLE 4.4-3
(Sheet 1 of 1)

REPRESENTATIVE TRANSIENT MCPR VALUES

Transient	Delta CPR
Turbine Trip without Bypass:	0.38*
Loss of 100°F Feedwater Heating:	0.12*
Feedwater Controller Failure:	0.36*
Fuel Loading Error (Mislocated)	0.22
Fuel Loading Error (Rotated)	0.23
Local Rod Withdrawal Error: (no credit for APRM Rod Block)	
APRM Status	
1. If any 2 LPRM assemblies which are input to the APRM System and are separated in distance by less than 3 times the control rod pitch contain a combination of 3 out of 4 detectors located in either the A and B or C and D levels which are failed or bypassed.	0.22
2. If any LPRM input to the APRM System at the B, C, or D level is failed or bypassed or any APRM channel is inoperable or bypassed.	**
3. All B, C and D LPRM inputs to the APRM System are operating and no APRM Channels are inoperable or bypassed.	**

* Does not include statistical adder
** No longer calculated since RWE transient is not limiting DCPR

TABLE 4.4-4
(Sheet 1 of 1)

BYPASS FLOW PATHS

	Flow Path Description	Driving Pressure	Number of Paths
1a.	Between Fuel Support and the Control Rod Guide Tube (Upper Path)	Core Support Plate Differential	One/Control Rod
1b.	Between Fuel Support and the Control Rod Guide Tube (Lower Path	Core Support Plate Differential	One/Control Rod
2.	Between Core Support Plate and the Control Rod Guide Tube	Core Support Plate Differential	One/Control Rod
3.	Between Core Support Plate and the In-Core Support Instrument Guide Tube	Core Support Plate Differential	One/Instrument
4.	Between Core Support Plate and Shroud	Core Support Plate Differential	One
5.	Between Control Rod Guide Tube and Control Rod Drive Housing	Core Support Plate Differential	One/Control Rod
6.	Between Fuel Support and Lower Tieplate	Channel Wall Differential Plus Lower Tieplate Differential	One/Channel
7.	Control Rod Drive Coolant	Independent of Core	One/Control Rod
8.	Between Fuel Channel and Lower Tieplate	Channel Wall Differential	One/Channel
9.	Holes in Lower Tieplate	Channel Wall Differential Plus Lower Tieplate Grid Differential	Two/Fuel Assembly (where applicable)

TABLE 4.4-5
(Sheet 1 of 1)

GENERAL OPERATING CONDITIONS, RATED POWER

TYPICAL POWER DISTRIBUTION

Fuel Design Thermal Output	1930 MWt
Reactor Pressure (steam dome)	1035 psia
Steam Flow Rate	7.259×10^6 1b/hr
Recirculation Flow Rate	61.0×10^6 lb/hr
Fraction of Power Appearing as Heat Flux	0.967
Power Density	40.2 kw/liter
Core Inlet Enthalpy	517.5 Btu/lb
Core Average Void Fraction, Active Coolant	39.4 percent (variable)
Core Average Exit Quality	0.118
Minimum Critical Power Ratio	>1.55

4.5 REACTOR MATERIALS

4.5.1 Control and Drive System Structural Materials

A detailed description of the Control Rod Drive Mechanisms is presented in Subsection of 4.6.1.1. All drive components exposed to the reactor coolant are either of 300 series stainless steel (BWR/2-5 CRDs) or of XM-19 stainless steel (BWR/6 CRD), except the following:

a. Seals and bushings on the drive piston and stop piston are Graphitar 14 (BWR/2-5 CRD only).

b. All springs and members requiring spring-action (collet fingers, coupling spud and spring washers) are made of heat-treated nickel base alloy.

c. The ball check valve is a Haynes Stellite cobalt base alloy.

d. Elastomeric O-ring seals are ethylene propylene.

e. Collet piston rings are cobalt base alloy.

f. Certain wear surfaces are hard faced with nickel base.

g. Nitriding by a proprietary Malcomizing process, electrolizing (a vapor deposition of chromium) and chromium plating are used in certain areas where resistance to abrasion is necessary.

h. The piston head is made of a heat-treated high nickel alloy.

All portions of the drive forming the external pressure shell or barrier are designed according to ASME Codes, and materials used conform to the appropriate ASME material specifications. Drive parts not in contact with coolant, chiefly the position indicator probe, are constructed of materials meeting the particular design requirements of the part. The probe, for instance, must be non magnetic, hence aluminum extrusions, beryllium copper clips and a 300 series stainless sheath are employed.

Significant drive parts and the factors determining the choice of materials are listed below:

a. The index tube must withstand the locking and unlocking action of the collet fingers. A compatible bearing combination is provided which is able to withstand moderate misalignment forces. The reactor environment seriously limits the choice of materials suitable for corrosion resistance. The column and tensile loads are satisfied by an annealed 300 series stainless steel (BWR/2-5 CRDs) or XM-19 stainless steel (BWR/6 CRD). The wear and bearing requirements are provided by Malcomizing the completed tube. To obtain suitable corrosion resistance, a carefully controlled process of surface preparation is employed.

b. The coupling spud is subjected to severe service conditions. Nickel base alloy, aged to produce maximum physical strength, is used to provide the required strength and corrosion resistance. As misalignments tend to produce a chafing in the semi spherical contact area, the entire part is protected by a thin vapor

deposited chromium plating (electrolizing process). This plating also serves to prevent galling of the threads attaching the coupling spud to the index tube.

c. Heat treated nickel base alloy is used for the collet fingers, which must function as leaf springs when cammed open to the unlocked position. A hard facing is applied to the area contacting the index tube and unlocking cam surface of the guide cap. Hard facing provides the long wearing surface which is adequate for design life.

d. Graphitar 14 has been selected for seals and bushings on the BWR/2-5 CRD drive piston and stop piston. The material is inert and has a low friction coefficient when water lubricated. Some loss of strength is experienced at higher temperatures, so the drive is normally supplied with cooling water to hold temperatures below 250°F. The Graphitar is relatively soft, which is advantageous when an occasional particle of dirt reaches a seal. Resulting scratches reduce sealing efficiency until "worn in" again, but the drive design allows for considerable leakage. These seals determine the practical service life of a drive mechanism, and the frequency of maintenance is based on average seal life.

4.5.2 Reactor Internal Materials

4.5.2.1 Design Basis of Reactor Vessel Internal Structures

The internal components of the reactor vessel (in conjunction with the reactor vessel) allow adequate core cooling to be maintained during normal operation and accident conditions and will not fail during normal operation and accident conditions. The reactor internals are designed to withstand a design basis earthquake.

To meet the above, the internal components of the reactor are designed to:

a. Provide support for the fuel, steam separators, dryers, etc., during normal operation and accident condition.

b. Maintain required fuel configurations and clearances during normal operation and accident conditions.

c. Circulate reactor coolant to cool the fuel.

d. Provide adequate separation of steam from water.

4.5.2.2 Description of Reactor Vessel Internal Structures

The reactor vessel internal structures are shown in Figure 4.5-1. Subsection 3.9.5.1 provides the description of the reactor vessel internals design arrangements.

The core shroud is an austenitic stainless steel cylinder. All of the core structure, except the springs in the fuel assemblies, the shroud support and the zircaloy fuel rod cladding, is fabricated from austenitic stainless steel. The shroud support is fabricated from solid nickel base alloy. The control rod guide tubes are constructed of austenitic stainless steel pipe, which is adequate for an external pressure of 100 psi.

4.5.2.3 Bases and Design Evaluation

The core structural components are designed to accommodate the loadings applied during normal operation and maneuvering transients considering both stress and deflection. Deflections are limited so that the normal functioning of the components under these conditions will not be impaired. Where deflection is not the limiting factor, the ASME Boiler and Pressure Vessel Code, Section III is used as a guide to determine limiting stress intensities and cyclic loadings for the core internal structure where code use is not mandatory.

The internal structural components are designed to also accommodate earthquake conditions. Structural earthquake loading on the internals is based on the calculated reactor vessel acceleration response of 0.20g to 0.30g, depending upon vertical elevation on the vessel, due to the specified ground motion of 0.11g. Deflections of components supporting the fuel and control systems are limited to allow safe operation during this transient. Stress intensity in these components is also maintained within the ASME Pressure Vessel Code, Section III.

The reactor internals are designed to preclude failure which would result in any part being discharged through the steam line in the event of a steam line break outside of the steam line isolation valve.

The structural components which guide the control rods are examined to determine the loadings which would occur in a Loss-of-Coolant Accident (including a steam line break). The core structural components are designed so that deformations produced by accident loadings will not prevent insertion of control rods.

4.5.2.4 Surveillance and Testing

Rigid quality control requirements insured that the design specifications of the vessel internal components were met. These quality control methods were utilized during the fabrication of the individual components as well as during the assembly process.

Preoperational performance tests of the core spray sparger demonstrated satisfactory operation of the system.

4.6 FUNCTIONAL DESIGN OF REACTIVITY CONTROL SYSTEMS

4.6.1 Information for Control Rod Drive System

4.6.1.1 Control Rod Drive Mechanism

The existing control rod drive (CRD) mechanisms employed at Oyster Creek are identified as type BWR/2-5. During Cycle 12R, thirty (30) of the BWR/2-5 type CRD's were replaced by redesigned type BWR/6 CRD's. During Cycle 13R, an additional twenty two (22) BWR/2-5 CRDs were replaced with BWR/6 CRDs. Future CRD replacements will also be of the BWR/6 design. The BWR/2-5's and BWR/6's are locking piston type, double acting hydraulic piston mechanisms. Both types of drives operate on the identical hydraulic principal and require the same supporting equipment to function. The hydraulic flow and pressure required for both drive designs to insert and withdraw the rod are the same.

4.6.1.1.1 BWR/2-5 CRD

The control rod drives are of the locking piston type. This type of drive has been used in the Dresden 1, SENN in Italy, Humboldt Bay 3 and Big Rock Point plants. The control rod drive mechanism as described herein is the same as that used on other General Electric BWRs. A simplified component illustration of the control rod drive system is shown in Figure 4.6-1 and a simplified sketch of a control rod drive mechanism is shown in Figure 4.6-2. A simplified sketch of the Oyster Creek control rod drive hydraulic control units, scram discharge volumes and scram air valving is shown in Figure 4.6-3. The drive mechanisms are mounted vertically in housings which are welded into the reactor bottom head penetrations. The low end of each housing terminates in a special flange which contains ports for attaching the hydraulic system lines, and a machined face which mates with a corresponding flange at the lower end of the drive mechanism itself. The operating principles and construction details of the mechanism are described in Subsection 3.9.4.

At the top end of the drive index tube (the movable element), a coupling is provided which engages and locks into a socket at the base of the control rod (Figure 4.6-4). The weight of the control rod alone will engage and lock this coupling. Once locked, the drive and rod form an integral unit which must be manually unlocked by specific procedures before a drive or rod can be removed from the reactor. These procedures are established to prevent an accidental separation of the control rod from the control rod drive.

The drives position the control rods in six inch increments of stroke and hold them in these discrete latch positions until actuated for movement by the hydraulic system to a new position. Visible indication of the position of each drive is displayed in the Control Room by means of illuminated numerals which correspond with the respective latched positions (see Chapter 7). In addition, indication is provided that shows whether insert and withdraw travel limits of the drive or an overtravel withdraw limit on the drive have been reached. Control rod seating at the lower end of the stroke prevents the overtravel withdrawal limit from being reached unless the control rod is uncoupled from the drive. This allows the coupling to be checked. These indicators and those for the incore monitors are grouped together and displayed on a control panel and arranged on the board to correspond to relative rod and incore monitor positions in the core.

During reactor operation, individual control rod drive mechanisms can be actuated to demonstrate functional performance.

During reactor shutdown the shutdown margin can be verified by withdrawing a maximum worth rod and ascertaining that the reactor is substantially subcritical.

4.6.1.1.2 BWR/6 CRD

The functional operation of the BWR/6 CRD is identical to the existing drives used in BWR/2-5s. The major difference between the BWR/6 CRD and the BWR/2-5 CRD lies in the buffer design. The buffer stroke for the BWR/6 CRD is about 2.5 inches as compared to 8.5 inches for the BWR/2-5 CRDs. The BWR/6 buffer design, unlike the BWR/2-5 design, isolates the higher buffer pressure from the Graphitar seals during the deceleration phase of the scram stroke. The interfacing internal components, such as the index tube, piston tube, drive piston and stop piston, were redesigned to incorporate this new buffer design.

In addition to the buffer modification, the major system differences between the BWR/2-5 system and the BWR/6 system are those changes associated with achieving a faster scram performance in the BWR/6 design.

The BWR/6 type CRD is designed for higher pressures and scram loads to support the BWR/6 fast scram requirements. The existing principal of operation, pressures and scram loads are maintained and no modification to the BWR/2-5 CRD supporting system is required.

Use of the BWR/6 type CRDs for Oyster Creek does not require any changes to the supporting equipment, hydraulic parameters or operational procedures. The scram and insert/withdrawal performance times remains the same with the application of the BWR/6 type CRD in Oyster Creek.

4.6.1.2 Hydraulic Control System

Under normal operation, the Hydraulic Control System (Drawing GE197E871) uses unheated condensate supplied by one of two drive system pumps as the working fluid to accomplish hydraulic positioning of the control rod drives and their attached control rods. These pumps take suction from the condensate system at low pressures and temperatures and discharge to the pressure control stations through filters. Water for charging scram accumulators is provided at approximately 1400 psig while the water required for cooling and positioning the drives is provided at a constant differential of approximately 250 psig above normal operating reactor pressure. Water for drive cooling is further adjusted at the control station to maintain cooling water flow through the individual drive in the reactor vessel. Pressure and flow for fast insertion (scram) of the control rods is supplied by stored energy in the scram accumulators or by reactor pressure. The operation of the hydraulic system is described in Subsection 3.9.4.

The Hydraulic Control System is so arranged that the equipment common to each drive can be packaged in modular form (Figure 4.6-6), one module for each drive. Any failure of the scram system within a particular module would, therefore, affect only its associated drive. Areas which are necessary to the scram system and common to all modules include the accumulator charging header, the scram discharge headers, volumes and instrumentation volumes, drive water header, cooling water header and exhaust header, as well as scram valve pilot air header.

If for any reason the accumulator charging header supply pressure should fail (this failure would be alarmed to the operator), the stop check valves supplying pressure to the accumulator in each module would close and hold the charged pressure so that scram capability is not lost.

The North and South scram discharge volumes, which receive water ejected from the drives during a scram, are provided with diverse and redundant instrumentation on their scram discharge instrument volumes. This instrumentation initiates SDV not drained alarm, hi-level rod block and hi-hi level scram signals at appropriate setpoints utilizing one-out-of-four twice logic for scram. This arrangement assures that adequate volume is available in the system to receive the water discharged during a scram.

The common point in the system where an accident, such as a plugged line, could affect the scram time of more than one drive would be in the scram discharge volumes. As the headers making up the scram discharge volumes are much larger than the individual lines feeding into them, and since they were thoroughly checked during preoperational tests, it is extremely unlikely that they could become plugged. Further, action of the drive during a scram is such that it will develop a pressure in excess of 2000 psig if its discharge is restricted. This pressure should be capable of expelling any conceivable line restriction. The system is designed to accommodate such pressures.

Also, because of the unique design of the locking piston drive, an automatic scram occurs if both the insert and withdrawal lines to the control rod drive or only the withdrawal line is severed at any point with the reactor at pressure.

During reactor shutdown, and with fuel loaded into the core, all control rods are normally inserted. A shutdown criteria has been established so that the reactor is still subcritical with the highest worth rod withdrawn. Interlocks are provided which prevent the inadvertent withdrawal of more than one control rod with the mode switch in the refuel position.

Rapid shutdown of the reactor is accomplished through actuation of the Reactor Protection System (or via manual scram) which opens the scram valves and permits water under pressure to be applied to the drive mechanism. The action exerts a pressure on the control rod drive piston mechanisms, and causes all rods to be fully inserted into the core. Any rod which is fully withdrawn will be fully inserted in approximately five seconds. Pressure for rod insertion is also available from the reactor, as discussed in Subsection 3.9.4.

4.6.2 Evaluation of the Control Rod Drive System

4.6.2.1 Rate of Response

The Reactivity Control System is designed such that under conditions of normal operation means are provided for continuous regulation of the core excess reactivity and reactivity distribution. The movement of control rods must not perturb the reactor beyond the capability of an operator to respond to the disturbance. This requirement prevents unnecessary operation of the Reactor Protection System. The maximum rate at which the rods can be moved and the incremental distance between control drive notches is such that under normal operating conditions a single notch increment of control withdrawal at the maximum withdrawal rate will result in a reactor period of not less than 20 seconds.

Under conditions of expected abnormal reactor system disturbances the Reactivity control System provides a sufficient rate of negative reactivity insertion, upon a signal from Reactor Protection System, to prevent fuel damage. Expected abnormal reactivity disturbances and resulting power transients in the core can arise from any of three sources:

a. Reactor system induced disturbances of core parameters such as coolant flow or pressure.

b. Single operator errors.

c. Single equipment malfunctions.

The design philosophy for the safety and control systems requires that, for those accidental power transients with the potential for endangering the health and safety of the public, the system provides, in addition to containment, at least a double level of automatic or inherent protection. For other incidents, at least a single level of automatic or inherent protection is provided by these systems.

For the incidents requiring single level protection, the Reactor Protection System described in Section 7.2 senses the disturbances and under certain specified conditions initiates a scram signal. Upon receipt of a scram signal the Reactivity Control System is required to render the reactor subcritical at a rate sufficient to prevent the initiating disturbance from causing fuel damage. An extensive program has been conducted to determine the characteristics of the Control Rod Drive System for BWRs. Part of this program has been the measurement of rod position versus time after pilot valve actuation. Many drives have been tested and the data has been treated statistically to arrive at a position versus time curve which is based on 95 percent confidence that 99.5 percent of all scram times are less than that time. The mean rod insertion times will be no greater than:

a. 10 percent of travel in 0.7 seconds

b. 50 percent of travel in 2.05 seconds

c. 90 percent of travel in 5.0 seconds.

The above does not include any delays due to instrumentation or scram circuitry. The actual reactivity inserted during scram would correspond to the mean of the data which shows much better characteristics than the specifications.

The analyses performed to evaluate nonsteady state behavior of the reactor fall into two categories:

a. Analysis of the transients associated with single equipment malfunction or single operator error

b. Evaluation of prompt excursions.

In analyzing the transients of the first category, a conservative activity or rod position versus time curve has been used. The evaluation of the second category, prompt excursions, is not sensitive to the scram rate since the excursion is terminated by the Doppler effect. Therefore, in these calculations, a simple ramp reactivity insertion was assumed. For both categories of transients, the scram system response used satisfied the above requirements. Therefore, the response of the Reactor Protection System in combination with the size, heat transfer features and inherent dynamic response characteristics of the core, prevent fuel damage resulting from a reactivity insertion accident due to any single equipment malfunction or single operator error.

The inherent safety features of the reactor design as described in Section 4.3, in combination with engineered safeguards relating to the Reactivity Control System are such that the consequences of a potential nuclear excursion accident, caused by any single component failure within the reactivity control system itself, does not result in damage either by motion or rupture to the reactor primary coolant system or impair operation of required safeguards. These features and safeguards thus limit potential power excursions such that the design bases are satisfied.

Certain postulated rapid reactivity insertion accidents are evaluated for the purpose of determining whether damage to the primary system would result. To damage the primary system from pressure and momentum effects, rapid (few millisecond) rates of heat transfer from fuel to reactor coolant are required. To achieve the heat transfer rate which could lead to damage requires, at the least, rapid fuel rod rupture and dispersal of some quantity of hot fuel into the surrounding coolant. The Oyster Creek reactor is designed on the basis of avoiding sudden rupture of a significant number of fuel rods in any accidental excursion resulting from component or procedural failure within the reactivity control system. The threshold for this type of fuel rupture is estimated to correspond to a fuel energy content of 425 calories/gm UO_2. This limit is discussed further in Subsection 4.3.2.5.

The magnitude of an accidental nuclear excursion is limited first by the strong, negative Doppler coefficient of reactivity inherent in the reactor design and secondly by the rate at which positive reactivity is added to the system by the malfunction. By the maximum worth that an individual rod can assume and the maximum rate at which it can drop from the core, potential accidental reactivity insertion rates can be held below values which could cause primary system damage. An engineered safeguard, the Rod Velocity Limiter described in Subsection 4.6.4, limits the rate at which a rod can fall from the core, and operating procedures, backed up by the Rod Worth Minimizer described in Section 7.6, limit the maximum control rod worth.

As a design basis, reactivity addition rates are limited by these devices to well below the selected limit, i.e., that value which could result in any significant amount of fuel reaching the sudden clad rupture range during an accidental excursion. The nominal value for excursion consequences used in setting control system and engineered safeguards design is 280 cal/gm. Thus, the Rod Worth Minimizer shall restrict control rod worths to such that in the event of a component failure and/or operator error, the consequences would not exceed a fuel enthalpy of 280 cal/gm. Since the control rod withdrawal rates are about 20 times slower than the control rod drop rate and the rod drop rate is safe, there will be no safety or operating limit on normal control rod withdrawal speed.

4.6.2.2 Rod Withdrawal Errors

Design features provided to minimize the probability of inadvertent continuous control rod withdrawal and to limit potential power transients in the event they should occur include the following:

a. The control system is designed so that only one rod can be withdrawn at a time.

b. Normal rod operation is a step (notch) at a time. Two control switches must be held open at the same time to withdraw a rod continuously.

c. The maximum continuous rod withdrawal rate is limited by the control rod hydraulic flow system to 3.6 inches per second.

d. Interlocks prevent rod withdrawal if the neutron flux monitors are not in a condition to provide the required protection, or if the rod withdrawal timing relay should fail.

e. Preplanned withdrawal patterns and procedural controls are used to prevent abnormal configurations giving high rod worths.

f. A Rod Worth Minimizer backs up operating procedures in preventing withdrawal of rods which would, in the event of a rod drop from the core, yield consequences in excess of 280 cal/gm.

g. Intermediate and power level scrams limit power excursions from low reactor power levels. During power operation, incore monitor alarms warn the operator if local neutron flux levels approach preset limits, an incore monitor averaging circuit blocks rod withdrawal and the simultaneous trip of two Average Power Range Monitors (one in each scram logic circuit), initiates reactor scram.

4.6.2.2.1 Continuous Control Rod Withdrawal at Power

Procedural controls require control rod patterns which minimize the worth of individual control rods. If heat fluxes (and critical power ratios) approached fuel damage levels, Average Power Range Monitors would block rod withdrawal. Starting at rated power, if a fully inserted control rod were erroneously withdrawn near a region which is operating near the limit of the normal operating range, the average power range monitors would prevent the critical power ratio or the temperature of the highest power pellets from approaching damaging levels by blocking further rod withdrawal. Thus, procedures are supplemented by automatic protection to prevent fuel damage.

4.6.2.2.2 Control Rod Withdrawn With Low Recirculation Flow

The reactor power can be adjusted by controlling the recirculation flow rate. The power resulting at a given value of recirculation flow is determined by the void and Doppler reactivity defects. As the recirculation flow rate is reduced, the power level will reduce automatically such that the amount of power being generated can always be safely handled by the coolant flow. The critical power margin increases as the power is dropped by flow control. However, rod withdrawal from this low flow, low power operating state could result in a reduction of the minimum critical power ratio to undesirable values before the rated power high flux scram would occur. Consequently, the Average Power Range Monitors are set to block control withdrawal when the reactor power is above the established limit at any reduced flow condition.

4.6.2.2.3 Continuous Control Rod Withdrawal From Lower Levels

Continuous control rod withdrawals from levels below one percent power will be terminated by neutron flux scram at conditions even milder than those for rod withdrawal at power. In the intermediate and power range instrumentation, high neutron flux scrams are located at the top of each instrument range. Interlocks prevent rod withdrawal if the instruments are not reading on scale unless they are set on their lowest range (see Section 7.7). Therefore, abnormal power increases are terminated before the power could reach cladding damaging conditions. Rather than analyze rod withdrawals from each intermediate power level, a rod withdrawal from cold conditions, assuming a control rod worth of 0.025 DELTA k withdrawn at the maximum

withdrawal rate (giving a maximum reactivity insertion rate of 0.0019 DELTA k/sec assumed constant during the withdrawal) and assuming no intermediate level scram, was analyzed. The excursion was turned by the inherent negative Doppler reactivity defect from fuel heating. Only then was the reactor scram, at over 120 percent of full power, assumed to be effective. In actuality, the reactor would have been scrammed at a much lower power level either by the intermediate range or power range instrumentation. Even with these assumptions the minimum reactor period was calculated to be approximately 30 milliseconds and the maximum reactor power peaked at 4500 MW thermal. About 530 MW-second of energy is generated in the high power density region. This energy generation would increase the fuel temperature at the hottest point to 2500°F and no fuel melting or cladding damage would occur. If the rod withdrawal started from a higher initial power level, the excursion would have been even milder, and scrams would have terminated the excursions at a much lower level.

4.6.2.3 Failure Effects of the Control Rod Drive System

The effects of various failures within the control rod drive mechanism or the Control Rod Drive Hydraulic System are discussed in detail in Subsection 3.9.4.

4.6.3 Testing and Verification of the Control Rod Drive System

During production, control rods were statistically tested for dimensions. After installation, all rods and drive mechanisms were tested full stroke for operability.

Each time a control rod is withdrawn a notch, the operator observes the incore monitor indications to verify that the control rod is following the drive mechanism. All control rods that are partially withdrawn from the core are tested weekly for rod following by inserting or withdrawing a notch and returning to the original position, while the operator observes the incore monitor indications.

When the operator withdraws a control rod full out of the core, a test is made of the coupling integrity by trying to withdraw the rod drive mechanism to the overtravel position. Failure of the drive to overtravel demonstrates rod to drive coupling integrity.

During a refueling outage, each control rod is fully withdrawn and inserted to test for operability. The scram time for each control rod is tested during each refueling outage.

4.6.4 Information for Combined Performance of Reactivity Systems

4.6.4.1 Standby Liquid Control System

In addition to the Control Rod Drive System, a Standby Liquid Control System (Liquid Poison System) is provided as a redundant shutdown system. The system is designed to bring the reactor from rated power to a cold shutdown at any time in core life. It is required only to shut down the reactor at a steady rate and keep the reactor subcritical as it cools. A fast scram of the reactor or operational control of fast reactivity transients is neither possible, nor required to be accomplished by this system. The Standby Liquid Control System is manually initiated from the Control Room to pump a boron neutron absorber solution into the reactor if the Control Room operator believes the reactor cannot accomplish shutdown or sustain shutdown using the control rods. The system is never expected to be needed because of the large number of independent control rods available to shutdown the reactor.

The Standby Liquid Control System is described in detail in Subsection 9.3.5.

4.6.4.2 Recirculation Flow Control System

As discussed in Subsection 4.4.3.1, reactor power can be adjusted using the recirculation flow rate over a wide range, bounded at the lower end by the 20 percent recirculation pump speed line.

This method of power level control takes advantage of the reactor negative void coefficient. To increase reactor power, it is necessary only to increase the recirculation flow rate which sweeps some of the voids from the moderator, causing an increase in core reactivity. As the reactor power increases, more steam is formed and the reactor stabilizes at a new power level with the transient excess reactivity balanced by the new void formation. No control rods are moved to accomplish this power level change. Conversely, when a power reduction is required, it is necessary only to reduce the recirculation flow rate. When this is done, more voids are formed in the moderator, and the reactor power output automatically decreases to a new power level commensurate with the new recirculation flow rate. No control rods are moved to accomplish the power reduction.

Power level control through the use of variations in the recirculation flow rate (flow control) is advantageous relative to power level control by control rod positioning. Flow variations perturb the reactor uniformly in the horizontal planes, and thus allow operation with flatter power distribution and reduced transient allowances. As the flow is varied, the power and void distributions remain approximately constant at the steady-state end points for a wide range of flow variations. These constant distributions provide the important advantage that the operator can adjust the power distribution at a reduced power and flow by movement of control rods and then bring the reactor to rated conditions by increasing flow, with the assurance that the power distribution remains approximately constant. The Recirculation Flow Control System is described in detail in Sections 5.4 and 7.7.

4.6.4.3 Control Rod Velocity Limiter

4.6.4.3.1 Design Basis

The Rod Velocity Limiter, in conjunction with the Rod Worth Minimizer, is provided to limit reactivity addition to the core in the event of a control rod drop accident to a rate which would not jeopardize primary system integrity or impair operation of required safeguards equipment.

The Rod Velocity Limiter is an engineered safeguard designed to limit the free fall drop velocity of the control rod to 3.11 ft/sec with the moderator temperature at 70°F. The control rod drop accident can happen only in the event of simultaneous procedural violations and equipment malfunctions. This sequence of events involves the establishing of a high worth control rod pattern through gross violation of procedures, a separation or

mechanical failure in the drive line, sticking or binding of the control rod, the withdrawal of the detached control rod drive mechanism, and then the release of the control rod by some unspecified means. The Rod Velocity Limiter is designed to limit the consequences of a drop of the maximum worth control rod, without significantly hindering the normal function of the system. The consequences of the control rod drop accident are reactivity rate dependent. The most probable threshold for potential mechanical damage to the reactor core or other primary cooling system components is a peak fuel enthalpy in excess of 425 cal/gm. By reducing the

velocity of a free falling rod to 3.11 ft/sec and assuring that excessive rod worth patterns cannot be established, the control rod drop accident will result in peak fuel enthalpy values in the range from 200 to 280 cal/gm. The associated energy release results in no system damage and presents minimal offsite dose consequences.

4.6.4.3.2 Description

The Rod Velocity Limiter is an integral part of the bottom of each control rod. It is designed as a large clearance piston which travels in the control rod guide tube over the entire control rod stroke (see Figures 4.6-7 and 4.6-8).

The velocity limiter assembly consists of two conical elements machined from a single Type 304 stainless steel casting. The lower conical element is a 15° angle relative to the upper conic element and the two elements are separated by four spacers 90° apart. There are no moving parts in the velocity limiter.

The Rod Velocity Limiter provides a streamlined profile in the scram (upward) direction and a nonstreamlined profile in the dropout (downward) direction. It may be regarded as a nozzle type limiter since during its downward motion a high percentage of the total water directly below the limiter flows up through the center of the limiter body and is ejected radially outward into the limiter guide tube annulus at an oblique angle due to its conical configuration.

The Rod Velocity Limiter has a nominal diameter of 9.255 inches and operates inside a control rod guide tube with an inside diameter of 10.420 inches. This configuration results in an annulus between the limiter and the guide tube of 0.582 inches, nominal. The casting terminates at the top of a cruciform section which matches the blade shape. The casting is machined to obtain the required outside diameter, roller lug and backseat. Consistent Rod Velocity Limiter performance is assured by machining the outside diameter and maintaining control of the hydraulically important casting tolerances. The backseat and coupling are identical to designs employed in the locking piston drive. The rollers are stellite to prevent galling against the interior of the guide tube and are held in place by pins made of

Haynes 25 alloy. A nominal radial gap of 0.125 inch is provided between the rollers and guide tube wall to allow for tolerance variation and misalignment.

The Rod Velocity Limiter is welded to the control rod blade at the cruciform to become an integral part of the control rod assembly. A coupling release handle is located in the cruciform section immediately below the control rod blade; raising the handle permits the control rod assembly to be separated from the locking piston drive.

The Rod Velocity Limiter acts within a cylindrical guide tube. The guide tube has a backseat on its lower end which rests on the control rod drive housing. This seat restricts water flowing out of the guide tube during the postulated accident and also restricts water flowing to the interior of the guide tube which would bypass the fuel elements during normal reactor operation. A close fit is provided between the top of the guide tube and the core plate to restrict flow through this joint, which would also bypass the fuel elements. There is normally an external pressure of 15 psi on the guide tube due to the core pressure drop.

4.6.4.3.3 Design Analysis

The Rod Velocity Limiter must limit the free fall velocity without significantly reducing the scram time of the rod. The hydraulic drag forces on the rod are approximately proportional to the square of the rod velocity and are essentially negligible during normal withdrawal or insertion functions. However, during the scram stroke, the rods reach relatively high velocities and, therefore, the drag forces become appreciable. As an additional design consideration, the Rod Velocity Limiter must not produce unacceptable loadings on the drive mechanism during the scram condition. Various full scale tests were performed at cold and operating conditions to evaluate the effect of the Rod Velocity Limiter on the control rod scram time. These tests included full scale prototypes using production design components under simulated reactor operating conditions. Results of these tests showed the Rod velocity Limiter caused an increase in scram time of approximately 0.25 seconds for 90 percent of stroke. The required design scram time for this distance is 3.4 seconds. The increased scram time due to the Rod Velocity Limiter is not prohibitive and the overall performance satisfies the design basis requirements.

Numerous model tests were performed in order to determine the sensitivity of the design to changes in its various dimensions (angles, gaps, etc). Wood and metal models were fabricated so that separate parameters could be variedwhile holding the remaining portions constant. Approximately 100 drop tests were performed. As a result of these tests the following principles were established for the design of the production rod velocity limiter:

a. The upper and lower cones must converge at 45° and 30°, respectively, from the horizontal and must be separated by about one inch at their periphery.

b. The upper cone must have an appreciable thickness.

c. The top of the upper cone requires a gentle radius at its outer edge and must terminate with a sharp edge.

d. A long, straight or tapered skirt attached to the lower cone decreases the effectiveness of this design.

e. The contours through the nozzle area need not be streamlined.

f. The relative angle between the cones, the outside diameter of the cones, and the top cone outer edge radius are dimensions which must be controlled.

Normal rod insertion and withdrawal operations were performed to detect possible abnormalities which could be attributed to the presence of the rod velocity limiter. As can be seen in the tabulated data, the pressure drop across the piston and the travel times are within the test guideline tolerance.

Insert	Reactor Pressure Atmospheric	1000 psi	Test
DELTA P across piston, psi	82	80	90 ± 15
Average velocity, ips	3.26	2.9	3 ± 0.3

Withdraw	Reactor Pressure Atmospheric	1000 psi	Test
DELTA P across piston, psi	120	120	110 ± 10
Average velocity, ips	3.0	3.3	3 ± 0.3

Inasmuch as these pressure drops and velocities are within tolerance, it is apparent that the Rod Velocity Limiter is not detrimentally affecting the drive system during normal operation due to excessive weight, mechanical interference, restricted flow paths, etc. Upon disassembly, the blade, limiter guide tube and other mechanical components were not observed to have sustained deformation or other adverse effects. Other components of the rod such as the coupling, handle and rollers are identical to the ones used in plants without velocity limiters.

The Rod Velocity Limiter performance satisfies the design basis and is in agreement with calculated predictions based on model tests. The average velocity for the hot and cold conditions is 2.86 ft/sec and 2.46 ft/sec, respectively. This is well within the design upper limit of 3.11 ft/sec.

4.6.4.3.4 Surveillance and Testing

Because the Rod Velocity Limiter is an integral part of the control rod assembly, it requires no specific retesting after installation. In addition to the close surveillance during the fabrication of the rod velocity limiter and control rod assembly manufacture, random control rod assemblies underwent shop testing which included rod drop tests. The control rod assemblies were tested during preoperational tests prior to reactor startup and after initial installation. These preoperational tests confirmed the operation of the individual control rod assemblies for normal operation and scram conditions.

4.6.5 Evaluation of Combined Performance

As discussed in Subsection 3.9.4, the control rod drive system contains 137 separate and independent control rod drive mechanisms and hydraulic control units. The only portions of the system common to these individual units are the hydraulic supply subsystem, which supplies and controls the pressure and flow requirements of the hydraulic control units, and certain scram subsystem lines, including the scram discharge volume headers and the scram discharge instrumentation volumes. The hydraulic control units are designed so that a failure in the hydraulic supply subsystem will not affect their ability to safely shut down the reactor. As a result of the Brown's Ferry Unit 3 incomplete scram incident, safety features were added to preclude potential single failure mechanisms in the scram subsystem lines, in conformance with

Nuclear Regulatory Commission I.E. Bulletin 80-17 and subsequent supplements. Both the North (69 CRD's) and the South (68 CRD's) banks were provided with separate scram discharge volumes (SDV's) and scram discharge instrumentation volumes (see Figure 4.6-3). Diverse and redundant scram initiation instrumentation was added to the scram discharge instrumentation volumes to ensure the SDV's will be sufficiently free of water to properly receive the water discharged from the control rod drive mechanisms during a scram. Redundant valves in series were provided on the SDV vent and drain lines to ensure the isolation of the reactor coolant contained in the SDV's during a scram. Refer to Subsection 3.9.4 for a more complete description of these features.

A Standby Liquid Control System (Liquid Poison System) is provided to shut down the reactor and maintain it subcritical if the control rods are unable to do so. The Control Rod Drive System and Standby Liquid Control System are designed to preclude potential common mode failures, as discussed in Subsections 3.9.4 and 9.3.5, respectively.