CHAPTER 4

REACTOR

4.1 Summary Description

This chapter describes the mechanical components of the reactor and reactor core, including the fuel rods and fuel assemblies, the nuclear design, and the thermal-hydraulic design.

The reactor contains a matrix of fuel rods assembled into mechanically identical fuel assemblies along with control and structural elements. The assemblies, containing various fuel enrichments, are configured into the core arrangement located and supported by the reactor internals. The reactor internals also direct the flow of the coolant past the fuel rods. The coolant and moderator are light water at a normal operating pressure of 2250 psia. The fuel, internals, and coolant are contained within a heavy walled reactor pressure vessel. An AP600 fuel assembly consists of 264 fuel rods in a 17x17 square array. The center position in the fuel assembly has a guide thimble that is reserved for in-core instrumentation. The remaining 24 positions in the fuel assembly have guide thimbles. The guide thimbles are joined to the top and bottom nozzles of the fuel assembly and provide the supporting structure for the fuel grids.

The fuel grids consist of an egg-crate arrangement of interlocked straps that maintain lateral spacing between the rods. The grid straps have spring fingers and dimples that grip and support the fuel rods. The middle grids also have coolant mixing vanes. The flow mixer grid straps contain support dimples and coolant mixing vanes only. The top and bottom grids do not contain mixing vanes.

The AP600 fuel assemblies are similar to 17x17 VANTAGE 5 HYBRID (VANTAGE 5H) fuel assemblies. The VANTAGE 5H fuel assembly design evolved from the previous VANTAGE 5 fuel assembly, optimized fuel assembly, and standard fuel assembly designs. It is based on the substantial design and operating experience with these designs. The VANTAGE 5H design is described and evaluated in WCAP-10445-NP-A (Reference 1). Design features from each of these previous designs are included as VANTAGE 5H and AP600 fuel assembly design features. The most significant difference between the VANTAGE 5H design and the VANTAGE 5 design is the use of the 0.374-inch outside diameter standard fuel rod.

The AP600 fuel assembly design includes: low pressure drop intermediate grids, four intermediate flow mixing (IFM) grids, reconstitutable top nozzle (RTN), and extended burnup capability. The bottom nozzle is a debris filter bottom nozzle (DFBN) that minimizes the potential for fuel damage due to debris in the reactor coolant.

The AP600 fuel assembly design supports the use of one additional (for a total of five) intermediate flow mixing grid as an option.

The fuel rods consist of enriched uranium, in the form of cylindrical pellets of uranium dioxide, contained in Zircaloy-4 tubing or ZIRLO™ (Reference 2) tubing. The tubing is plugged and seal welded at the ends to encapsulate the fuel. An axial blanket comprised of fuel pellets with reduced enrichment may be placed at each end of the enriched fuel pellet stack to reduce the neutron leakage and to improve fuel utilization.

A second type of fuel rod may be used to varying degrees within some fuel assemblies. These rods use an integral fuel burnable absorber (IFBA) containing a thin boride coating on the surface of the fuel pellets. The boride-coated fuel pellets provide a burnable absorber integral to the fuel.

Fuel rods are pressurized internally with helium during fabrication to reduce clad creep down during operation and, thereby, to increase fatigue life. The fuel rods in the AP600 fuel assemblies contain additional gas space, compared to VANTAGE 5H and previous designs, below the fuel pellets to allow for increase fission gas production due to extended fuel burnups.

Depending on the position of the assembly in the core, the guide thimbles are used for rod cluster control assemblies (RCCAs), gray rod cluster assemblies (GRCAs), neutron source assemblies, or non-integral discrete burnable absorber (BA) assemblies.

For the initial core design, discrete wet annular burnable absorbers (WABA) and integral fuel burnable absorbers (IFBA) are used. Discrete absorber designs, integral fuel burnable absorber designs, or both may be used in subsequent reloads.

The bottom nozzle is a box-like structure that serves as the lower structural element of the fuel assembly and directs the coolant flow distribution to the assembly. The size of flow passages through the bottom nozzle limits the size of debris that can enter the fuel assembly. The top nozzle assembly serves as the upper structural element of the fuel assembly and provides a partial protective housing for the rod cluster control assembly or other components.

The rod cluster control assemblies consist of 24 absorber rods fastened at the top end to a common hub, or spider assembly. Each absorber rod consists of an alloy of silver-indium-cadmium, which is clad in stainless steel. The rod cluster control assemblies are used to control relatively rapid changes in reactivity and to control the axial power distribution.

The gray rod cluster assemblies consist of 24 rodlets fastened at the top end to a common hub or spider. Geometrically, the gray rod cluster assembly is the same as a rod cluster control assembly except that 20 of the 24 rodlets are fabricated of stainless steel, while the remaining 4 are silver-indium-cadmium clad with stainless steel.

The gray rod cluster assemblies are used in load follow maneuvering. The assemblies provide a mechanical shim reactivity mechanism to minimize the need for changes to the concentration of soluble boron.

The reactor core is cooled and moderated by light water at a pressure of 2250 psia. Soluble boron in the moderator/coolant serves as a neutron absorber. The concentration of boron is varied to control reactivity changes that occur relatively slowly, including the effects of fuel burnup. Burnable absorbers are also employed in the initial cycle to limit the amount of soluble boron required and, thereby, to maintain the desired negative reactivity coefficients.

The nuclear design analyses establish the core locations for control rods and burnable absorbers. The analyses define design parameters, such as fuel enrichments and boron concentration in the coolant.

The nuclear design establishes that the reactor core and the reactor control system satisfy design criteria, even if the rod cluster control assembly of the highest reactivity worth is in the fully withdrawn position.

The core has inherent stability against diametral and azimuthal power oscillations. Axial power oscillations, which may be induced by load changes, and resultant transient xenon may be suppressed by the use of the rod cluster control assemblies.

The control rod drive mechanisms used to withdraw and insert the rod cluster control assemblies and the gray rod cluster assemblies are described in subsection 3.9.4.

The thermal-hydraulic design analyses establish that adequate heat transfer is provided between the fuel clad and the reactor coolant. The thermal design takes into account local variations in dimensions, power generation, flow distribution, and mixing. The mixing vanes incorporated in the fuel assembly spacer grid design and the fuel assembly intermediate flow mixers induce additional flow mixing between the various flow channels within a fuel assembly, as well as between adjacent assemblies.

The reactor internals direct the flow of coolant to and from the fuel assemblies and are described in subsection 3.9.5.

The performance of the core is monitored by fixed neutron detectors outside the core, fixed neutron detectors within the core, and thermocouples at the outlet of selected fuel assemblies. The ex-core nuclear instrumentation provides input to automatic control functions.

Table 4.1-1 presents a summary of the principal nuclear, thermal-hydraulic, and mechanical design parameters of the AP600 fuel. A comparison is provided to the fuel design used in a licensed Westinghouse-designed plant using VANTAGE 5 fuel. A four-loop plant is used for the comparison, since no domestic, Westinghouse-designed two-loop plants use 17x17 VANTAGE 5H fuel.

Table 4.1-2 tabulates the analytical techniques employed in the core design. The design basis must be met using these analytical techniques. Enhancements may be made to these techniques provided that the changes are bounded by NRC-approved methods, models, or criteria. In addition, application of the process described in WCAP-12488-A, (Reference 3) allows the Combined License holder to make fuel mechanical changes. Table 4.1-3 tabulates the mechanical loading conditions considered for the core internals and components. Specific or limiting loads considered for design purposes of the various components are listed as follows: fuel assemblies in subsection 4.2.1.5; neutron absorber rods, gray rods, burnable absorber rods, and neutron source rods, in subsection 4.2.1.6. The dynamic analyses, input forcing functions, and response loadings for the control rod drive system and reactor vessel internals are presented in subsections 3.9.4 and 3.9.5.

4.1.1 Principal Design Requirements

The fuel and rod control rod mechanism are designed so the performance and safety criteria described in Chapter 4 and Chapter 15 are met. [*The mechanical design and physical arrangement of the reactor components, together with the corrective actions of the reactor control, protection, and emergency cooling systems (when applicable) are designed to achieve these criteria, referred to as Principal Design Requirements:*

- *Fuel damage, defined as penetration of the fuel clad, is predicted not to occur during normal operation and anticipated operational transients.*
- *Materials used in the fuel assembly and in-core control components are selected to be compatible in a pressurized water reactor environment.*
- *For normal operation and anticipated transient conditions, the minimum DNBR calculated using the WRB-2 correlation is greater than or equal to 1.17.*
- *The calculated center-line temperature of the fuel, at the overpower limit, is less than or equal to 4700°F.*
- *The maximum fuel rod cladding temperature following a loss-of-coolant accident is calculated to be less than 2200°F.*
- *For normal operation and anticipated transient conditions, the calculated core average linear power, including densification effects, is less than or equal to 4.11 kw/ft for the initial fuel cycle.*
- *For normal operation and anticipated transient conditions, the calculated total heat flux hot channel factor, F_Q, is less than or equal to 2.60 for the initial fuel cycle.*
- *Calculated rod worths provide sufficient reactivity to account for the power defect from full power to zero power and provide the required shutdown margin, with allowance for the worst stuck rod.*
- *Calculations of the accidental withdrawal of two control banks using the maximum reactivity change rate predict that the peak linear heat rate and DNBR limits are met.*

*NRC Staff approval is required prior to implementing a change in this material; see DCD Introduction Section 3.5.

- *The maximum rod control cluster assembly and gray rod speed (or travel rate) is 45 inches per minute.*
- *The control rod drive mechanisms are hydrotested after manufacture at a minimum of 150 percent of system design pressure.*
- *For the initial fuel cycle, the fuel rod temperature coefficient is calculated to be negative for power operating conditions.*
- *For the initial fuel cycle, the moderator temperature coefficient is calculated to be negative for power operating conditions.*]*

4.1.2 Combined License Information

This section contains no requirement for additional information to be provided in support of Combined License.

4.1.3 References

1. Davidson, S. L., (Ed.), "VANTAGE 5H Fuel Assembly," Addendum 2-A, WCAP-10444-A (Proprietary) and WCAP-10445-NP-A (Non-Proprietary), February 1989.

2. Davidson, S. L., and Nuhfer, D. L., (Ed.), "VANTAGE+ Fuel Assembly Reference Core Report," WCAP-12610-P-A (Proprietary) and WCAP-14342-A (Non-Proprietary), April 1995.

[*3. Davidson, S. L. (Ed.), "Fuel Criteria Evaluation Process," WCAP-12488-A (Proprietary) and WCAP-14204-A (Non-Proprietary), October 1994.*]*

*NRC Staff approval is required prior to implementing a change in this material; see DCD Introduction Section 3.5.

Table 4.1-1 (Sheet 1 of 3)

REACTOR DESIGN COMPARISON TABLE

Thermal and Hydraulic Design Parameters	AP600	Typical Four-Loop[a]
Reactor core heat output (MWt)	1933	3411
Reactor core heat output (10^6 Btu/hr)	6596	11,639
Heat generated in fuel (%)	97.4	97.4
System pressure, nominal (psia)	2250	2250
System pressure, minimum steady-state (psia)	2200	2200
Minimum departure from nuclear boiling (DNBR) for design transients		
Typical flow channel	>1.23	>1.24
Thimble (cold wall) flow channel	>1.22	>1.23
Departure from nucleate boiling (DNB) correlation[b]	WRB-2	WRB-2
Coolant Flow[c]		
Total vessel thermal design flow rate (10^6 lbm/hr)[d]	72.9	142.1
Effective flow rate for heat transfer (10^6 lbm/hr)	66.3	130.2
Effective flow area for heat transfer (ft^2)	38.5	54.13
Average velocity along fuel rods (ft/s)	10.6	15.1
Average mass velocity, (10^6 lbm/hr-ft^2)	1.72	2.41
Coolant Temperature[e]		
Nominal inlet (°F)	532.8	558.7
Average rise in vessel (°F)	69.6	59.4
Average rise in core (°F)	75.8	64.2
Average in core (°F)	572.6	592.6
Average in vessel (°F)	567.6	588.4
Heat Transfer		
Active heat transfer surface area (ft^2)	44,884	57,505
Avg. heat flux (BTU/hr-ft^2)	143,000	197,200
Maximum heat flux for normal operation (BTU/hr-ft^2)[f]	372,226	493,000
Average linear power (kW/ft)[g]	4.11	5.45
Peak linear power for normal operation (kW/ft)[f][g]	10.7	13.6
Peak linear power (kW/ft)[f][h] (Resulting from overpower transients/operator errors, assuming a maximum overpower of 118%)	22.5	22.5

Table 4.1-1 (Sheet 2 of 3)

REACTOR DESIGN COMPARISON TABLE

Thermal and Hydraulic Design Parameters	AP600	Typical Four-Loop[a]
Heat flux hot channel factor (F_Q)	2.60	2.50
Peak fuel center line temperature (°F) (For prevention of center-line melt)	4700	4700
Fuel assembly design	17x17	17x17
Number of fuel assemblies	145	193
Uranium dioxide rods per assembly	264	264
Rod pitch (in.)	0.496	0.496
Overall dimensions (in.)	8.426 x 8.426	8.426 x 8.426
Fuel weight, as uranium dioxide (lb)	167,360	204,231
Zircaloy clad weight (lb)	35,555	45,914
Number of grids per assembly		
Top and bottom - (Ni-Cr-Fe Alloy 718)	2[i]	2
Intermediate - (Zircaloy-4)	7 or 7 ZIRLO™	6
Intermediate flow mixing - (Zircaloy-4)	4 or 4 ZIRLO™	3
Loading technique, first cycle	3 region nonuniform	3 region nonuniform
Fuel Rods		
Number	38,280	50,952
Outside diameter (in.)	0.374	0.360
Diametral gap (non-IFBA) (in.)	0.0065	0.0062
Clad thickness (in.)	0.0225	0.0225
Clad material	Zircaloy-4 or ZIRLO™	Zircaloy-4
Fuel Pellets		
Material	UO_2 sintered	UO_2 sintered
Density (% of theoretical)	95	95
Diameter (in.)	0.3225	0.3088
Length (in.)	0.387	0.370

Table 4.1-1 (Sheet 3 of 3)

REACTOR DESIGN COMPARISON TABLE

Rod Cluster Control Assemblies	**AP600**	**Typical Four-Loop**
Neutron absorber		
RCCA	24 Ag-In-Cd rodlets	Ag-In-Cd
GRCA	20 304 SS rodlets 4 Ag-In-Cd rodlets	N/A
Cladding material	Type 304 SS, cold-worked	Type 304 SS, cold-worked
Clad thickness, (Ag-In-Cd)	0.0185	0.0185
Number of clusters	45 RCCAs	53 RCCAs
	16 GRCAS	
Core Structure		
Core barrel, ID/OD (in.)	133.75/137.75	148.0/152.5
Thermal shield	None	Neutron Panel
Baffle thickness (in.)	Radial reflector	0.88
Structure Characteristics		
Core diameter, equivalent (in.)	115.0	132.7
Core height, cold, active fuel (in.)	144.0	144.0
Fuel Enrichment First Cycle (Weight Percent)		
Region 1	1.90	2.10
Region 2	2.80	2.60
Region 3	3.70	3.10

a. 17x17 VANTAGE 5 core
b. See subsection 4.4.2.2.1 for the use of the W-3 correlation
c. Flow rates are based on 10 percent steam generator tube plugging for the AP600 design
d. Inlet temperature = 532.8°F
e. Coolant temperatures based on thermal design flow with 10% steam generator tube plugging
f. Based on F_Q of 2.60 for AP600
g. Based on densified active fuel length
h. See subsection 4.3.2.2.6
i. The top grid may be fabricated of either nickel-chromium-iron Alloy 718 or ZIRLO™

Table 4.1-2 (Sheet 1 of 2)

ANALYTICAL TECHNIQUES IN CORE DESIGN

Analysis	Technique	Computer Code	Subsection Referenced
Mechanical design of core internals loads, deflections, and stress analysis	Static and dynamic modeling	BLOWDOWN code, FORCE, finite element structural analysis code, and others	3.7.2.1 3.9.2 3.9.3
Fuel rod design Fuel performance characteristics (such as, temperature, internal pressure, and clad stress)	Semi-empirical thermal model of fuel rod with considerations such as fuel density changes, heat transfer, and fission gas release.	Westinghouse fuel rod design model	4.2.1.1 4.2.3.2 4.2.3.3 4.3.3.1 4.4.2.11
Nuclear design Cross-sections and group constants	Microscopic data; macroscopic constants for homogenized core regions	Modified ENDF/B library with PHOENIX-P or LEOPARD/ CINDER (ARK)type	4.3.3.2
X-Y and X-Y-Z power distributions, fuel depletion, critical boron concentrations, X-Y and X-Y-Z xenon distributions, reactivity coefficients	2-group diffusion theory, 2-group nodal theory	ANC (2-D or 3-D) or TURTLE (2-D) (TORTISE)	4.3.3.3
Axial power distributions, control rod worths, and axial xenon distribution	1-D, 2-group diffusion theory	APOLLO	4.3.3.3
Fuel rod power	Integral transport theory	LASER	4.3.3.1
Effective resonance temperature	Monte Carlo weighing function	REPAD	4.3.3.1
Criticality of reactor and fuel assemblies	3-D, Monte Carlo theory	AMPX system of codes, KENO-Va	4.3.2.6

Table 4.1-2 (Sheet 2 of 2)

ANALYTICAL TECHNIQUES IN CORE DESIGN

Analysis	Technique	Computer Code	Subsection Referenced
Vessel irradiation	Multigroup spatial dependent transport theory	DOT	4.3.2.8
Thermal-hydraulic design steady state	Subchannel analysis of local fluid conditions in rod bundles, including inertial and cross-flow resistance terms; solution progresses from core-wide to hot assembly to hot channel.	THINC-IV WESTAR	4.4.4.5.2
Transient departure from nucleate boiling	Subchannel analysis of local fluid conditions in rod bundles during transients by including accumulation terms in conservation equations; solution progresses from core-wide to hot assembly to hot channel.	THINC-I (THINC-III)	4.4.4.5.4

Table 4.1-3

DESIGN LOADING CONDITIONS FOR REACTOR CORE COMPONENTS

- Fuel assembly weight and core component weights (burnable absorbers, sources, RCCA, GRCA)
- Fuel assembly spring forces and core component spring forces
- Internals weight
- Control rod trip (equivalent static load)
- Differential pressure
- Spring preloads
- Coolant flow forces (static)
- Temperature gradients
- Thermal expansion
- Interference between components
- Vibration (mechanically or hydraulically induced)
- Operational transients listed in Table 3.9.1-1
- Pump overspeed
- Seismic loads (safe shutdown earthquake)
- Blowdown forces (due to pipe rupture)