4.0 Reactor

4.1 Summary Description

The U.S. EPR is an evolutionary pressurized water reactor (PWR) with a rated thermal power of 4590 MW that is cooled and moderated using light water at a normal operating pressure of 2250 psia. A summary of the reactor design and performance characteristics is presented in Table 4.1-1.

The fuel rods consist of pellets contained in a seamless M5™ zirconium alloy tube, with M5™ end plugs welded at each end. The fuel rods are pressurized with helium during fabrication. The fuel pellets contain uranium dioxide (UO_2) or uranium dioxide + gadolinia (UO_2+Gd_2O_3) with enrichments as high as 5 weight percent (wt%) U-235.

The rods (pins) are combined together using spacer grids and end grids in a 17x17 array to form a fuel assembly. The assembly utilizes 10 spacer grids that, with the 24 guide tubes and a top and a bottom nozzle, provide the structural skeleton for supporting the 265 fuel rods. The fuel rods are supported by the top and bottom high mechanical performance (HMP) end spacer grids and eight high thermal performance (HTP) intermediate spacer grids. Major features of the fuel assembly include the following items:

- 17x17 array of pins (24 guide tube locations and 265 fuel rods).
- Top-loaded in-core detectors that occupy selected guide tube locations.
- M5 MONOBLOC™ guide tubes.
- Alloy M5™ fuel rods.
- [*62,000 MWD/MTU fuel rod burnup limit.*]*
- Nominal fuel length is 13.8 ft.
- Gadolinia rods up to 8 weight percent Gd_2O_3 (possible range of 0 to 28 rods per assembly).

Details of fuel pellets, rods, and assemblies are provided in Section 4.2, Section 4.3, and U.S. EPR Fuel Assembly Mechanical Design Topical Report (Reference 1).

The core is formed by arranging 241 fuel assemblies into a pattern that approximates a right circular cylinder. The core periphery is surrounded by a heavy reflector, which is a large steel structure 4 to 8 in thick that reduces fast neutron leakage and flattens the core power distribution. Details of the heavy reflector are presented in Section 4.3. The heavy reflector also reduces the fast flux on the reactor pressure vessel. Details of reactor pressure vessel internals are provided in Section 3.9.5.

Mechanical loading conditions considered for the core internals and components are listed below:

- Fuel assembly weight.
- Fuel assembly spring forces.
- Internals weight.
- Control rod trip (i.e., equivalent static load).
- Differential pressure.
- Spring preloads and operational loads.
- Static coolant flow forces.
- Temperature gradients.
- Differences in thermal expansion (because of temperature differences or expansion of different materials).
- Interference between components.
- Shipping and handling.
- Mechanically or hydraulically induced vibration.
- One or more loops out of service.
- Operational transients.
- Pump overspeed.
- Operating basis and safe shutdown earthquake seismic loads.
- LOCA blowdown forces (i.e., cold or hot leg break).

The initial core loading consists of seven different fuel assembly neutronic designs with up to three rod types. Each fuel assembly neutronic design for the initial core uses a uniform distribution of uranium and gadolinia bearing fuel rods. The core is loaded by placing the lowest enriched fuel on the core periphery to enhance neutron economy, while distributing the remainder of the fuel in the core interior to establish a favorable radial power distribution.

During reactor operation, U-235 depletion of the fuel assemblies and buildup of fission products occur. To compensate for these effects, the core design for each cycle must have an initial fuel loading with excess reactivity. To control the excess reactivity in the initial fuel loading, two reactivity control methods are used: rod cluster control assemblies (RCCA) and soluble neutron poison in the reactor coolant system (RCS).

Each RCCA contains 24 rods with annular absorbers fastened to a spider assembly. The absorber rods are constructed of stainless steel tubing containing annular neutron absorbing material (80 wt% Ag, 15 wt% In, and 5 wt% Cd). Details of RCCAs and control components are provided in Section 4.2.

RCCAs are located in 89 of the 241 fuel assemblies. For assemblies containing an RCCA, all 24 guide tubes are occupied by control rods; therefore, those assemblies do not have in-core instrumentation. Movement of the RCCAs is provided by the control rod drive system (CRDS). Details of the CRDS are presented in Section 3.9.4, and reactivity control systems are discussed in Section 4.6.

Soluble boron (B-10 enriched) is used as a neutron poison in the RCS. The concentration in the coolant is adjusted as the reactivity of the core changes. The boron concentration in the coolant is varied to control the slow reactivity changes needed during power operation (e.g., xenon poisoning, burn-up effects). The boron concentration also compensates for the large reactivity changes associated with temperature variations during cooldown or heatup phases. The chemical and volume control system (CVCS), described in Section 3.9.4, adds or removes the soluble boron from the RCS as needed during normal operations. The extra borating system (EBS) also uses soluble boron for reactivity control and is described in Section 6.8.

As the soluble boron concentration in the RCS is increased, the moderator temperature coefficient (MTC) becomes less negative. The use of a soluble neutron absorber alone could result in a positive moderator coefficient at beginning-of-life (BOL) in the initial core. Therefore, integral burnable absorbers in the fuel are used in the first core to reduce the soluble boron concentration so that the MTC is negative for power operating conditions. The integral gadolinia burnable absorbers are also strategically located to provide a favorable radial power distribution.

Other reactor design details are presented in Sections 4.3, 4.4, and 4.5. Computer codes used for core design are presented in Table 4.1-2

Control of the core is performed through the use of instrumentation to predict and measure the nuclear power level and distribution. Core instrumentation consists of ex-core and in-core instruments:

- Ex-Core Instrumentation: During power operation, the power level is measured principally by a four-fold redundant primary heat balance that relies on temperature measurements in the cold and hot legs of the RCS loops. This primary heat balance is used with ex-core neutron flux measurements (power range) that have a short response time to provide an efficient system for fast and slow core power change detection. The core is also monitored and protected when operated at very low power levels or in subcritical conditions.

- In-Core Instrumentation: In-core instrumentation is top-mounted and consists of an aeroball measurement system (AMS) as the movable reference core instrumentation, a fixed power density detector system (PDDS), and core outlet thermocouples (COTC).

The AMS is a simple and reliable method to assess power distribution. AMS is based on a pneumatic system that inserts steel balls containing vanadium into the 40 detector locations in the core. Upon demand, a nitrogen gas driving medium transports the Aeroball stacks to the core where they are irradiated. After a defined irradiation time, the Aeroballs leave the core and pass into the measuring table in the AMS room inside the containment. The activation is proportional to the power; therefore, a detailed axial and radial power distribution can be inferred. The advantage of AMS is the reduced time (minutes rather than hours) required to take a core flux map, compared to other moveable in-core detector systems.

The AMS probes are distributed over the core by 12 in-core instrumentation lances. Each in-core lance contains one self-powered neutron detector (SPND) finger and either three or four aeroball fingers, depending on the instrumentation lance yoke type. AMS probes are distributed in 40 radial locations, each divided into 36 axial segments, and produce accurate three-dimensional flux and power distribution maps. Additional details for AMS are provided in Section 4.4.6.

The fixed in-core instrumentation consists of SPNDs and COTCs. At twelve radial locations, six SPNDs are placed axially in a power density detector (PDD) finger to provide full core monitoring capability. Each yoke within the instrumentation lance system contains one PDD finger that is replaceable should a detector become defective. The number and distribution of SPNDs within the core allow the system to detect and assess local power density increases caused by flux and power redistributions that can occur under either steady-state or non-steady-state conditions. The advantage of SPNDs is their rapid response time and the three-dimensional representation they provide.

4.1.1 Principal Design Requirements

Fuel and reactor design requirements are presented in Chapter 4, and the design is analyzed in Chapter 15. The principal design requirements are:

- Fuel damage is not expected during normal operation or anticipated operational occurrences (AOO).

- [*For the initial fuel loading, the fuel Doppler temperature reactivity coefficient is negative.*

- *For the initial fuel loading, the MTC is negative for power operating conditions.*
- *Power oscillations that could result in conditions exceeding fuel design limits are not possible, or can be reliably and readily detected and suppressed.*]*
- Instrumentation and controls (I&C) are provided to monitor variables and systems that can affect the fission process over anticipated ranges for normal operation, AOOs, and postulated accident (PA) conditions, and maintain the variables and systems within prescribed operating ranges.
- Reactivity control systems automatically initiate so that fuel design limits are not exceeded as a result of AOOs. This requires automatic operation of safety-related systems and components under accident conditions.
- No single malfunction of the reactivity control systems (excluding rod ejection) causes violation of the fuel design limits.
- Two independent reactivity control systems of different design are provided.
- Reactivity control systems have a combined capability, in conjunction with poison addition by the safety injection system (SIS), of reliably controlling reactivity changes under PA conditions, with appropriate margin for stuck rods.
- Fuel damage during PAs will not be severe enough to prevent control rod insertion when it is required.
- The effects of postulated reactivity accidents neither result in damage to the reactor coolant pressure boundary greater than limited local yielding, nor cause sufficient damage to significantly impair core coolability.
- Core coolability will be maintained, even after PAs.
- The reactor can be brought to a safe state, and the core can be kept sub-critical with acceptable heat transfer following a PA with only a small fraction of fuel rods damaged.
- Reactor materials are selected to be compatible with operating conditions.

4.1.2 References

1. ANP-10285P, Revision 1, “U.S. EPR Fuel Assembly Mechanical Design Topical Report,” AREVA NP Inc., May 2013.
2. ANP-10263P-A, Revision 0, “Codes and Methods Applicability Report for the U.S. EPR,” AREVA NP Inc., August 2007.
3. BAW-2241P-A, Revision 2, “Fluence and Uncertainty Methodologies,” Worsham, J.R., et al., Lynchburg, Virginia, April 2006.
4. BAW-10231P-A, Revision 1, “COPERNIC Fuel Rod Design Computer Code,” Framatome ANP, January 2004.

5. BAW-10156-A, Revision 1, "LYNXT Core Transient Thermal-Hydraulic Program," B&W Fuel Company, August 1993.

6. BAW-10227P-A, Rev. 1, "Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel," June 2003.

Table 4.1-1—Summary of U.S. EPR Reactor Design and Performance Characteristics
Sheet 1 of 3

Design Parameter	Value
UO_2 Pellet Parameters	
Outside diameter[1]	0.3225 in
Length[1]	0.531 in
Density (% of theoretical)	96.0
Fissile enrichment (less enrichment tolerance)	≤4.95 wt% U-235
UO_2-Gd_2O_3 Pellet Parameters	
Outside diameter[1]	0.3225 in
Length[1]	0.531 in
Density (% of theoretical)	96.0
Fuel Rod Parameters	
Cladding material	M5™
Cladding outside diameter[1]	0.3740 in
Cladding inside diameter[1]	0.3291 in
Fuel column length[1]	165.354 in
Overall fuel rod length[1]	179.134 in
Fuel Assemblies in Core	
Number	241
Rod array	17x17
Rods per fuel assembly	265
Rod pitch[1]	0.496 in
Overall transverse dimensions[1]	8.426 x 8.426 in
Nominal fuel weight per assembly[1]	536.1 kg U
Number of grids per assembly	10
Peak pin exposure core design criteria for UO_2 rods	[*62.0 GWD/MTU*]*
Peak pin exposure core design criteria for Gd_2O_3 rods	[*55.0 GWD/MTU*]*
Max Gd rods per bundle	28
Gd_2O_3 concentration[1]	2, 4, 6, or 8 w/o Gd_2O_3
Control rods[1]	Absorber composed of annular slugs consisting of silver (80wt%), indium (15wt%), and cadmium (5wt%)

Table 4.1-1—Summary of U.S. EPR Reactor Design and Performance Characteristics
Sheet 2 of 3

Design Parameter	Value
RCCAs	
Number of rods per RCCA	24
Maximum RCCA bank withdrawal speed[1]	29.5 in/minute
Silver-indium-cadmium bar overall length[1]	166.929 in
Functional rod length[1]	177.972 in
Overall rod length[1]	178.331 in
RCCA total height[1]	185.726 in
Core Design Criteria	
Rated core thermal power	4590 MW_t
Number of loops	4
System pressure[1]	2250 psia
Thermal design flow[1]	478,768 gpm
Best estimate flow[1]	498,964 gpm
Mechanical design flow[1]	538,648 gpm
Nominal inlet temperature[1]	563.4°F
Average temperature rise in vessel[1]	60.6°F
Average temperature rise in core[1]	62.7°F
Average temperature in core[1]	596.8°F
Average temperature in vessel[1]	594°F
Average linear power density (includes gamma energy deposition)	5.22 kW/ft
Peak linear power for normal operating conditions with uncertainty (includes gamma energy deposition)	13.6 kW/ft
Peak linear power protection threshold	17.2 kW/ft
DNB limiting condition of operation	2.50
Core Description	
Equivalent diameter of active core[1]	148.3 in
Height-to-diameter ratio of active core[1]	1.115
Total cross section area of active core[1]	119.95 ft^2

Table 4.1-1—Summary of U.S. EPR Reactor Design and Performance Characteristics
Sheet 3 of 3

Design Parameter	Value
Reflector Thickness and Composition Used in Neutronic Design	
Top – water plus steel[1]	11.81 in
Bottom – water plus steel[1]	11.81 in
Side – water plus steel[1]	≈4 in minimum; ≈8 in maximum

Note:

1. Nominal value.

Table 4.1-2—Core Design Analytical Techniques
Sheet 1 of 2

Analysis	Technique	Computer Code
Calculation of microscopic burnup in burnable absorber rods (in particular for gadolinia-bearing fuel rods) and generation of the burnable absorber cross section data required by CASMO-3.	Multi-group, one dimensional transmission probability code used to calculate microscopic burnup in rods containing neutron absorber material that is initially homogeneously distributed.	MICBURN-3 (Reference 2)
Fuel assembly calculations and cross section data generation required by the reactor core simulator methodology (PRISM and NEMO-K).	Nodal expansion method to solve the two-group neutron diffusion theory representation of the reactor core.	CASMO-3 (Reference 2)
Reactor core calculations including radial and axial power and xenon distributions, fuel depletion, critical boron concentrations, reactivity coefficients, and control rod worths.	Three-dimensional core simulator code. The code uses a nodal expansion method to solve the two-group neutron diffusion theory representation of the reactor core.	PRISM (Reference 2)
Reactor core kinetics calculations during fast (rod ejection) as well as slower (rod drop) transients.	Three-dimensional, reactor kinetics code incorporating time-dependent solutions for neutronics, fuel temperature, and coolant properties into the steady-state NEMO code.	NEMO-K (Reference 2)
Vessel irradiation.	Two-dimensional discrete ordinates transport calculation that computes the flux distributions throughout the reactor.	DORT (Reference 3)
Fuel rod design.	General purpose fuel rod design and licensing code that simulates the various mechanisms at work in a fuel rod and calculates parameters (e.g., fuel centerline temperatures, fuel rod internal pressure, cladding strains).	COPERNIC (References 2 and 4)
Thermal and hydraulic design.	Fuel assembly/subchannel code for calculating steady-state and transient local coolant conditions/departure from nucleate boiling and fuel temperatures in rod arrays for a wide variety of conditions.	LYNXT (References 2 and 5)

Table 4.1-2—Core Design Analytical Techniques
Sheet 2 of 2

Analysis	Technique	Computer Code
Creep analysis of fuel rod cladding.	Code for evaluating the resistance of the fuel rod cladding to creep collapse. Inputs to the analysis include differential pressure, temperature gradients, and fast flux. The enveloping power histories from the COPERNIC thermal-hydraulic analysis are used to initialize the creep collapse code.	CROV (Reference 6)
Fuel assembly analysis.	General purpose finite element code used for fuel assembly component structural evaluations, and for guide tube and fuel rod buckling.	ANSYS and CASAC (Reference 1)

Next File