Status report 75 - Advanced Passive pressurized water reactor (AP-600)

Overview

Full name	Advanced Passive pressurized water reactor
Acronym	AP-600
Reactor type	Pressurized Water Reactor (PWR)
Coolant	Light Water
Moderator	Light water
Neutron spectrum	Thermal Neutrons
Thermal capacity	1940.00 MWth
Electrical capacity	600.00 MWe
Design status	Basic Design
Designers	Westinghouse
Last update	04-04-2011

Description

Introduction

The Westinghouse Advanced Passive PWR AP-600 is a 600 MWe pressurized water reactor (PWR) with advanced passive safety systems and extensive plant simplifications to enhance the construction, operation, and maintenance of the plant. The plant design utilizes proven technology which builds on approximately 40 years of operating PWR experience. PWRs represent 74 percent of all Light Water Reactors around the world, and the majority of these are based on Westinghouse PWR technology.

The AP-600 is designed to achieve a high safety and performance record. Safety systems maximize the use of natural driving forces such as pressurized gas, gravity flow and natural circulation flow. Safety systems do not use active components (such as pumps, fans or diesel generators) and are designed to function without safety-grade support systems (such as AC power, component cooling water, service water, HVAC). The number and complexity of operator actions required to control the safety systems are minimized; the approach is to eliminate a required operator action rather than to automate it. The net result is a design with significantly reduced complexity and improved operability.

The AP-600 standard design complies with all applicable U.S.NRC criteria. Extensive safety analysis has been completed and documented in the Standard Safety Analysis Report (SSAR) and Probabilistic Risk Analysis (PRA) submittals to the NRC. An extensive testing programme has been completed, and verifies that the plant's innovative features will perform as designed and analyzed. PRA results predict a very low core damage frequency which meets the goals established for advanced reactor designs and a low frequency of release due to improved containment isolation and cooling. The AP-600 design received Final Design Approval from the U.S.NRC in September 1998 and Design Certification in December 1999.

An important aspect of the AP-600 design philosophy focuses on plant operability and maintainability. These factors have been incorporated into the design process.

The AP-600 design includes features such as simplified system design to improve operability while reducing the number of components and associated maintenance requirements. In particular, simplified safety systems reduce surveillance requirements by enabling significantly simplified technical specifications.

Selection of proven components has been emphasized to ensure a high degree of reliability with a low maintenance requirement. Component standardization reduces spare parts, minimizes maintenance training requirements, and allows shorter maintenance durations. Built-in testing capability is provided for critical components.

Plant layout ensures adequate access for inspection and maintenance. Laydown space for staging of equipment and personnel, equipment removal paths, and space to accommodate remotely operated service equipment and mobile units have been considered as part of the plant design. Access platforms and lifting devices are provided at key locations, as are service provisions such as electrical power, demineralized water, breathing and service air, ventilation and lighting.

The AP-600 design also incorporates radiation exposure reduction principles to keep worker dose as low as reasonably achievable (ALARA). Exposure length, distance, shielding and source reduction are fundamental criteria that are incorporated into the design.

Various features have been incorporated in the design to minimize construction time and total cost by eliminating components and reducing bulk quantities and building volumes. Some of these features include the following:

- The flat, nuclear island common basemat design effectively minimizes construction cost and schedule.
- Utilization of the integrated protection system, the advanced control room, distributed logic cabinets, multiplexing, and fiber optics, significantly reduces the quantity of cables, cable trays, and conduits.
- The stacked arrangement of the Class 1E battery rooms, the dc switchgear rooms, the integrated protection system rooms, and the main control room. This stacked arrangement eliminates the need for the upper and lower cable spreading rooms that are required in the current generation of PWR plants.
- Application of the passive safety systems replaces and/or eliminates many of the conventional mechanical safety systems that are typically located in the Seismic Category I buildings in current PWR plants.

The AP-600 is designed with environmental consideration as a priority. The safety of the public, the power plant workers, and the impact to the environment have all been addressed as specific design goals, as follows:

- Operational releases have been minimized by design features;
- Aggressive goals for worker radiation exposure have been set and satisfied;
- Total radwaste volumes have been minimized;
- Other hazardous waste (non-radioactive) have been minimized.

The AP-600 Nuclear Power Plant was designed by Westinghouse under the sponsorship of the US Department of Energy (DOE) and the Electric Power Research Institute (EPRI). The design team included a number of US and foreign companies and organizations, such as Bechtel, Burns & Roe, Initec (Spain), UTE (Spain), and Ansaldo (Italy) as architect engineers, Avondale Industries (module design), CBI Services, Inc. (containment vessel design), M-K Ferguson Co. (constructability, schedule, and cost estimation), Southern Electric International (turbine island buildings and systems), ENEA Energy Research Center of Italy (tests of the automatic depressurization system), SIET, SPES Facility in Italy (full-pressure integral passive safety system tests), and Oregon State University (low-pressure integral passive safety system tests).

EPRI has, with a broad international participation of organizations from numerous countries, developed a Utility Requirements Document (URD) for ALWRs, taking into account the wealth of information related to nuclear power plant safety and operations that has been generated worldwide with commercial nuclear power. The purpose of the URD is to delineate utility desires for their next generation of nuclear plants, and to this end, it consists of a comprehensive set of design requirements for future plants.

Incorporation of the ALWR URD has given the AP-600 a well-defined design basis that is confirmed through thorough engineering analyses and testing and is in conformance with the URD. Some of the high-level design characteristics, and operational and safety goals of the plant are:

- Net electrical power of at least 600 MWe; and a thermal power of 1940 MWt;
- Rated performance is achieved with up to 10% of the steam generator tubes plugged and with a maximum hot leg temperature of 600°F (315.6°C);
- Core design is robust with at least a 15% operating margin on core power parameters;
- Short lead time (five years from owner's commitment to commercial operation) and construction schedule (3 years);
- No plant prototype is needed since proven power generating system components are used;
- Major safety systems are passive; they require no operator action for 72 hours after an accident, and maintain core and containment cooling for a protracted time without ac power;
- Predicted core damage frequency of 1.7E-07/yr is well below the 1E-05/yr utility requirement, and frequency of significant release of 3.0E-08/yr is well below the 1E-06/yr utility requirement;
- Standard design is applicable to anticipated US sites;
- Occupational radiation exposure expected to be below 0.7 man-Sv/yr (70 man-rem/yr);
- Core is designed for a 24-month fuel cycle assuming an 87% capacity factor; capable of an 18-month cycle;
- Refuelling outages can be conducted in 17 days or less;
- Plant design life of 60 years without replacement of the reactor vessel.
- Overall plant availability greater than 93%, including forced and planned outages; the goal for unplanned reactor trips is less than one per year.

2.1. Primary circuit and its main characteristics

The primary circuit of the AP-600 reactor has retained most of the general design features of current designs, but some evolutionary features that enhance the safety and maintainability of the system have been adopted. The coolant loops (Figure 2.1-1) consist of two hot leg and four cold leg pipes, and the reactor coolant pumps are installed directly onto the steam generators, eliminating the primary piping between pumps and steam generator; these features significantly contribute to safety and maintainability. Also, a simplified support structure for the primary systems reduces in-service inspections and improves accessibility for maintenance.

The reactor coolant system pressure boundary provides a barrier against the release of radioactivity generated within the reactor and is designed to provide a high degree of integrity throughout operation of the plant.

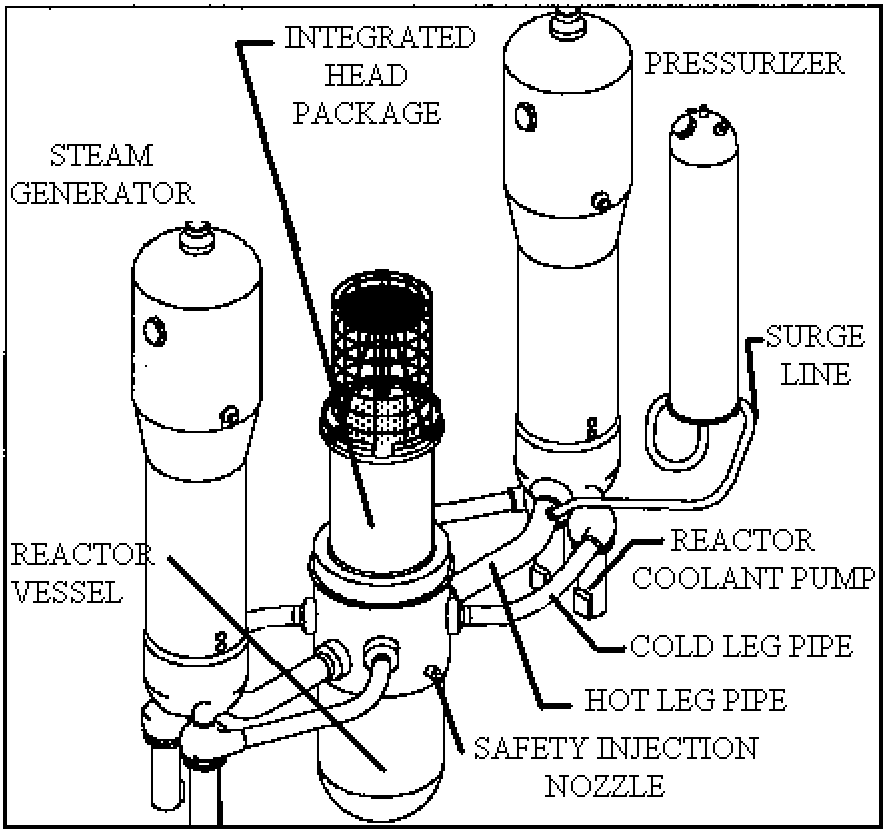

FIG. 2.1-1. Isometric view of NSSS

2.2. Reactor core and fuel design

The core, reactor vessel, and reactor internals of the AP-600 are similar to those of earlier Westinghouse PWRs. Several important features based on existing technology measurably enhance performance characteristics as compared with earlier Westinghouse PWRs. The reactor core is a low-power density design that uses a 12 feet (366 cm), 17´17 fuel assembly. This configuration results in core power density and average linear power density reductions of about 25 percent over earlier Westinghouse plants of the same power rating by making the core larger than earlier Westinghouse 600 MWe designs, with the number of fuel assemblies increased from 121 to 145. This results in lower fuel enrichments, less reliance on burnable absorbers, and longer fuel cycles.

Another feature that contributes to lowering fuel cycle cost and extending reactor life is the use of a stainless steel radial neutron reflector. This reflector reduces neutron leakage, thereby improving core neutron utilization and allowing for reduced fuel enrichment. The radial reflector, see Reactor Internals, has the added benefit of reducing the damaging neutron fluence on the reactor vessel, an important factor in achieving the 60-year design life objective.

The combination of the radial reflector and the low-power density core results in a fuel-cycle cost savings of 15 to 20 percent compared with a standard Westinghouse design of the same power rating.

Another core design feature is the use of reduced-worth control rods (termed "gray" rods) to achieve daily load follow without requiring daily changes in the soluble boron concentration. The use of gray rods, in conjunction with an automated load follow control strategy, eliminates the need for processing thousands of gallons of water per day to change the soluble boron concentration sufficiently to achieve a daily load follow schedule. As a result, systems are simplified through the elimination of the evaporator, and other boron processing equipment (such as pumps, valves, and piping). With the exception of the neutron absorber materials used, the design of the gray rod assembly is identical to that of a normal control rod assembly.

The core consists of three radial regions that have different enrichments; the enrichment of the fuel ranges from 2 to 4%. The temperature coefficient of reactivity of the core is highly negative. The core is designed for a fuel cycle of 18 to 24 months with discharge burnups as high as 55,000 MWd/t.

2.3. Fuel handling and transfer systems

Refuelling of the reactor is performed in the same way as for current plants. After removing the vessel head, fuel handling takes place from above, using the refuelling machine. During refuelling, one third of the core inventory is replaced.

New fuel storage - New fuel is stored in a high density rack that includes integral neutron absorbing material to maintain the required degree of sub-criticality. The rack is designed to store fuel of the maximum design basis enrichment. The rack in the new fuel pit consists of an array of cells interconnected to each other at several elevations and to supporting grid structures at the top and bottom elevations. The new fuel rack includes storage locations for 56 fuel assemblies. Minimum separation between adjacent fuel assemblies is sufficient to maintain a subcritical array even in the event the building is flooded with un-borated water or fire extinguishing aerosols or during any design basis event.

Spent fuel storage - Spent fuel is stored in high density racks which include integral neutron absorbing material to maintain the required degree of subcriticality. The racks are designed to store fuel of the maximum design basis enrichment. The spent fuel storage racks include storage locations for 619 fuel assemblies. The modified 10´7 rack module additionally contains integral storage locations for five defective fuel storage containers. The design of the rack is such that a fuel assembly can not be inserted into a location other than a location designed to receive an assembly.

2.4. Primary components

Reactor pressure vessel -The reactor vessel (Figure 2.4-1) is the high pressure containment boundary used to support and enclose the reactor core. The vessel is cylindrical, with a hemispherical bottom head and removable flanged hemispherical upper head.

The reactor vessel is approximately 38 feet (11.7 m) long and has an inner diameter at the core region of 157 inches (3.988 m). The total weight of the vessel is approximately 400 tons. Surfaces which can become wetted during operation and refuelling are clad with stainless steel welded overlay. The AP-600 reactor vessel is designed to withstand the design environment of 2500 psia (17.2 MPa) and 650°F (343 ^{o}C) for 60 years. The major factor affecting vessel life is radiation degradation of the lower shell.
As a safety enhancement, there are no penetrations below the top of the core. This eliminates the possibility of a loss of coolant accident by leakage from the reactor vessel which could allow the core to be uncovered. The core is positioned as low as possible in the vessel to limit reflood time in accident situations.

Reactor internals - The reactor internals, the core support structures, the core shroud, the downcomer and flow guiding structure arrangement, and the above-core equipment and structures, are very similar to those in current plants.

The reactor internals consist of two major assemblies - the lower internals and the upper internals. The reactor internals provide the protection, alignment and support for the core, control rods, and gray rods to provide safe and reliable reactor operation.

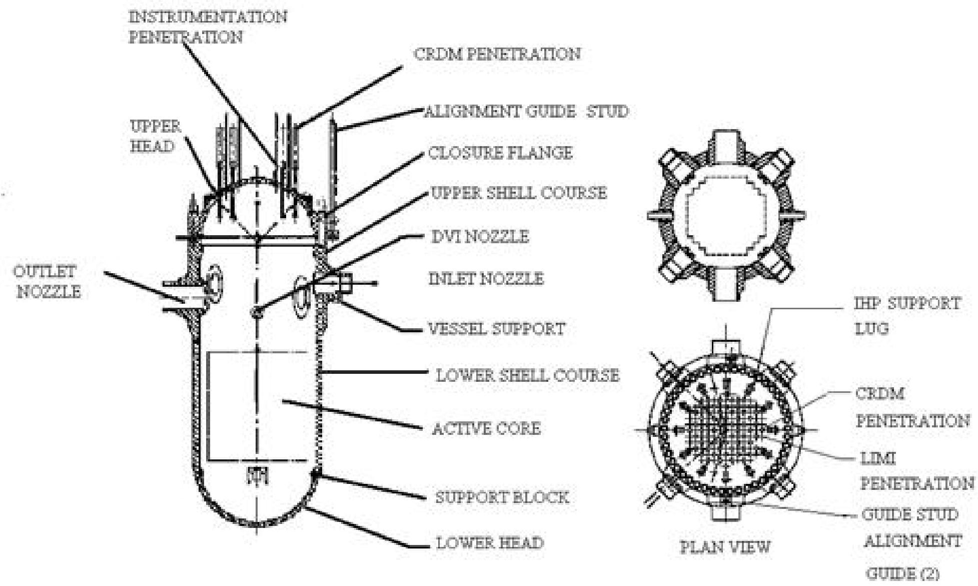

FIG. 2.4-1. Reactor pressure vessel

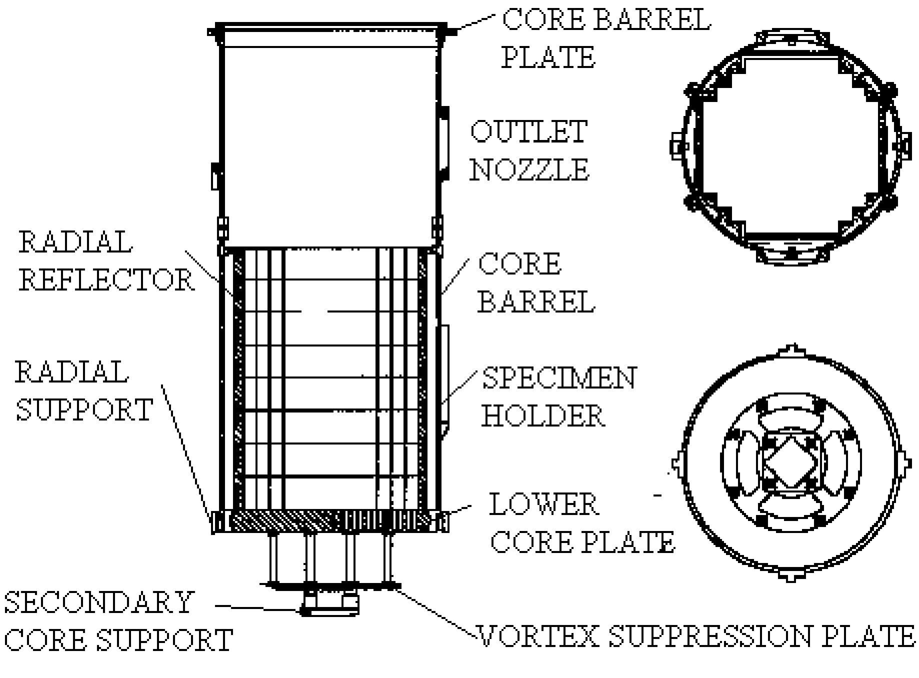

FIG. 2.4-2. Lower core support assembly

The major containment and support member of the reactor internals is the lower core support assembly. This assembly (Figure 2.4-2) consists of the core barrel, lower core support plate, secondary core support, vortex suppression plate, radial reflectors, radial supports, and related attachment hardware. The major material for this structure is 300 series austenitic stainless steel.

A key feature of the AP-600 plant is the radial reflector. The radial reflector is located inside the core barrel and above the lower core support to form the radial periphery of the core. The reflector provides a transition from the round core barrel to the square fuel assemblies. The reflector assemblies have longitudinal holes that allow cooling water to flow through, while still providing sufficient material to perform the neutron reflection and radiation shielding functions. This results in lower neutron loss from the core and decreased fluence on the reactor pressure vessel. Each reflector assembly (ring) is sized in height so that adjoining sections meet at a fuel grid elevation.

Steam generators - The model Delta-75 steam generator (Figure 2.4-3) is a vertical shell and U-tube evaporator with integral moisture separating equipment based on standard Westinghouse Model F technology. There are currently 84 Model F steam generators operating in 25 nuclear plants with a wide range of operating environments. To date, they have accumulated over 1200 steam-generator-years of operation. The 25 Model F-type replacement steam generators have an impressive record with less than one tube plugged per steam generator for every four years of operation. The basic Delta-75 steam generator design and features have been proven in tests and in previous steam generators including replacement steam generator designs.

Design enhancements include nickel-chromium-iron Alloy 690 thermally treated tubes on a triangular pitch, broached tube support plates, improved antivibration bars, single-tier separators, enhanced maintenance features, and a primary-side channel head design that allows for easy access and maintenance by robotic tooling.

All tubes in the steam generator are accessible for sleeving, if necessary. The design enhancements are based on proven technology. The steam generators operate on all volatile treatment secondary side water chemistry.

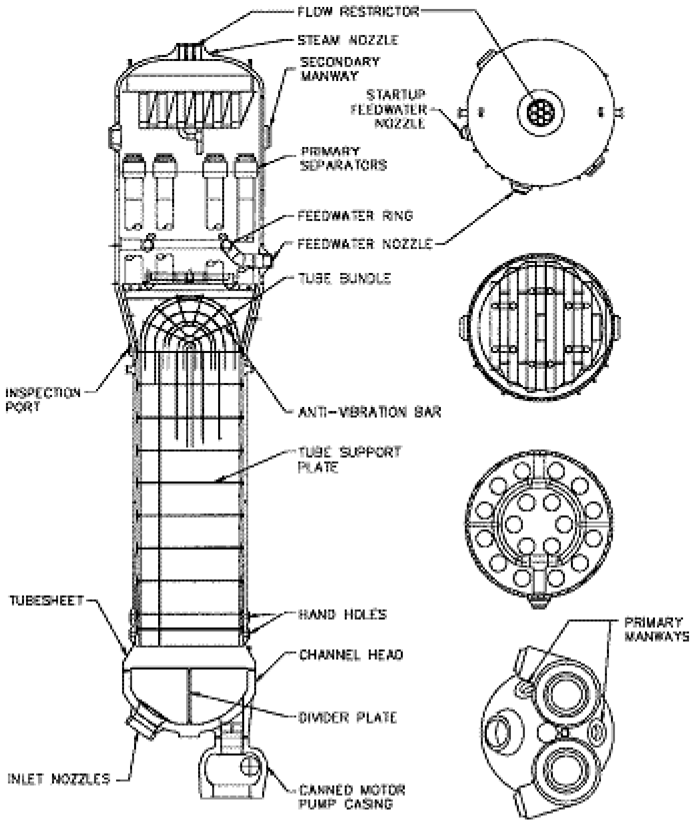

FIG. 2.4-3. Steam generator

Pressurizer - The pressurizer is based on proven Westinghouse technology and years of operating experience. The pressurizer is about 30 percent larger than that normally used in a Westinghouse plant of comparable power rating. The larger pressurizer increases transient operation margins, resulting in a more reliable plant with fewer reactor trips, and avoiding challenges to the plant and operator during transients. It also eliminates the need for fast-acting power-operated relief valves, which are a possible source of RCS leakage and maintenance.

Reactor coolant pumps - The reactor coolant pumps (Figure 2.4-4) are high-inertia, high-reliability, low-maintenance, hermetically sealed canned motor pumps that circulate the reactor coolant through the reactor vessel, loop piping, and steam generators.

The pumps are integrated into the steam generator channel head. The integration of the pump suction into the bottom of the steam generator channel head eliminates the cross-over leg of coolant loop piping; reduces the loop pressure drop; simplifies the foundation and support system for the steam generator, pumps, and piping; and reduces the potential for uncovering of the core by eliminating the need to clear the loop seal during a small loss-of-coolant accident (LOCA). The AP-600 design uses four pumps; two pumps are coupled with each steam generator.

Since the pumps have no seals, they cannot cause a seal failure LOCA. This is a significant safety enhancement, as seal failure LOCA is a major industry issue. Maintenance is also enhanced, since seal replacement is eliminated.

The pumps are mounted in the inverted (motor-below-casing) position. Inverted canned motors have been in operation for over 34 years in marine and fossil boiler circulation systems. These pumps have better operating reliability than upright units because the motor cavity is self-venting into the pump casing, avoiding the potential for gas pockets in the bearing and water regions.

One modification of the pumps from commercial and marine canned motor pump practice is the use of a flywheel to increase the pump rotating inertia. The increased inertia provides a slower rate-of-flow coastdown to improve core thermal margins following the loss of electric power. Extensive testing has validated the manufacturability and operability of the pump flywheel assembly.

Main coolant lines - Reactor coolant system (RCS) piping is configured with two identical main coolant loops, each of which employs a single 31-inch (790 mm) inside diameter hot leg pipe to transport reactor coolant to a steam generator. The two reactor coolant pump suction nozzles are welded directly to the outlet nozzles on the bottom of the steam generator channel head. Two 22-inch (560 mm) inside diameter cold leg pipes in each loop (one per pump) transport reactor coolant back to the reactor vessel to complete the circuit.

The RCS loop layout contains several important features that provide for a significantly simplified and safer design. The reactor coolant pumps mount directly on the channel head of each steam generator. This allows the pumps and steam generator to use the same structural support, greatly simplifying the support system and providing more space for pump and steam generator maintenance. The combined steam generator/pump vertical support is a single pinned column extending from the cell floor to the bottom of the channel head. The steam generator channel head is a one-piece forging with manufacturing and inspection advantages over multipiece, welded components. The integration of the pump suction into the bottom of the steam generator channel head eliminates the crossover leg of coolant loop piping, thus avoiding the potential for core uncovery due to loop seal venting after a small loss-of-coolant accident.

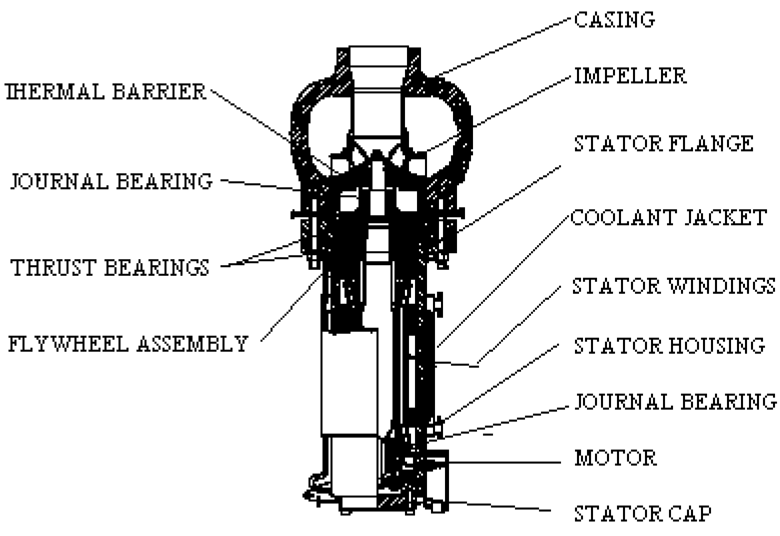

FIG. 2.4-4. Reactor coolant pump

The simplified, compact arrangement of the RCS also provides other benefits. The two cold leg lines of the two main coolant loops are identical (except for instrumentation and small line connections) and include bends to provide a low-resistance flow path and flexibility to accommodate the expansion difference between the hot and cold leg pipes. The piping is forged and then bent by a hot induction forming process. The use of a pipe bend reduces costs and in-service inspection requirements by eliminating welds. The loop configuration and material selection yield sufficiently low pipe stresses so that the primary loop and large auxiliary lines meet the requirements to demonstrate leak-before-break. Thus, pipe rupture restraints are not required, greatly simplifying the design and providing enhanced access for maintenance. The simplified RCS loop configuration also allows for a significant reduction in the number of snubbers, whip restraints, and supports. Field service experience and utility feedback have indicated the high desirability of these features.

2.5. Reactor auxiliary systems

Chemical and volume control system - The chemical and volume control system consists of regenerative and letdown heat exchangers, demineralizers and filters, makeup pumps, tanks, and associated valves, piping, and instrumentation.

The chemical and volume control system is designed to perform the following major tasks:

- **Purification** - maintain reactor coolant purity and activity level within acceptable limits;
- **Reactor coolant system inventory control and makeup** - maintain the required coolant inventory in the reactor coolant system; maintain the programmed pressurizer water level during normal plant operations;
- **Chemical shim and chemical control** - maintain the reactor coolant chemistry conditions by controlling the concentration of boron in the coolant for plant startups, normal dilution to compensate for fuel depletion and shutdown boration and provide the means for controlling the reactor coolant system pH by maintaining the proper level of lithium hydroxide;
- **Oxygen control** - provide the means for maintaining the proper level of dissolved hydrogen in the reactor coolant during power operation and for achieving the proper oxygen level prior to startup after each shutdown;
- **Filling and pressure testing of the reactor coolant system** - provide the means for filling and pressure testing of the reactor coolant system. The chemical and volume control system does not perform hydrostatic testing of the reactor coolant system, which is only required prior to initial startup and after major, nonroutine maintenance, but provides connections for a temporary hydrostatic test pump;
- **Borated makeup to auxiliary equipment** - provide makeup water to the primary side systems which require borated reactor grade water;
- **Pressurizer Auxiliary Spray** - provide pressurizer auxiliary spray water for depressurization.

Normal residual heat removal system - The normal residual heat removal system consists of two mechanical trains of equipment, each comprising one pump, one heat exchanger and associated valves, piping and instrumentation. The two trains of equipment share a common suction line from the reactor coolant system and a common discharge header. The major functions of the system are:

- **Shutdown Heat Removal.** The normal residual heat removal system removes both residual and sensible heat from the core and the reactor coolant system. It reduces the temperature of the reactor coolant system during the second phase of plant cooldown. The first phase of cooldown is accomplished by transferring heat from the reactor coolant system via the steam generators to the main steam system;
- Following cooldown, the normal residual heat removal system removes heat from the core and the reactor coolant system during the plant shutdown, until the plant is started up;
- The normal residual heat removal system reduces the temperature of the reactor coolant system from 350 to 120°F (177 to 48.9 ^{o}C) within 96 hours after shutdown. The system maintains the reactor coolant temperature at or below 120°F for the plant shutdown;
- **Shutdown Purification.** The normal residual heat removal system provides reactor coolant system flow to the chemical and volume control system during refueling operations;
- **In-Containment Refuelling Water Storage Tank Cooling.** The normal residual heat removal system provides cooling for the in-containment refuelling water storage tank (IRWST) during operation of the passive residual heat removal heat exchanger or during normal plant operations when required. The system is manually initiated by the operator. The normal residual heat removal system limits the IRWST water temperature to less than 212°F (100 ^{o}C) during extended operation of the passive residual heat removal system and to not greater than 120°F during normal operation;
- **Low Pressure Reactor Coolant System Makeup and Cooling**. The normal residual heat removal system provides low pressure makeup from the IRWST to the reactor coolant system and provides additional margin for core cooling. The system is manually initiated by the operator following receipt of an automatic depressurization signal. If the system is available, it provides reactor coolant system makeup once the pressure in the reactor coolant system falls below the shutoff head of the normal residual heat removal system pumps;
- **Low Temperature Overpressure Protection.** The normal residual heat removal system provides a low temperature overpressure protection function for the reactor coolant system during refuelling, startup, and shutdown operations;
- **Long-Term, Post-Accident Containment Inventory Makeup Flowpath.** The normal residual heat removal system provides a flow path for long term post-accident makeup to the reactor containment inventory, under design assumptions of containment leakage.

2.6. Operating characteristics

The plant control scheme is based on the "reactor follows plant loads". A grid fluctuation can be compensated for through turbine control valves in case of a frequency drop. A decrease in pressure at the turbine would require an increase in reactor power.

The AP-600 is designed to withstand the following operational occurrences without the generation of a reactor trip or actuation of the safety related passive engineered safety systems:

- ± 5%/minute ramp load change within 15% and 100% power,
- ± 10% step load change within 15% and 100% power,
- 100% generator load rejection,
- 100-50-100% power level daily load follow over 90% of the fuel cycle life,
- Grid frequency changes equivalent to 10% peak-to-peak power changes at 2%/minute rate,
- 20% power step increase or decrease within 10 minutes,
- Loss of a single feedwater pump.

The logic and setpoints for all of the AP-600 Nuclear Steam Supply System (NSSS) control systems are developed in order to meet the above operational transients without reaching any of the protection system setpoints.

3.1. Safety requirements and design philosophy

The AP-600 design provides for multiple levels of defense for accident mitigation (defense-in-depth), resulting in extremely low core damage probabilities while minimizing the occurrences of containment flooding, pressurization, and heat-up situations. This defense-in-depth capability includes multiple levels of defense for a very wide range of plant events. Defense-in-depth is integral to the AP-600 design, with a multitude of individual plant features capable of providing some degree of defense of plant safety. Six aspects of the AP-600 design contribute to defense-in-depth:

3.1.1. Stable Operation.

In normal operation, the most fundamental level of defense-in-depth ensures that the plant can be operated stably and reliably. This is achieved by the selection of materials, by quality assurance during design and construction, by well-trained operators, and by an advanced control system and plant design that provide substantial margins for plant operation before approaching safety limits.

3.1.2. Physical Plant Boundaries.

One of the most recognizable aspects of defense-in-depth is the protection of public safety through the physical plant boundaries. Releases of radiation are directly prevented by the fuel cladding, the reactor pressure boundary, and the containment pressure boundary.

3.1.3. Passive Safety-Related Systems.

The highest level of defense includes the AP-600 safety-related passive systems and equipment. The safety-related passive systems are sufficient to automatically establish and maintain core cooling and containment integrity for the plant following design basis events, assuming that the most limiting single failure occurs. These systems maintain core cooling and containment integrity after an event, without operator action and onsite and offsite ac power sources, for an indefinite amount of time (see section 3.2 for a description of AP-600 passive safety systems).

3.1.4. Diversity within the Safety-Related Systems.

An additional level of defense is provided through the diverse mitigation functions within the passive safety-related systems themselves. This diversity exists, for example, in the residual heat removal function. The PRHR HX is the passive safety-related heat exchanger for removing decay heat during a transient. In case of multiple failures in the PRHR HX, defense-in-depth is provided by the passive safety injection and automatic depressurization (passive feed and bleed) functions of the PXS.

3.1.5. Non-safety Systems.

The next level of defense-in-depth is the availability of certain non-safety systems for reducing the potential for events leading to core damage. For more probable events, these defense-in-depth non-safety systems automatically actuate to provide a first level of defense to reduce the likelihood of unnecessary actuation and operation of the safety-related systems. These non-safety-related systems establish and maintain safe shutdown conditions for the plant following design basis events, provided that at least one of the non-safety-related ac power sources is available.

3.1.6. Containing Core Damage.

The AP-600 design provides the operators with the ability to drain the in-containment refueling water storage tank (IRWST) water into the reactor cavity in the event that the core has uncovered and is melting. The objective of this cavity flooding action is to prevent reactor vessel failure and subsequent relocation of molten core debris into the containment. Retention of the debris in the vessel provides for a high confidence that containment failure and radioactive release to the environment will not occur due to ex-vessel severe accident phenomena.

AP-600 defense-in-depth features enhance safety such that no severe release of fission products is predicted to occur from an initially intact containment for more than 100 hours after the onset of core damage, assuming no actions for recovery. This amount of time provides for performance of accident management actions to mitigate the accident and prevent containment failure. The frequency of severe release as predicted by PRA is 3.0E-08 per reactor year.

3.2. Safety systems and features (active, passive, and inherent)

The use of passive safety systems provides significant and measurable improvements in plant simplification, safety, reliability, and investment protection. The AP-600 uses passive safety systems to improve the safety of the plant and to satisfy safety criteria of regulatory authorities. The passive safety systems require no operator actions to mitigate design basis accidents. These systems use only natural forces such as gravity, natural circulation, and compressed gas to make the systems work. No pumps, fans, diesels, chillers, or other active machinery are used. A few simple valves align and automatically actuate the passive safety systems. To provide high reliability, these valves are designed to actuate to their safety positions upon loss of power or upon receipt of a safety actuation signal. However, they are also supported by multiple, reliable power sources to avoid unnecessary actuations.

The AP-600 passive safety-related systems include:

- The passive core cooling system (PXS),
- The passive containment cooling system (PCS),
- The main control room habitability system (VES),
- Containment isolation.

These passive safety systems provide a high degree of plant safety and investment protection. They establish and maintain core cooling and containment integrity indefinitely, with no operator or ac power support requirements. The passive systems are designed to meet the single-failure criteria, and probabilistic risk assessments (PRAs) are used to verify their reliability.

The AP-600 passive safety systems are significantly simpler relative to safety systems of earlier Westinghouse plants since they contain significantly fewer components, reducing the required tests, inspections, and maintenance, require no active support systems, and their readiness is easily monitored. The passive safety systems have one-third the number of remote valves as typical active safety systems, and they contain no pumps. In addition, these passive safety systems do not require a radical departure in the design of the rest of the plant, core, RCS, or containment. This design approach allows the licensing safety criteria to be satisfied with a greatly simplified plant design.

Emergency core cooling system - The passive core cooling system (PXS) (Figure 3.2-1) protects the plant against reactor coolant system (RCS) leaks and ruptures of various sizes and locations. The PXS provides the safety functions of core residual heat removal, safety injection, and depressurization. Safety analyses (using U.S.NRC-approved codes) demonstrate the effectiveness of the PXS in protecting the core following various RCS break events, even for breaks as severe as the 8-inch (200 mm) vessel injection lines. The PXS provides approximately a 400°F (220°C) margin to the maximum peak clad temperature limit for the double-ended rupture of a main reactor coolant pipe.

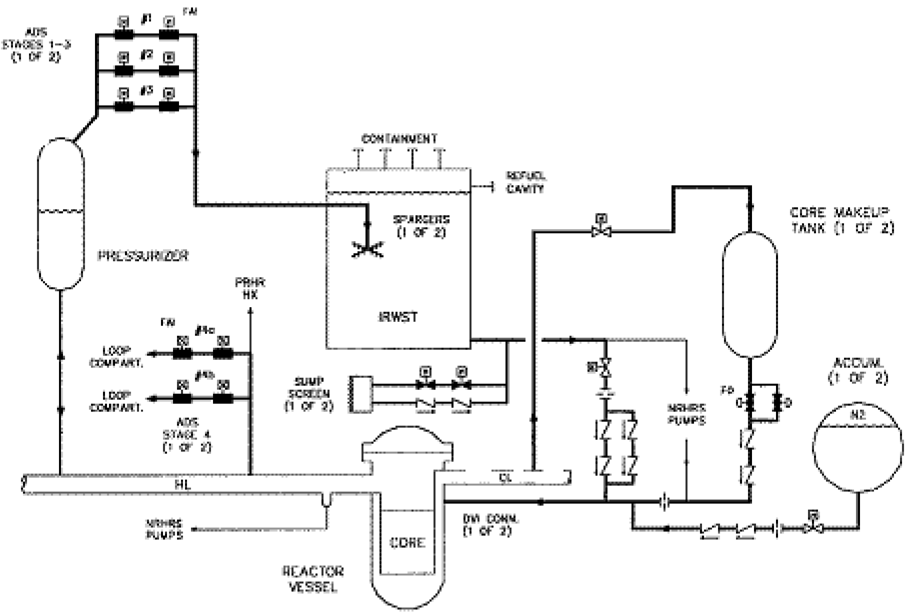

FIG. 3.2-1. AP-600 Passive core cooling system

Safety injection and depressurization - The PXS uses three passive sources of water to maintain core cooling through safety injection. These injection sources include the core makeup tanks (CMTs), the accumulators, and the in-containment refuelling water storage tank (IRWST). These injection sources are directly connected to two nozzles on the reactor vessel so that no injection flow can be spilled for the larger break cases.

Long-term injection water is provided by gravity from the IRWST, which is located in the containment just above the RCS loops. Normally, the IRWST is isolated from the RCS by check valves. This tank is designed for atmospheric pressure. As a result, the RCS must be depressurized before injection can occur.

The depressurization of the RCS is automatically controlled to reduce pressure to about 12 psig (0.18 MPa); at which point, the head of water in the IRWST overcomes the low RCS pressure and the pressure loss in the injection lines. The PXS provides for depressurization using the four stages of the ADS to permit a relatively slow, controlled RCS pressure reduction.

Passive residual heat removal - The PXS includes a 100% capacity passive residual heat removal heat exchanger (PRHR HX). . The PRHR HX is designed to match the core decay heat at 15 minutes after reactor shutdown. Following a loss of main feed water, with no credit for actuation of the startup feed water pumps and with safety analysis conservatisms; the PRHR HX heat removal rate, together with the steam generator secondary side inventory, is sufficient to maintain the RCS fluid subcooled and maintain acceptable pressurizer pressure and level increase. Thus the PRHR HX protects the plant against transients that upset the normal steam generator feedwater and steam systems and satisfies the safety criteria for loss of feedwater, feedwater line breaks, and steam line breaks.

The IRWST provides the heat sink for the PRHR HX. The IRWST water volume is sufficient to absorb decay heat for more than 1 hour before the water begins to boil. Once boiling starts, steam passes to the containment. This steam condenses on the steel containment vessel and, after collection, drains by gravity back into the IRWST. The PRHR HX and the passive containment cooling system provide indefinite decay heat removal capability with no operator action required.

Passive containment cooling system - The passive containment cooling system (PCS) (Figure 3.2-2) provides the safety-related ultimate heat sink for the plant. As demonstrated by computer analyses and extensive test programs, the PCS effectively cools the containment following an accident such that the design pressure is not exceeded and the pressure is rapidly reduced.

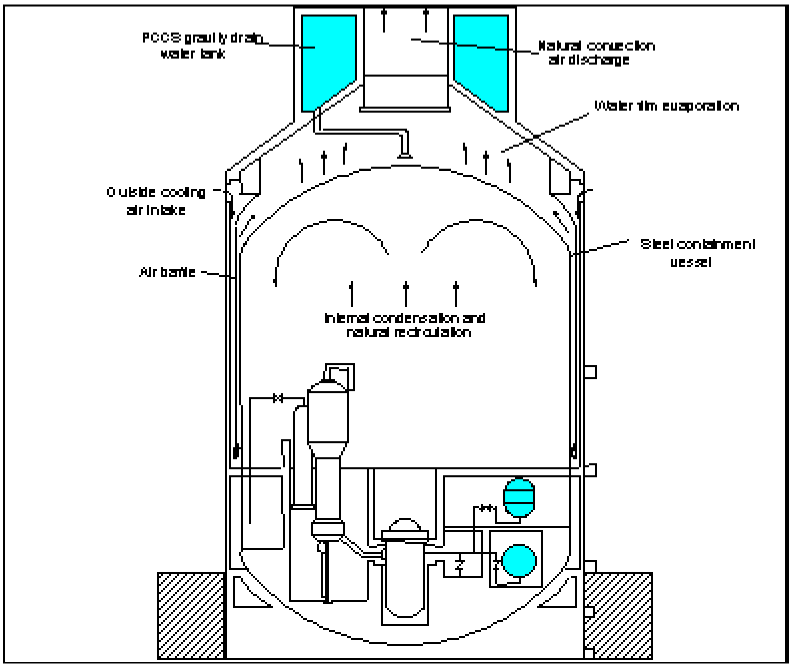

FIG. 3.2-2. Passive containment cooling system

The steel containment vessel provides the heat transfer surface that removes heat from inside the containment and rejects it to the atmosphere. Heat is removed from the containment vessel by continuous natural circulation flow of air. During an accident, the air cooling is supplemented by evaporation of water. The water drains by gravity from a tank located on top of the containment shield building.

Westinghouse has calculated the AP-600 to have a significantly reduced frequency of release of large amounts of radioactivity following a severe accident core damage scenario. This analysis shows that with only the normal PCS air cooling, the containment stays well below the predicted failure pressure. Other factors include improved containment isolation and reduced potential for LOCAs outside of containment. This improved containment performance supports the technical basis for simplification of offsite emergency planning.

Main control room habitability system - The main control room habitability system (VES) provides fresh air, cooling, and pressurization to the main control room (MCR) following a plant accident. Operation of the VES is automatically initiated upon receipt of a high MCR radiation signal, which isolates the normal control room ventilation path and initiates pressurization. Following system actuation, all functions are completely passive.

The VES air supply is contained in a set of compressed air storage tanks. The VES also maintains the MCR at a slight positive pressure, to minimize the infiltration of airborne contaminants from the surrounding areas.

Containment isolation - AP-600 containment isolation is significantly improved. One major improvement is the large reduction in the number of penetrations. Furthermore, the number of normally open penetrations is reduced by 60 percent. There are no penetrations required to support post-accident mitigation functions (the canned motor reactor coolant pumps do not require seal injection, and the passive residual heat removal and passive safety injection features are located entirely inside containment).

Long-term accident mitigation - A major safety advantage of the AP-600 is that long-term accident mitigation is maintained without operator action and without reliance on offsite or onsite ac power sources. The passive safety systems are designed to provide long-term core cooling and decay heat removal without the need for operator actions and without reliance on the active nonsafety-related systems. For the limiting design basis accidents, the core coolant inventory in the containment for recirculation cooling and boration of the core is sufficient to last for at least 30 days, even if inventory is lost at the design basis containment leak rate.

3.3. Severe accidents (beyond design basis accidents)

In-vessel retention of molten core debris

In-vessel retention (IVR) of core debris by cooling from the outside is a severe accident mitigation attribute of the AP-600 design. With the reactor vessel intact and debris retained in the lower head, phenomena that may occur as a result of core debris being relocated to the reactor cavity are prevented. The AP-600 is provided with reactor vessel insulation that promotes in-vessel retention and surface treatment that promotes wetability of the external surface.

The design features of the AP-600 containment promote flooding of the containment cavity region during accidents, submerging of the reactor vessel lower head in water. Liquid effluent released through the break during a LOCA event is directed to the reactor cavity. The AP-600 design also includes a provision for draining the in-containment refuelling water storage tank (IRWST) water into the reactor cavity through an operator action.

Proliferation resistance

See Design Description for the AP-1000

Safety and security (physical protection)

See Design Description for the AP-1000

Description of turbine-generator systems

6.1. Turbine generator plant

The AP-600 turbine consists of a double-flow, high-pressure cylinder and two double-flow, low-pressure cylinders that exhaust to individual condensers. It is a four flow tandem-compound, 1800 rpm machine. The turbine generator is intended for base load operation and also has load follow capability. Mechanical design of the turbine root and rotor steeple attachments uses optimized contour to significantly reduce operational stresses. Steam flow to the high-pressure turbine is controlled by two floor-mounted steam chests. Each contains two throttle/stop valve assemblies, and two load-governing valves. The single moisture separator reheater (MSR) has a single stage of reheat.

The condenser and circulating water systems have been optimized. The condenser is a twin-shell, multipressure unit with one double-flow, low-pressure turbine exhausting into the top of each shell.

The turbine-generator and associated piping, valves, and controls are located completely within the turbine building and are accessible under operating conditions. There are no safety-related systems or components located within the turbine building. The probability of destructive overspeed condition and missile generation, assuming the recommended inspection frequency, is less than $1´10^{-5}$ per year. In addition, orientation of the turbine-generator is such that high-energy missiles would be directed away at right angles to safety-related structures, systems, or components. Failure of turbine-generator equipment does not preclude safe shutdown of the reactor. The single direct-driven generator is gas-cooled and rated at 856 MVA at 22 kV, and a power factor of 0.9. Other related system components include a complete turbine-generator bearing lubrication oil system, a digital electrohydraulic (DEH) control system with supervisory instrumentation, a turbine steam sealing system, overspeed protective devices, turning gear, a generator hydrogen and seal oil system, a generator CO_2 system, an exciter cooler, a rectifier section, an exciter, and a voltage regulator.

6.2. Condensate and feedwater systems

The condensate and feedwater system supplies the steam generators with heated feedwater in a closed steam cycle using regenerative feedwater heating. The condensate and feedwater system is composed of the condensate system, the main and startup feedwater system, and portions of the steam generator system.

The feedwater train consists of seven stages of feedwater heating with two parallel string, low-pressure feedwater heaters located in the condenser neck with the next two single-string, low-pressure heaters, deaerator, and the high-pressure heaters located within the turbine building. The condenser hotwell and deaerator storage capacity provides margin in the design. This margin, coupled with three 50 percent condensate pumps, provides greater flexibility and the ability for an operator to control feedwater and condensate transients.

7.1. I&C Systems

The I&C system design for AP-600 integrates individual systems using similar technology. The heart of the system is the portions used for plant protection and for operation of the plant.

The integrated AP-600 I&C system provides the following benefits, relative to previous Westinghouse LWR designs:

- Control wiring is reduced by 80 percent compared to equivalent hard wired plants without passive safety features;
- Cable spreading rooms are eliminated;
- Duplicate sensors, signal conditioners, and cables are eliminated;
- Maintenance is simplified;
- Plant design changes have little impact on I&C design;
- Accurate, drift-free calibration is maintained;
- Operating margins are improved.

The AP-600 man-machine interfaces have been simplified compared to existing Westinghouse plants. The probability of operator error is reduced and operations, testing, and maintenance are simplified. An automatic signal selector in the control system selects from a redundant sensor for control inputs in lieu of requiring manual selection by the control board operator. Accident monitoring and safety parameters are displayed on safety qualified displays with a co-ordinated set of graphics generated by the qualified data processor. The major benefits of the improved man-machine interfaces are:

- Reduced quantity of manual actions is required,
- Reduced quantity of data is presented to operator,
- Number of alarms is reduced,
- Improved quality of data is presented to operator,
- Data is interpreted for the operator by system computer,
- Maintenance is simplified.

7.1.1. Design concept, including control room

The instrumentation and control architecture for the AP-600 is arranged in a hierarchical manner to provide a simplified, structured design that is horizontally and vertically integrated.

Above the monitor bus are the systems that facilitate the interaction between the plant operators and the I&C. These are the operations and control centers system (OCS) and the data display and monitoring system (DDS). Below the monitor bus are the systems and functions that perform the protective, control, and data monitoring functions.
These are the protection and safety monitoring system (PMS) (Section 7.1.2) the plant control system (PLS), the

special monitoring system (SMS), the diverse actuation system and the in-core instrumentation system (IIS).

The plant control system (PLS) has the function of establishing and maintaining the plant operating conditions within prescribed limits. The control system improves plant safety by minimizing the number of situations for which some protective response is initiated and relieves the operator from routine tasks.

The purpose of the diverse actuation system (DAS) is to provide alternative means of initiating the reactor trip and emergency safety features. The hardware and software used to implement the DAS are different from the hardware and software used to implement the protection and safety monitoring system. The DAS is included to meet the anticipated transient without (reactor) trip (ATWT) rule and to reduce the probability of a severe accident resulting from the unlikely coincidence of a transient and common mode failure of the protection and safety monitoring. The protection and safety monitoring system is designed to prevent common mode failures; however, in the low-probability case where a common mode failure could occur, the DAS provides diverse protection.

Main control room - The operations and control centers system includes the complete operational scope of the main control room, the remote shutdown workstation, the waste processing control room, and partial scope for the technical support center. With the exception of the control console structures, the equipment in the control room is part of the other systems (for example, protection and safety monitoring system, plant control system, data and display processing system). The conceptual arrangement of the main control room is shown in Figure *7.1.1-2*.

The boundaries of the operations and control center system for the main control room and the remote shutdown workstation are the signal interfaces with the plant components. These interfaces are via the plant protection and safety monitoring system processor and logic circuits, which interface with the reactor trip and engineered safety features plant components; the plant control system processor and logic circuits, which interface with the non-safety-related plant components; and the plant monitor bus, which provides plant parameters, plant component status, and alarms.

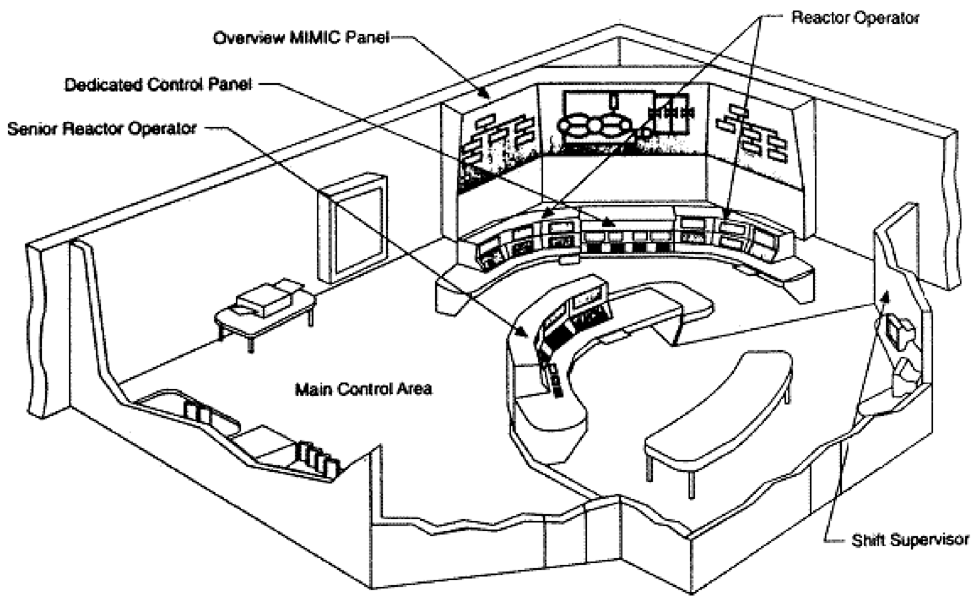

FIG. 7.1.1-2. AP-600 Main control room sketch

7.1.2. Reactor protection system and other safety systems

The AP-600 provides instrumentation and controls to sense accident situations and initiate engineered safety features. The occurrence of a limiting fault, such as a loss-of-coolant accident or a secondary system break, requires a reactor trip plus actuation of one or more of the engineered safety features. This combination of events prevents or mitigates damage to the core and reactor coolant system components, and provides containment integrity.

The protection and safety monitoring system (PMS) provides the safety-related functions necessary to control the plant during normal operation, to shut down the plant, and to maintain the plant in a safe shutdown condition. The protection and safety monitoring system controls safety-related components in the plant that are operated from the main control room or remote shutdown workstation.

7.2. Electrical systems

The AP-600 on-site power system includes the main AC power system and the DC power system. The main AC power is a non-Class 1E system. The DC power system consists of two independent systems, one Class 1E and one non-Class 1E. The on-site power system is designed to provide reliable electric power to the plant safety and non-safety equipment for normal plant operation, startup, and normal shut down, and for accident mitigation and emergency shutdown.

The main generator is connected to the off-site power system via three single-phase main step-up transformers. The normal power source for the plant auxiliary AC loads is provided from the 22 kV isophase generator buses through the two unit auxiliary transformers of identical ratings. In the event of a loss of the main generator, the power is maintained without interruption from the preferred power supply by an auto-trip of the main generator breaker. Power then flows from the transformer area to the auxiliary loads through the main and unit auxiliary transformers.

Off-site power has no safety-related function due to the passive safety features incorporated in the AP-600 design. Therefore, redundant off-site power supplies are not required. The design provides a reliable offsite power system that minimizes challenges to the passive safety system.

7.2.1. Operational power supply systems

The main AC power system is a non-Class 1E system that does not perform any safety function. The standby power supply is included in the on-site standby power system.

The power to the main AC power system normally comes from the station main generator through unit auxiliary transformers. The plant is designed to sustain a load rejection from 100 percent power with the turbine generator continuing stable operation while supplying the plant house loads. The load rejection feature does not perform any safety function

The on-site standby AC power system is powered by the two on-site standby diesel generators and supplies power to selected loads in the event of loss of normal, and preferred AC power supplies.
The plant DC power system comprises two independent Class 1E and non-Class 1E DC power systems. Each system consists of ungrounded stationary batteries, DC distribution equipment, and uninterruptible power supplies.

7.2.2. Safety-related systems

The Class 1E DC power system includes four independent divisions of battery systems. Any three of the four divisions can shut down the plant safely and maintain it in a safe shutdown condition. Divisions B and C have two battery banks. One of these battery banks is sized to supply power to selected safety-related loads for at least 24 hours, and the other battery bank is sized to supply power to another set of selected safety-related loads for at least 72 hours following a design basis event (including the loss of all AC power).

For supplying power during the post-72 period, provisions are made to connect an ancillary ac generator to Class 1E voltage regulating transformers (Divisions B and C only).

8.1. Liquid, Gaseous and Solid Waste management

Radioactive waste management - The radioactive waste management systems include systems to deal with liquid, gaseous and solid waste. These systems are closely integrated with the chemical and volume control system (CVS).

The liquid waste management systems include the systems that may be used to process for disposal liquids containing radioactive material. These include the following:

- Steam generator blowdown processing system
- Radioactive waste drain system
- Liquid radwaste system

The liquid radwaste system (WLS) uses ion exchangers to process and discharge all wastes from the reactor coolant system. To enhance ion exchange performances, the WLS is divided in two trains to separate borated reactor water from mixed liquid waste.

The gaseous radwaste system is a once-through, ambient-temperature, charcoal delay system. The system consists of a drain pot, a gas cooler, a moisture separator, an activated charcoal-filled guard bed, and two activated charcoal-filled delay beds. Also included in the system are an oxygen analyzer subsystem and a gas sampling subsystem.

The radioactive fission gases entering the system are carried by hydrogen and nitrogen gas. The primary influent source is the liquid radwaste system degasifier. The degasifier extracts both hydrogen and fission gases from the chemical and volume control system letdown flow.

The solid waste management system is designed to collect and accumulate spent ion exchange resins and deep bed filtration media, spent filter cartridges, dry active wastes, and mixed wastes generated as a result of normal plant operation, including anticipated operational occurrences. The system is located in the auxiliary and radwaste buildings. Processing and packaging of wastes are by mobile systems in the auxiliary building loading bay and the mobile systems facility which is a part of the radwaste building. The packaged waste is stored in the annex, auxiliary and radwaste buildings until it is shipped offsite to a licensed disposal facility.

9.1. Buildings and structures, including plot plan

A typical site plan for a single unit AP-600 is shown on Figure 9.1-1. The power block complex consists of five principal building structures; the nuclear island, the turbine building, the annex building, the diesel generator building and the radwaste building. Each of these building structures is constructed on individual basemats. The nuclear island consists of the containment building, the shield building, and the auxiliary building, all of which are constructed on a common basemat.

The plant arrangement contains conventional and unique features that facilitate and simplify operational and maintenance activities. For example, accessibility to the containment during an outage is extremely important to those maintenance activities that can be performed only during the outage. The AP-600 containment contains a 22-feet (6.7 m) diameter main equipment hatch and a personnel airlock at the operating deck level, and a 16-feet (4.9 m) diameter maintenance hatch and a personnel airlock at grade level. These large hatches significantly enhance accessibility to the containment during outages and, consequently, reduce the potential for congestion at the containment entrances. These containment hatches, located at the two different levels, allow activities occurring above the operating deck to be unaffected by activities occurring below the operating deck.

The containment arrangement provides large laydown areas inside containment at both the operating deck level and the maintenance floor level. Additionally, the auxiliary building and the adjacent annex building provide large staging and laydown areas immediately outside of both large equipment hatches.

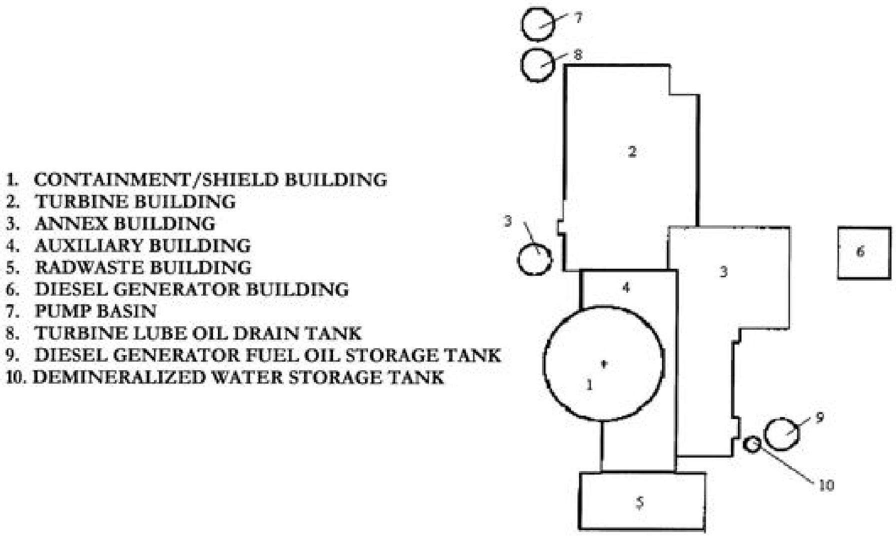

FIG. 9.1-1. AP-600 Site layout

The AP-600 consists of the following five principal structures. Each of these buildings is constructed on individual basemats:

- Nuclear island (Containment, Shield building, Auxiliary building),
- Annex building,
- Diesel generator building,
- Radwaste building,
- Turbine building.

9.2. Reactor building

The reactor building of the AP-600 basically coincides with the shield building surrounding the containment .

9.3. Containment

Containment building - The containment building is the containment vessel and all structures contained within the containment vessel. The containment building is an integral part of the overall containment system with the functions of containing the release of airborne radioactivity following postulated design basis accidents and providing shielding for the reactor core and the reactor coolant system during normal operations.

The containment vessel is an integral part of the passive containment cooling system. The containment vessel and the passive containment cooling system are designed to remove sufficient energy from the containment to prevent the containment from exceeding its design pressure following postulated design basis accidents.

The principal system located within the containment building is the reactor coolant system, the passive core cooling system, and the reactor coolant purification portion of the chemical and volume control system.

Shield building - The shield building is the structure and annulus area that surrounds the containment building. During normal operations, a primary function of the shield building is to provide shielding for the containment vessel and the radioactive systems and components located in the containment building. The shield building, in conjunction with the internal structures of the containment building, provides the required shielding for the reactor coolant system and all the other radioactive systems and components housed in the containment. During accident conditions, the shield building provides the required shielding for radioactive airborne materials that may be dispersed in the containment as well as radioactive particles in the water distributed throughout the containment.

The shield building is also an integral part of the passive containment cooling system. The passive containment cooling system air baffle is located in the upper annulus area. It is attached to the cylindrical section of the containment vessel. The function of the passive containment cooling system air baffle is to provide a pathway for natural circulation of cooling air in the event that a design basis accident results in a large release of energy into the containment. In this event the outer surface of the containment vessel transfers heat to the air between the baffle and the containment shell. This heated and thus, lower density air flows up through the air baffle to the air diffuser and cooler and higher density air is drawn into the shield building through the air inlet in the roof of the shield building.

Another function of the shield building is to protect the containment building from external events. The shield building protects the containment vessel and the reactor coolant system from the effects of tornadoes and tornado produced missiles.

9.4. Turbine building

The turbine building houses the main turbine, generator, and associated fluid and electrical systems. It provides weather protection for the laydown and maintenance of major turbine/generator components. The turbine building also houses the makeup water purification system. No safety-related equipment is located in the turbine building.

9.5. Other buildings

9.5.1. Auxiliary building

The primary function of the auxiliary building is to provide protection and separation for the seismic Category I mechanical and electrical equipment located outside the containment building. The auxiliary building provides protection for the safety-related equipment against the consequences of either a postulated internal or external event and also provides shielding for the radioactive equipment and piping that is housed within the building.

The most significant equipment, systems and functions contained within the auxiliary building are the following:

9.5.2. Main control room.

The main control room provides the man-machine interfaces required to operate the plant safely under normal conditions and to maintain it in a safe condition under accident conditions.

9.5.3. Instrumentation and control systems.

The instrumentation and control systems provide monitoring and control of the plant during startup, ascent to power, powered operation, and shutdown.

9.5.4. Class 1E direct current system.

The Class 1E DC system provides 125 volts power for safety-related and vital control instrumentation loads including monitoring and control room emergency lighting.

9.5.5. Fuel handling area.

The primary function of the fuel handling area is to provide for the handling and storage of new and spent fuel.

9.5.6. Mechanical equipment areas.

The mechanical equipment located in radiological control areas of the auxiliary building are the normal residual heat removal pumps and heat exchangers, the spent fuel cooling system pumps and heat exchangers, the liquid and gaseous radwaste pumps, tanks, demineralizers and filters, the chemical and volume control pumps and heating, ventilating and air conditioning exhaust fans.

9.5.7. Containment penetration areas.

The auxiliary building contains all of the mechanical, electrical and I&C containment penetration areas from the shield building.

9.5.8. Main steam and feedwater isolation valve compartment.

The main steam and feedwater isolation valve compartment is contained within the auxiliary building. The auxiliary building provides an adequate venting area for the main steam and feedwater isolation valve compartment in the event of a postulated leak in either a main steam line or feedwater line.

9.5.9. Annex building

The annex building provides the main personnel entrance to the power generation complex. It includes access ways for personnel and equipment to the clean areas of the nuclear island in the auxiliary building and to the radiological control area. The building includes the health physics facilities for the control of entry to and exit from the radiological control area as well as personnel support facilities such as locker rooms.

9.5.10. Diesel generator building

The diesel generator building houses two identical slide along diesel generators separated by a three hour fire wall. These generators provide backup power for plant operation in the event of disruption of normal power sources. No safety-related equipment is located in the diesel generator building.

9.5.11. Radwaste building

The radwaste building includes facilities for segregated storage of various categories of waste prior to processing, for processing by mobile systems, and for storing processed waste in shipping and disposal containers. No safety-related equipment is located in the radwaste building.

Plant performance

AP-600 economics is derived from simplification and state-of-the-art construction techniques.

10.1. Simplification

AP-600 is an advance passive nuclear power plant that has been designed to meet globally recognized requirements. A concerted effort has been made to simplify systems and components to facilitate construction, operation and maintenance and to reduce the capital and generating costs. The use of passive systems allows the plant design to be significantly simpler than current generation pressurized water reactor plants. In addition, the passive safety systems do not require the large network of safety support systems found in current generation nuclear power plants (e.g., Class 1E ac power, safety HVAC, safety cooling water systems and associated seismic buildings). The AP-600 uses 50% fewer valves, 80% less pipe (safety grade), 70% less control cable, 35% fewer pumps, and 45% less seismic building volume than an equivalent conventional Westinghouse plant. Simplicity reduces the cost for reasons other than reduction of the number of items to be purchased. With a fewer number of components, the overall cost of installation is reduced and construction time is minimized.

10.2. Construction Schedule

The AP-600 has been designed to make use of modern modular construction techniques. Not only does the design incorporate vendor designed skids and equipment packages, it also includes large structural modules and special equipment modules. Modularization allows construction tasks that were traditionally performed in sequence to be completed in parallel. The modules, constructed in factories, can be assembled at the site for a planned construction schedule, as predicted by Westinghouse, of 3 years – from ground-breaking to fuel load. This duration has been verified by experienced construction managers through 4D (3D models plus time) reviews of the construction sequence.

10.3. Availability and O&M Costs

The AP-600 combines proven Westinghouse PWR technology with utility operating experience to enhance reliability and operability. Model F steam generators, canned motor pumps and rugged turbine generators are proven performers with outstanding operating records. The digital on-line diagnostic instrumentation and control system features an integrated control system, which avoids reactor trips due to single channel failure. In addition, the plant design provides large margins for plant operation before reaching the safety limits. This assures a stable and reliable plant operation with a reduced number of reactor trips (less than one reactor trip per year). Based on the above, and considering the short planned refueling outage (17 days) and plans to use a 24-month fuel cycle, the AP-600 is expected to largely exceed the 93% availability goal. For AP-600 availability is enhanced by the simplicity designed into the plant, as described above. There are fewer components which result in lower maintenance costs, both planned and unplanned. In addition, the great reduction in safety-related components results in a large reduction in inspection and testing requirements. Simplicity is also reflected in the reduced AP-600 staffing requirements.

Deployment status and planned schedule

The AP-600 design received Final Design Approval from the U.S.NRC in 1998 and Design Certification in 1999.

Technical data

General plant data

Reactor thermal output	1940 MWth
Power plant output, net	600 MWe
Power plant efficiency, net	31 %
Mode of operation	Baseload and Load follow
Plant design life	60 Years
Plant availability target >	93 %
Seismic design, SSE	0.3
Primary coolant material	Light Water
Secondary coolant material	Light Water
Moderator material	Light water
Thermodynamic cycle	Rankine
Type of cycle	Indirect

Safety goals

Core damage frequency <	1.7E-7 /Reactor-Year
Large early release frequency <	3.0E-8 /Reactor-Year
Occupational radiation exposure <	0.7 Person-Sv/RY
Operator Action Time	0.5 Hours

Nuclear steam supply system

Steam flow rate at nominal conditions	1063 Kg/s
Steam pressure	5.74 MPa(a)
Steam temperature	272.7 °C
Feedwater flow rate at nominal conditions	1063 Kg/s
Feedwater temperature	224 °C

Reactor coolant system

Primary coolant flow rate	9940 Kg/s
Reactor operating pressure	15.513 MPa(a)
Core coolant inlet temperature	279.5 °C
Core coolant outlet temperature	315.6 °C
Mean temperature rise across core	36.1 °C

Reactor core

Active core height	3.658 m
Equivalent core diameter	2.921 m
Average linear heat rate	13.5 KW/m
Average fuel power density	28.89 KW/KgU
Average core power density	78.82 MW/m^3
Fuel material	Sintered UO2
Cladding material	ZIRLO™
Outer diameter of fuel rods	9.5 mm
Rod array of a fuel assembly	Square , 17x17
Lattice geometry	Square
Number of fuel assemblies	145
Enrichment of reload fuel at equilibrium core	4.8 Weight %
Fuel cycle length	24 Months
Average discharge burnup of fuel	55000 MWd/Kg

Burnable absorber (strategy/material)	Wet annular Burnable Absorber, Integral Fuel Burnable Absorber
Control rod absorber material	Ag-In-Cd(Black), Ag-In-Cd /304SS(Gray)
Soluble neutron absorber	H3BO3

Reactor pressure vessel

Inner diameter of cylindrical shell	3988 mm
Wall thickness of cylindrical shell	203 mm
Design pressure	17.1 MPa(a)
Design temperature	343.3 °C
Base material	Carbon Steel
Total height, inside	11708 mm

Steam generator or Heat Exchanger

Type	Delta-75, U-Tube, Vertical
Number	2
Total tube outside surface area	6986 m^2
Number of heat exchanger tubes	6307
Tube outside diameter	17.5 mm
Tube material	Inconel 690-TT
Transport weight	365.5 t

Reactor coolant pump (Primary circulation System)

Pump Type	Seal-less
Number of pumps	4
Pump speed	1800 rpm
Head at rated conditions	73 m
Flow at rated conditions	2.485 m^3/s

Pressurizer

Total volume	45.31 m^3
Steam volume (Working medium volume): full power/zero power	14.16 m^3
Heating power of heater rods	1600 kW

Primary containment

Type	Dry, free standing, steel
Overall form (spherical/cylindrical)	Cylindrical
Dimensions - diameter	39.6 m
Dimensions - height	57.6 m
Design pressure	0.316 MPa
Design temperature	137.8 °C
Design leakage rate	0.12 Volume % /day

Residual heat removal systems

Active/passive systems	Passive

Safety injection systems

Active/passive systems	Passive

Turbine

Number of turbine sections per unit (e.g. HP/MP/LP)	1HP / 2LP
Turbine speed	1800 rpm

Generator

Rated power	880 MVA
Active power	675 MW
Voltage	22 kV
Frequency	60 Hz

Condenser

Type	Multipressure (cooling towers) or Single pressure (direct cooling)
Condenser pressure	9.1 kPa

Plant configuration and layout

Plant configuration options	Ground-based

Feedwater pumps

Type	Motor Driven
Number	2