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STUDY OF COST EFFECTIVE LARGE ADVANCED PRESSURIZED WATER REACTORS THAT EMPLOY PASSIVE SAFETY FEATURES

Final Report for the Period August 15, 2000 - August 14, 2003

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PREPARED FOR THE UNITED STATES DEPARTMENT OF ENERGY OFFICE OF NUCLEAR ENERGY, SCIENCE, AND TECHNOLOGY

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EXECUTIVE SUMMARY

On December 16, 1999, The United States Nuclear Regulatory Commission issued Design Certification of the AP600 standard nuclear reactor design. This culminated an 8-year review of the AP600 design, safety analysis and probabilistic risk assessment. The AP600 is a 600 MWe reactor that utilizes passive safety features that, once actuated, depend only on natural forces such as gravity and natural circulation to perform all required safety functions. These passive safety systems result in increased plant safety and have also significantly simplified plant systems and equipment, resulting in simplified plant operation and maintenance. The AP600 meets NRC deterministic safety criteria and probabilistic risk criteria with large margins.

The large safety margins of the AP600 can be attributed to the performance of the passive safety systems in response to accidents. An extensive AP600 test program was performed to provide confidence in the ability to adequately predict the performance characteristics of the passive safety systems as required by 10 CFR 50. This test program consisted of separate effects and integral systems tests of the passive safety systems and is well documented in NUREG-1512, Final Safety Evaluation Report Related to Certification of the AP600 Standard Design. Westinghouse used the test programs to develop analytical computer codes that can predict with adequate certainty, the performance of the passive safety systems in response to design basis and beyond design basis accidents. In addition to the extensive test program conducted by Westinghouse, the NRC also performed confirmatory tests and analyses at both the APEX test facility at Oregon State University and the ROSA test facility at the Japan Atomic Energy Research Institute. As a result, the Westinghouse computer codes were validated as sufficient for use in performing accident analyses in accordance with the requirements of 10 CFR Part 50 and Part 52. In addition, the NRC performed independent analyses of the AP600 using different analysis codes to confirm the adequacy of the AP600 design as well as the AP600 safety analysis presented in the AP600 Standard Safety Analysis Report. These independent analyses also confirmed the large safety margins exhibited in the AP600.

Westinghouse is developing a larger version of the AP600 called the AP1000. The AP1000 design is based largely on the AP600. It employs passive systems that operate in the same manner as the AP600 passive systems. The AP1000 is being designed to meet NRC regulatory criteria in a similar manner to that found to be acceptable for the AP600. The AP1000 is being designed to meet NRC deterministic safety criteria and probabilistic risk criteria with large margins.

Westinghouse is certifying the AP1000 standard plant design under the provisions of 10 CFR Part 52. To that end, Westinghouse submitted an application for Design Certification of AP1000 to the United States Nuclear Regulatory Commission (NRC) on March 28, 2002. The NRC staff reviewed the design certification application and issued the Draft Safety Evaluation Report (DSER) on June 16, 2003. The DSER contains more than 170 open items. The NRC staff and Westinghouse have focused on resolution of the identified open items to forward preparation of the Final Safety Evaluation Report.

This final report provides summary information on those areas of:

- Comparisons of the AP1000 design features to the AP600 or operating reactors to demonstrate the proven basis for the AP1000 design features selected
- The design and the safety analyses of the AP1000, that are included in the Design Certification application, which were partially supported by the Nuclear Energy Research Initiative.

A complete description of these areas as they relate to Design Certification is included in the AP1000 Design Control Document (DCD) (APP-GW-GL-700).

A summary comparison of key passive safety system design features is provided in Table 1. These key features are discussed due to their importance in affecting the key thermal-hydraulic phenomenon exhibited by the passive safety systems in critical areas. The scope of some of the design changes to the AP600 is described. These changes are the ones that are important in evaluating the passive plant design features embodied in the certified AP600 standard plant design. These design changes are incorporated into the AP1000 standard plant design that Westinghouse is certifying under 10 CFR Part 52.

The reactor coolant pumps for AP1000 are canned motor pumps similar to those used by the United States naval nuclear program and are part of the AP600 design. A pump specification was developed to meet AP1000 requirements while minimizing the pump design extension from current practice. The steam generators for AP1000 are larger than those for AP600 and a unique specification was prepared for AP1000 thermal hydraulic operating conditions and for the channel head mounted reactor coolant pumps. AP1000 is designed based upon more current versions of the ASME Code and other applicable national consensus standards than AP600.

Safety analyses on LOCA, Non-LOCA and Containment Response were performed for the AP1000 using the AP600 validated analysis codes and models of the AP1000 plant. These analyses are not a complete set of analyses as prescribed by 10 CFR 50. Based on the results of these analysis assessments and others described in the AP1000 DCD, the AP1000 provides large safety margins for its postulated design basis accidents and transient events.

In conclusion, this report describes the results of the representative design certification activities that were partially supported by the Nuclear Energy Research Initiative. These activities are unique to AP1000, but are representative of research activities that must be driven to conclusion to realize successful licensing of the next generation of nuclear power plants in the United States.

Table 1 Comparison of Passive Safety System Design Features

	AP600	AP1000	Comment
Core Makeup Tanks Number Volume, ft^3 Line Resistance, % Design Flow Rate, %	 2 2000 100% 100%	 2 2500 64% 125%	Core makeup tank volume and flow rate is increased to provide additional safety injection flow. CMT elevations are maintained at the AP600 level. The duration of CMT injection is maintained similar to AP600.
Accumulators Number Volume, ft^3 Pressure, psig	 2 2000 700	 2 2000 700	The accumulators are the same as AP600. Accumulator sizing is based on LBLOCA performance and is determined largely on reactor vessel volume. The AP600 employs a 3-loop reactor vessel, while the AP1000 employs a 3XL vessel similar to Doel and Tihange plant. AP1000 LBLOCA performance will be similar to AP600.
IRWST Volume, gallons Water Level, ft Line Resistance Design Flow Rate, %	 557,000 130.00” 100% 100%	 590,000 131.58” 32% 184%	The IRWST level has been increased in the AP1000 by using more accurate level instruments. This permits a higher operating level.
Automatic Depressurization Stages 1-3 Location, Configuration, Vent Area, % Stage 4 Location, Configuration, Line size, nominal Vent Area, % Line Resistance Capacity	 Top pzr 6 paths 100% Hot Leg 4 paths 10-inch 100% 100% 100%	 Top pzr 6 paths 100% Hot Leg 4 paths 14-inch 176% 28% 189%	The first three stages of ADS are the same as AP600. Their sizing basis is to reduce pressure to permit adequate injection from the accumulators and to permit transition to 4^{th} stage ADS. The ADS 4^{th} stage vent area is increased more than the ratio of the core power. The 4^{th} stage ADS venting is the most important design feature to allow for stable IRWST/sump injection during long term core cooling.
Passive RHR Heat Exchanger Type Surface Area, % Design Flow Rate, % Design Heat Transfer, %	 C-Tube 100% 100% 100%	 C-Tube 122% 174% 172%	The AP1000 PRHR HX retains the AP600 configuration. The heat transfer surface area is increased by extending the length of the heat exchanger. The inlet and outlet piping has been increased resulting in higher flow rates.
Containment Diameter, ft Overall Height, ft Shell Thickness, 1A Design Pressure, psig Net Free Volume, ft^3	 130 189.83 1.625 45 1.73 E06	 130 215.33 1.75 59 2.07 E06	The AP1000 containment volume and design pressure are increased to accommodate higher mass and energy releases.

Table 1 Comparison of Passive Safety System Design Features			
	AP600	**AP1000**	**Comment**
Passive Containment Cooling System Water Storage Tank Volume (Top of Overflow), gallons	580,000	800,000	The PCS water storage tank was increased to accommodate higher flow rates. The PCS flow rates have been increased based on the increase in core power.

1 INTRODUCTION

Westinghouse Electric Company has designed an advanced 600 MWe (1933 MWt) nuclear power plant called the AP600. The AP600 uses passive safety systems to enhance plant safety and to satisfy US licensing requirements. The use of passive safety systems provides significant and measurable improvements in plant simplification, safety, reliability, investment protection and plant costs. These systems use only natural forces such as gravity, natural circulation, and compressed gas to provide the driving forces for the systems to adequately cool the reactor core following an accident. The AP600 received Design Certification by the Nuclear Regulatory Commission in December 1999.

Westinghouse has initiated development of the AP1000 standard nuclear reactor design based closely on the AP600 design. The AP1000, with a power output of approximately 1000 Mwe (3400 MWt), maintains the AP600 design configuration, use of proven components and licensing basis by limiting the changes to the AP600 design to as few as possible.

The AP1000 reactor and passive safety features retain the same configuration as the AP600. The capacities of the major reactor components have been increased to support the increased power rating. The approach to designing the passive safety features (core cooling and containment cooling) is to evaluate each feature to determine if changes are necessary to provide proper safety margins at the higher power rating. Preliminary safety evaluations have shown that the AP1000 passive safety systems provide adequate performance during limiting design basis accidents.

Westinghouse submitted an application for Design Certification of AP1000 to the United States Nuclear Regulatory Commission (NRC) on March 28, 2002. This report provides information on those areas of *design features, the design and safety analyses*, that are included in the Design Certification application, which were partially supported by the Nuclear Energy Research Initiative. A complete description of these areas as they relate to Design Certification is included in the AP1000 Design Control Document (DCD) (APP-GW-GL-700).

Section 2 of this report describes some of the AP1000 plant design features on the nuclear island, as well as major differences between the AP1000 and AP600.

AP1000 uses a canned motor pump for its reactor coolant pump. Canned motor pumps are used in the United States naval nuclear program and are part of the AP600 design. The pump and motor size required for AP1000 is an extension from current practice. The plant designers worked with the pump designers to develop a pump specification that met plant requirements while minimizing the pump design extension. Section 3 of this report describes the design efforts performed to date to establish the larger AP1000 canned motor pump design based on that of AP600 design.

AP1000 should be based upon more current versions of the ASME Code and other applicable national consensus standards. Section 4 of this report includes a discussion of which version of the ASME Code will be the bass for AP1000 and what technical basis was provided to NRC for use of this version.

Section 5 of this report describes the design work done for the steam generators for AP1000. They are larger than those for AP600 and are based upon the replacement units for Arkansas Nuclear Unit 1. A unique specification was prepared to account for the AP1000 thermal hydraulic operating conditions and for the channel head mounted reactor coolant pumps.

Safeguards data packages were prepared to support safety analyses and the safety analyses themselves were performed. Section 6 and 7 of this report, respectively, provide examples of the nature and results of these two efforts for the AP1000.

Westinghouse developed analytical computer codes that can predict with adequate certainty, the performance of the AP600 passive safety systems in response to design basis and beyond design basis accidents. These computer codes were validated by the extensive test program conducted by Westinghouse and confirmatory tests and analyses performed by the NRC. They are considered sufficient for use in performing accident analyses in accordance with the requirements of 10 CFR Part 50 and Part 52. The NRC approved to apply these analysis codes to the AP1000 in a letter from J. Lyons to W. E. Cummins dated on March 25, 2002. The AP1000 safety analyses are performed by using the same computer codes in the same manner as those of AP600. Section 8,9 and 10 of this report, respectively, describes safety analyses on LOCA, Non-LOCA and Containment Response. Based on the results of these analysis assessments, it appears that the analysis results for the AP1000 will provide large safety margins for the range of postulated accidents and transient events.

In conclusion, this report describes the results of the representative design certification activities that were partially supported by the Nuclear Energy Research Initiative. These activities are unique to AP1000, but are representative of research activities that must be driven to conclusion to realize successful licensing of the next generation of nuclear power plants in the United States.

2 AP1000 DESIGN DESCRIPTION

2.1 PLANT OVERVIEW

2.1.1 Design Origin and Overall Plant Description

The AP1000 is a two-loop, 1000 MWe pressurized water reactor (PWR) with passive safety features and extensive plant simplifications to enhance the construction, operation, and maintenance. The AP1000 design is derived directly from the AP600, a two-loop, 600 MWe PWR. The AP1000 retains the AP600 approach of using proven PWR technology and safety features that rely on natural forces.

The AP1000 passive safety systems are the same as those for the AP600, except for some changes in component capacities. The 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 alternating current [ac] power, component cooling water, service water, or heating, ventilation, and air-conditioning [HVAC]). The number and approach is to eliminate required operator action rather than to automate it. The net result is a design with reduced complexity and improved operability.

The approach in uprating the AP600 to the AP1000 was to increase the power capability of the plant within the space constraints of the AP600, while retaining the credibility of proven components and substantial safety margins. Therefore, the AP1000 retains the AP600 licensing basis.

Some of the high-level design characteristics of the AP1000 are as follows:

- Net electrical power is approximately 1090 MWe, and nuclear steam supply system (NSSS) thermal power is 3415 MWt.
- Rated performance is achieved with up to 10 percent of the steam generator tubes plugged and with a maximum hot leg temperature of 615°F.
- Major safety systems are passive; they require no operator action for 72 hours after an accident, and maintain core and containment cooling for a protected time without ac power.
- Predicted core damage frequency is similar to AP600 (1.7E-07 / yr) and is well below the 1E-04/ yr requirement. The frequency of significant release is similar to AP600 (1.8E-08/ yr) and is well below the 1E-06/ yr requirement.
- Occupational radiation exposure is expected to be below 0.7 man-Sv / yr (70 man-rem/yr).
- The core is designed for an 18-month fuel cycle.
- Overall plant availability is greater than 93 percent, including forced and planned outages; the goal for unplanned reactor trips is less than one per year.
- The plant is designed to accept a 100-percent load rejection from full power to house loads without reactor trip or operation of the pressurizer or steam generator safety valves. The design provides for a turbine capable of continued stable operation at house loads.

- The plant design objective is 60 years without the planned replacement of the reactor vessel, which itself has a 60-year design objective based on conservative assumptions. The design provides for the replaceability of other major components, including the steam generators.

2.1.2 Plant Design Features

The design approach for the AP1000 was to utilize design features and components that have been proven in currently operating plants or are based on such proven components. The AP1000 incorporates both design features that are the same as in current operating plants, and those that are based heavily on proven technology. The major design features which are based on proven designs in current plants are discussed here. They include the core design, steam generator design, reactor coolant pump motors.

CORE DESIGN

The AP1000 core design incorporates 157 fuel assemblies. This core design is the same as in V.C. Summer, Doel 3, and Tihange 4. The active fuel region in the Doel and Tihange plants is 14 feet, just as in the AP1000. However, the liner power density of the AP1000 core is approximately the same as the V.C. Summer core, although the active length of the V.C. Summer core is only 12 feet. Thus, the Doel and Tihange plants provide operating experience with the 157 assembly core and the longer fuel assembly mechanical design. The V.C. Summer plant provides operating experience with this core arrangement at the higher AP1000 linear power density compared to Doel and Tihange.

STEAM GENERATOR DESIGN

Design Features and Power Rating

The AP1000 steam generator is a vertical U-tube design with a triangular pitch tube arrangement called Delta 125. Many of the design features of the Delta 125 units have been incorporated from the operating replacement Delta 75 and Delta 94 steam generators. Operating experience with these generators has been obtained in the V.C. Summer and Shearon Harris plants (Delta 75) and the South Texas plant (Delta 94). These generators operate at a lower power rating than those of the AP1000. However, the replacement steam generators for the Arkansas Nuclear Unit 1, provide experience in the power range of the AP1000. The steam generators for the San Onofre and Waterford units are also rated at the same 1700 MWt as the AP1000.

Inconel Tubes

In the past, steam generator tube integrity has been linked with tube material and the reactor coolant system hot leg temperature. The AP1000 steam generator design utilizes Inconel-690 tubes and has a hot leg temperature of 615°F. Among the current operating plants which have steam generators with Inconel-690 tubes, three plants have hot leg operating temperatures higher than the temperature proposed for the AP1000.

REACTOR COOLANT PUMP

The AP1000 reactor coolant pump utilizes a hermetically sealed canned motor of proven design. The addition of a uranium alloy flywheel to provide the rotating inertia needed for flow coast-down is based on the design and testing of these features for the AP600 reactor coolant pumps.

The AP1000 pump incorporates the hydraulics scaled down from the hydraulics developed for the Tsuruga 3 and 4 reactor coolant pumps. Thus, the AP1000 reactor coolant pumps are based on components with extensive operating history and previous design work.

2.2 REACTOR SYSTEM DESIGN

The reactor system consists of those major items of equipment constituting the operating nuclear reactor. The system is defined to include the reactor vessel, core, reactor internals, control rod drive mechanisms (CRDM) and the integrated head package. These components are described below.

2.2.1 Core Design

The AP1000 core design consist of 157 17 x 17 fuel assemblies with a 14 foot active fuel length.

2.2.1.1 Fuel Assemblies

Each fuel assembly consists of 264 fuel rods, 24 guide thimbles, and 1 instrumentation tube arranged within a supporting structure. The instrumentation thimble is located in the center position and provides a channel for insertion of an in-core neutron detector, if the fuel assembly is located in an instrumented core position. The guide thimbles provide channels for insertion of either a rod cluster control assembly, depending on the position of the particular fuel assembly in the core. The AP1000 incorporates the Westinghouse ROBUST fuel assembly design that includes guide thimbles with increased wall thickness and an improved grid design.

The fuel rods are loaded into the fuel assembly structure so that there is clearance between the fuel rod ends and the top and bottom nozzles. The fuel rods are supported within the fuel assembly structure by ten grids. The top and bottom grids are fabricated from nickel-chromium-iron Alloy 718, while the intermediate grids are fabricated from Zircaloy-4. Top, bottom, and intermediate grids provide axial and lateral support to the fuel rods. In addition, four intermediate flow mixer (IFM) grids located near the center of the fuel assembly and between the intermediate grids provide additional fuel rod restraint.

The fuel assembly structure consists of a bottom nozzle, top nozzle, fuel rods, guide thimbles, and grids which are discussed below.

FUEL RODS

The fuel rods consist of uranium dioxide ceramic pellets contained in ZIRLO™ tubing, which is plugged and seal-welded at the ends to encapsulate the fuel. The fuel pellets are right circular cylinders consisting of slightly enriched uranium dioxide powder which has been compacted by cold pressing and then sintered to the required density. The ends of each pellet are dished slightly, to allow greater axial expansion at the pellet centerline and to increase the void volume for fission gas release. The ends of each pellet also have a small chamfer at the outer cylindrical surface which improves manufacturability, and mitigates potential pellet damage due to fuel rod handling.

Void volume and clearances are provided within the rods to accommodate fission gases released from the fuel, differential thermal expansion between the clad and the fuel, and fuel density changes during irradiation. To facilitate the extended burnup capability necessitated by longer operating cycles, the fuel rod is designed with two plenums (upper and lower) to accommodate the additional fission gas release. The upper plenum volume is maintained by a fuel pellet hold-down spring. The lower plenum volume is maintained by a stand off assembly.

The AP1000 fuel rod design may also include axial blankets. The axial blankets consist of fuel pellets of a reduced enrichment at each end of the fuel rod pellet stack. Axial blankets reduce neutron leakage axially and improve fuel utilization.

The AP1000 fuel rods include integral fuel burnable absorbers. The integral fuel burnable absorber coated fuel pellets are identical to the enriched uranium dioxide pellets except for the addition of a thin boride coating less than 0.001 inch in thickness on the pellet cylindrical surface. Coated pellets occupy the central portion of the fuel column.

BOTTOM NOZZLE

The bottom nozzle serves as the bottom structural element of the fuel assembly and directs the coolant flow distribution to the assembly. The nozzle is fabricated from Type 304 stainless steel and consists of a perforated plate, and casting which incorporates a skirt and four angle legs with bearing pads. The legs and skirt form a plenum to direct the inlet coolant flow to the fuel assembly. The perforated plate also prevents accidental downward ejection of the fuel rods from the fuel assembly. The bottom nozzle is fastened to the fuel assembly guide thimbles by locked thimble screws, which penetrate through the nozzle and engage with a threaded plug in each guide thimble.

TOP NOZZLE

The top nozzle functions as the upper structural component of the fuel assembly and, in addition, provides a partial protective housing for the rod cluster control assembly, wet annular burnable absorber, or other core components. The basic components of the welded top nozzle include the adapter plate, enclosure, and top plate. The top nozzle assembly includes four sets of hold-down springs and associated spring screws and clamps, which are secured to the top nozzle top plate. The springs are made of nickel-chromium-iron Alloy 718. The other top nozzle components are made of Type 304 stainless steel.

GUIDE THIMBLES

The guide thimble are structural members that provide channels for the neutron absorber rods, burnable absorber rods, neutron source rods, or other assemblies. Each guide thimble is fabricated from Zircaloy-4 or ZIRLO™ tubing having two different diameters. The larger tube diameter at the top section provides a relatively large annular area necessary to permit rapid control rod insertion during a reactor trip, as well as to accommodate the flow of coolant during normal operation. Holes are provided on the guide thimble above the dashpot to reduce the rod drop time. The lower portion of the guide thimble is swaged to a smaller diameter, which results in a dashpot action near the end of the control rod travel during normal trip operation. The dashpot is closed at the bottom by means of an end plug, which is provided with a small flow port to avoid fluid stagnation in the dashpot volume during normal operation.

GRIDS

The fuel rods are supported at intervals along their lengths by grid assemblies which maintain the lateral spacing between the rods throughout the design life of the assembly. Each fuel rod is given support at six contact points within each grid by the combination of support dimples and springs.

The grid assembly consists of individual slotted straps assembled and interlocked into an egg-crate type arrangement with the straps permanently joined at their points of intersection. The straps may contain springs, support dimples, and mixing vanes; or any such combination.

Two types of structural grid assemblies are used on the AP1000 fuel assembly. One type, with mixing vanes projecting from the edges of the straps into the coolant stream, is used in the high heat flux region of the fuel assemblies to promote mixing of the coolant. The other type, located at the top and bottom of the assembly, does not contain mixing vanes on the internal straps.
The outside straps on the grids contain mixing vanes that, in addition to their mixing function, aid in guiding the grids and fuel assemblies past projecting surfaces during handling or during loading and unloading of the core.

2.2.1.2 In-Core Control Components

Reactivity control is provided by neutron absorbing rods, gray rods, burnable absorber rods, and a soluble chemical neutron absorber (boric acid). The boric acid concentration is varied to control long-term reactivity changes such as fuel and burnable absorber depletion, fission product buildup, and zero power reactivity changes.

ROD CLUSTER CONTROL ASSEMBLIES

The rod cluster control assemblies are divided into two categories: control and shutdown. The rod cluster assemblies consists of 24 rodlets fastened at the top end to a common hub or spider. The absorber material used in the control rods is silver-indium-cadmium alloy, which is essentially “black” to thermal neutrons and has sufficient additional resonance absorption to significantly increase worth. The absorber material is in the form of solid bars sealed in cold-worked stainless steel tubes. The material used in the absorber rod end plugs is Type 308 stainless steel.

GRAY ROD CLUSTER ASSEMBLIES

The mechanical design of the gray rod cluster assemblies plus the control rod drive mechanism and the interface with the fuel assemblies and guide thimbles are identical to the rod cluster control assembly. Geometrically, the gray rod cluster assembly is the same as a rod cluster control assembly except that 20 of 24 rodlets are stainless steel while the remaining four contain the same silver-indium-cadmium absorber material clad with stainless steel as the rod cluster control assemblies.

The gray rod cluster assemblies are used in load follow maneuvering and provide a mechanical shim to replace the use of changes in the concentration of soluble boron, that is, a chemical shim, normally used for this purpose. The AP1000 uses 53 rod cluster control assemblies and 16 gray rod cluster assemblies.

BURNABLE ABSORBER ASSEMBLY

Each burnable absorber assembly consists of wet annular burnable absorber rods attached to a hold-down assembly. The wet annular burnable absorber rods consist of annular pellets of alumina-boron carbide material contained within two concentric zirconium alloy tubes. These zirconium alloy tubes, which form the inner and the outer clad for the wet annular burnable absorber rod, are plugged, pressurized with helium, and seal-welded at each end to encapsulate the annular stack of absorber material. The absorber stack length is positioned axially within the wet annular burnable absorber rod by the use of a zirconium alloy bottom-end spacer. An annular plenum is provided within the rod to accommodate and retain the helium gas released from the absorber material as it depletes during irradiation. The reactor coolant flows inside the inner tube and outside the outer tube of the annular rod.

NEUTRON SOURCE ASSEMBLIES

The purpose of a neutron source assembly is to provide a base neutron level to give confidence that the detectors are operational and responding to core multiplication neutrons. The source assembly also permits detection of changes in the core multiplication factor during core loading, refueling, and approach to criticality.

Four source assemblies are typically installed in the reactor core: two primary source assemblies and two secondary source assemblies. Each primary source assembly contains one primary source rod and a number of burnable absorber rods. Each secondary source assembly contains a symmetrical grouping of four to six secondary source rods.

2.2.1.3 Difference Between AP1000 and AP600

The major differences in the AP1000 core design compared to the AP600 core design are the addition of 12 fuel assemblies, an increase in the length of the fuel assemblies, and additional control assemblies. The extra assemblies and increase in length along with an increase in the linear power density in the core enabled the core power rating to be increased from 1,933 MWt to 3,400 MWt within the same diameter reactor vessel. The number of rod control clusters was increased to 53 in the AP1000 compared to 45 in the AP600. The AP1000 core also incorporates the Westinghouse ROBUST fuel assembly design compared to the Vantage 5-H design of the AP600. The ROBUST design includes guide tubes with increased wall thickness.

2.2.2 Reactor Vessel and Internals Design

2.2.2.1 Reactor Vessel

The reactor vessel is cylindrical, with a hemispherical bottom head and removable flanged hemispherical upper head. The vessel inside diameter at the core region is 157 inches. The vessel is fabricated by welding together the lower head, the lower shell and the upper shell. The upper shell contains the penetrations from the inlet and outlet nozzles and direct vessel injection nozzles. The closure head is fabricated with a head dome and bolting flange. The upper head has penetrations for the control rod drive mechanisms, the incore instrumentation, head vent, and support lugs for the integrated head package. The removable flanged hemispherical closure head is attached to the vessel (consisting of the upper shell-lower shell-bottom hemispherical head) by studs. Two metal o-rings are used for sealing the two assemblies. Inner and outer monitor tubes are provided through the upper shell to collect any leakage past the o-rings.

Surfaces are clad to a nominal 0.22 inches of thickness with stainless steel welded overlay which includes the upper shell top surface but not the stud holes. The AP1000 reactor vessel’s design objective is to withstand the design environment of 2500 psi and 650°F for 60 years. An evaluation of the reactor vessel fluence indicates that the increase in fluence from the AP600 to the AP1000 will not have a major impact on the AP1000 reactor vessel.

The closure head has a 77.5-inch inner spherical radius and a 188.0-inch O.D. outer flange. Cladding is extended across the bottom of the flange for refueling purposes. Forty-five, seven-inch diameter studs attach the head to the lower vessel and two metal o-rings are used for sealing. There are 69 penetrations in the removable flanged hemispherical head (closure head) that are used to provide access for the control rod drive mechanisms. Each control rod drive mechanism is positioned in its opening and welded to the closure head penetration. In addition there are penetrations in the closure head used to provide access for in-core and core exit instrumentation. A tube is inserted into each of the penetrations and is welded to the closure head penetration.

The vessel upper shell is a large ring forging. Included in this forging are four 22-inch inner diameter inlet nozzles, two 31-inch inner diameter outlet nozzles and two 6.81-inch inner diameter direct vessel injection nozzles (8-inch schedule 160 pipe connections). These nozzles are forged into the ring or are fabricated by "set in" construction. The inlet and outlet nozzles are offset axially in different planes by 17.5 inches. This offset allows pump maintenance without discharging the core. The injection nozzles are 100 inches down from the main flange and the outlet nozzles are 80 inches down and the inlet nozzles are 62.5 inches below the mating surface.

There are no penetrations in the reactor vessel below the core. This eliminates the possibility of a loss-of-coolant accident by leakage from the reactor vessel that would allow the core to be uncovered.

2.2.2.2 Reactor Vessel Internals Design

The reactor internals are the structural assemblies that support the core within the reactor vessel and provide the proper flow path for the circulation of the coolant through the core. The internals consist of two major assemblies: the lower internals and the upper internals.

The lower internals consists of the core barrel, lower support plate, vortex suppression plate, radial reflector, radial support and the related attachment hardware. During reactor operation the core barrel serves to direct the coolant flow from the reactor vessel inlet nozzles through the downcomer annulus, and into the lower plenum below the lower support plate. The flow then turns and passes upward through the lower core support plate into the core region.

The upper internals assembly consists of the upper support plate, upper support columns, upper core plate, guide tubes and the related attachment hardware. During operation coolant flows up from the core through the upper core plate and out through the outlet nozzles.

2.2.2.3 Integrated Head Package Design

The purpose of the integral head package is to help reduce the outage time and minimize personnel radiation exposure by combining operations associated with movement of the reactor vessel head during the refueling outage.

The integrated head package consists of the following components:

- Shroud Assembly and Cooling System
- Lifting System
- Mechanism Seismic Support Structure
- Messenger Tray and Cable Support Structure
- Cable Bridge
- Cables
- Incore Instrumentation Conduits

With the integrated head package concept, the control rod drive mechanisms (CRDMs), and rod position indicators (RPI) remain with the reactor vessel head within the cooling shroud assembly at all times. The shroud assembly is a carbon steel structure which encloses the CRDMs above the reactor head. During normal operation it provides for flow of cooling air for the CRDMs and RPI coil stacks.

2.2.2.4 Differences Between AP1000 and AP600

The AP1000 reactor vessel has the same overall diameter and number and size of nozzles as the AP600 vessel. The overall length of the AP1000 vessel has been increased to accommodate the increase in core length to 14 feet. The AP1000 reactor vessel internals are of the same design as the AP600 vessel internals except that the length of the lower internals has increased because of the longer core design. Also, the thickness of the lower support plate has increased to accommodate the heavier AP1000 core which has both additional fuel assemblies (12) and heavier assemblies due to the longer length. AP1000 reactor vessel internals design does not include a radial reflector concept similar to that of the AP600. A reflector is not required to achieve a 60 year reactor vessel life.

The AP1000 integrated head package design is the same as that of the AP600 except that the overall height has increased to accommodate the longer control rod drives and incore components required for the 14-foot AP1000 core. Internally, the AP1000 integrated head package also accommodates an additional eight control rod assemblies.

2.3 REACTOR COOLANT SYSTEM DESIGN

2.3.1 Overall System Design

The reactor coolant system consists of two heat transfer circuits, each with a steam generator, two reactor coolant pumps, a single hot leg and two cold legs, for circulating reactor coolant between the reactor and the steam generators. In addition, the system includes a pressurizer, automatic depressurization system, interconnecting piping, valves, and instrumentation necessary for operational control and safeguards actuation. The automatic depressurization system consists of four different valve stages that open sequentially to reduce reactor coolant system pressure so that long term cooling can be provided from the passive core cooling system. All system equipment is located in the reactor containment.

The reactor coolant system includes the following:

- The reactor vessel, including control rod drive mechanism housings.
- The reactor coolant pumps, consisting of four canned motor pumps that pump fluid through the entire reactor coolant and reactor systems and two pumps that are coupled with each steam generator.
- The portion of the steam generators containing reactor coolant, including the channel head, tubesheet, and tubes.
- The pressurizer which is attached by the surge line to one of the reactor coolant hot legs with a combined steam and water volume, the pressurizer maintains the reactor system within a narrow pressure range.
- The safety and automatic depressurization system valves.

- The reactor vessel head vent isolation valves.
- The interconnecting piping and fittings between the preceding principal components.
- The piping, fittings, and valves leading to connecting auxiliary or support systems.

The major components are described in more detail below.

2.3.2 Steam Generator Design

2.3.2.1 Design Bases

The AP1000 utilizes a Model Delta-125 steam generator. This generator design is base on the following proven designs:

- Delta 75 – Replacement steam generator for V.C. Summer plant and other plants
- Delta 94 – Replacement steam generator for South Texas plant
- ANO (Arkansas) – Replacement steam generator (1500 MWt per steam generator)
- San Onofre and Waterford – Steam generator capacities similar to the 1700 MWt capacity of the AP1000 generators

2.3.2.2 Design Description

The AP1000 steam generator is a vertical-shell U-tube evaporator with a triangular pitch tube bundle and integral moisture separating equipment.

On the primary side, the reactor coolant flow enters the primary chamber via the hot leg nozzle. The lower portion of the primary chamber is spherical and merges into a cylindrical portion, which mates to the tubesheet. This arrangement provides enhanced access to all tubes, including those at the periphery of the bundle, with robotics equipment. The head is divided into inlet and outlet chambers by a vertical divider plate extending from the apex of the head to the tubesheet.

The reactor coolant flow enters the inverted U-tubes, transferring heat to the secondary side during its traverse, and returns to the cold leg side of the primary chamber. The flow exits the steam generator via two cold leg nozzles to which the canned-motor reactor coolant pumps are directly attached.

A passive residual heat removal (PRHR) nozzle attaches to the bottom of the channel head of the loop 1 steam generator on the cold leg portion of the head. This nozzle provides recirculated flow from the passive residual heat removal heat exchanger to cool the primary side under emergency conditions.

The steam generator channel head has provisions to drain the head. To minimize deposits of radioactive corrosion products on the channel head surfaces and to enhance the decontamination of these surfaces, the channel head cladding is machined or electropolished for a smooth surface.

The tubes are fabricated of nickel-chromium-iron Alloy 690. The tubes undergo thermal treatment following tube-forming operations. The tubes are tack-rolled, welded, and hydraulically expanded essentially over the full depth of the tubesheet. Westinghouse has used this practice in F-type steam generators. The process was selected because of its capability to control secondary water ingress to the tube-to-tube-sheet crevice. Residual stresses smaller than from other expansion methods result from this process and are minimized by tight control of the pre-expansion clearance between the tube and tubesheet hole.

Support of the tubes is provided by ferritic stainless steel tube support plates. The holes in the tube support plates are broached with a hole geometry to promote high-velocity flow along the tube and to provide an appropriate interface between the tube support plate and the tube. Anti-vibration bars installed in the U-bend portion of the tube bundle minimize the potential for excessive vibration.

The Delta-125 steam generator incorporates a separate startup feedwater nozzle. The startup feedwater nozzle is located at an elevation that is just below the main feedwater nozzle and is circumferentially rotated 60 degrees clockwise with respect to the main feedwater nozzle. A spray system independent of the main feedwater feedring is used to introduce startup feedwater into the steam generator.

2.3.2.3 Difference Between AP1000 and AP600

There are differences between the Delta-75 steam generators used in the AP600 and the Delta-125 steam generators used in the AP1000 both in number of tubes and size of the steam generator shell. Both units are vertical-shell U-tube evaporators with a triangular pitch tube bundle and integral moisture separating equipment. To accommodate the higher thermal output of the AP1000 more heat transfer surface is required, thus increasing the shell diameter and height to enclose the larger tube bundle and larger moisture separation equipment required for the higher steam flow.

The secondary side volume of the Delta-125 steam generator is also larger than that of the Delta-75. This increased water mass results in a greater heat transfer capability from the reactor coolant system and is credited at the time of a design basis accident.

2.3.3 Reactor Coolant Pump Design

2.3.3.1 Design Description

The AP1000 reactor coolant pump is a single stage, hermetically sealed, high-inertia, centrifugal canned-motor pump. The AP1000 pump is based on the AP600 canned-motor pump design with provisions to provide more flow and a longer flow coast down. The motor size is minimized through the use of a variable speed controller to reduce motor power requirements during cold coolant conditions. The variable speed controller is not used during power operations. The pump hydraulics are based on the high-efficiency hydraulics developed and tested for the Tsuruga 3 and 4 reactor coolant pumps.

A canned motor pump contains the motor and all rotating components inside a pressure vessel. The pressure vessel consists of the pump casing, thermal barrier, stator shell, and stator cap, which are designed for full reactor coolant system pressure. The stator and rotor are encased in corrosion-resistant cans that prevent contact of the rotor bars and stator windings by the reactor coolant. Because the shaft for the impeller and rotor is contained within the pressure boundary, seals are not required to restrict leakage out of the pump into containment. A gasket and canopy seal type connection between the pump casing, the stator flange, and the thermal barrier is provided. This design provides definitive leak protection for the pump closure. To access the internals of the pump and motor, the canopy seal weld is severed. When the pump is reassembled a canopy seal is rewelded. Canned-motor reactor coolant pumps have a long history of safe, reliable performance in military and commercial nuclear plant service.

The reactor coolant pump driving motor is a vertical, water-cooled, squirrel-cage induction motor with a canned rotor and a canned stator. The motor is cooled by component cooling water circulating through a cooling jacket on the outside of the motor housing and through a thermal barrier between the pump casing and the rest of the motor internals.

Flywheel assemblies provide rotating inertia that increases the coastdown time for the pump. Each flywheel assembly is a composite of a uranium alloy flywheel casting or forging contained within a welded nickel-chromium-iron alloy enclosure. Surrounding the flywheel assembly is the thick cylindrical motor end closure and the heavy wall of the stator shell and main flange.

The materials in contact with the reactor coolant and cooling water (with the exception of the bearing material) are austenitic stainless steel, nickel-chromium-alloy, or equivalent corrosion-resistant material.

2.3.3.2 Differences Between AP1000 and AP600

The same basic canned-motor pump design is employed in the AP1000 as in the AP600. However, the higher thermal power and core power density of the AP1000 requires higher flow and longer coastdown from the AP1000 pumps compared to the AP600 pumps. A variable speed controller was added to the AP1000 pumps to reduce the motor power required when pumping cold reactor coolant. To provide the larger flow rates, the AP1000 pumps include high efficiency hydraulics which were scaled down from Tsuruga 3 and 4 reactor coolant pump design. A longer coastdown is obtained in the AP1000 pumps through increased inertia in the flywheel.

2.4 Containment Vessel Design

The containment vessel is an ASME metal containment. It serves both to limit releases in the event of an accident and to provide the safety-related ultimate heat sink. It is a free-standing, cylindrical steel vessel with ellipsoidal upper and lower heads. The containment vessel has the following design characteristics:

Diameter: 130feet
Height: 215 feet 4 inches
Design Code: 1998 edition of the ASME Code (with 1999 and 2000 addenda), Section III, Div. 1
Material: SA738 Grade B
Design Pressure: 59 psig
Design Temperature: 300°F
Design External Pressure: 2.9 psid

The wall thickness of the cylinder and the heads is 1.75 inches. The heads are ellipsoidal with a major diameter of 130 feet and a height of 37 feet 7.5 inches.

The containment vessel supports most of the containment air baffle. The air baffle is arranged to permit inspection of the exterior surface of the containment vessel. Flow distribution weirs are welded to the dome as part of the water distribution system of the passive containment cooling system.

2.4.1 Containment Vessel Support

The bottom head is embedded in concrete, with concrete up to elevation 100 ft (grade level) on the outside and approximately elevation 108 ft on the inside. The containment vessel is assumed as an independent, free-standing structure above elevation 100 ft. The thickness of the lower head is the same as that of the upper head. There is no reduction in shell thickness even though credit could be taken for the concrete encasement of the lower head.

Vertical and lateral loads on the containment vessel and internal structures are transferred to the basemat below the vessel by friction and bearing. Seals are provided at the top of the concrete on the inside and outside of the vessel to prevent moisture between the vessel and concrete.

2.4.2 Coatings

The containment vessel is coated with an inorganic zinc coating, except for those portions fully embedded in concrete. The inside of the vessel below the operating floor and up to 8 feet above the operating floor also has a phenolic top coat. Below elevation 100 ft the vessel is fully embedded in concrete with the exception of the few penetrations at low elevations. Embedding the steel vessel in concrete protects the steel from corrosion.

The exterior of the vessel is embedded at elevation 100 ft and concrete is placed against the inside of the vessel up to elevation 107 ft-2 in. Above this elevation the inside and outside of the containment vessel are accessible for inspection of the coating. The vessel is coated with an inorganic zinc primer to a level just below the concrete.

2.4.3 Differences Between AP1000 and AP600

Although there are several differences between the AP1000 and AP600 containment vessels, the diameter of the vessel remains unchanged. The containment free volume was increased to 2.07×10^6 ft^3 in the AP1000 by increasing the containment vessel overall height to 215 ft-4 in.

The vessel pressure capability was increased by changing the vessel material and increasing the wall thickness of the vessel.

3 REACTOR COOLANT PUMP DESIGN SPECIFICATION

3.1 TASK SUMMARY

There are four reactor coolant pumps in the AP1000 with two pumps integrally attached to each of two steam generator channel heads. To control the AP1000 reactor coolant pump design efforts and develop the required pump parameters for use in both safety analyses and in parameter tables in the AP1000 Design Control Document (DCD), a pump design specification has been developed. This specification is a markup of the AP600 reactor coolant pump specification has been developed. This specification is a markup of the AP600 reactor coolant pump specification to account for differences in specific design requirements between AP600 and AP1000. The major areas of changes in the design specification between AP600 and AP1000 are:

- Design flow
- Design head
- Flywheel assembly inertia
- Addition of variable frequency drive
- Motor voltage

3.1.1 Design Flow and Head

The higher core power in the AP1000 requires increased reactor coolant flow compared to the AP600. The design pump head also increases because of the increased pressure drop through both the steam generators and reactor vessel. The best estimated flow for AP1000 is 78,750 gpm/pump and pump developed head for AP1000 is 365 ft.

3.1.2 Flywheel Assembly Inertia

The AP1000 pump coastdown characteristics are maintained the same as the AP600. Since the AP1000 pump normal flowrate is higher and the system pressure drops higher, more inertia is required in the flywheel assembly to maintain the same coastdown curve as for the AP600. The required flywheel assembly inertia for the AP1000 pump is 16,500 lb-ft^2.

3.1.3 Variable Frequency Drive

The AP1000 pump motor size is minimized through the use of a variable frequency drive to provide speed control in order to reduce motor power requirements during pump startup from cold conditions. The variable frequency drive is used only during heatup and cooldown when the reactor coolant system temperature is less than 450°F. During power operations the drive is isolated and the pump is run at constant speed.

3.1.4 Motor Voltage

The design motor voltage for the AP1000 reactor coolant pumps has been increased to 6600 volts. The voltage was increased to enable a more efficient and smaller pump motor design for the AP1000.

3.2 DOCUMENTS AND DESIGN CONTROL DOCUMENT

The initial issue of the AP1000 pump specification is :

APP-MP01-M2-001, Revision A, "AP1000 RCP Design Specification"

Subsequent meetings with the AP1000 pump designers and evolution of the AP1000 plant design have resulted in additional changes reactor coolant pump parameter changes which are documented in:

APP-MP01-Z0R-001, Revision 0, "AP1000 RCP Hydraulic/Flywheel/Missile Containment/Critical Speed Evaluations"

The following sections of the AP1000 DCD contain information relative to the reactor coolant pump:

- Section 5.1.3.3 Reactor Coolant Pumps – General Description
- Section 5.4.1 Reactor Coolant Pump Assembly – Design Basis, Pump Assembly description, Design Evaluation, and Test and Inspections

4 ASME CODE VALIDATION FOR THE CONTAINMENT VESSEL

4.1 TASK SUMMARY

This task developed key features of the design for the AP1000 containment vessel. The task evaluated the ASME design rules for containment vessels and the suitability of alternate higher strength materials. Based on these studies SA738, Grade B was selected and design was specified to the 1998 edition of the ASME Code, Section III, Subsection NE, Metal Containment, including the 1999 and 2000 Addenda. A code case was requested and the material has now been approved for containment vessels by Code Case N655. A change is being processed by ASME to include the material for containment vessels in the 2002 edition of the ASME code.

This task included a series of preliminary evaluations of the AP1000 containment vessel to support selection of the material and to provide information for inclusion in the AP1000 Design Control Document (DCD). The vessel was designed by ASME code rules for internal and external pressure and penetration reinforcement thickness was established for major openings. Finite element axisymmetric models were developed for internal pressure analyses and for seismic analyses.

4.2 DOCUMENTS AND DESIGN CONTROL DOCUMENT

The initial issue of the AP1000 containment vessel design specification is:

APP-MV50-Z0-001, Revision A, "AP1000 Containment Vessel Design Specification"

The preliminary studies of the containment vessel design are described in:

APP-MV50-S3R-002, Revision A, "AP1000 Containment Vessel Design Report"

The initial issues of the containment vessel outline drawings are:

APP-MV50=V1-001 to 007, Revision 0

The AP1000 DCD Section 3.8.2 contains information relative to the containment vessel.

5 STEAM GENERATOR DESIGN SPECIFICATION

5.1 TASK SUMMARY

The AP1000 Delta 125 steam generator has been configured to incorporate the most recent advancements in steam generator design, evolving from the AP600 steam generator. The advancements have been directed to meet the performance and maintenance requirements the AP1000 plant, including its 60-year design life. The principal areas of focus have been: thermal/hydraulic analysis, design layout, and integration with the plant.

5.1.1 Thermal/Hydraulic Design

The AP1000 tube bundle was reconfigured to meet the power level requirements associated with the change from the AP600 to AP1000. The surface area of the AP1000 steam generator was increased to 123,540 ft^2. Associated with the steam pressure reduction in the AP1000 were reductions to the upper shell by approximately 10” in size. The reduction in upper shell size led to re-evaluation of the moisture separator arrangement and to additional evaluations of steam inlet qualities into the primary and secondary separators. Moisture carryover evaluations were performed utilizing recent steam generator results to provide a prediction of AP1000 carryover values.

In addition, evaluations were performed of the required tube spans to avoid tube vibration in the straight leg region. Modifications to the AP1000 SG design led to a selection of ten tube support plates, evenly spaced from the secondary face of the tubesheet.

5.1.2 Design Layout

The layout of the AP1000 steam generators was updated to conform it to the AP1000 cubicle arrangement and to required piping layouts. The location of the channel head weld was revised to position the weld assembly of the channel head to the tubesheet further away from the primary face of the tubesheet. This provides an ISI enhancement, and reduces heat input to the tubesheet during postweld heat treatment. The channel head was modified to increase the spacing between the pump nozzles, which was necessary to accommodate an increase in the pump casing diameter. In addition, the CVS nozzle in the channel head was further design, and several fabrication options were considered in configuring its attachment.

Scoping analyses were made of the channel head loading, and weight and center-of-gravity calculations for the unit were completed. Following calculations of basic shell thicknesses, a seismic model was completed to provide stiffnesses required for loop modeling.

5.1.3 Design and Maintenance Enhancements

Design and maintenance enhancements were made in several areas. Additional input was solicited from plants which have recently implemented very large (20”) primary manways, and on this basis, it was recommended that the primary manway be reduced to and 18” size.

A recirculation nozzle was added in the upper shell region to provide for ingress of chemicals to support SG cooldown, wet layup, and chemical cleaning activities. Inspection openings were reviewed for their suitability for the increased tube bundle size. A groove on the secondary side of the tubesheet provides for rnhanced ability for tubesheet sludge lancing and drainage. The drain locations and sizes were re-evaluated, as well as the blowdown size.

5.2 DOCUMENTS AND DESIGN CONTROL DOCUMENTS

Calculations supporting safety analysis related to the steam generator are:

APP-SGS-M3C-001, Revision 1, "Development of AP1000 Delta-125 Version H Steam Generator Model Performance Predictions"

APP-SGS-M3C-003, Revision 0, "AP1000 Steam Generator Center of Gravity – Preliminary Estimate"

APP-SGS-M3C-005, Revision 0, "AP600 and AP1000 Steam Generator Tube Metal Heat Capacity

APP-SGS-M3C-006, Revision 0, "Tube Schedule for AP1000 Delta-125 Steam Generators"

APP-SGS-M3C-007, Revision 0, "AP1000 Delta-125 Steam Generator Moisture Separatoe Evaluations"

APP-SGS-M3C-008, Revision 0, "ATHOS 3-D Thermal-Hydraulic Analysis of the AP1000 Delta-125 Steam Generator"

APP-SGS-M3C-011, Revision 0, "AP1000 Bounding Calculation of Main Steam Volumes on Nuclear Island"

APP-SGS-M3C-012, Revision 0, "AP1000 Steam Generator Local Shell Stiffness for Concentrated Moment Loading Preliminary Estimate"

The following sections of the AP1000 DCD contain information relative to the steam generators:

- Section 5.1.3.2 AP1000 Steam Generator
- Section 5.4.2 Steam Generators

6 SAFEGUARDS DATA PACKAGES FOR ANALYSIS ACTIVITIES

6.1 TASK SUMMARY

Westinghouse performed the complete set of Chapter 15 accident analysis to submit to the U.S. Nuclear Regulatory Commission in the application for Design Certification of the AP1000. This task was to perform the necessary engineering calculations, analyses and assessments to collect the inputs necessary to perform the accident analyses. These inputs are developed in accordance with proper regulatory and quality assurance procedures, and represent the engineering data that represent the AP1000 standard plant.

The AP1000 Plant Parameters document includes relevant plant information that is used as inputs to the AP1000 accident analysis. It is under formal change control and includes the following:

Overview of Plant Parameters – Major NSSS and BOP parameters including major component information related to the reactor coolant pumps, steam generators, pressurizer, overpressure components, loop piping, turbine and generator, key auxiliary system information.

Core and Core Component Parameters – Significant core design information including limiting peaking factors, core loading and fuel assembly parameters. Also includes design information related to numbers of control rods, gray rods, and core support structures.

Performance Parameters – Key NSSS performance parameters including operating pressures, temperatures, loop flow rates, pressure drop information, steam pressure and temperature provided to the turbine.

Vessel Parameters – Significant vessel geometric data, pressure drop information and opearting temperatures.

Reactor Coolant Pump Design Data – Homologous curve information including pump head, flow rate, torque and speed data.

Building Design Parameters – Collection of miscellaneous structural and site-related information including seismic design class for various buildings, key dimensions and design data for containment and applicable site parameters.

Passive Safety System Design Information – Includes passive system design information such as system piping resistance data, system flow rates, tank dimensions and volumes, heat exchanger performance characteristics, and valve characteristics.

6.2 DOCUMENTS

The AP1000 Plant Parameters document has been issued as APP-GW-G0-002. This document is prepared and controlled in accordance with 10 CFR 50 Appendix B.

7 SAFETY CODE AND MODEL DEVELOPMENT

7.1 TASK SUMMARY

This task was to develop the computer accident analysis codes used in the various Chapter 15 accident analyses. Summaries of some of the principal computer codes used in the AP1000 accident analyses are given as follows.

7.1.1 FACTRAN Computer Code

FACTRAN calculates the transient temperature distribution in a cross section of a metal-clad UO2 fuel rod and the transient heat flux at the surface of the cladding using as input the nuclear power and the time-dependent coolant parameters (pressure, flow, temperature, and density). The code uses a fuel model which simultaneously exhibits the features to handle fast transients such as rod ejection accidents, and to handle post-DNB transients: film boiling heat transfer correlations, zircaloy-water reaction, and partial melting of the materials.

7.1.2 LOFTRAN Computer Code

The LOFTRAN program is used for studies of transient response of a pressurized water reactor system to specified perturbations in process parameters. LOFTRAN simulates a multiloop system by a model containing reactor vessel, hot and cold leg piping, steam generator (tube and shell sides), and pressurizer. The pressurizer heaters, spray, and safety valves are also considered in the program. Point model neutron kinetics, and reactivity effects of the moderator, fuel, boron, and rods are included. The secondary side of the steam generator uses a homogeneous, saturated mixture for the thermal transients and a water level correlation for indication and control. The protection and safety monitoring system is simulated to include reactor trips on high neutron flux, overtemperature ?T, high and low pressure, low flow, and high pressurizer level. Control systems are also simulated, including rod control, steam dump, feedwater control, and pressurizer level and pressure control. The emergency core cooling system, including the accumulators, is also modeled.

LOFTRAN is a versatile program suited to both accident evaluation and control studies as well as parameter sizing. LOFTRAN also has the capability of calculating the transient value of DNBR based on the input from the core limits.

The LOFTRAN code is modified to allow the simulation of the passive residual heat removal (PRHR) heat exchanger, core makeup tanks, and associated protection and safety monitoring system actuation logic. LOFTTR2 is a modified version of LOFTRAN with a more realistic break flow model, a two-region steam generator secondary side, and an improved capability to simulate operator actions during a steam generator tube rupture (SGTR) event. These modifications were approved by NRC for application to AP1000.

7.1.3 TWINKLE Computer Code

The TWINKLE program is a multidimensional spatial neutron kinetics code, which is patterned after steady-state codes currently used for reactor core design. This is the standard, NRC approved, Westinghouse neutron kinetic code. The code uses an implicit finite-difference method to solve the two-group transient neutron diffusion equations in one, two, and three dimensions. The code uses six delayed neutron groups and contains a detailed multiregion fuel-clad-coolant heat transfer model for calculating pointwise Doppler and moderator feedback effects. Aside from basic cross-section data and thermal-hydraulic parameters, the code accepts as input basic driving functions, such as inlet temperature, pressure, flow, boron concentration, control rod motion, and others. Various edits are provided (for example, channelwise power, axial offset, enthalpy, volumetric surge, point-wise power, and fuel temperatures). The TWINKLE code is used to predict the kinetic behavior of a reactor for transients that cause a major perturbation in the spatial neutron flux distribution.

7.1.4 COAST Computer Program

The COAST computer program is used to calculate the reactor coolant flow coastdown transient for any combination of active and inactive pumps and forward or reverse flow in the hot or cold legs. The equations of conservation of momentum are written for each of the flow paths of the COAST model assuming unsteady one-dimensional flow of an incompressible fluid. Pressure losses due to friction, and geometric losses are assumed proportional to the flow velocity squared. Pump dynamics are modeled using a head-flow curve for a pump at full speed and using four-quadrant curves, which are parametric diagrams of pump head and torque on coordinates of speed versus flow, for a pump at other than full speed.

7.1.5 NOTRUMP

The NOTRUMP computer code is used in the analysis of LOCAs due to small-breaks in the reactor coolant system. The NOTRUMP computer code is a one-dimensional, general network code, which includes a number of advanced features. Among these features are the calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. The version of NOTRUMP used in AP1000 small-break LOCA calculations has been validated against applicable passive plant test data.

7.1.6 WCOBRA/TRAC

Westinghouse applies the WCOBRA/TRAC computer code to perform best-estimate large-break LOCA analyses in compliance with 10 CFR 50. WCOBRA/TRAC is a thermal-hydraulic computer code that calculates realistic fluid conditions in a PWR during the blowdown and reflood of a postulated large-breake LOCA.

The AP1000 safety-related systems are designed to provide adequate cooling of the reactor indefinitely. Following a LOCA in the long-term, the water in containment rises in temperature toward the saturation temperature. Long-term heat removal from the reactor and containment is by heat transfer through the containment shell to atmosphere. To support the AP1000 LOCA analysis, WCOBRA/TRAC is also used to perform a long-term cooling analysis to demonstrate that the passive systems provide adequate emergency core cooling system performance during the IRWST injection/containment recirculation time scale. The long-term cooling analysis is performed to verify that the passive injection system is providing sufficient flow to the reactor vessel to cool the core and to preclude boron precipitation.

The AP1000 long-term cooling analysis is supported by the series of tests at the Oregon State University APEX Test Facility. This test facility is designed to represent the AP600 reactor safety-related systems and nonsafety-related systems at quarter-scale during long-term cooling. The data obtained during testing at this facility has been shown to apply to the AP1000. These tests were modeled using WCOBRA/TRAC with an equivalent noding scheme to that used for AP1000 in order to validate the code for long-term cooling analysis.

7.2 DOCUMENTS

Under this task, the use of these analysis codes and their applicability to the AP1000 was documented in APP-GW-GSC-003, AP1000 Code Applicability Report. This report was submitted to the US NRC as topical report WCAP-15644. The NRC approved the use of the AP1000 analysis codes.

8 LOCA ANALYSES

8.1 ANALYSES SUMMARY

A loss-of-coolant accident (LOCA) which would resulting from a spectrum of postulated piping breaks within the reactor coolant system pressure boundary was analyzed. For the analyses reported here, a major pipe break (large break) is defined as a rupture with a total cross-sectional area equal to or greater than 1.0 ft^2. This event is not expected to occur during the lifetime of the plant but is postulated as a conservative design basis.

A minor pipe break (small break) is defined as a rupture of the reactor coolant pressure boundary with a total cross-sectional area less than 1.0 ft^2 in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. This event is an infrequent fault that may occur during the life of the plant.

The acceptance criteria for the LOCA are described in 10 CFR 50.46 as follows:

- The calculated maximum fuel element cladding temperature shall not exceed 2200°F.
- Localized cladding oxidation shall not exceed 17 percent of the total cladding thickness before oxidation.
- The amount of hydrogen generated from fuel element cladding reacting chemically with water or steam shall not exceed 1 percent of the total amount if all metal cladding were to react.
- The core remains amenable to cooling for any calculated change in core geometry.
- The core temperature is maintained at a low value, and decay heat is removed for the extended period of time required by the long-lived radioactivity remaining in the core.

For the AP1000, the small breaks yield results with more margin than large breaks.

8.2 Basis and Methodology for LOCA Analyses

Should a major break occur, depressurization of the reactor coolant system results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low-pressure trip setpoint is reached. A safeguards actuation (“S”) signal is generated when the appropriate setpoint is reached. These measures limit the consequences of the accident in two ways:

- Reactor trip and borated water injection complement void formation in causing rapid reduction of power to a residual level corresponding to fission product decay heat. Insertion of control rods to shut down the reactor is neglected in the large-break analysis.
- Injection of borated water provides core cooling and prevents excessive cladding temperatures.

8.2.1 Description of Large-break LOCA Transient

Before the break occurs, the unit is in an equilibrium condition. At the beginning of the blowdown phase, the entire reactor coolant system contains subcooled liquid, which transfers heat from the core by forced convection with some fully developed nucleate boiling. After the break, the core heat transfer is based upon local fluid conditions. Transition boiling and dispersed flow boiling are the major heat transfer mechanism.

The heat transfer between the reactor coolant system and the secondary system may be in either direction, depending upon the relative temperatures. In the case of continued heat addition to the secondary, secondary system pressure increases and the main steam safety valves may lift to limit the pressure. The safety injection signal actuates a feedwater isolation signal, which isolates normal feedwater flow by closing the main feedwater isolation valves.

The reactor coolant pumps trip automatically during the accident following an “S” signal. The effects of pump coastdown are included in the blowdown. The blowdown phase of the transient ends when the reactor coolant system pressure (Initially assumed at 2250 psia) falls to a value approaching that of the containment atmosphere.

When the “S” signal occurs, the core makeup tank valves in the cold leg pressure balance line are opened. The core makeup tank begins to inject subcooled borated water into the reactor vessel through the direct vessel injection lines.

Calculations show that the effective post-LOCA long-term cooling is provided by passive means.

8.2.2 Description of Small-break LOCA Transient

During a small-break LOCA, the AP1000 reactor coolant system depressurizes to the pressurerizer low-pressure setpoint, actuating a reactor trip signal. The passive safety design approach of the AP1000 is to depressurize the reactor coolant system if the break or leak is greater than the makeup capability of the charging system. The passive core cooling system is aligned for delivery following the generation of an “S” signal when the pressurizer low-pressure stepoint is reached. The passive core cooling system includes two core makeup tanks, two accumulators, a large IRWST, and the PRHR heat exchanger.

The core makeup tanks are located above the reactor coolant loops, and they provide high-pressure safety injection in the event of a small-break LOCA. The injection is provided by gravity head of the colder water in the core makeup tanks.

The pressurized accumulators provide additional borated water to the reactor coolant system in the event of a LOCA. Nominally, these 2000-ft^3 tanks are filled with 1700 ft^3 of water and 300 ft^3 of nitrogen at an initial pressure of 700 psig.

The IRWST at a minimum provides an additional 78,900 ft^3 of borated water for long-term core cooling. To attain injection from the IRWST, the reactor coolant system pressure must be lowered to approximately 13 psi above containment pressure. For this pressure to be achieved during a small-break LOCA, the ADS system is initiated. When the system is depressurized below the IRWST delivery pressure, flow from the IRWST continues to maintain the core in a coolable state.

The ADS consists of a series of valves, connected to the pressurizer and hot legs, which provide a phased depressurization of the reactor coolant system. When the level in the core makeup tank drops to the 67.5-percent level, the ADS valves open to accelerate the reactor coolant system depressurization rate. The ADS Stage 1 4-inch valves open at the 67.5-percent level; the 8-inch Stage 2 and the 8-inch Stage 3 valves open in a timed sequence thereafter. The flow from the first three stages of the ADS is discharged into the IRWST through a sparger system. The fourth stages of the ADS are connected to the reactor coolant system hot legs and discharge to containment atmosphere. The ADS Stage 4 valves are activated when the core makeup tank level reaches the 20-percent level.

Calculations indicate that acceptable core cooling is provided for the small-break LOCA transients and show that the effective post-LOCA long-term cooling is provided by passive means.

8.3 Radiological Consequences

Although the analysis of the core response during a LOCA shows that core integrity is maintained, for the evaluation of the radoilogical consequences of the accident, it is assumed that major core degradation and melting occur. The release of activity to the containment consists of two parts. The initial release is the activity contained in the reactor coolant system. This is followed by the release of core activity.

The reactor coolant is assumed to have activity levels consistent with operation at the Technical Specification limits of 280 µCi/gm dose equivalent Xe-133 and 1.0 µCi/gm dose equivalent I-131. The AP1000 is a leak-before-break plant, and water in the reactor coolant system is assumed to blow down into the containment over a period of 10 minutes based on NUREG-1465. As the reactor coolant enters the containment, the noble gases and half of the iodine activity are assumed to be released into the containment atmosphere.

The release of activity from fuel takes place in two stages. First is the gap release which is assumed to occur at the end of the primary coolant release phase (i.e., at 10 minutes into the accident) and continue over a period of 0.5 hour. After the gap activity release phase, there is the second stage of the in-vessel core melt in which the bulk of the activity releases associated with the accident occur and which lasts for 1.3 hours.

The core fission product inventory at the time of the accident is based on operation near the end of a fuel cycle at 102-percent power. There are three groups of nuclides considered in the gap activity releases: noble gases, iodines, and alkali metals. For the core melt phase, there are five additional nuclide groups: tellurium group, the noble metals group, the cerium group, the lanthanide group, and barium and strontium.

The AP1000 does not include active systems for the removal of activity from the containment atmosphere. Elemental iodine is removed by deposition onto surfaces. Particulates are removed by sedimentation, diffusiophoresis (deposition driven by steam condensation), and thermophoresis (deposition driven by heat transfer). No removal of organic iodine is assumed.

The majority of the releases due to the LOCA are the result of containment leakage. The containment is assumed to leak at its design leak rate for the first 24 hours and at half rate for the remainder of the analysis period. During the initial part of the accident, before the containment is isolated, it is assumed that containment purge is in operation and that activity is released through this pathway until the purge valves are closed. The activity releases are assumed to be ground level releases.

The LOCA radiological consequences analysis assumptions include a number of conservatisms. For example, as for core release source term, the assumed core melt is a major conservatism associated with the analysis. In the event of a postulated LOCA no major core damage is expected and release of activity from the core is limited to a fraction of the core gap activity. The primary coolant source term is based on operation with the design fuel defect level of 0.25 percent; whereas, the expected fuel defect level is far less.

The exclusion area boundary dose is calculated for the 2-hour period over which the highest doses would be accrued by an individual located at the exclusion area boundary. The low population zone boundary dose is calculated for the nominal 30-day duration of the accident.
For both the exclusion area boundary and low population zone boundary, the calculated doses are within the dose guideline of 25 rem TEDE from 10 CFR Part 50.34.

The doses calculated for the main control room personnel due to air borne activity entering the main control room, due to direct shine from the activity in the adjacent buildings and due to sky-shine from the radiation that streams out the top of the containment shield building and is reflected back down by air-acattering. The total of the three calculated doses are within the dose criteria of 5 rem TEDE.

8.4 Core and System Performance

8.4.1 Large-break LOCA Analysis

8.4.1.1 Large-break LOCA Analysis Methodology

Westinghouse applies the WCOBRA/TRAC computer code to perform best-estimate large-break LOCA analyses. In the methodology used for the AP1000 analysis, three major components of uncertainty are considered. The initial conditions uncertainty component addresses variations and uncertainties in the initial fluid conditions in the reactor coolant system and the emergency core cooling system boundary conditions. The power distribution uncertainty component addresses variations and uncertainties in power-related parameters, such as peaking factors and axial power distributions. The model uncertainty component addresses uncertainties in the code models that affect the overall system transient ("global" models), as well as those which affect the hot rod only ("local" models). The WCOBRA code is used to calculate the effects of initial conditions, power distributions, and global models, and the HOTSPOT code is used to calculate the effects of local models.

In addition to the code uncertainty estimates quantified in the model uncertainty component, a separate code uncertainty has been estimated based on direct comparisons of WCOBRA/TRAC predictions to experimental data. This estimate is considered in the appropriate step of the methodology used to develop the overall uncertainty. Finally, the uncertainty of the experimental data has been quantified. This uncertainty is also considered in the appropriate step of methodology used to develop the overall uncertainty.

The plant boundary conditions for WCOBRA/TRAC, including the initial operating conditions and the core power distribution, are bounded in a conservative manner based on initial sensitivity studies investigating the range of AP1000 possible values. The modeling bias and uncertainty is then evaluated. This component accounts for uncertainties in the ability of the WCOBRA/TRAC code to accurately predict important phenomena affecting the overall system response ("Global" parameters) and the local fuel rod response ("local" parameters). The code and model bias is the difference between the reference transient PCT, which assumes nominal values for the global and local parameters, and the average PCT, taking into account the possible values of global and local parameters.

A WCOBRA/TRAC LOCA calculation is initiated from a point at which the flows, temperatures, powers, and pressures are at their approximate steady-state values before the postulated break occurs. The values used to set the steady-state plant conditions reflect the AP1000 parameters for reactor coolant pump flows, core power, and steam generator tube plugging levels. The fuel parameters provide the steady-state fuel temperatures, pressures, and gap conductances as a function of fuel burnup and linear power. The calculated fuel temperatures from WCOBRA/TRAC are adjusted to match the specified fuel data by adjusting the gap heat transfer coefficient between the pellet and the cladding. Once the vessel fluid temperatures, flows, pressures, loop pressure drop, and core parameters are in agreement with the desired values and are steady, a suitable initial condition is achieved. Once the steady-state calculation is found to be acceptable, the semi-implicit pipe break model is added to the desired break location and the transient calculation is initiated.

8.4.1.2 Large-break LOCA Analysis Results

The global model matrix of calculations and the final 95-percent uncertainty calculations have been performed for AP1000. The global model run matrix developed for the approved best-estimate LOCA methodology was analyzed to evaluate the effect of broken loop resistance, break discharge coefficient, and condensation rate on the PCT for the guillotine break. Cold-leg breaks are analyzed because the hot-leg break location is nonlimiting in the large-break LOCA best-estimate methodology. The DECLG (guillotine) break was shown to be more limiting than the limiting size split break. In all cases analyzed, the bounding core design values of Fq (2.60) and FdH (1.65) are applied to the hot rod, and 102 percent of nominal core power is assumed.

AP1000 Large-Break LOCA Transient

The break was modeled to occur in one of the cold legs in the loop containing the core makeup tanks. Shortly after the break opens, the vessel rapidly depressurizes and the core flow quickly reverses. The hot assembly fuel rods dry out and begin to heat up during the flow reversal.

The steam generator secondaries are assumed to be isolated immediately at the inception of the break to maximize their stored energy. The massive size of the break causes an immediate, rapid pressurization of the containment. At 2.2 seconds of the transient, credit is taken for receipt of an “S” signal due to high-2 containment pressure. Applying the pertinent signal processing delay means that the valves isolating the core makeup tanks from the direct vessel injection line and the PRHR begin to open at 4.2 seconds into transient. The reactor coolant pumps are presumed to trip immediately following the break. Core shutdown occurs due to voiding; no credit is taken for the control rod reactivity effect.

The system depressurizes rapidly as the initial mass inventory is depleted due to break flow. The pressurizer drains completely approximately 25 seconds into the transient, and accumulator injection commences 15 seconds into the transient. Accumulator actuation shuts off core makeup tank (CMT) flow, which has been delivering since the isolation valve opened. The CMT liquid level remains well above the ADS Stage 1 actuation setpoint throughout the AP1000 DECLG LOCA cladding temperature excursion, even though CMT injection begins again at 215 seconds.

Top of core liquid flow is relatively stagnant for the first few seconds; once the upper head begins to flash, liquid drains directly down the guide tubes and that fraction that is able to penetrate into the core does so, at a maximum flow rate exceeding 2000 lbm/sec of total liquid flow between 5 and 18 seconds. At that point, the flow entering the guide tubes in the upper head is largely steam; residual liquid is supplied to the guide tube fuel assemblies at a constant or decreasing rate out to 42 seconds.

Between 11 and 18 seconds, the combined flow of continuous and entrained liquid is 600 to 1500 lbm/sec; the entrained liquid flow continues to be significant until 30 seconds. After 10 seconds of transient, the downflow pattern in the open hole/support column locations and the guide tubes is established to the extent that vapor downflow is also predicted. Thus, there exists good flow of liquid into the top of the core at these locations from before 10 seconds to after 20 seconds. The flow in the open hole and guide tube assemblies is sufficient to quench the fuel in each respective assembly.

Liquid downflow is delayed into the hot assembly. By 10 seconds into the transient, liquid that has built up in the global region above the upper core plate begins to flow through the plate at the hot assembly location and then proceeds into the core. Significant flow of continuous and/or entrained droplet liquid into the hot assembly exists from 10 to 22 seconds. The liquid flow is not enough to quench the hot rod and hot assembly rod at all elevations, although effective cooling is achieved.

Liquid downflow exists through the top of the peripheral core assemblies from 2 seconds through about 26 seconds in the reference DECLG transient. The power of the fuel in this region is almost identical to that in the open hole and guide tube locations, so the cladding temperature profiles are similar.

About 15 seconds into the transient, the accumulator begins to inject water into the upper downcomer region, most of which is initially bypassed to the break. At approximately 25 seconds, accumulator water begins to flow into the lower plenum. Break flow rates through the loop and vessel sides of the break diminish as the transient progresses. At approximately 70 seconds, the lower plenum fills to the point that water begins to reflood the core from below. The void fraction at the core bottom begins to decrease, and as time passes, core cooling increases substantially. The cladding temperature begins to decrease once the core water level has risen high enough in the core.

Large-Break LOCA Conclusions

Based on the analysis, the Weastinghouse Best-Estimate Large-Break LOCA methodology has shown that there is a high level probability that the following acceptance criteria of 10 CFR 50.46 are satisfied for AP1000.

1. The calculated maximum fuel element cladding temperature (i.e., peak cladding temperature (PCT)) will not exceed 2200°F.

2. The calculated total oxidation of the cladding (i.e., maximum cladding oxidation) will nowhere exceed 0.17 times the total cladding thickness before oxidation.

3. The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam (i.e., maximum hydrogen generation) will not exceed 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

4. The calculated changes in core geometry are such that the core remains amenable to cooling. Note that criterion 4 has historically been satisfied by adherence to criteria 1 and 2, and by assuring that fuel deformation due to combined LOCA and seismic loads is specifically addressed. Criteria 1 and 2 are satisfied for best-estimate large-break LOCA applications. The approved methodology specifies that effects of LOCA and seismic loads on core geometry do not need to be considered unless grid crushing extends beyond the assemblies in the low power channel as defined in the WCOBRA/TRAC model. This situation has not been calculated to occur for the AP1000. Therefore, acceptance criteria 4 is satisfied.

5. After successful initial operation of the emergency core cooling system (ECCS), the core temperature will be maintained at an acceptably low value and decay heat will be removed for the extended period of time required by the long-lived radioactivity remaining in the core.

   Criterion 5 is satisfied if a coolable core geometry is maintained and the core is cooled continuously following the LOCA. The AP1000 passive core cooling system provides effective core cooling following a large-break LOCA event, even assuming the limiting single failure of a core makeup tank delivery line isolation valve. The large-break LOCA transient has been extended beyond fuel rod quench until 1800 seconds, a time at which the CMT liquid level has decreased to the low-2 setpoint that actuates the fourth-stage ADS valves and IRWST injection. A significant increase in safety injection flow rate occurs when the IRWST becomes active. The analysis performed demonstrates that CMT injection is sufficient to maintain the mass inventory in the core and downcomer, from the period of fuel rod quench until IRWST injection. The AP1000 passive core cooling system provides effective post-LOCA long-term core cooling.

8.4.2 Small-break LOCA Analyses

8.4.2.1 Small-break LOCA Analyses Methodology

The NOTRUMP computer code is used in the analysis of LOCAs due to small-breaks in the reactor coolant system. The NONTRUMP computer code is a one-dimensional, general network code, which includes a number of advanced features. Among these features are the calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes, and regime-dependent heat transfer correlations. The version of NONTRUMP used in AP1000 small-break LOCA calculations has been validated against applicable passive plant test data.

The use of NOTRUMP in the analysis involves the representation of the reactor core as heated control volumes with an associated bubble rise model to permit a transient mixture height calculation. The multi-node capability of the program enables an explicit and detailed spatial representation of various system components. A steady-state input deck for AP1000 was set up to comply, where appropriate, with the standard small-break LOCA Evaluation Model methodology.

Active single failures of the passive safeguards systems are considered. The limiting failure is judged to be one out of four ADS Stage 4 valves failing to open on demand, the failure that most severely impacts depressurization capability. The safety design approach of the AP1000 is to depressurize the reactor coolant system to the containment pressure in an orderly fashion such that the large reservoir of water stored in the IRWST is available for core cooling. The mass inventory plots provided for the breaks show the minimum inventory condition generally occurs at the start of IRWST injection. Penalizing the depressurization is the most conservative approach in postulating the single failure for such breaks.

The small-break LOCA spectrum analyzed for AP1000 includes a break that exhibits a minimum reactor vessel inventory early in the transient, before the accumulators become active: the double-ended direct vessel injection (DEDVI) line break. In this transient, the early mass inventory decrease is terminated by injection flow from the intact accumulator, and depressurization through the break enables accumulator injection to begin with no contribution from the actuation of ADS Stage 1, 2, and 3. For consistency, the conservative failure of one of the ADS Stage 4 valves located off the PRHR inlet pipe, which adversely affects the depressurization necessary to achieve IRWST injection in small-break LOCAs, is assumed in all cases.

8.4.2.2 Small-break LOCA Analyses Results

A series of small-break LOCA calculations are performed to assess the AP1000 passive safety system design performance. In these calculations, the decay heat used is the value of the ANS-1971 Standard plus 20 percent for uncertainty as specified in 10 CFR 50, Appendix K. This maximizes the core steam generation to be vented. The results demonstrate that the minimum reactor coolant system mass inventory condition occurs for the relatively large system pipe breaks. Smaller breaks exhibit a greater margin-to-core uncovery. The small-break LOCA transients analyzed using NOTRUMP include the following:

Inadvertent ADS Actuation

A “no-break” small-break LOCA calculation that uses an inadvertent opening of the 4-inch nominal size ADS Stage 1 valves is a situation that minimizes the venting capability of the reactor coolant system. The inadvertent opening of the ADS Stage 1 transient confirms the minimum venting area capability to depressurize the reactor coolant system to the IRWST pressure. The analysis indicates that the ADS sizing is sufficient to depressurize the reactor coolant system assuming the worst single failure as the failure of a Stage 4 ADS path to open and decay heat equal to the ANS-1971 plus 20 percent, which over estimates the core steam generation rate. Even under these limiting conditions, IRWST injection is obtained, and the core remains covered such that no cladding heatup occurs.

2-inch Cold Leg Break in the Core Makeup Tank Loop

The small size of the break leads to a long period of recirculatory flow from the cold leg into the core makeup tank. This delays the formation of a vapor space in the core makeup tank and therfore the actuation of the ADS. This case models a 2-inch break occurring in the bottom of cold leg connected to the balance line of CMT-1. The analysis indicates that the core never uncovers throughout this transient, and that the peak cladding temperature occurs at the inception of the event for this transient. The 2-inch break cases exhibit large margin-to-core uncovery.

Double-ended Rupture of the Direct Vessel Injection Line

The injection line break evaluates the ability of the plant to recover from moderately sized break with only half of the total emergency core cooling system capacity available. The vessel side of the break of the DEDVI line break is 4 inches in equivalent diameter. The double-ended nature of this break means that there are effectively two breaks modeled:

- Downcomer to containment. The direct vessel injection nozzle includes a venturi, which limits the available break area.
- Direct vessel injection line into containment from the cold leg balance line and the broken loop core makeup tank.

The containment pressure was conservatively assumed to pressurize to 25 psia in the NOTRUMP simulations. This pressure was selected from the generated pressure history curves obtained from the WGOTHIC runs. WGOTHIC code calculates the containment pressure response. The analysis indicates that the core uncovery is not predicted for this transient and that no cladding heatup occurs.

10-inch Cold Leg Break

This case models a break size that approaches the upper limit size for small-break LOCAs and a 10-inch break occurring in the bottom of a cold leg connected to the balance line of CMT-1. During the initial portion of the 10-inch break, both liquid and steam flow out the top of the core as the void fraction in the core increases. The break in the cold leg draws fluid from the bottom of the core, and insufficient liquid remains in the core and upper plenum to sustain the mixture level. The mixture level falls to a minimum and then starts to recover as accumulator flows enter the downcomer. During this initial blowdown period, a portion of the core exhibits the potential for core dryout to occur without the prediction of a traditional core uncovery period. To conservatively account for this potential core dryout period, a composite core mixture level was created which collapses to the minimum of the actual core/upper plenum two-phase mixture level and the bottom of the lowest core node that exceeds the core dryout onset conditions. A 90-percent quality limit was chosen as the indicator of the onset of core dryout indicative of the critical heat flux; dryout is assumed at core qualities above this value. To conservatively estimate the effects of this dryout period, an adiabatic heat-up calculation was performed, and the resulting peak cladding temperature is determined to be approximately 1370°F. Even under these conservative adiabatic heat-up assumptions, the AP1000 plant design exhibits large margins to the 10 CFR 50.46 Appendix-K limits of 2200°F for the 10-inch break.

Conclusions

The small-break LOCA analyses performed show that the performance of the AP1000 plant design to small-break LOCA scenarios is excellent and that the passive safeguards systems in the AP1000 are sufficient to mitigate LOCAs. Specifically, it is concluded that:

- The primary side can be depressurized by the ADS to allow stable injection into the core.
- Injection from the core makeup tanks, accumulators, and IRWST prevents excessive cladding heatup for small-break LOCAs analyzed, including double-ended ruptures in the passive safeguards system lines.

The 10-inch cold leg break exhibits the limiting minimum inventory condition that occurs during the initial blowdown period and is terminated by accumulator injection. The AP1000 design is such that the minimum inventory occurs just prior to IRWST injection for all breaks except the 10-inch cold leg break. All breaks simulated in the break spectrum produce results that demonstrate significant margin to peak cladding temperature regulatory limits.

8.4.3 Post-LOCA Long-Term Cooling Analysis

8.4.3.1 Post-LOCA Long-Term Cooling Analysis Methodology

The AP1000 safety-related system are designed to provide adequate cooling of the reactor indefinitely. Initially, this is achieved by discharging water from the IRWST into the vessel. When the low-3 level setpoint is reached in the IRWST, the containment recirculation subsystem isolation valves open and water from the containment reactor coolant system (RCS) compartment can flow into the vessel through the passive core cooling system (PXS) piping. The water in containment rises in temperature toward the saturation temperature. Long-term heat removal from the reactor and containment is by heat transfer through the containment shell to atmosphere.

The purpose of the long-term cooling analysis is to demonstrate that the passive system provide adequate emergency core cooling system performance during the IRWST injection/containment recirculation time scale. The long-term cooling analysis is performed using the WCOBRA/TRAC Computer code to verify that the passive injection system is providing sufficient flow to the reactor vessel to cool the core and to preclude boron precipitation. The AP1000 long-term cooling analysis is supported by the series of tests at the Oregon State University APEX Test Facility. These tests were modeled using WCOBRA/TRAC in order to validate the code for long-term cooling analysis. The data obtained during testing at this facility has been shown to apply to the AP1000.

A DEDVI line break is analyzed because it is the most limiting long-term cooling case in the relationship between decay power and available liquid driving head. Because the IRWST spills directly onto the containment floor in a DEDVI break, this event has the highest core decay power when the transfer to sump injection is initiated. In postulated DEDVI break cases, the compartment water level exceeds the elevation at which the DVI line enters the reactor vessel, so water can flow from the containment into the reactor vessel through the broken DVI line; this in-flow of water through the broken DVI line assists in the heat removal from the core. The steam produced by boiling in the core vents to the containment through the ADS valves and condenses on the inner surface of the steel containment vessel. The condensate is collected and drains to the IRWST to become available for injection into the reactor coolant system. The WCOBRA/TRAC analysis presented analyzes the DEDVI small-break LOCA event from a time (3000 seconds) at which IRWST injection is fully established to beyond the time of containment recirculation. During this time, the head of water to drive the flow into the vessel for IRWST injection decreases from the initial level to its lowest value at the containment recirculation switchover time. PXS Room B is the location of the break in the DVI line. At this location, liquid level in containment at the time of recirculation is a minimum.

A continuous analysis of the post-LOCA long term cooling is provided from the time of stable IRWST injection through the time of sump recirculation for the DEDVI break. Maximum design resistances are applied in WCOBRA/TRAC for both the ADS Stage 4 flow paths and the IRWST injection and containment recirculation flow paths.

The break modeled is a double-ended guillotine rupture of one of the direct vessel injection lines. The long-term cooling phase begins after the simultaneous opening of the isolation valves in the IRWST DVI lines and the opening of ADS Stage 4 squib valves, when flow injection from the IRWST has been fully established.

8.4.3.2 Post-LOCA Long-Term Cooling Analysis Results

DEDVI Line Break with ADS Stage 4 Single Failure, Passive Core Cooling System Only Case; Continuous Case

This subsection presents the results of a DEDVI line break analysis during IRWST injection phase continuing into sump recirculation. Initial conditions at the start of the case are prescribed based on the NOTRUMP DEDVI break results to allow a calculation to begin shortly after IRWST injection begins in the small break long-term cooling transient. This calculation uses boundary conditions taken from a WGOTHIC analysis of this event. During the calculation, which is carried out for 10,000 seconds until a quasi-steady-state sump recirculation condition has been established, the IRWST water level is decreased continuously until the sump recirculation setpoint is reached.

In the analysis, one of the two ADS Stage 4 valves in the PRHR loop is assumed to have failed. The core makeup tanks do not contribute to the DVI injection during this phase of the transient. The reactor coolant pumps are tripped and not rotating.

In this transient, the IRWST provides a hydraulic head sufficient to drive water into the downcomer through the intact DVI nozzle. Also, water flows into the downcomer from the RCS loop compartment through the broken DVI line once the liquid level is adequate to support flow. The water flows down the downcomer and up through the core, into the upper plenum. Steam produced in the core and liquid flow out of the reactor coolant system via the ADS Stage 4 valves. The venting provided by the ADS-4 paths enables the liquid flow through the core to maintain core cooling.

Boiling in the core produces steam and a two-phase mixture, which flows into the upper plenum. The core is 14 feet high, and the core average collapsed liquid level is shown from the start of long-term cooling. The boiling process causes a variable rate of steam production and resulting pressure changes, which in turn causes oscillations in the liquid flow rate at the bottom of the core and also variations in the core collapsed level and the flow rates of liquid and vapor out of the top of the core. The average void fraction of these upper core cells is about 0.8 during long-term cooling, during IRWST injection, and into the containment recirculation period. There is a continuous flow of two-phase fluid into the hot legs, and mainly vapor flow toward the ADS Stage 4 valve occurs at the top of the pipe. The hot legs on average are more than 50-percent full.

The upper plenum pressure fluctuation that occurs is due to the ADS Stage 4 water discharge. The PCT at the top of the hot assembly rod follows saturation temperature, which demonstrates that the calculated core collapsed liquid level is adequate to provide enough liquid at the top of the core that no uncovery and no cladding temperature excursion occurs. Analysis shows the injection flow is outward through the broken DVI line at the start of the long-term cooling period, and it increases to a maximum average value of about 52 lbm/s after the compartment water level has increased above the nozzle elevation to permit liquid injection into the reactor vessel. In contrast, the intact DVI line flow falls from 170 lbm/s with a full IRWST to about 65 lbm/s flow from the containment at the end of the calculation. The recirculation core liquid throughput is more than adequate to preclude any boron buildup on the fuel.

DEDVI Break and Wall-to Wall Floodup; Containment Recirculation

This subsection presents a DEDVI line break analysis with wall-to-wall flooding due to leakage between compartments, using the window mode methodology. All containment free volume beneath the level of the liquid is assumed filled in this calculation to generate the minimum water level condition during containment recirculation. The time identified for this calculation is 14 days in to the event, and the core power is calculated accordingly. Containment recirculation is simulated during the time window. The calculation is carried out over a time period long enough to establish a quasi-steady-state solution; after 1000 seconds of problem time, the flow dynamics are quasi-steady-state and the predicted results are independent of the assumed initial conditions. The liquid level is simulated constant at 28.2 feet above the bottom inside surface of the reactor vessel during the time window, and the liquid temperature in containment is 205°F. The identified containment pressure is 19.0 psia. The single failure of an ADS Stage 4 flow path is assumed.

Focusing on the 1000- to 4000-second time interval of this case, the containment liquid provides a hydraulic head sufficient to drive water into the downcomer through the DVI nozzles. The water introduced into the downcomer flows down the downcomer and up through the core, into the upper plenum. Steam produced in the core entrains liquid and flows out of the reactor coolant system via the ADS Stage 4 valves. The DVI flow and the venting provided by the ADS paths provide a liquid flow through the core that enables the core to remain cool.

Boiling in the core produces steam and a two-phase mixture, which flows out of the core into the upper plenum. Pressure spikes produced by boiling in the core can cause the mass flow of the DVI flow into the vessel to stop momentarily, but the injection flow is quickly reestablished.

The PCT does not rise appreciably above the saturation temperature at the top of the hot rod. The flow through the core and out of the reactor coolant system is more than sufficient to provide adequate flushing to preclude concentration of the boric acid solution. Liquid collects above the upper core plate in the upper plenum, where the average collapsed liquid level is about 3.6 feet. There is no significant flow through the cold legs into either the broken or the intact loops, and there is no significant quantity of liquid residing in any of the cold legs.

Long-Term Core Boron Concentration

For the AP1000, water carryover out the ADS Stage 4 lines limits the potential core boron concentration buildup following a cold leg LOCA. The higher the ADS Stage 4 vent quality, the higher the core boron concentration buildup. Analyses have been performed to bound the maximum core boron concentration buildup.

These analyses demonstrate that highest ADS Stage 4 vent qualities result from the following:

- Highest decay heat levels
- Lowest PXS injection/ADS 4 vent flows, including high line resistances and low containment water levels

The ADS Stage 4 vent quality resulting from this analysis is less than 40 percent at the beginning of IRWST injection and reaches a maximum of less than 50 percent around the initiation of recirculation. It decreases after this peak, dropping to a value less than 8 percent at 14 days.

With high decay heat values, the ADS Stage 4 vent flows and velocities are high. These high vent velocities result in flow regimes that are annular out through at least 14 days and slug/churn after that time. Such flow regimes can move water up and out the ADS Stage 4 lines. The limiting condition for core boron buildup is with high decay heat that leads to the highest ADS Stage 4 vent qualities.

With the maximum ADS Stage 4 vent qualities, the maximum core boron concentration peaks at a value less than 7400 ppm at the time of recirculation initiation. After this time, the core boron concentration decreases as the ADS Stage 4 vent quality decreases, reaching 5000 ppm about 6 hours after the accident. The core boron solubility temperature reaches a maximum of 58°F(at 7400 ppm) and quickly drops to 40°F(at 5000 ppm). With these low core boron solubility temperatures, there is no concern with cold PXS injection water causing boron precipitation in the core. With the IRWST located inside containment, its water temperature is greater than 120°F considering the minimum IRWST temperature permitted by the Technical Specification (50°F) and the heatup of the injection by steam condensation and pickup of sensible heat from the reactor vessel, core barrel, and lower support plate.

The boron concentration water in the containment is initially about 2980 ppm. As the core boron concentration increases, the containment concentration decreases slightly. The minimum boron concentration in containment is greater than 2950 ppm. The solubility temperature of the containment water at its maximum boron concentration is 32°F.

Conclusions

Calculations of AP1000 long-term cooling performance have been performed using the WCOBRA/TRAC model developed for AP1000. The DEDVI case was chosen because it reaches sump recirculation at the earliest time (and highest decay heat). A window mode case at the minimum containment water level postulated to occur 2 weeks into long-term cooling was also performed.

The DEDVI small-break LOCA exhibits no core uncovery due to its adequate reactor coolant system mass inventory condition during the long-term cooling phase from initiation into containment recirculation. Adequate flow through the core is provided to maintain a low cladding temperature and to prevent any buildup of boric acid on the fuel rods. The wall-to-wall floodup case using the window mode technique demonstrates that effective core cooling is also provided at the minimum containment water level. The results of these cases demonstrate the capability of the AP1000 passive systems to provide long-term cooling for a limiting LOCA event.

8.5 DOCUMENTS AND DESIGN CONTROL DOCUMENTS

The methodology used for the AP1000 analysis is documented in:

> WCAP-12945-P, Revision 2, “Code Qualification Document for Best Estimate Analysis”
>
> WCAP-14171, Revision 2, “WCOBRA/TRAC Applicability to AP600 Large-Break Loss-of-Coolant Accident”

NONTRUMP’s applicability is documented in:

> WCAP-15644-P, Revision 1, “AP1000 Code Applicability Report”

The AP1000 DCD Section 15.6.5 contains information relative to loss-of-coolant accident resulting from a spectrum of postulated piping breaks within the reactor coolant pressure boundary.

9 NON-LOCA ANALYSES

9.1 INCREASE IN HEAT REMOVAL FROM THE PRIMARY SYSTEM

A number of events that could result in an increase in heat removal from the reactor coolant system are postulated. Detailed analyses are performed for the events that have been identified as limiting cases. Summaries of some of the reactor coolant system cooldown events analyzed for the AP1000 are given as follows. The most severe radiological consequences are reported only for those resulted from the main steam line break accident.

9.1.1 Feedwater System Malfunctions that Result in a Decrease in Feedwater Temperature

A fault in the feedwater heaters section of the Feedwater System causes a reduction in feedwater temperature that increases the thermal load on the primary system. The maximum reduction in feedwater temperature, due to a single failure in the feedwater system, is lower than 79.5°F. This reduction results in an increase in heat load on the primary system of less than 10-percent full power. These analysis results indicate that evaluation criteria for the decrease in feedwater temperature is met.

9.1.2 Feedwater System Malfunctions that Result in an Increase in Feedwater Flow

In the case of an accidental full opening of one feedwater control valve with the reactor at zero power, the maximum reactivity insertion rate is less than the maximum reactivity insertion rate analyzed for an uncontrolled rod cluster control assembly (RCCA) bank withdrawal from a subcritical or low-power startup condition.

The full-power case (maximum reactivity feedback coefficients, automatic rod control) results in the greater power increase. Assuming the rod control system to be in the manual control mode results in a slightly less severe transient. Transient results show the power level rises by a maximum of about 12 percent above nominal during the excessive feedwater flow incident, the fuel temperature also rises until after reactor trip occurs. Departure from nucleate boiling (DNB) does not occur at any time during the excessive feedwater flow incident. The minimum DNBR is predicted to occur before the reactor trip and the reactor coolant pump coastdown caused by the loss of offsite power. The minimum DNBR predicted is 2.14, which is well above the design limit. Thus, the capability of the primary coolant to remove heat from the fuel rods is not reduced and the fuel cladding temperature does not rise significantly above its initial value during the transient.

9.1.3 Excessive Increase in Secondary Steam Flow

An excessive increase in secondary system steam flow (excessive load increase incident) results in a power mismatch between the reactor core power and the steam generator load demand. The plant control system is designed to accommodate a 10-percent step load increase or a 5-percent-per-minute ramp load increase in the range of 25- to 100-percent full power. Any loading rate in excess of these values may cause a reactor trip actuated by the protection and safety monitoring system.

The transient analysis demonstrates that for a 10-percent step load increase, the DNBR remains above the design limit. The plant rapidly reaches a stabilized condition following the load increase. Thus, DNB does not occur during the excessive load increase transients, the capability of the primary coolant to remove heat from the fuel rod is not reduced.

9.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

The most severe core conditions resulting from an accidental depressurization of the main steam system are associated with an inadvertent opening of a single steam dump, relief, or safety valve. The steam release, as a consequence of this accident, results in an initial increase in steam flow which decreases during the accident as the steam pressure falls. The energy removal from the reactor coolant system causes a reduction of coolant temperature and pressure. In the presence of a negative moderator temperature coefficient, the cooldown results in an insertion of positive reactivity.

The assumed steam release of 520 pounds per second at 1200 psia is typical of the capacity of any single steam dump, relief, or safety valve. Core makeup tank injection and the associated tripping of the reactor coolant pumps are initiated automatically by the low T_{cold} “S” signal. Boron solution at 3400 ppm enters the reactor coolant system, providing enough negative reactivity to prevent a significant return to power and core damage. Later in the transient, as the reactor coolant pressure continues to fall, the accumulators actuate and inject boron solution at 2600 ppm. The analysis demonstrates that no significant return to power occurs and, therefore, DNB does not occur.

9.1.5 Steam System Piping Failure

Transient Analysis Results

The steam release arising from a rupture of a main steam line results in an initial increase in steam flow, which results in an insertion of positive reactivity. The major rupture of a steam line is the most limiting cooldown transient and the analysis is performed at zero power with no decay heat, assuming the most reactive stuck RCCA in its fully withdrawn position and assuming a single failure in the engineered safety features system.

During the course of the event, the reactor protection system initiates a trip of the reactor coolant pumps in conjunction with actuation of the core makeup tanks. Uncontrolled steam release from more than one steam generator is prevented by automatic trip of the main steam isolation valves in the steam lines by high containment pressure signals or by low steam line pressure signals. The main steam isolation valves fully close in less than 10 seconds from receipt of a closure signal.

The core attains criticality with the RCCAs inserted before boron solution at 3400 ppm (from core makeup tanks) or 2600 ppm (from accumulators) enters the reactor coolant system. A peak core power significantly lower than the nominal full-power value is attained. At no time during the analyzed steam line break event does the core makeup tank level approach the setpoint for actuation of the automatic depressurization system. During non-LOCA events, the core makeup tanks remain filled with water. The volume of injection flow leaving the core makeup tank is offset by an equal volume of recirculation flow that enters the core makeup tanks via the reactor coolant system cold leg balance lines. The analysis shows that no DNB occurs for the main steam line rupture.

Radiological Consequences

The evaluation of the radiological consequences of a postulated main steam line break outside containment assumes that the reactor has been operating with the design basis fuel defect level (0.25 percent of power produced by fuel rods containing cladding defects) and that leaking steam generator tubes have resulted in a buildup of activity in the secondary coolant.

Following the rupture, startup feedwater to the faulted loop is isolated and the steam generator is allowed to steam dry. Any radioiodines carried from the primary coolant into the faulted steam generator via leaking tubes are assumed to be released directly to the environment. It is conservatively assumed that the reactor is cooled by steaming from the intact loop.

The only significant radionuclide releases due to the main steam line break are the iodines and alkali metals that become airborne and are released to the environment as a result of the accident. The analysis considers two different reactor coolant iodine source terms, both of which consider the iodine spiking phenomenon. The iodine spike is assumed to be initiated by the accident with the spike causing an increasing level of iodine in the reactor coolant. The second case assumes that the iodine spike occurs prior to the accident and that the maximum resulting reactor coolant iodine concentration exists at the time the accidents occurs.

The calculated total effective dose equivalent (TEDE) doses for the case with accident-initiated iodine spike are determined to be less than 0.9 rem at the site boundary for the limiting 2-hour interval (0 to 2 hours) and 2.0 rem at the low population zone outer boundary. These doses are small fractions of dose guideline of 25 rem TEDE. The TEDE doses for the case with pre-existing iodine spike are determined to be less than 0.8 rem at the site boundary for the limiting 2-hour interval (0 to 2 hours) and 0.8 rem at the low population zone outer the boundary. These doses are within the dose guidelines of 10 CFR Part 50.34.

9.1.6 Inadvertent Operation of the PRHR Heat Exchanger

The inadvertent actuation of the PRHR heat exchanger causes an injection of relatively cold water into the reactor coolant system. This produces a reactivity insertion in the presence of a negative Moderator temperature coefficient. The overpower/overtemperature protection functions are intended to prevent a power increase which could lead to a DNBR less than the safety analysis limit. In addition, because the cold leg temperature is reduced which depressurizes the reactor coolant system during this event, the low cold leg temperature or low pressurizer pressure protection functions could generate a reactor trip. These protection functions do not terminate operation of the PRHR heat exchanger.

The inadvertent operation of the PRHR heat exchanger incident is an overpower transient for which the fuel temperature rises. Assuming a reactor trip does not occur, the plant reaches a new equilibrium condition at a higher power level corresponding to the increase in power demanded by the system. In the limiting case analyzed, the plant power stabilizes at about 108 percent of its nominal value.

The results of the analysis show that the DNBRs encountered for an inadvertent actuation of the PRHR heat exchanger at power are above the design limit values. The ability of the primary coolant to remove heat from the fuel rods is therefore not reduced.

9.2 DECREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

A number of transients and accidents that could result in a reduction of the capacity of the secondary system to remove heat generated in the reactor coolant system are postulated. Analyses are performed for the events that are identified as more limiting than the others. Summaries of the events analyzed for the AP1000 are given as follows. The radiological consequences of the accidents in this section are bounded by the radiological consequences of a main steam line break.

9.2.1 Loss of External Electrical Load

A major load loss on the plant can result from loss of electrical load due to an electrical system disturbance. The ac power remains available to operate plant components such as the reactor coolant pumps; as a result, the standby onsite diesel generators do not function for this event. Following the loss of generator load, an immediate fast closure of the turbine control valves occurs. The automatic turbine bypass system accommodates the excess steam generation. Reactor coolant temperature and pressure do not significantly increase if the turbine bypass system and pressurizer pressure control system function properly. If the condenser is not available, the excess steam generation is relieved to the atmosphere. Additionally, main feedwater flow is lost if the condenser is not available.

For a loss of electrical load without subsequent turbine trip, no direct reactor trip signal is generated. The plant trips from the protection and safety monitoring system if a safety limit is approached. A continued steam load of approximately 5 percent exists after total loss of external electrical load because of the steam demand of plant auxiliaries.

The protection and safety monitoring system may be required to terminate core heat input and to prevent departure from nucleate boiling (DNB). Depending on the magnitude of the load loss, pressurizer safety valves and/or steam generator safety valves may open to maintain system pressures below allowable limits. No single active failure prevents operation of any system required to function. Normal plant control systems and engineered safety systems are not required to function. The passive residual heat removal (PRHR) system may be automatically actuated following a loss of main feedwater, further mitigating the effects of the transient.
The results of the turbine trip event analysis bound those expected for the loss-of-external-load event.

9.2.2 Turbine Trip

The turbine stop valves close rapidly (about 0.15 seconds) on loss of trip fluid pressure actuated by one of a number of possible turbine trip signals. Upon initiation of stop valve closure, steam flow to the turbine stops abruptly. Sensors on the stop valves detect the turbine trip and initiate turbine bypass. The loss of steam flow results in a rapid increase in secondary system temperature and pressure, with a resultant primary system transient for the loss-of-external-load event. A slightly more severe transient occurs for the turbine trip event due to the rapid loss of steam flow caused by the abrupt valve closure. A turbine trip is a more limiting than a loss-of-external-load event, loss of condenser vacuum, and other events which result in a turbine trip.

In the analysis, the transient responses of the unit are evaluated for a complete loss of steam load from 100 percent of full power, without rapid power reduction. The eight analysis cases are performed assuming minimum and maximum reactivity feedback, with and without credit for pressurizer spray, and with and without offsite power available. In each case, the analyses results show that the reactor is tripped and the reactor coolant system pressure is maintained below 110 percent of the design value, and that turbine trip presents no challenge to the integrity of the reactor coolant system or the main steam system.
The analyses show that the predicted DNBR is greater than the design limit at any time during the transient. Thus, the departure from nucleate boiling design basis is met.

9.2.3 Loss of ac Power to the Plant Auxiliaries

The loss of power to the plant auxiliaries is caused by a complete loss of the offsite grid accompanied by a turbine-generator trip. The onsite standby ac power system remains available but is not credited to mitigate the accident. From the decay heat removal point of view, in the long term this transient is more severe than the turbine trip event, because the decrease in heat removal by the secondary system is accompanied by a reactor coolant flow coastdown, which further reduces the capacity of the primary coolant to remove heat from the core.

Upon the loss of power to the reactor coolant pumps, coolant flow necessary for core cooling and the removal of residual heat is maintained by natural circulation in the reactor coolant and PRHR loops. During a plant transient, core decay heat removal is normally accomplished by the startup feedwater system if available, which is started automatically when low levels occur in either steam generator. If that system is not available, emergency core decay heat removal is provided by the PRHR heat exchanger. The heat exchanger is located above the core to provide natural circulation flow when the reactor coolant pumps are not operating. The IRWST provides the heat sink for the heat exchanger. The PRHR heat exchanger, in conjunction with the passive containment cooling system, keeps the reactor coolant subcooled indefinitely. After the IRWST water reaches saturation (in about two and half hours), steam starts to vent to the containment atmosphere. The condensation that collects on the containment steel shell (cooled by the passive containment cooling system) returns to the IRWST, maintaining fluid level for the PRHR heat exchanger heat sink.

The analysis is performed to demonstrate the adequacy of the protection and safety monitoring system, the PRHR heat exchanger, and the reactor coolant system natural circulation capability in removing long term (approximately 36,000 seconds) decay heat. This analysis also demonstrates the adequacy of these systems in preventing excessive heatup of the reactor coolant system with possible reactor coolant system overpressurization or loss of reactor coolant system water.
The analysis results show that sufficient long-term reactor coolant heat removal capability exists, via natural circulation and the PRHR heat exchanger, following reactor coolant pump coastdown to prevent fuel or cladding damage and reactor coolant system overpressure.

9.2.4 Loss of Normal Feedwater Flow

A loss of normal feedwater (from pump failures, valve malfunctions, or loss of ac power sources) results in a reduction in the capability of the secondary system to remove the heat generated in the reactor core. If startup feedwater is not available, the safety-related PRHR heat exchanger is automatically aligned by the protection and safety monitoring system to remove decay heat.

An analysis of the system transient is performed to show that, following a loss of normal feedwater, the PRHR heat exchanger is capable of removing the stored and decay heat to prevent either overpressurization of the reactor coolant system or loss of water from the reactor coolant system.

Results of the analysis show that a loss of normal feedwater does not adversely affect the core, the reactor coolant system, or the steam system. The heat removal capacity of the PRHR heat exchanger is such that reactor coolant water is not relieved from the pressurizer safety valves. DNBR always remains above the design limit values, and reactor coolant system and steam generator pressures remain below 110 percent of their design values.

9.2.5 Feedwater System Pipe Break

A major feedwater line rupture is a break in a feedwateer line large enough to prevent the addition of sufficient feedwater to the steam generators in order to maintain shell-side fluid inventory in the steam generators. If the break is postulated in a feedwater line between the check valve and the steam generator, fluid from the steam generator may also be discharged through the break.

Depending upon the size of the break and the plant operating conditions at the time of the break, the break could cause either a reactor coolant system cooldown (by excessive energy discharge through the break) or a reactor coolant system heatup. Potential reactor coolant system cooldown resulting from a secondary pipe rupture is evaluated in subsection 9.1.5. Therefore, only the reactor coolant system heatup effects are evaluated for a feedwater line rupture. The case analyzed assumes a double-ended rupture of the largest feedwater pipe at full power.

The results show that the pressures in the reactor coolant system and main steam system remain below 110 percent of the respective design pressure. Pressurizer pressure decreases after reactor trip on the low steam generator water level due to the loss of heat input.

After the trip, the core makeup tanks are actuated on low steam line pressure in the ruptured loop while the PRHR heat exchanger is actuated on a low steam generator water level wide range. The addition of the PRHR heat exchanger and the core makeup tanks flow rates helps to cool down the primary system and to provide sufficient fluid to keep the core covered with water.

Pressurizer safety valves open due to the mismatch between decay heat and the heat transfer capability of the PRHR heat exchanger. In the first part of the transient, there is a cooling effect due to the core makeup tanks that inject cold water into the reactor coolant system and receive hot water from the cold leg. This effect decreases due to the heatup of the core makeup tanks from recirculation flow. Also, the injection driving head is lowered as the core makeup tanks heat up.

Reactor coolant system temperatures are low and, in this condition, the PRHR heat exchanger cannot remove the entire decay heat load. Reactor coolant system temperatures increase until an equilibrium between decay heat power and heat absorbed by the PRHR heat exchanger is reached. After about 11,300 seconds, the heat transfer capability of the PRHR heat exchanger exceeds the decay heat power and the reactor coolant system temperatures, pressure, and pressurizer water volumes start to steadily decrease. Core cooling capability is maintained throughout the transient because reactor coolant system inventory is increasing due to core makeup tank injection.

Results of the analyses demonstrate that for the postulated feedwater line rupture, the capacity of the PRHR heat exchanger is adequate to remove decay heat, to prevent overpressurizing the reactor coolant system, and to maintain the core cooling capability.

9.3 DECREASE IN REACTOR COOLANT SYSTEM FLOW RATE

A number of faults that could result in a decrease in the reactor coolant system flow rate are postulated. Detailed analyses are performed for the most limiting cases of the reactor coolant system flow decrease events. Summaries of the four limiting flow rate decrease events analyzed for the AP1000 are given as follows. The most severe radiological consequences are reported only for those resulted from the reactor coolant pump shaft seizure accident.

9.3.1 Partial Loss of Forced Reactor Coolant Flow

A partial loss of coolant flow accident can result from a mechanical or an electrical failure of a reactor coolant pump or from a fault in the power supply to the pump or pumps. If the reactor is at power at the time of the event, the immediate effect of the loss of coolant flow is a rapid increase in the coolant temperature. Normal power for the pump is applied through four buses connected to the generator. When a generator trip occurs, the buses are supplied from offsite power. The pumps continue to operate.

The effects of a loss of offsite power are considered in evaluating partial loss of forced reactor coolant flow transients. The loss of offsite power is considered to be a potential consequence of the event due to disruption of the electrical grid following a turbine trip during the event. A delay of 3 seconds is assumed between the turbine trip and the loss of offsite power. In addition, turbine trip occurs 5.0 seconds following a reactor trip condition being reached. This delay on turbine trip is a feature of the AP1000 reactor trip system. The primary effect of the loss of offsite power is to cause the remaining operating reactor coolant pumps to coast down.

The plant is tripped by the low-flow trip rapidly enough so that the capability of the reactor coolant to remove heat from the fuel rods is not greatly reduced. The average fuel and cladding temperatures do not increase significantly above their initial values.

The analyses show that the affected reactor coolant pumps continue to coast down, and the core flow reaches a new equilibrium value. In the event that a loss of offsite power occurs as a consequence of a turbine trip during a partial loss of reactor coolant flow, the remaining two operating reactor coolant pumps start coasting down, reactor trip has already been initiated, core heat flux has started decreasing, and DNB is increasing. The analyses results indicate that, for the partial loss of reactor coolant flow, the DNBR does not decrease below the design basis value at any time during the transient.

9.3.2 Complete Loss of Forced Reactor Coolant Flow

A complete loss of flow accident may result from a simultaneous loss of electrical supplies to the reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of a loss of coolant flow is a rapid increase in the coolant temperature.

The complete loss of flow transient is analyzed for a loss of power to four reactor coolant pumps. A loss of forced primary coolant flow can result from a reduction in the reactor coolant pump motor supply frequency. Following the loss of power supply to all pumps at power, a reactor trip is actuated by the reactor coolant pump underspeed trip.

The analysis shows that the reactor coolant pumps continue to coast down, and natural circulation flow is established. With the reactor tripped, a stable plant condition is attained. The analysis result demonstrates that, for the complete loss of forced reactor coolant flow, the DNBR does not decrease below the design basis limit value at any time during the transient.

9.3.3 Reactor Coolant Pump Shaft Seizure (Locked Rotor)

Transient Analyses Results

The accident postulated is an instantaneous seizure of a reactor coolant pump rotor. Flow through the affected reactor coolant loop is rapidly reduced, leading to a reactor trip on a low-flow signal. Following the reactor trip, heat stored in the fuel rods continue to be transferred to the coolant, causing the coolant temperature to increase and expand. The rapid expansion of the coolant in the reactor core, combined with reduced heat transfer in the steam generators, causes an insurge into the pressurizer and a pressure increase throughout the reactor coolant system. The insurge into the pressurizer compresses the steam volume, actuates the automatic spray system, and opens the pressurizer safety valves, in that sequence. For conservatism, the pressure-reducing effect of the spray is not included in the analysis.

The accident is evaluated for both cases with and without offsite power available. For the case without offsite power available, power is lost to the unaffected pumps at 3.0 seconds following turbine/generator trip. Turbine trip occurs 5.0 seconds following a reactor trip condition being reached. This delay on turbine trip is a feature of the AP1000 reactor trip system.

The transient analyses results, for one locked rotor with four reactor coolant pumps in operation with and without offsite power available, show that the without-offsite-power case bounds the results for the case with offsite power. The peak reactor coolant system pressure reached during the transient is less than that which causes stresses to exceed the faulted condition stress limits of the ASME Code, Section III. Also, the peak cladding surface temperature is considerably less than 2700°F. The cladding temperature is conservatively calculated, assuming that DNB occurs at the initiation of the transient.

Radiological Consequences

The evaluation of the radiological consequences of a postulated locked reactor coolant pump rotor accident assumes that the reactor has been operating with the design basis fuel defect level and that leaking steam generator tubes have resulted in a buildup of activity in the secondary coolant. Two separate accident scenarios are assumed for the radiological evaluation.

As a result of the accident analyses, it is determined that no fuel rods are damaged such that the activity contained in the fuel-cladding gap is released to the reactor coolant. However, a conservative analysis has been performed assuming 10 percent of the rods are damaged. Activity carried over to the secondary side because of primary-to-secondary leakage is available for release to the environment via the steam line safety valves or the power-operated relief valves.

The significant radionuclide releases due to the locked rotor accident are the iodines, alkali metals and noble gases. The reactor coolant iodine source term assumes a pre-existing iodine spike. The initial reactor coolant noble gas and alkali metal concentrations are assumed to be those associated with the design basis fuel defect level. The initial reactor coolant activities are of secondary importance compared to the release of the gap inventory of fission products from the portion of the core assumed to fail because of the accident. The initial secondary coolant activity is assumed to be 10 percent of the maximum equilibrium primary coolant activity for iodines and alkali metals.

The calculated total effective dose equivalent (TEDE) doses are determined to be less than 0.7 rem at the exclusion area boundary for the limiting 2-hour interval (0 to 2 hours) and less than 0.4 rem at the low population zone outer boundary for the scenario in which there is no feedwater available to maintain water level in the steam generators. The dose for the scenario in which it is assumed that water level in the steam generators is maintained are 0.5 rem at the exclusion area boundary for the limiting 2-hour interval of 7 to 8 hours and 0.8 rem at the low population zone outer boundary. These doses are small fractions of dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34.

9.3.4 Reactor Coolant Pump Shaft Break

The accident is postulated as an instantaneous failure of a reactor coolant pump shaft. Flow through the affected reactor coolant loop is rapidly reduced, though the initial rate of reduction of coolant flow is greater for the reactor coolant pump rotor seizure event. Reactor trip occurs on a low-flow signal in the affected loop.

With a failed shaft, the impeller could be free to spin in a reverse direction as opposed to being fixed in position as is the case when a locked rotor occurs. This results in a decrease in the end point (steady-state) core flow. For both the shaft break and locked rotor incidents, reactor trip occurs very early in the transient. In addition, the locked rotor analysis conservatively assumes that DNB occurs at the beginning of the transient. The calculated results for the locked rotor analysis bound the reactor coolant pump shaft break event.

9.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

A number of faults that could result in reactivity and power distribution anomalies are postulated. Reactivity changes could be caused by control rod motion or ejection, boron concentration changes, or addition of cold water to the reactor coolant system. Power distribution changes could be caused by control rod motion, misalignment, or ejection, or by static means such as fuel assembly mislocation. Analyses are performed for the most limiting cases of these events. Summaries of some of the limiting events analyzed for the AP1000 are given as follows. The most severe radiological consequences have been determined for those resulted from the complete rupture of a control rod drive mechanism housing. Radiological consequences are reported only for the limiting case.

9.4.1 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal from a Subcritical or Low-power Startup Condition

An RCCA withdrawal accident is an uncontrolled addition of reactivity to the reactor core caused by the withdrawal of RCCAs which results in a power excursion. Such a transient can be caused by a malfunction of the reactor control or rod control systems. This can occur with the reactor subcritical, at hot zero power, or at power.

Although the reactor is normally brought to power from a subcritical condition by RCCA withdrawal, initial startup procedures with a clean core use boron dilution. The maximum rate of reactivity increase in the case of boron dilution is less than that assumed in this analysis. The maximum reactivity insertion rate analyzed is that occurring with the simultaneous withdrawal of the combination of two sequential RCCA banks having the maximum combined worth at maximum speed.

The analysis of the uncontrolled RCCA bank withdrawal from subcritical accident is performed in three stages: first, an average core nuclear power transient calculation; then, an average core heat transfer calculation; and finally, the departure from nucleate boiling ratio (DNBR) calculation.

The transient results for the uncontrolled RCCA bank withdrawal from subcritical incident show that the accident is terminated by reactor trip at 35 percent of nominal power. With the reactor tripped, the plant returns to a stable condition. The energy release and the fuel temperature increase are relatively small. There is margin to DNB during the transient because the rod surface heat flux remains below the critical heat flux value, and there is a high degree of subcooling at all times in the core. The minimum DNBR at all times remains above the design limit value. Thus, no fuel or cladding damage is predicted as a result of DNB.

9.4.2 Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power

Uncontrolled RCCA bank withdrawal at power results in an increase in the core heat flux. Because the heat extraction from the steam generator lags behind the core power generation until the steam generator pressure reaches the relief or safety valve setpoint, there is a net increase in the reactor coolant temperature. Unless terminated by manual or automatic action, the power mismatch and resultant coolant temperature rise could eventually result in DNB. Therefore, to avert damage to the fuel cladding, the protection and safety monitoring system is designed to terminate any such transient before the DNBR falls below the design limit.

The transient response analyses performed for uncontrolled RCCA bank withdrawal at-power accidents include the following cases:

- A representative rapid RCCA withdrawal incident starting from full power with offsite power lost as a consequence of turbine trip
- A representative slow RCCA withdrawal incident starting from full power with offsite power lost as a consequence of turbine trip
- RCCA withdrawal incidents for minimum and maximum reactivity feedback, starting from initial full-power operation
- RCCA withdrawal incidents for minimum and maximum reactivity feedback, starting at 60-percent and 10-percent power, respectively

In each case, the result of the analysis shows that the transient is always terminated by reactor trip functions by either power range neutron flux instrumentation, overtemperature ??, or high pressurizer pressure. The minimum DNBR is greater than the design limit value.

9.4.3 Rod Cluster Control Assembly Misalignment (System Malfunction or Operator Error)

RCCA misoperation accidents include:

- One or more dropped RCCAs within the same group
- Statically misaligned RCCA
- Withdrawal of a single RCCA

A drop of one or more RCCAs from the same group results in an initial reduction in the core power and a perturbation in the core radial power distribution. Depending on the worth and position of the dropped rods, this may cause the allowable design power peaking factors to be exceeded. For cases of dropped RCCAs or dropped banks, including inadvertent drops of the RCCAs in those groups selected to be inserted as part of the rapid power reduction system, the transient response shows that the DNBR remains greater than the safety analysis limit value and, therefore, the DNB design basis is met.

The most severe misalignment situations with respect to DNBR arise from cases in which one RCCA is fully inserted, or where the mechanical shim or axial offset rod banks are inserted up to their insertion limit with one RCCA fully withdrawn while the reactor is at full power. This case is analyzed and the result shows that the DNBR remains above the safety analysis limit value.

The accidental withdrawal of a single RCCA, with the reactor in the automatic or manual control mode and initially operating at full power with the mechanical shim or axial offset banks at their insertion limits, results in both an increase in core power and coolant temperature and an increase in the local hot channel factor in the area of the withdrawing RCCA. The increased local power peaking in the area of the withdrawn RCCA results in lower minimum DNBRs than for the withdrawn bank cases. Depending on initial bank insertion and location of the withdrawn RCCA, automatic reactor trip may not occur sufficiently fast to prevent the minimum DNBR from falling below the safety analysis limit value. The probability of such a combination of conditions is considered low such that the limiting consequences may include slight fuel damage. Evaluation of this case at the power and coolant conditions at which the overtemperature ?? trip is expected to trip the plant shows that an upper bound of the number of fuel rods with a DNBR less than the safety analysis limit value is 5 percent of the total fuel rods in the core.

9.4.4 Chemical and Volume Control System Malfunction that Results in a Decrease in the Boron Concentration in the Reactor Coolant

Other than control rod withdrawal, the principal means of positive reactivity insertion to the core is the addition of unborated, primary-grade water from the demineralized water transfer and storage system into the reactor coolant system through the reactor makeup portion of the chemical and volume control system. Normal boron dilution with these systems is manually initiated under strict administrative controls requiring close operator surveillance. An inadvertent boron dilution is caused by the failure of the demineralized water transfer and storage system or chemical and volume control system, either by controller, operator or mechanical failure.

Boron dilutions during refueling, cold shutdown, hot shutdown, hot standby, startup, and power modes of operation are considered in this analysis. Conservative values for necessary parameters are used (high reactor coolant system critical boron concentrations, high boron worths, minimum shutdown margins, and lower-than-actual reactor coolant system volumes). These assumptions result in conservative determinations of the time available for operator or automatic system response after detection of a dilution transient in progress.

Inadvertent boron dilution events are prevented during refueling and automatically terminated during cold shutdown, safe shutdown, and hot stanby modes. Inadvertent boron dilution events during startup or power operation, if not detected and terminated by the operators, result in an automatic reactor trip. Following reactor trip, automatic termination of the dilution occurs and post-trip return to criticality is prevented.

9.4.5 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

Fuel and core loading errors can inadvertently occur, such as those arising from the inadvertent loading of one or more fuel assemblies into improper positions, having a fuel rod with one or more pellets of the wrong enrichment, or having a full fuel assembly with pellets of the wrong enrichment. This leads to increased heat fluxes if the error results in placing fuel in core positions calling for fuel of lesser enrichment. Also included among possible core-loading errors is the inadvertent loading of one or more fuel assemblies requiring burnable poison rods into a new core without burnable poison rods.

An error in enrichment, beyond the normal manufacturing tolerances, can cause power shapes more peaked than those calculated with the correct enrichment. A 5-percent uncertainty margin is included in the design value of power peaking factor assumed in the analysis of Condition I and Condition II transients. The power distortion due to a combination of misplaced fuel assemblies could significantly increase peaking factors and is readily observable with the online core monitoring system.

The following core loading error cases are analyzed:

- A Region 1 assembly is interchanged with a Region 3 assembly. The particular case considered is the interchange of two assemblies near the periphery of the core.
- A Region 1 assembly is interchanged with a neighboring Region 2 fuel assembly. For the particular case considered, the interchange is assumed to take place close to the core center and with burnable poison rods located in the correct Region 2 position, but in a Region 1 assembly mistakenly loaded in the Region 2 position.
- Enrichment error – A Region 2 fuel assembly is loaded in the core central position.
- A Region 2 fuel assembly instead of a Region 1 assembly is loaded near the core periphery.

Fuel assembly enrichment errors are prevented by administrative procedures implemented in fabrication.

In the event that a single pin or pellet has a higher enrichment than the nominal value, the consequences in terms of reduced DNBR and increased fuel and cladding temperatures are limited to the incorrectly loaded pin or pins and perhaps the immediately adjacent pins.

Fuel assembly loading errors are prevented by administrative procedures implemented during core loading. In the unlikely event that a loading error occurs, analyses in this section confirm that resulting power distribution effects are either readily detected by the online core monitoring system or cause a sufficiently small perturbation to be acceptable within the uncertainties allowed between nominal and design power shapes.

9.4.6 Spectrum of Rod Cluster Control Assembly Ejection Accidents

Transient Analyses Results

This accident is defined as the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of an RCCA and drive shaft. The consequence of this mechanical failure is a rapid positive reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

Failure of a control rod housing, due either to longitudinal or circumferential cracking, does not cause damage to adjacent housings that increase the severity of the initial accident.

Because the control rod insertion limits for the AP1000 are multidimentional, a significant number of rodded configurations are evaluated to determine the most limiting cases. The hot zero power cases and hot full power cases assume that the mechanical shim and axial offset control RCCAs are inserted to their insertion limits before the event. The limiting RCCA ejection cases, for both the beginning and end of cycle at zero and full power, are presented below.

- Beginning of cycle, full power – The peak hot spot cladding average temperature is 2265°F. The peak hot spot fuel center temperature reaches melting at 4900°F. However, melting is restricted to less than 10 percent of the pellet at the hot spot.
- Beginning of cycle, zero power – The peak hot spot cladding average temperature is 1907°F, and the peak hot spot fuel center temperature is 3018°F.
- End of cycle, full power – The peak hot spot cladding average temperature is 2151°F. The peak hot spot fuel center temperature reaches melting at 4800°F. However, melting is restricted to less than 10 percent of the pellet at the hot spot.
- End of cycle, zero power – The peak hot spot cladding average temperature is 2122°F, and the peak hot spot fuel center temperature is 3263°F.

The calculated sequence of events for the limiting case rod ejection accidents shows that reactor trip occurs early in the transients, after which the nuclear power excursion is terminated. The ejection of an RCCA constitutes a break in the reactor coolant system, located in the reactor pressure vessel head. Following the RCCA ejection, the plant response is the same as a LOCA.

Radiological Consequences

The evaluation of the radiological consequences of a postulated rod ejection accident assumes that the reactor is operating with the design basis fuel defect level and that leaking steam generator tubes result in a buildup of activity in the secondary coolant.

As a result of the accident analyses, 10 percent of the fuel rods are assumed to be damaged such that the activity contained in the fuel-cladding gap is released to the reactor coolant. In addition, a small fraction of fuel is assumed to melt and release core inventory to the reactor coolant. Activity released to the containment via the spill from the reactor vessel head is assumed to be available for release to the environment because of containment leakage. Activity carried over to the secondary side due to primary-to-secondary leakage is available for release to the environment through the steam line safety valves or the power-operated relief valves.

The significant radionuclide releases due to the rod ejection accident are the iodines, alkali metals and noble gases. The reactor coolant iodine source term assumes a pre-existing iodine spike. The initial reactor coolant noble gas and alkali metal concentrations are assumed to be those associated with the design basis fuel defect level. These initial reactor coolant activities are of secondary importance compared to the release of fission products from the portion of the core assumed to fail.

The calculated total effective dose equivalent (TEDE) doses are determined to be less than 2.9 rem at the site boundary for the limiting 2-hour interval (0 to 2 hours) and less than 5.5 rem at the low population zone outer boundary. These doses are well within the dose guideline of 25 rem TEDE identified in 10 CFR Part 50.34.

9.5 INCREASE IN REACTOR COOLANT INVENTORY

9.5.1 Inadvertent Operation of the Core Makeup Tanks During Power Operation

Spurious core makeup tank operation at power could be caused by an operator error, a false electrical actuation sinal, or a valve malfunction. A spurious signal may originate from any of the safeguards (“S”) actuation channels. The AP1000 protection logic is such that a single failure cannot actuate both core makeup tanks without also actuating the passive residual heat removal (PRHR) heat exchanger. A scenario such as this is the spurious “S” signal event. However, if one core makeup tank is inadvertently actuated by a single failure, the event may progress with the plant at power until a reactor trip is reached. For the plant under automatic rod control, a reactor trip on high-3 pressurizer water level reactor trip is expected to occur followed by the PRHR actuation and eventually by an "“” signal, which would then actuate the second core makeup tank. When a consequential loss of offsite power is assumed, this event is more conservative than the spurious “S” signal event.

The inadvertent opening of the core makeup tank discharge valves, due to operator error or valve failure, results in significant core makeup tank injection flow leading to a boration. If the automatic rod control system is operable, it will begin to withdraw rods from the core to counteract the reactivity effects of the boration. As a result, the core makeup tank will continue injection and slowly raise the pressurizer level until the high-3 pressurizer level trip setpoint is reached. A loss of offsite power is assumed to occur as a consequence of reactor trip. The primary effect of this assumption is the coastdown of the reactor coolant pumps. The core makeup tank injection will increase as the steam generator outlet temperature increases resulting in a lower density in the CMT balance line. This event is more limiting primarily due to the higher pressurizer level at the time of reactor trip and to the significant heat up of the injected fluid during the pre-trip phase of the accident. Thus, the inadvertent core makeup tank actuation event with a consequential loss of offsite power is analyzed here.

Upon receipt of the high-3 pressurizer level reactor trip signal, the reactor is tripped; then the turbine is immediately tripped, and after a 3-second delay, a consequential loss of offsite power is assumed. The high-3 pressurizer level signal also actuates the PRHR heat exchanger and blocks the pressurizer heaters, but a 15-second delay is built in to prevent unnecessary actuation of the PRHR heat exchanger if offsite power is maintained.

Following reactor trip, the reactor power drops and the average reactor coolant system temperature decreases with subsequent coolant shrinkage. However, due to the assumed loss of offsite power, the reactor coolant cold leg temperature, in the loop without PRHR, increases and the core makeup tank starts injecting cold water into the reactor coolant system at a much higher rate. The primary coolant system shrinkage is counteracted by the core makeup tank injection, and the pressurizer water volume starts to increase because of the heatup of the cold injected fluid by the decay heat. The high-3 pressurizer level setpoint is once again reached, and after a 15-second delay, the signal is sent to actuate the PRHR heat exchanger and block the pressurizer heaters.

Eventually, the core makeup tank heats up and the gravity-driven recirculation is significantly reduced. The PRHR heat exchanger continues to extract heat from the reactor coolant system, and the pressurizer water volume starts to decrease. Ultimately, the core makeup tank stops recirculating, the PRHR heat removal matches decay heat and the reactor coolant system cooldown begins eventually leading to a “S” signal on a Low T_{cold} setpoint.

The cold injection flow from the second CMT initially results in a fast decrease in temperature and shrinkage of the reactor coolant. However, as the temperature decreases, the PRHR heat removal capability diminishes and a moderate heat up occurs followed by the increase of pressurizer water level. The second CMT injection rate is much lower than that experienced during the first part of the transient from the first CMT. Due to the colder cold leg temperatures, the density in balance line is much higher than during the first part of the transient, resulting in a reduction of the total buoyancy driving head. Ultimately, the PRHR heat removal once again matches the decay heat and the final reactor coolant system cooldown begins.

The plant response to an inadvertent core makeup tank actuation is analyzed. The results of this analysis show that inadvertent operation of the core makeup tanks during power operation does not adversely affect the core, the reactor coolant system, or the steam system. The PRHR heat removal capacity is such that reactor coolant water is not relieved from the pressurizer safety valves. DNBR always remains above the design limit values, and reactor coolant system and steam generator pressures remain below 110 percent of their design values.

9.5.2 Chemical and Volume Control System Malfunction that Increases Reactor Coolant Inventory

The increase of reactor coolant system coolant inventory may be due to the spurious operationof one or both of the chemical and volume control system pumps or by the closure of the letdown path. If the chemical and volume control system is injecting highly borated water into the reactor coolant system, the reactor experiences a negative reactivity excursion due to the injected boron, causing a decrease in reactor power and subsequent coolant shrinkage. The load decreases due to the effect of reduced steam pressure after the turbine throttle valve fully opens.

At high chemical and volume control system boron concentration, low reactivity feedback conditions, and reactor in manual rod control, an “S” signal will be generated by either the Low T_{cold} or low steamline pressure setpoints before the chemical and volume control system can inject a significant amount of water into the reactor coolant system. In this case, the chemical and volume control system malfunction event proceeds similarly to, and is only slightly more limiting than, a spurious “S” signal event. If the automatic rod control is modeled and the pressurizer spray functions properly to prevent a high pressure reactor trip signal, no “S” signals are generated and this specific event is terminated by automatic isolation of the chemical and volume control system on the safety-related high-2 pressurizer level setpoint.

The analyses are performed for the transient response to a chemical and volume control system malfunction that results in an increase of reactor coolant system inventory. Neutron flux slowly decreases due to boron injection, but steam flow does not decrease until later in the transient when the turbine throttle valves are wide open.

As the chemical and volume control system injection flow increases reactor coolant system inventory, pressurizer water volume begins increasing while the primary system is cooling down. At about 1,009 seconds, the Low T_{cold} setpoint is reached, the reactor trips, and the control rods start moving into the core.

Immediately following reactor trip, the turbine is tripped and after a 3-second delay, a consequential loss of offsite power is assumed and the reactor coolant pumps trip. Soon after reactor trip, the pressurizer heaters are blocked and the main feedwater lines, steam lines, and chemical and volume control system are isolated. After a conservative 17-second delay, the PRHR heat exchanger is actuated and the core makeup tank discharge valves are opened. The core makeup tanks work in recirculation mode, meaning they are always filled with water because cold borated water injected through the injection lines is replaced by hot water coming from the cold leg balance lines.

The operation of the PRHR heat exchanger and the core makeup tanks cools down the plant. Due to the swelling of the core makeup tank water, the pressurizer level is still increasing. As the reactor coolant system average temperature goes below 490°F, the cooling effect due to the core makeup tanks is decreasing. In this condition, the PRHR heat exchanger cannot remove the entire decay heat. Reactor coolant system temperature tends to increase until anequilibrium between decay heat power and heat absorbed by the PRHR heat exchanger is reached.

When the PRHR heat flux matches the core decay heat, the pressurizer water volume stops increasing, and the pressurizer safety valves close. Then the core makeup tanks essentially stop injecting.

The results of this analysis demonstrate that a chemical and volume control system malfunction does not adversely affect the core, the reactor coolant system, or the steam system. The PRHR heat removal capacity is such that reactor coolant water is not relieved from pressurizer safety valves. DNBR remains above the design limit values, and reactor coolant system and steam generator pressures remain below 110 percent of their design valves.

9.6 DESIGN CONTROL DOCUMENTS

The following sections of the AP1000 DCD contain information relative to Non-LOCA Analyses:

- Section 15.1 Increase in Heat Removal From the Primary System
- Section 15.2 Decrease in Heat Removal by the Secondary System
- Section 15.3 Decrease in Reactor Coolant System Flow Rate
- Section 15.4 Reactivity and Power Distribution Anomalies
- Section 15.5 Increase in Reactor Coolant Inventory

10 CONTAINMENT RESPONSE ANALYSES

10.1 ANALYSES SUMMARY

The containment system is designed such that for all break size, up to and including the double-ended severance of a reactor coolant pipe or secondary side pipe, the containment peak pressure is below the design pressure.

This capability is maintained by the containment system assuming the worst single failure affecting the operation of the passive containment cooling system (PCS). For primary system breaks, loss of offsite power is assumed. For secondary system breaks, offsite power is assumed to be available when it maximizes the mass and energy released from the break.

The single failure postulated for the containment pressure/temperature calculations is the failure of one of the valves controlling the cooling water flow for the PCS. Failure of one of these valves would lead to cooling water flow being delivered to the containment vessel through two of three delivery headers. This results in reduced cooling flow for PCS operation. No other single failures are postulated in the containment analysis.

The containment integrity analyses for the AP1000 employ a multivolume lumped parameter model to study the long-term containment response to postulated Loss of Coolant Accidents (LOCA) and Main Steam Line Break (MSLB) accidents.

The analyses are based on assumptions that are conservative with respect to the containment and its heat removal systems, such as minimum heat removal, and maximum initial containment pressure.

10.2 ANALYSES RESULTS

For the LOCA events, two double-ended guillotine reactor coolant system pipe breaks are analyzed. The breaks are postulated to occur in either a hot or a cold leg of the reactor coolant system. The hot leg break results in the highest blowdown peak pressure. The cold leg break results in the higher post-blowdown peak pressure. The cold leg break analysis includes the long term contribution to containment pressure from the sources of stored energy, such as the steam generators.

For the MSLB event, a representative pipe break spectrum is analyzed. Various break sizes and power levels are analyzed with the WGOTHIC code. The MSLB mass and energy releases are used for these calculations.

The results of the LOCA and MSLB postulated accidents analyses demonstrate that the containment pressure response for the peak pressure is below design pressure (59 psig) in each case. For LOCA, the calculated results show that the peak containment pressure is 50.0 psig for double-ended hot leg guillotine and 57.8 psig for double-ended cold leg guillotine. The peak containment temperature is 416.5°F for double-ended hot leg guillotine and 284.9°F for double-ended cold leg guillotine. For MSLB, the calculated results of main steamline double-ended rupture with MSIV failure condition are that the peak containment pressure is 57.3 psig for 30% power and 53.7 psig for 101% power. The peak containment temperature is 373.9°F for 30% power and 375.3°F for 101% power.

10.3 CONTAINMENT SUBCOMPARTMENTS

10.3.1 Summary of Subcompartment Pipe Break Analyses

Subcompartments within containment are designed to withstand the transient differential pressures of a postulated pipe break. These subcompartments are vented so that differential pressures remain within structural limits.

Each subcompartment is analyzed for effects of differential pressures resulting from the break of the most limiting line in the subcompartment which has not been evaluated for LBB. The subcompartment analysis demonstrates that the wall differential pressuress resulting from the most limiting high energy line break within the subcompartments are within the design capability.

10.3.2 Pipe Breaks Evaluation

The subcompartment analysis for the steam generator compartment is performed assuming a double-ended guillotine break in a 3-inch inside diameter reactor cooling system hot leg or cold leg pipe or a 4-inch double ended steam generator blowdown line, or a 4-inch pressurizer spray line break.

The analysis for the pressurizer compartment pipe and valve room is performed assuming a double-ended guillotine break in a 4-inch inside diameter reactor coolant system spray line. This break envelopes the branch lines that could be postulated to rupture in this area.

The analysis for the steam generator vertical access area is performed assuming a double-ended guillotine break in a 3-inch inside diameter reactor coolant system cold-leg pipe. This break envelopes the branch lines that could be postulated to rupture in this area.

The analysis for the maintenance floor and operating deck compartments are performed assuming a one square foot rupture of a main steamline pipe. This break envelopes the branch lines that could be postulated to rupture in these areas.

The analysis for the main chemical and volume control system room is performed assuming a single-ended guillotine break in a 3-inch diameter reactor coolant system cold-leg pipe. This break envelopes the branch lines that could be postulated to rupture in this area.

The analysis for the pipe tunnel in the chemical and volume control system room is performed assuming a double-ended guillotine break in a 4-inch diameter steam generator blowdown line. This double-ended break envelopes the branch lines that could be postulated to rupture in this area.

10.4 MASS AND ENERGY RELEASE ANALYSES FOR POSTULATED PIPE RUPTURES

Mass and Energy releases are documented for two different types of transients. The first type is used to calculate the releases for the subcompartment differential pressure analysis (referred to as the short term analysis). These releases are used for the subcompartment response.

The second type is used to determine the releases for the containment pressure and temperature calculations (referred to as the long term analysis). These releases are used for the containment integrity analysis.

The short term analysis considers only the initial stages of the blowdown transient, and takes into consideration the application of LBB methodology. Since LBB is applicable to reactor coolant system piping that is 6 inches in diameter and greater, the mass and energy release analysis for sub-compartments postulates the complete DEG severance of 3-inch and 4-inch pipe. The mass and energy release postulated for a ruptured steam line is for a one square foot break.

Conversely, the limiting break size for containment integrity analysis considers as its LOCA design basis the complete DEG severance of the largest reactor coolant system pipe.

The containment system receives mass and energy releases following a postulated rupture of the reactor coolant system. The release rates are calculated for pipe failure at two locations: the hot leg and the cold leg. These break locations are analyzed for both the short-term and the long-term transients. Because the initial operating pressure of the reactor coolant systems approximately 2250 psi, the mass and energy are released extremely rapidly when the break occurs. As the water exits from the broken pipe, a portion of it flashed to steam because of the differences in pressure and temperature between the reactor coolant system and containment. The reactor coolant system depressurizes rapidly since break flow exits from both sides of pipe in a DEG severance.

10.5 MASS AND ENERGY RELEASE ANALYSIS FOR POSTULATED SECONDARY-SYSTEM PIPE RUPTURE INSIDE CONTAINMENT

Steam line ruptures occurring inside a reactor containment structure may result in significant releases of high-energy fluid to the containment environment, possibly resulting in high containment temperatures and pressures. The quantitative nature of the releases following a steam line rupture is dependent upon the configuration of the plant steam system, the containment design as well as the plant operating conditions and the size of the rupture. The methods used in determining the containment responses to a variety of postulated pipe breaks encompassing variations in plant operation are described in AP1000 Design Control Document (DCD).

10.6 DOCUMENTATION

The following section of the AP1000 DCD contains more information relative to containment response analyses:

- Section 6.2 Containment Systems