1 INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

On March 28, 2002, Westinghouse Electric Company (hereinafter referred to as Westinghouse or the applicant) tendered its application for certification of the AP1000 standard nuclear reactor design with the U.S. Nuclear Regulatory Commission (the NRC or staff). The applicant submitted this application in accordance with Subpart B, "Standard Design Certifications," of Title 10 of the Code of Federal Regulations (10 CFR) Part 52, and Appendix O of 10 CFR Part 52, "Standardization of Design: Staff Review of Standard Designs." The application included the AP1000 design control document (DCD) and the AP1000 probabilistic risk assessment (PRA) report. The NRC formally accepted the application as a docketed application for design certification (Docket No. 52-006) on June 25, 2002. Information submitted before that date can be found under Project No. 711.

The applicant originally submitted the AP1000 DCD on March 28, 2002. The DCD Tier 1 information includes the inspections, tests, analyses, and acceptance criteria (ITAAC), and Tier 2 information describes the design of the facility. Subsequently, the applicant supplemented the information in the DCD by providing revisions to that document. The applicant submitted the most recent DCD Revision 5 to the Commission on May 19, 2003. Similarly, the applicant originally submitted the PRA on March 28, 2002. It has been revised through Revision 2 (submitted by letter dated April 15, 2003). In addition, throughout the course of the review, the NRC staff requested that the applicant submit additional information to clarify the description of the AP1000 design. Some of the applicant's responses to these requests for additional information (RAIs) are discussed throughout this report. Appendix E of the final safety evaluation report will provide a listing of the issuance and response dates. The DCD, PRA, Tier 1 information, and all other pertinent information and materials are available for public inspection at the NRC Public Document Room and added to the Agencywide Documents Access and Management System Public Electronic Reading Room (ADAMS).

This report summarizes the staff's safety review of the AP1000 design against the requirements of Subpart B of 10 CFR Part 52, "Standard Design Certifications," and delineates the scope of the technical details considered in evaluating the proposed design. Sections 1.2 and 1.3 of this report summarize the AP1000 design. Section 1.4 identifies agents and contractors. Section 1.5 provides a discussion of the principal matters that were the subjects of the staff's review. The staff has not completed its review of the design, and has identified open and confirmatory items (in Sections 1.6 and 1.7 of this report) that must be satisfactorily resolved before the staff can complete its review.

Unless otherwise noted, this report is based on DCD Revision 3, dated February 6, 2003. Westinghouse has provided Revision 4 to the DCD dated April 15, 2003, and Revision 5 to the

DCD dated May 19, 2003. The staff's review of these revisions to determine their impact on the conclusions in this report is Open Item 1.1-1

The staff has not issued RAIs regarding the security portion of the AP1000 design. The staff will issue a schedule for the security review of the AP1000, including milestones and target dates for issuance of security-related RAIs, under separate correspondence.
The review documented in Section 14.2 of Chapter 14 reflects the staff assessment of the Initial Plant Test Program with the exception of certain aspects of the testing scope, general test methods, and acceptance criteria. Pending completion of the review, this is Open Item 14.2-1.

The staff's evaluation documented in Chapter 21 concentrates on the differences between the AP1000 and the AP600 design with the understanding that the AP600 testing and computer codes were found to be acceptable for the AP600 design in accordance with the staff's evaluation documented in Chapter 21 of NUREG-1512, "Final Safety Evaluation Report Related to Certification of the AP600 Standard Design," September 1998. Chapter 21 currently contains references to NUREG-1512, which provides the basis for accepting the AP600 testing and computer codes. Prior to issuing the final safety evaluation report for the AP1000, the staff will remove these references and replace the references with the basis for its conclusion that the testing and computer codes are acceptable for the AP1000. This is Open Item 21.1-1.

The staff will then issue a final safety evaluation report after resolving the open and confirmatory items identified in this report.

This report presents the status of the staff's review of information submitted to the NRC through April 21, 2003, the cutoff-date for consideration in this report. In order to close the open and confirmatory items listed in this report, the staff requires the additional information identified in this report. Additional information to close these items may already have been provided in material received by the staff after the April 21, 2003, cutoff-date. The staff will continue its review of material already submitted, as well as any new information submitted by the applicant, in an effort to close the open and confirmatory items consistent with the schedule for the final report provided in a letter dated July 12, 2002. The staff plans to reevaluate the schedule after this report is issued. Maintaining the schedule will depend on timely, high-quality responses by the applicant to the items identified in this report. While meetings and submittals providing increased detail or clarity are expected after April 21, 2003, submittal of significant new information after that date could result in a delay of the schedule for issuance of the final report.

1.1.1 Metrication

This report conforms with the Commission's Policy Statement on metrication (61 FR 31169). Therefore, all measures are expressed as metric units, followed by English units in parentheses.

1.1.2 Proprietary Information

This report references several of Westinghouse's reports. Some of these reports contain information that the applicant requested be withheld, exempt from public disclosure, as provided by 10 CFR Part 2.790. For each such report, the applicant provided a nonproprietary version, similar in content except for the omission of the proprietary information. The staff predicated its findings on the proprietary versions of these documents, so the staff refers only to the proprietary versions throughout this report.

1.1.3 Combined License Applicants Referencing the AP1000 Design

Applicants who reference the AP1000 standard design in the future for specific facilities will retain architect-engineers, constructors, and consultants, as needed. As part of its review of an application for a combined license (COL) that references the AP1000 design, the staff will evaluate, for each plant-specific application that references the AP1000 design, the technical competence of the COL applicant and its contractors to manage, design, construct, and operate a nuclear power plant. The plant-specific applicants will also be required to satisfy the requirements of 10 CFR Part 52, Subpart C and any requirements resulting from the staff's review of this standard design. The applicant has identified matters to be addressed by plant-specific applicants as "Combined License Information" throughout the DCD. The staff has also identified such matters, referred to as "COL Action Items," throughout this report. A cross-reference of the COL action items and the COL information is provided in Appendix F of this report.

1.1.4 Additional Information

Appendix A to this report provides a chronology of the principal actions, submittals, and amendments related to the processing of the application. Appendix B provides a list of references identified in this report. Appendix C provides a list containing definitions of the acronyms and abbreviations used throughout this report. Appendix D lists the principal technical reviewers who evaluated the AP1000 design. Appendix E, F and G will be provided in the final safety evaluation report. Appendix E will provide an index of the staff's RAIs and the applicant's responses. Appendix F will provide a cross-reference of the COL information in the DCD, FSER, and COL action items. Appendix G will provide the report issued by the Advisory Committee on Reactor Safeguards (ACRS).

The NRC's licensing project managers assigned to the AP1000 standard design review are Mr. John P. Segala, Mrs. Joelle Starefos and Mr. Joseph Colaccino. They may be reached by calling (301) 415-7000, or by writing to the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, DC 20555-0001.

1.2 General Design Description

1.2.1 Scope of the AP1000 Design

The requirement that governs the scope of the AP1000 design is in 10 CFR Part 52.47(b)(2)(i)(A)(4), which requires that an applicant for certification provide a complete design scope, except for site-specific elements. Therefore, the scope of the AP1000 design must include all of the plant structures, systems, and components that can affect the safe operation of the plant, except for its site-specific elements. The applicant described the AP1000 standard design scope in DCD Tier 2 Section 1.8, including the site-specific elements that are either partially or wholly outside of the standard design scope. The applicant also described interface requirements (see DCD Tier 2 Table 1.8-1) and representative conceptual designs, as required by 10 CFR Part 52.47(a)(1)(vii) and 10 CFR Part 52.47(a)(1)(ix), respectively.

1.2.2 Summary of the AP1000 Design

The AP1000 design has a nuclear steam supply system (NSSS) power rating of 3415 Mwt with an electrical output of at least 1000 Mwe. The plant is designated for rated performance with up to 10 percent of the steam generator (SG) tubes plugged and with a maximum hot leg temperature of 321.1°C (610°F). The plant is designed to accept a step load increase or decrease of 10 percent between 25 and 100 percent power without reactor trip or steam dump system actuation, provided that the rated power level is not exceeded. In DCD Tier 2 Section 1.2, the applicant also indicates that the plant is designed to accept a 100-percent load rejection from full power to house loads without a reactor trip or operation of the pressurizer or SG safety valves. The goal for the overall plant availability is projected to be greater than 90 percent, considering all forced and planned outages, with a rate of less than one unplanned reactor trip per year. The applicant states that the plant has a design objective of 60 years without a planned replacement of the reactor vessel. However, the design does provide for replaceability of other major components, including the SG. The following is a general description of the AP1000 design. A detailed description of each system is provided in the section of this report that discusses the given system.

1.2.2.1 Reactor Coolant System Design

The AP1000 reactor coolant system (RCS) is designed to effectively remove or enable removal of heat from the reactor during all modes of operation, including shutdown and accident conditions.

The system consists of two heat transfer circuits, each with the following components:

- an SG
- two Reactor coolant pumps (RCPs)
- a single hot leg
- two cold legs

In addition, the system includes a pressurizer, interconnecting piping, valves, and the instrumentation necessary for operational control and safeguards actuation. All of the system equipment is located within the reactor containment. Figure 1.2-1 shows a diagram of the AP1000 RCS.

The reactor system pressure is controlled by operation of the pressurizer. Overpressure protection for the RCS is provided by the spring-loaded safety valves installed on the pressurizer. These safety valves discharge to the containment atmosphere. The valves for the first 3 stages of automatic depressurization are also mounted on the pressurizer. These valves discharge steam through spargers to the in-containment refueling water storage tank (IRWST) of the passive core cooling system (PXS). The discharged steam is condensed and cooled by mixing with water in the tank.

The following auxiliary systems interface with the RCS:

- chemical and volume control system (CVS)
- component cooling water system
- liquid radwaste system
- primary sampling system
- PXS
- spent fuel pit cooling system
- SG system

1.2.2.2 Reactor Design

An AP1000 fuel assembly consists of 264 fuel rods in a 17x17 square array. The fuel grids consist of an eggcrate arrangement of interlocked straps that maintain lateral spacing between the rods. The fuel rods consist of enriched uranium, in the form of cylindrical pellets of uranium dioxide, contained in ZIRLO tubing. The tubing is plugged and seals welded at the ends to encapsulate the fuel. An axial blanket comprised of fuel pellets with reduced enrichment may be placed at each end of the enriched fuel pellet stack to reduce the neutron leakage and improve fuel utilization. Other types of fuel rods may be used to varying degrees within some fuel assemblies. One type uses an integral fuel burnable absorber containing a thin boride coating on the surface of the fuel pellets. Another type uses fuel pellets containing gadolinium oxide mixed with uranium oxide. The boride-coated fuel pellets provide burnable absorber integral to the fuel.

The applicant states that the reactor core is designed for a 18-month fuel cycle. A core design is maintained for projected fuel cycles. The reactor core is located low in the vessel to minimize core temperature during a postulated loss-of-coolant accident (LOCA). The core is designed to have a moderator temperature coefficient that is non-positive over the entire fuel cycle and at any power level, with the reactor coolant at the normal operating temperature. The core design provides an adequate margin so that departure from nucleate boiling will not occur with a 95 percent probability and 95 percent confidence basis for all Condition I and II events. No vessel penetrations exist below the top of the core because the AP1000 does not use bottom-mounted in-core instrumentation. In addition, the design employs an integrated head package that consists of the following components:

- control rod drive mechanisms
- integrated head cooling fans
- instrument columns
- insulation
- seismic support
- package lift rig

A permanent, welded-seal ring is used to provide the seal between the vessel flange and the refueling cavity floor.

1.2.2.3 Steam Generator Design

The AP1000 design uses the Model Delta 125 SG, which employs thermally-treated nickel-chromium-iron Alloy 690 tubes and a steam separator area sludge trap with clean-out provisions. The channel head is designed to directly attach the two RCPs, and to allow both manual and robotic access for inspection, plugging, sleeving, and nozzle dam placement operations.

1.2.2.4 Reactor Coolant Pump Design

The four AP1000 RCPs are hermetically sealed canned pumps. Two RCPs are attached directly to the SG channel head with the motor located below the channel head to simplify the loop piping and eliminate fuel uncovery during postulated small-break LOCA scenarios. Each RCP includes sufficient internal rotating inertia to permit coastdown to avoid departure from nucleate boiling following a postulated loss-of-coolant flow accident. Each pump impeller and diffuser vane is ground and polished to minimize radioactive crud deposition and maximize pump efficiency. The RCPs are designed such that they are not damaged due to a loss of all cooling water until a safety-related pump trip occurs on high bearing water temperature. This automatic protection is provided to protect the RCPs from an extended loss of coolant water.

1.2.2.5 Pressurizer and Loop Arrangement

The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads. One spray nozzle and two nozzles for connecting the safety and depressurization valve inlet headers are located in the top head. Electrical heaters are installed through the bottom head. The piping layouts for the AP1000 is designed to provide adequate thermal expansion flexibility, assuming a fixed vessel and a free-floating SG/RCP support system. The reactor coolant loop and surge line piping are designed to leak before break criteria. The pressurizer itself is designed such that, with design spray flow rates, the power-operated relief valve function is neither required nor provided.

1.2.2.6 Steam and Power Conversion System Design

Turbine Generator

The AP1000 turbine generator design consists of a double-flow, high-pressure cylinder (high-pressure turbine) and three double-flow, low-pressure cylinders (low-pressure turbines) that exhaust to the condenser. It is a six-flow, tandem-compound, 1800-rpm machine. The turbine system includes the following components:

- stop, control, and intercept valves directly attached to the turbine and in the steam flow path
- crossover and cross under piping between the turbine cylinders and the moisture separator reheaters

The high-pressure turbine has extraction connections for one stage of feedwater heating, and its exhaust provides steam for one stage of feedwater heating in the deaerator. The low-pressure turbines have extraction connections for four stages of feedwater heating.

There are two moisture separator reheaters located between the high-pressure turbine exhaust and the low-pressure turbine inlet. The moisture separator reheater is an integral component of the turbine system, which extracts moisture from the steam, and reheats the steam to improve turbine system performance. The reheater has a single stage of reheat.

The turbine is oriented in a manner that minimizes potential interactions between turbine missiles and safety-related structures and components.

Main Steam System

The main steam system is designed to supply steam from the SG to the high-pressure turbine over a range of flows and pressures for the entire plant operating range. The main steam system is also designed to dissipate the heat generated by the nuclear steam supply system to the condenser through the steam dump valves, or to the atmosphere through power-operated atmospheric relief valves or spring-loaded main steam safety valves, when either the turbine generator or the condenser is not available. There are two steam headers with each one utilizing six SG safety valves.

Main Feedwater and Condensate System

The main feedwater system is designed to supply the SGs with adequate feedwater during all modes of plant operation, including transient conditions. The condensate system is designed to condense and collect steam from the low-pressure turbines and turbine bypass systems, and then transfer this condensate from the main condenser to the deaerator. The applicant states that the main feedwater and condensate systems are designed for increased availability and improved dissolved oxygen control.

1.2.2.7 Engineered Safeguards Systems Design

The engineered safeguards systems consist of the following systems and components. Figure 1.2-2 shows some of the passive safety features, including the containment, the passive containment cooling system, and the passive core cooling system.

- The containment vessel is a free-standing cylindrical steel vessel. Its engineered safety feature (ESF) function is to contain the release of radioactivity following a postulated design basis accident (DBA). It provides shielding for the reactor core and reactor coolant system during normal operation. It also functions as the safety-related ultimate heat sink for the removal of the reactor coolant system sensible heat, core decay heat, and stored energy.

- The passive containment cooling system (PCS) consists of the following components:

    - a passive containment cooling water storage tank that is incorporated in the shield building structure above the containment

    - an air baffle that is located between the steel containment vessel and the concrete shield building

    - air inlet and exhaust paths that are incorporated in the shield building structure

    - a water distribution system

- an ancillary water storage tank and two recirculation pumps for onsite storage of additional PCS cooling water

On actuation, the PCS delivers water to the top, external surface of the steel containment shell, which forms a film of water over the dome and side walls of the containment structure. Air is induced to flow over the containment as it is heated, causing a chimney effect. This air flow and cooling water evaporation removes the heat generated within the containment and expels it to the outside air. The applicant states that the passive containment cooling system maintains the containment pressure and temperature within the appropriate design limits for both design basis and severe accident scenarios. Figure 1.2-3 shows the passive containment cooling system.

- The major function of the containment isolation system is to provide containment isolation to allow the normal or emergency passage of fluids through the containment boundary while preserving the integrity of the containment boundary. This prevents or limits the escape of fission products that may result from postulated accidents. The containment isolation provisions are designed so that fluid lines that penetrate the primary containment boundary are isolated in the event of an accident. The system consists of the piping, valves, and actuators that isolate the containment.

- The containment hydrogen control system controls the hydrogen concentration in the containment so that containment integrity is not endangered. It consists of the hydrogen monitoring system, passive autocatalytic hydrogen recombiners, and hydrogen ignitors.

- The PXS provides emergency core cooling following postulated design basis events. The PXS is comprised of the following components:

  – two core makeup tanks
  – two accumulators
  – the IRWST
  – a passive residual heat removal (PRHR) heat exchanger
  – pH adjustment baskets
  – associated piping and valves

- The automatic depressurization system, which is part of the RCS, and also provides important passive core cooling functions by depressurizing the RCS. The PXS system provides emergency core cooling following a postulated DBA by providing (1) RCS makeup water and boration when the normal makeup supply is lost or insufficient, (2) safety injection to the RCS to provide adequate core cooling during a postulated DBA, and (3) core decay heat removal during transients and accidents. Figure 1.2-4 shows the safety injection systems.

- The main control room (MCR) emergency habitability system is comprised of a set of storage tanks connected to a main and an alternate air delivery line. Components common to both lines include a manual isolation valve, a pressure regulating valve, and a flow metering orifice. This system is designed to provide the ventilation and pressurization requirements to maintain a habitable environment in the MCR for 72 hours following any DBA.

In DCD Tier 2 Section 1.2.1.4.1 the applicant states that the engineered safeguards systems are designed to mitigate the consequences of DBAs with a single failure. With the exception of the MCR emergency habitability system, the passive safety systems are designed to cool the RCS from normal operating temperatures to safe-shutdown conditions. In addition, all of these systems are designed to maximize the use of natural driving forces, such as pressurized nitrogen, gravity flow, and natural circulation flow. They do not rely on active components such as pumps, fans, or diesel generators to function. They do, however, use valves to initially align the safety systems when activated. In addition, the safety systems are designed to function without safety-related support systems, such as alternating current; component cooling water; service water; or heating, ventilation, and air conditioning (HVAC).

The number and complexity of operator actions required to control the safety systems are minimized. In meeting this objective, the approach was to eliminate the required action and not to automate them.
An automatic reactor coolant system depressurization feature is included in the design and meets the following criteria:

- The reliability (redundancy and diversity) of the automatic depressurization system valves and controls satisfies single failure criteria as well as the failure tolerance required by the low core melt frequency goals.

- The design provides for both real demands (such as reactor coolant system leaks and failure of the CVS makeup pumps) and spurious instrumentation signals. The probability of significant flooding of the containment due to the use of the automatic depressurization system is less than once in 600 years.

The design is such that, for small break loss-of-coolant accidents up to 8 inches in diameter, the core remains covered.

Non-Safety-Related Systems Designs

The applicant states that the non-safety-related systems used in the AP1000 are not relied on to provide safety functions required to mitigate DBAs. The AP1000 includes active systems that provide defense-in-depth (DID) (or investment protection) capabilities for reactor coolant system makeup and decay heat removal. These active systems are the first line of defense to reduce challenges to the passive systems in the event of transients or plant upsets. Most active systems in the AP1000 are designated as non-safety-related.

Examples of non-safety-related systems that provide DID capabilities for the AP1000 design include the CVS, normal residual heat removal system, and the startup (backup) feedwater system. For these DID systems to operate, the associated systems and structures to support these functions must also be operable, including non-safety-related standby diesel generators, the component cooling water system, and the service water system. The AP1000 also includes other active systems, designated as non-safety-related, such as the HVAC system, that remove heat from the instrumentation and control (I&C) cabinet rooms and the main control room to limit challenges to the passive safety capabilities for these functions.

In existing plants, and in the evolutionary advanced light water reactor (ALWR) designs, many of these active systems are designated as safety-related. However, by virtue of their designation in the AP1000 design as non-safety-related, credit is generally not taken for the active systems in the Chapter 15 licensing DBA analyses except in certain cases where operation of a non-safety-related system could make an accident worse.

The residual uncertainties associated with passive safety system performance increase the importance of active non-safety-related systems in providing DID functions to the passive systems. The staff does not require that these active systems meet all of the criteria imposed on safety-related systems, but expects a high level of confidence that active systems which have a significant safety role will be available when challenged. As discussed in SECY-94-084, “Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems in Passive Plant Design,” March 28, 1994, a process was developed for maintaining appropriate regulatory oversight of these active systems in the passive ALWR designs. In an SRM dated June 30, 1994, the Commission approved the recommendations made in SECY-94-084 concerning the issue of regulatory treatment of non-safety-related systems (RTNSS). The staff’s evaluation of RTNSS is contained in Chapter 22 of this report.

1.2.2.8 Instrumentation and Control System and Electrical System Designs

Control and Protection Systems Designs

The AP1000 control and protection systems are significantly different from I&C systems in operating reactor designs. In particular, the AP1000 employs digital microprocessor-based I&C systems, instead of the analog electronics, relay logic, and hard-wired systems currently used in most operating plants. In DCD Tier 2 Section 1.2.1.5.1 the applicant states that the design of the control and protection system ensures that a single failure in the I&C system will not result in a reactor trip or ESF actuation during normal operation. The design is intended to reduce the potential for a reactor trip and for a safeguards actuation because of failures in the reactor control or protection systems as compared to current operating plants.

The number of measured plant variables used for reactor trip and for safeguards actuation is minimized relative to current operating plants. The margin between the normal operating condition and the protection system setpoints is increased relative to current operating plants. The potential for interaction between the protection and safety monitoring system and the plant control system is reduced relative to current operating plants by incorporating a signal selector function that selects signals for control and for protection.

The AP1000 I&C systems is comprised of the following major systems:

- protection and safety monitoring system (PMS)
- special monitoring system (SMS)
- plant control system (PLS)
- diverse actuation system (DAS)
- operation and control centers system (OCS)
- data and display processing system (DDS)
- incore instrumentation system (IIS)

The PMS (1) monitors plant processes using a variety of sensors; (2) performs calculations, comparisons, and logic functions based on those sensor inputs; and (3) actuates a variety of equipment. The PMS provides the safety-related functions necessary to control the plant during normal operation, to shutdown the plant, and to maintain the plant in safe shutdown condition. The PMS is also used to operate safety-related systems and components.

The SMS consists of specialized subsystems that interface with the instrumentation and control architecture to provide diagnostic and long-term monitoring functions.

The PLS (1) controls and coordinates the plant during start-up, ascent to power, power operation, and shutdown conditions; (2) integrates the automatic and manual control of the reactor, reactor coolant, and various reactor support processes for required normal and off-normal conditions; (3) controls the non-safety-related decay heat removal systems during shutdown; and (4) permits the operator to control plant components from the MCR or remote shutdown workstation.

The DAS (1) provides a backup to the PMS for some specific diverse automatic actuation, (2) provides diverse indications and controls to assist in operator manual actions, and (3) is a DID system that is also designed to provide essential protection functions in the event of a postulated
common-mode failure of the PMS.

The OCS includes the complete operational scope of the MCR, remote shutdown workstation, technical support center, local control stations, and the emergency operations facility.

The DDS comprises the equipment used for processing data that results in nonsafety-related alarms and displays for both normal and emergency plant operations.

The IIS provides a three-dimensional flux map of the reactor core. It also provides the protection and safety monitoring system with in-core thermocouple signals for postaccident inadequate core cooling monitor.

Alternating and Direct Current Power Designs

All safety-related electrical power is provided from the Class 1E direct current (dc) power system. The AP1000 does not include a separate safety-related alternate current (ac) power system. Safety-related dc power is provided to support reactor trip and engineered safeguards actuation. Batteries are sized to provide the necessary dc power and uninterruptable ac power for items such as the protection and safety monitoring system actuation, control room functions including habitability, dc-powered valves in the passive safety systems, and containment isolation.

Main Control Room (MCR) Design

The MCR controls the plant during normal and anticipated transients as well as DBAs. It includes indications and controls that are capable of monitoring and controlling the plant safety systems and the non-safety-related control systems. The MCR contains the safety-related instrumentation and controls to allow the operator to achieve and maintain safe shutdown following any DBA.

The MCR is serviced by redundant non-safety-related power sources and HVAC systems during normal operation. In the event that either the normal power source or HVAC system becomes unavailable, there are passive systems (batteries and compressed air) available that the applicant states will support MCR operation for up to 3 days. The safety-related power sources and passive cooling system are designed to provide a habitable environment for the operating staff, assuming that no ac power is available. The safety-related instrumentation (equipment racks) is maintained at acceptable ambient conditions for 3 days following a loss of all ac power by using a passive cooling system. After 3 days, the applicant states that it will be possible to continue operation with the control room cooled and ventilated with the natural circulation of outside air.

The operating staff can transfer control from the MCR to the remote shutdown workstation should they be required to leave the MCR. The remote shutdown workstation contains the safety-related indications and controls that allow an operator to achieve and maintain safe shutdown of the plant following an event when the MCR is unavailable.

1.2.2.9 Plant Arrangement

The AP1000 plant is arranged with the following principal building structures:

- the nuclear island
- the turbine building
- the annex building
- the diesel generator building
- the radwaste building

The nuclear island is structurally designed to meet seismic Category I requirements as defined in Regulatory Guide 1.29 Seismic Design Classification. The nuclear island consists of the following:

- a free-standing steel containment building
- a concrete shield building
- an auxiliary building

The nuclear island is designed to withstand the effects of postulated internal events such as fires and flooding without loss of capability to perform safety functions.

Figure 1.2-5 shows the AP1000 building layout.

The containment building is the containment vessel and the structures contained within the containment vessel. The shield building comprises the structure and annulus area that surrounds the containment building. The containment building is an integral part of the overall containment system with the functions of containing the release of airborne radioactivity following postulated DBA and providing shielding for the reactor coolant system during normal operations. The containment and shield buildings are an integral part of the passive containment cooling system. The auxiliary building protects and separates all of the seismic

Category I mechanical and electrical equipment located outside the containment building. It contains the MCR, I&C systems, dc system, fuel handling area, mechanical equipment areas, containment penetration areas, and main steam and feedwater isolation valve compartment.

The turbine building houses the main turbine, generator, and associated fluid and electrical systems. It also houses the makeup water purification system. No safety-related equipment is located in the turbine building.

The annex building provides the main personnel entrance to the power generation complex. The building includes the health physics area, the non-Class 1E ac and dc electric power systems, the ancillary diesel generators and their fuel supply, other electrical equipment, the technical support center, and various HVAC. No safety-related equipment is located in the annex buildings.

The diesel generator building houses two diesel generators and their associated HVAC equipment. No safety-related equipment is located in the diesel generator building. The building is a nonseismic structure designed for wind and seismic loads in accordance with the Uniform Building Code.

The radwaste building contains facilities for segregated storage of various categories of waste prior to proceeding, for processing by mobile systems, and for storing processed waste in shipping and disposal containers. No safety-related equipment is located in the radwaste building. It is a nonseismic structure designed for wind and seismic loads in accordance with the Uniform Building Code. The foundation for the building is a reinforced concrete mat on grade.

The overall plant arrangement utilizes building configurations and structural designs to minimize the building volumes and quantities of bulk materials (concrete, structural steel, and rebar) consistent with safety, operational, maintenance, and structural needs. The plant arrangements provides separation between safety-related and nonsafety-related systems to preclude adverse interaction between safety-related and nonsafety-related equipment. Separation between redundant safety-related equipment and systems provides confidence that the safety design functions can be performed. In general this separation is provided by partitioning an area with concrete walls.

1.3 Comparison With Similar Facility Designs

The AP1000 standard design contains many features that are not found in current operating reactor designs. For example, a variety of engineering and operational improvements provide additional safety margins and address the Commission’s severe accident, safety goal, and standardization policy statements. The most significant improvement to the design is the use of safety systems that use passive means, such as gravity, natural circulation, condensation and evaporation, and stored energy for accident prevention and mitigation. DCD Tier 2 Table 1.3-1 provides a detailed comparison of the principal design features of the AP1000 standard design with the certified AP600 and a typical two-loop plant.

1.4 Identification of Agents and Contractors

Westinghouse is the principal AP1000 designer. The following organizations provided the principal subcontracting services for the design of the AP1000:

- Avondale Industries, Incorporated
- Bechtel North American Power Corporation
- Burns & Roe Company
- Chicago Bridge & Iron Services, Incorporated
- MK-Ferguson Company
- Southern Electric International

Westinghouse received additional support from the following organizations:

- Badan Tenaga Atom Nasional of Indonesia
- BPPT of Indonesia
- European Nuclear Energy Association of Italy
- Ente Nazionale per l'Enerfia Elettrica of Italy
- FIAT of Italy
- Empresa Nacional de Ingeniería y TTecnología, S.A. of Spain
- Oregon State University
- PLN of Indonesia
- Società Progettazione Reattori Nucleari, SpA (SOPREN)/ANSALDO of Italy
- UNESA of Spain
- University of Western Ontario of Canada
- UTE of Spain
- EdF of France
- SNERDI of China
- MHI of Japan
- UAK of Switzerland
- DTN of Spain
- Fortum of Finland

1.5 Summary of Principal Review Matters

The procedure for certifying a design is conducted in accordance with the requirements of Subpart B of 10 CFR Part 52, and carried out in two stages. The technical review stage is initiated by an application filed in accordance with the requirements of 10 CFR Part 52.45, continues with reviews by the NRC staff and the ACRS, and concludes with the issuance of an final safety evaluation report (FSER) that discusses the staff's conclusions related to the acceptability of the design. The administrative review stage begins with the publication of a Federal Register notice that initiates rulemaking, in accordance with 10 CFR Part 52.51, and provides a proposed standard design certification rule. The rulemaking will be conducted by the Commission and also provides an opportunity for an informal hearing before an Atomic Safety and Licensing Board. The Board may also request authority from the Commission to use additional procedures, such as direct and cross examination by the parties, or may request that the Commission convene a formal hearing under Subpart G of 10 CFR Part 2 on specific and substantial disputes of fact, necessary for the Commission's decision, that cannot be resolved with sufficient accuracy except in a formal hearing. The rulemaking culminates with the denial or issuance of a design certification rule.

The staff performed its technical review of Westinghouse's application for certification of the AP1000 standard design in accordance with the requirements of 10 CFR Part 52 Sections 52.47, 52.48, and 52.53. The staff evaluated the technical information required by 10 CFR Part 52.47(a)(1)(i) in accordance with the standard review plan (NUREG-0800); that evaluation is the subject of this report. In addition to these safety standards, the staff followed Commission guidance provided in the SRMs for all applicable Commission papers, including those referenced throughout this report. In particular, SECY-93-087, "Policy, Technical, and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Reactor (ALWR) Designs," April 2, 1993; SECY-94-084, and SECY-95-132, "Policy and Technical Issues Associated with the Regulatory Treatment of Non-Safety Systems (RTNSS) in Passive Plant Designs (SECY-94-084)," May 22, 1995, identified staff positions generic to passive light-water reactor (LWR) design certification policy issues, and SECY-96-128, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design,'' June 12, 1996; SECY-97-044, "Policy and Key Technical Issues Pertaining to the Westinghouse AP600 Standardized Passive Reactor Design,'' February 19, 1997; and SECY-98-161, "The Westinghouse AP600 Standard Design as it Relates to the Fire Protection and the Spent Fuel Pool Cooling Systems,'' July 1, 1998; identified staff positions on issues specific to the AP600 design. In SRMs dated July 21, 1993; June 30, 1994; June 28, 1995; January 15, 1997; and June 30, 1997, the Commission provided its guidance on these matters as they pertain to passive plant designs. Unless otherwise noted, the staff is reviewing the application using the newest codes and standards that have been endorsed by the NRC.

The staff's evaluation of the technically relevant unresolved safety issues, generic safety issues, and Three Mile Island requirements (Sections 52.47(a)(1)(ii) and (iv)) is discussed in Chapter 20 of this report. The evaluation of the site parameters required by 10 CFR Part 52.47(a)(1)(iii) is discussed in Chapter 2. The staff's evaluation of the design-specific PRA (10 CFR Part 52.47(a)(1)(v)) is discussed in Section 19.1. The evaluation of the ITAAC required by 10 CFR Part 52.47(a)(1)(vi) is discussed in Section 14.3 of this report.

The staff evaluation of interface requirements and representative conceptual designs (10 CFR Part 52.47(a)(1)(vii) through (ix)) are discussed throughout selected chapters of this report, and will also be discussed in Chapter 14. The staff also implemented the Commission's Severe Accident Policy Statement, dated August 8, 1985, 50 FR 32138, and the Commission's staff requirement memorandums (SRMs) on SECYs-90-016, "Evolutinary Light Water Reactor (LWR) Certification Issues and Their Relationship to Current Regulatory Requirements," January 12, 1990; 93-087, 94-084, 95-132, 96-128, and 97-044, in its resolution of severe accident issues. The staff's evaluation of severe accident issues is discussed in Section 19.2 of this report.

The regulations in 10 CFR Part 52.47(a)(2) describe the level of design information needed to certify a standard design. In addition, the December 4, 1990 SRM for SECY-90-377, "Requirements for Design Certification Under 10 CFR Part 52," February 15, 1991, set forth the Commission's position on the level of design information required for a certification application, and the staff followed that guidance in preparing this report. The staff also followed the guidance of SECY-92-053, "Use of Design Acceptance Criteria During 10 CFR Part 52 Design Certification Reviews," February 19, 1992. To allow for technology improvements and as-procured equipment characteristics, the staff based its safety determinations on the use of design acceptance criteria (DAC) for certain technical areas. The DAC are part of the Tier 1 information proposed for the AP1000 design. The staff's evaluation of the Tier 1 information, including DAC and ITAAC, is in Section 14.3 of this report.

As part of its technical review, the staff issued numerous RAIs to gain sufficient bases for its safety findings, thereby meeting the requirement in 10 CFR Part 52.47(a)(3) to advise the applicant on the staff's requirements for additional technical information. Appendix E provides an index of the applicant's responses to these RAIs.

Section 1.2.1 of this report discusses the scope of the design to be certified. Because of the unique nature of the AP1000 design, the applicant implemented an extensive testing program to provide data on the passive safeguards systems. This data validates the safety analysis methods and computer codes, and provides information to assess the design margins in the passive safety system performance. The staff's evaluation of the testing program required pursuant to 10 CFR Part 52.47(b)(2) is discussed in Chapter 21 of this report. Because the AP1000 is designed as a single unit, that is, no safety systems will be shared at a multi-unit site, GDC 5 of Appendix A to 10 CFR Part 50 and 10 CFR Part 52.47(b)(3) do not apply to this design. Any applicant wishing to construct multiple units at a single site will be required to address these regulations in its application.

1.6 Summary of Open Items

As a result of the staff's review of Westinghouse's application for certification of the AP1000 design, including the additional information provided to the NRC through April 21, 2003, the staff identified the following issues that remained open at the time this report was prepared. An issue is open if the applicant has not provided requested information and the staff is unaware of what will be included in the promised submittal. Each open item was assigned a unique identifying number, which identifies the section in this report where the open item is described. For example, Open Item 4.4-1 is discussed in Section 4.4 of this report.

Item	Description
1.1-1	Unless otherwise noted, this report is based on DCD Revision 3, dated February 6, 2003.
1.9-1	The staff has not yet identified all of the Tier 2* information pertaining to the AP1000 design. This effort will be completed to support the final safety evaluation report.
1.10-1	The staff has not yet completed the cross-reference of the COL action items.
2.3.4-1	A review of the hypothetical reference control room χ/Q values calculated by the applicant has identified unresolved issues related to adequate justification for assuming a diffuse release, estimation of initial sigma values, other release assumptions, building cross-sectional areas, and distances between release/receptor pairs.

2.5.1-1	The DCD, while listing certain site specific aspects of basic geologic and seismic information to be provided by a COL applicant referencing the AP1000 certified design, does not include some of the attributes necessary to determine whether geologic features underlying the site affects of the foundation design are acceptable.
2.5.2-1	The DCD should include a probabilistic seismic hazard analysis, including the definition of controlling earthquakes.
2.5.4-1	With regard to stability of subsurface materials and foundations, a number of items have been identified which need to be addressed.
2.5.4-2	The DCD should specify the allowable average and allowable peak bearing capacity values.
2.5.4-3	The DCD does not provide any quantitative justification as to why a basemat tilting of a few inches will not affect functionality of structures, systems and components.
3.3.1-1	The wind load design for the AP1000 makes it unsuitable for sites that fall under exposure Category D.
3.3.1-2	The applicant should clarify its inconsistent use of the ASCE 7-98 recommendations for wind load design.
3.3.2-1	The applicant needs to identify all the structures for which it has used a pressure drop lower than 13.8 kPa (2 psi).
3.3.2-2	The site interface criteria for wind and tornado do not make it clear that the COL applicant needs to follow the three acceptable criteria described in DCD Tier 2 Section 3.3.2.3 to ensure that structures outside the scope of the certified design do not compromise the function of safety-related structures or systems of the AP1000 plant.
3.3.2-3	Prior to issuing the final safety evaluation report for the AP1000, the staff will amend Section 3.3.2.3 with the basis for its conclusion that the AP600 wind tunnel test reports are acceptable for the AP1000.
3.5.1.3-1	The applicant has not provided a basis for the maximum undetected flaw size.
3.5.1.3-2	Because the stress corrosion cracking (SCC) growth rate data set is considerably smaller than the data set relied upon in past analyses, the applicant needs to expand the database by including available data on the same material from other sources in its current analysis. Further, the

specified values or units for the coefficients in the SCC growth equation appear incorrect.

3.6.3.4-1	The applicant needs to modify its DCD Tier 2 Section 3.6.4 on COL information to indicate that COL holders should implement inspection plans, evaluation criteria, and other types of measures imposed on or adopted by operating PWRs with currently approved leak-before-break (LBB) applications as part of the resolution of concerns regarding the potential for PWSCC in those units.
3.6.3.4-2	This open item concerning LBB piping system analysis will be included in a supplement to the DSER.
3.7.1.5-1	The applicant should further clarify the definition of allowable bearing capacity for a hard rock site.
3.7.2.1-1	The applicant did not perform calculations of total earth pressures for the various load cases to ensure that the load case will lead to the maximum wall moments and shears.
3.7.2.3-1	Depending on the outcome of the comparisons from the uplift effect and the stiffness reduction analyses, the design calculations may have to be revised.
3.7.2.3-2	The applicant needs to justify why the vertical acceleration at the containment vessel dome is reduced from 1.40g to 1.13g as a result of using different polar crane model.
3.7.2.3-3	The applicant agreed to review the references provided by the staff on the stiffness reduction in shear walls, and provide justification or correction as needed.
3.7.2.9-1	The effects of parameter uncertainty had not been explicitly considered. This will be addressed in conjunction with Open Item 3.7.2.3-3.
3.7.2.16-1	The applicant should commit in the DCD that the COL applicants should perform an analysis and an evaluation using the design basis earthquake ground motion and plant-specific site conditions to confirm the design adequacy of the AP1000 design.
3.8.2.1-1	The staff expected that the final detailed analyses for the AP1000 steel containment would be submitted for staff review as part of the design certification process. To complete the staff evaluation of the AP1000 steel containment design, the staff will need to audit the final detailed analyses.

3.8.2.2-2	Code Case N-284-1 has not been designated as Tier 2* material.
3.8.3.5-1	The staff has a concern regarding the lack of a documented method for considering out-of-plane wall and floor flexibility.
3.8.3.5-2	The design summary report for containment internal structures documenting that the structures meet the acceptance criteria has not been completed. Therefore, the staff has not perform its review of the report.
3.8.3.5-3	The staff reviewed the approach used to calculate the needed steel area of the in-containment refueling water storage tank (IRWST) structural walls and concluded that the applicant's approach for the analysis and design does not meet the criteria of Chapter 21.6, "Structural Walls, Diaphragms and Trusses," of ACI-349-01.
3.8.4.2-1	The need for boundary elements around openings and at intersections of reinforced concrete walls should be evaluated in accordance with Chapter 21.6 of ACI-349-01.
3.8.4.3-1	For subcompartment locations inside containment (except for the IRWST) and outside containment, the applicant needs to define any rapid thermal transients that can occur, and demonstrate that no unacceptable degradation would result from differential thermal expansion of the steel and concrete throughout the entire transient.
3.8.4.5-1	The design summary report for other Category I structures has not been completed for staff review.
3.8.4.5-2	During the review of the Wall 7.3 design calculation, the staff could not conclude that the corrected equation accurately calculates the necessary positive reinforcement.
3.8.5.1-1	The non-structural concrete mud mat cannot withstand the very high toe pressure predicted in the applicant's seismic analysis.
3.8.5.4-1	The staff determined that the potential uplift and slapping back of the containment internal structures foundation on the basemat through the steel containment vessel during a seismic event could affect both the seismic design loads and in-structure response spectra for all structures, systems and components associated with the containment internal structure, and could also affect the seismic response of the steel containment shell.

3.8.5.4-2	The specified shear wave velocity for a hard rock site condition should be 2438m/sec (8000 fps), instead of 1067m/sec (3500 fps).
3.8.5.4-3	the design summary report for the basemat foundation has not been completed for staff review.
3.8.5.5-1	The north-south (NS) overturning moment (moment about the east-west (EW) or long axis of the basemat) increases by 42.8 percent while the EW overturning moment (moment about the NS or short axis of the basemat) increases by 10.8 percent. The EW base shear increases by only 1.8 percent while the NS base shear increases by 8.1 percent. The reported increases are not consistent with the 9.9 percent increase in the NI mass or the 17 percent increase in simple equivalent static overturning moment. In addition, the applicant indicates in its response to RAI 220.018 that the safety factor against overturning for the NS earthquake increases for AP1000, compared to AP600, even though the overturning moment increases by 42.8 percent.
4.4-1	Upon installation of the actual instrumentation, the combined license (COL) applicant should evaluate the instrumentation uncertainties of the operating parameters, and confirm the design limit departure from nucleate boiling ratio values.
4.5.1-1	The information on preservice examinations provided in Westinghouse's response to RAI 252.001 should be addressed by a COL applicant, and should be reflected in DCD Tier 2 Section 5.3.6.
4.5.1-2	The COL applicant should perform analyses and inservice inspections and provide reports and notifications equivalent to those contained in Sections IV.A to IV.F of NRC Order EA-03-009, "Interim Inspection Requirements for Reactor Pressure Vessel Heads at PWRs."
5.2.3-1	The applicant needs to clarify in the DCD that the method used to calculate the δ-ferrite is based on Hull's equivalent factors or a method producing an equivalent level of accuracy; i.e., ± 6% deviation between the measured and calculated values.
5.3.3-1	The DCD does not provide limitations (values of RT_{NDT}) for the closure flange region of the reactor vessel and head to satisfy the requirements of 10 CFR Part 50, Appendix G.
5.4.2-1	The staff is requesting the rationale for assuming the alternating stress and associated fatigue usage induced by fluid-elastic coupling is negligible for the case where the fluid-elastic stability ratio is 0.75.

6.1.1-1	DCD Tier 2 Section 6.1.1 contains an incorrect reference to a hydrogen production analysis for a post accident analysis.
6.2.1.8.1-1	Westinghouse should justify the jet impingement model used to determine the boundaries of the zones from which fibrous material would be excluded.
6.2.1.8.2-1	Westinghouse should justify the analysis used to determine the capability of the AP1000 in containment refueling water storage tank screens to accommodate anticipated debris loadings.
6.2.1.8.3-1	Westinghouse did not address the potential for other sources of debris, such as fibrous debris and floatable debris, to enter the reactor coolant system through a pipe break and block requisite core cooling flowpaths.
6.2.1.8.3-2	Westinghouse should support the assumption that paint particles smaller than 200 mils are not a blockage concern for the containment recirculation screens.
6.2.1.8.3-3	Westinghouse should justify the analysis associated with the capability of the AP1000 containment recirculation screens to accommodate anticipated debris loadings.
6.2.5-1	The AP1000 design control document for the control of combustible gas in containment during accidents does not comply with current regulations.
6.2.6.4-1	Westinghouse should place the numerical value of P_a (the peak calculated containment internal pressure for the design basis loss of coolant accident) into the AP1000 technical specifications.
6.4-1	Information is needed to complete the staff's review of the dose analysis for main control room personnel during design-basis accidents.
8.2.3.1-1	The COL applicant should perform a failure modes and effects analysis to ensure that the design provides power to the reactor coolant pumps (RCPs) for a minimum of 3 seconds following a turbine trip. If 3 seconds cannot be maintained, then the DCD Tier 2 Chapter 15 analysis should be re-analyzed and provided to the staff for review.
9.4.-1	Insufficient information was available at the time of review concerning the codes and standards that are in effect as of the submittal date of the DCD Tier 2 for the references identified in DCD Tier 2 Sections 6.4.8, 9.4.13, and Appendix 1A.

9.5.1-1	Westinghouse needs to demonstrate that the performance of the composite steel/concrete barrier provides an equivalent level of safety to that provided by using concrete and masonry.
9.5.1-2	Westinghouse needs to demonstrate that the use of the fire-induced vulnerability evaluation methodology is an appropriate choice to model a fire within containment.
9.5.2-1	Westinghouse should revise the DCD to clarify the categorization of communication equipment and the requirement addressing this equipment in 10CFR 73.55 (f).
9.5.2-2	Westinghouse should revise DCD Tier 2 Section 9.5.2 to specify the testing requirements in 10CFR 73.55 (g) for certain communication systems.
9.5.2-3	Westinghouse should revise the DCD to specify that the COL applicant should address the issue of Bulletin BL-80-15 for recommendations of loss of the emergency notification system due to a loss of offsite power.
9.5.2-4	Westinghouse should revise the DCD to specify that (1) communications system will be protected from EMI/RFI effects, (2) there will be adequate testing and field measurements to demonstrate effective communications, and (3) effective communications will be sustained for maximum potential noise levels.
10.2.8-1	DCD Tier 2 Section 10.2.3.1 toughness and margin criteria not consistent with what is stated in DCD Tier 2 Section 10.2.3.4.
10.2.8-2	Westinghouse has not provided adequate justification to conclude that rotor resonant stresses resulting from passing through rotor critical speeds are insignificant.
10.2.8-3	DCD Tier 2 Section 10.2.3 is not consistent with the stress limit criterion of SRP 10.2.3, which stipulates that the combined stresses of a low-pressure turbine disk at design overspeed due to centrifugal forces, interference fit, and thermal gradients not exceed 0.75 of the minimum specified yield strength of the material.
13.3-1	The staff has identified the inability of the technical support center (TSC) to provide emergency habitability under accident conditions and habitability issues associated with the isolation of the main control room as an open item.

13.3-2	The staff has identified utilization of the emergency operations facility as an alternate TSC as an open item.
13.6-1	The staff has not completed the review the applicant's change to the security plan.
14.2-1	While the staff has completed its review of whether the Initial Test Program conforms to specified Regulatory Guides and certain other matters, as discussed below, the staff has not completed its review of certain aspects of the testing scope, general test methods, and acceptance criteria.
14.2.7-1	The NRC staff lacked sufficient information to determine if this exception to RG 1.41 was acceptable.
14.2.7-2	The NRC staff requested the applicant to clarify and justify the inconsistent natural circulation testing provisions in the exception to RG 1.68 and in Test Abstracts 14.2.10.3.6 and 14.2.10.3.7. Specifically, the staff asked the applicant to clarify under what circumstances natural circulating testing would be performed.
14.7.2-3	The NRC staff believes that the applicant should delete this exception to RG 1.68 in WCAP-15799 and state that the test abstract in DCD Tier 2 Section 14.2.10.4.28 meets the guidance in RG 1.68, Regulatory Position C.1, Appendix A.5, Test 5.d.d, or provide additional information to clarify this exception.
14.2.10-1	The NRC staff has determined that the applicant should clarify whether this test should be performed for every AP1000 plant or justify that this test is a first plant-only test as described in DCD Tier 2 Section 14.2.5.
14.2.10-2	RG 1.68, Appendix A, Item 5.e. recommends performance of pseudo-rod ejection testing during the power ascension test phase to validate the rod ejection accident analysis. The applicant states that this test is performed on the first plant only.
14.2.10-3	The NRC staff agrees that incore and excore neutron flux instrumentation testing should not be performed at a power level that could cause the plant to exceed thermal limits. However, the applicant should either perform the test at a higher power level than proposed, consistent with RG 1.68, or provide additional information to justify performing this test at a maximum of 50 percent power.
14.2.10-4	The NRC staff lacks sufficient information to conclude that the plant trip from 100 percent bounds the MSIV closure transient.

14.3.2-1	ITAAC: The staff finds that item No. 2 under the Design Description for the containment system states that the components identified in Table 2.2.1-1 and the piping identified in Table 2.2.1-2 are designed and constructed in accordance with ASME Code Section III requirements. However, during the April 2-5, 2003, design audit, the staff found that the applicant did not complete the final analyses and design of the containment vessel, including attached components and piping systems.
14.3.2-2	ITAAC: The phrase "structural integrity and" should be added in two places: (1) Design Description No. 5 for the containment system, and (2) Subitem No. 5.ii under the Acceptance Criteria of ITAAC Table 2.2.1-3. The sentence should read ". . . the seismic Category I equipment can withstand seismic design basis dynamic loads without loss of structural integrity and safety function."
14.3.2-3	ITAAC: The thickness of the steel containment vessel should be designated as Tier 1 information and specified in Section 2.2.1 or listed in Table 3.3-1.
14.3.2-4	ITAAC: The staff found that incomplete design commitments related to controls and displays exist in the current system-based ITAAC.
14.3.2-5	ITAAC: The design description for the equipment hatch hoist and the maintenance hatch hoist are not identified as single failure proof as they are in Tier 2, and Table 2.3.5.2 does not require a test, inspection, or analysis to demonstrate whether these items of equipment will meet their design criteria.
14.3.2-6	ITAAC: Section 2.3.9, "Containment Hydrogen Control System," must remain open because hydrogen control is an open item in this report. Briefly, this is because the AP1000 Tier 2 information is written in anticipation of a rule change to 10 CFR 50.44 that would relax requirements, but has not been finalized.
14.3.2-7	ITAAC: The applicant needs to provide appropriate ITAAC for all the communication systems.
14.3.2-8	ITAAC: The staff cannot complete its review of these ITAAC because the staff's review of the security program for AP1000 is not complete.
14.3.2-9	ITAAC: Please provide proposed ITAAC related to the geometry, fabrication, materials, accessibility for inspection, and operating conditions for control rod drive system penetrations, as motivated by recent operating experience.

14.3.2-10	ITAAC: Please provide proposed ITAAC to verify that all Alloy 600/690 components and welds in the reactor coolant pressure boundary are identified and are readily accessible for bare metal visual inspection
14.3.2-11	ITAAC: To verify that the design will be in accordance with the regulatory requirements associated with PTS, the applicant needs to provide an appropriate ITAAC.
14.3.2-12	ITAAC: Section 3.1, "Emergency Response Facilities," the staff finds this ITAAC unacceptable because it does not address the radiological habitability or the ventilation system for the technical support center; both of which should be the same as, or comparable to the main control room ITAAC.
14.3.2-13	ITAAC: The phenomenon of the foundation mat uplifting and shear wall cracking will directly affect the design adequacy of the nuclear island structures, systems and components, including the thickness of structural elements listed in Table 3.3-1 and safety-related piping systems.
14.3.2-14	ITAAC: The applicant should justify the use of the proposed tolerances for vessel wall inside diameter.
14.3.2-15	ITAAC: The staff found that the list of risk significant components in Table 3.7-1 was not updated to include all risk- significant structures, systems, and components (SSCs) from the list of risk significant SSCs identified in Tier 2 Section 17.4, Table 17.4-1, "Risk Significant SSCs within the Scope of D-RAP."
14.3.3-1	ITAAC: Section 2.5.1, "Diverse Actuation System," Table 2.5.1-1, "Functions Automatically Actuated by the DAS" should be modified to include "actuate core makeup tanks, and trip the reactor coolant pumps on low wide-range steam generator water level."
14.3.3-2	ITAAC: Section 2.5.1 design description item 2(c) should be modified to include "the DAS manual control bypasses the protection and safety monitoring system cabinets."
14.3.3-3	ITAAC: Section 2.5.1 design description item 3(e) should be modified to include "The DAS uses sensors that are separate from those being used by the PMS and the plant control system."
14.3.3-4	ITAAC: Section 2.5.2, "Protection and Safety Monitoring System," Table 2.5.2-1 and Figure 2.5.2-1 should be modified to include "two divisions of safety-related postaccident parameter displays" to be consistent with the Tier 1 Section 2.5.2 design description.

14.3.3-5 ITAAC: Tier 1 design description Item 6(c) should be modified to clarify that the functions listed on Table 2.5.2-4 are based on minimum inventory requirements.

14.3.3-6 ITAAC: Section 2.5.2 design description Item 8(b) should be modified to clarify that the control transfer function is implemented by multiple transfer switches.

14.3.3-7 ITAAC: Section 2.5.2, Table 2.5.2-7, “PMS Interlocks,” should be modified to include “Interlocks for the Accumulator Isolation Valves and IRWST Discharge Valve” to be consistent with Tier 2 information provided in Section 7.6.2.3.

14.3.3-8 ITAAC: Tier 1 Section 2.5.2, Table 2.5.2-6, “PMS Blocks,” should be modified to include (1) block automatic rod withdrawal (P-17) and (2) block automatic safeguards (P-4).

14.3.3-9 ITAAC: Section 2.5.2, Table 2.5.2-8, ITAAC 7(c) columns do not have sufficient criteria to verify that the design commitment is met.

14.3.3-10 ITAAC: Section 2.5.2, Table 2.5.2-8, ITAAC 7(d) columns may not be sufficient to verify the design commitment, especially the terminology “non-class 1E controls” in the performance of the operational tests.

14.3.3-11 ITAAC: Section 3.2, Table 3.2-1, “Acceptance Criteria for Design Commitment 3," should include the following as a last criterion: "Man-in-the loop engineering test reports" as one of the documents to indicate that the design of the OCS was conducted in conformance with the implementation plan.

14.3.3-12 ITAAC: Table 3.2-1, “Acceptance Criteria for Design Commitment 4,” should be changed to indicate that the verification and validation implementation plan includes specified activities.

14.3.3-13 ITAAC: Table 3.2-1, "Design Commitment" statement No. 5, should be changed to indicate that the verification and validation implementation plan includes specified activities.

14.3.3-14 ITAAC: Table 3.2-1, "Acceptance Criteria," for "Design Commitment" statement No. 5, should be changed to include a new, "a)" to indicate that, "a) Operational Conditions Sampling was conducted in accordance with the implementation plan." The remaining criteria should be re-lettered.

14.3.3-15 ITAAC: Table 3.2-1, "Inspections, Tests, Analyses," item "d)", should be changed to replace "design issues resolution" with "human engineering discrepancy resolution."

14.3.3-16 ITAAC: Table 3.2-1, "Acceptance Criteria," item "d)," should be changed to, "human engineering discrepancy resolution verification was conducted in accordance with the implementation plan and includes verification that human factors issues that were documented in the design issues tracking system and human engineering discrepancies that were identified in the design process have been addressed in the final design."

14.3.3-17 ITAAC: There is nothing in Tier 1, Subsection 2.6.3, that evaluates the adequacy/effectiveness/suitability of illumination levels for the facility or the workstations in the facilities. As part of evaluating a suitable work space environment for the MCR and RSR, there should be an assessment of auditory levels (noise) as well.

14.3.3-18 ITAAC: Table 3.2-1, item 10.i: Subsection 2.7.1 does not have an ITAAC related to RSR - there is nothing in the ITAAC that requires inspection, test, and analyses for the RSR and ventilation.

14.3.3-19 ITAAC: Without performing an evaluation of the LBB bounding curves using preliminary analysis results at the design certification stage, the question of whether there is sufficient margin in the piping to demonstrate that the probability of pipe rupture is extremely low would remain unresolved.

14.3.4-1 ITAAC: Control room χ/Q values are not provided in Table 5.0-1, "Site Parameters." In addition, the staff has not completed its evaluation, but has identified unresolved issues related to adequate justification for assuming a diffuse release, estimation of initial sigma values, other release assumptions, building cross-sectional areas, and distances between release/receptor pairs.

15.1.5-1 The COL applicant should evaluate the validity of the safety analysis documented in the AP1000 DCD using plant-specific setpoints and instrument uncertainties, including the SG mid-deck plate level measurement uncertainties. The COL applicants should submit in the plant specific applications the evaluation results. The applicant should include this COL Action Item in the DCD.

15.2.7-1 The information provided regarding core void distribution was not sufficiently detailed to draw the conclusion that adiabatic heating would be avoided. The applicant should provide more detailed information of the axial void distribution during LTC and show that the possibility of adiabatic fuel heating is excluded.

15.3-1 The staff has not completed its evaluation of the use of the AP600 aerosol removal coefficients. The staff is performing independent analyses to confirm information provided by the applicant.

15.3-2 The staff has not completed its evaluation of the hypothetical reference control room atmospheric dispersion factors, as discussed in Section 2.3.4 of this report. When this evaluation is completed, the staff will complete its review of the design basis accident control room habitability radiological consequences analyses.

15.3.6-1 The staff has not completed its evaluation of the LOCA radiological consequences analysis because of Open Items 15.3-1 and 15.3-2 discussed above. When these Open Items are resolved, the staff will complete its review of the design basis LOCA offsite and control room radiological consequences analyses.

15.3.7-1 The applicant has not provided sufficient analysis of the radiological consequences of the FHA taking into account the absence of a TS LCO for decay time before movement of fuel. The applicant should provide an analysis of the radiological consequences of an FHA that occurs 24 hours post-shutdown, including both the offsite and control room doses.

16.2-1 Westinghouse should explain why an equivalent surveillance for determining $\bar{E}$ (the average disintegration energy) using dose equivalent Iodine-131 is not proposed in the AP1000 technical specifications.

16.2-2 Westinghouse should clarify the technical specification Bases for the passive core cooling system limiting condition for operation.

16.2-3 Westinghouse should propose a decay time (the time interval between the time the reactor was last critical and the initial movement of an irradiated fuel assembly from the reactor core) in the AP1000 technical specifications.

17.3.2-1 The NRC staff plans to perform a QA test control implementation inspection to determine if additional testing activities performed at the test facility associated with the AP1000 design are accomplished, in accordance with the Westinghouse 10 CFR Part 50, Appendix B, QA program as described in Chapter 17 of the AP1000.

17.3.2-2 The staff plans to conduct an inspection of the implementation of the project-specific quality plan to verify that design activities conducted for

	the AP1000 project complied with the Westinghouse QMS and the requirements of 10 CFR Part 50, Appendix B.
17.3.2-3	The NRC staff determined that this exception to RG 1.28, Regulatory Position C.2, may not be acceptable since programmatic nonpermanent records could be discarded 3 years after issuance of a final design approval; therefore, these records may not be available to a future COL applicant.
17.3.2-4	Missing QA Related Information in DCD Tier 2 Section 17.6, References
17.5-1	In an effort to ensure that the COL action items in DCD 17.5, associated with D-RAP and O-RAP, are accomplished in a manner consistent with the guidance contained in SECY 95-132, the applicant should provide a COL action item to reflect conformance with the SECY 95-132 guidance.
18.3.3.1-1	Criterion 5, “Risk-important human actions”, states that the OER should identify risk-important human actions that have been identified as different or where errors have occurred. The human actions should be identified as requiring special attention during the design process to lessen their probability. Westinghouse should address this item in discussing developing the OER.
18.11.3.4-1	Criterion 3, “HED Documentation”, states that HEDs should be documented in terms of the HSI component involved and how the characteristics depart from a particular guideline.Westinghouse to provide details related to the process that will be used to identify, analyze, prioritize, evaluate, document, and determine and evaluate design solutions, etc. for HEDs using, the HED resolution review criteria in NUREG-0711 as a template.
18.11.3.5-1	Criterion 3, “Plant Personnel”, states that participants in validation tests should represent an unbiased sample; be representative of actual plant personnel; reflect characteristics of the population of plant personnel; include shift supervisors, reactor operators, shift technical advisors, etc., and minimum and normal crew configurations. Westinghouse should explain how validation tests address NUREG-0711 item.
18.11.3.5-2	Westinghouse should indicate the applicability of the general test plan to validation tests or provide further detail on this Criterion 6, “Test Design” in DCD Tier 2 Section 18.8.11 or in WCAP-15860.
18.11.3.5-3	Westinghouse should indicate the applicability of the general test plan (e.g., Section 2.4.2, “Measures and Analysis,”) to validation tests or

provide further detail on Criterion 7, “Data Analysis and Interpretation,” in DCD Tier 2 Section 18.8.11 or in WCAP-15860.

18.11.3.5-4 Westinghouse should indicate the applicability of the general test plan (e.g., DCD Tier 2 Section 2.4.2, “Measures and Analysis,” to validation tests or provide further detail on Criterion 8, “Validation Conclusions,” in DCD Tier 2, Section 18.8.11 or in WCAP-15860.

18.11.3.6-1 For Section 18.11.3.6, “Human Engineering Discrepancy Resolution”, Westinghouse described their general approach to human engineering discrepancy (HED) resolution in WCAP-15860. The applicant needs to provide further details related to the process that will be used to identify, analyze, prioritize, evaluate, document, and determine and evaluate design solutions, etc. for HEDs using, the HED resolution review criteria in NUREG-0711 as a template.

19.1.3.2-1 Westinghouse should include the probability of a pre-existing opening in containment large enough to constitute an isolation failure in the containment isolation fault trees for the AP1000 PRA.

19.1.3.2-2 Westinghouse should include the failure mode of plugging of the drains near the floor of the annulus around the containment shell in the AP1000 PRA.

19.1.10.1-1 Westinghouse should provide a comparison of important features between the common qualified platform and the safety-related protection and safety monitoring system modeled in the PRA.

19.1.10.1-2 The final list of “design certification requirements” related to the PRA should be in agreement with the resolution of the PRA open items identified in the AP1000 DSER.

19.1.10.1-3 Westinghouse should provide additional information related to the steps in the process for using PRA results to identify non-safety-related systems for regulatory oversight as well as the type and level of such oversight.

19.1.10.1-4 Westinghouse should address the impact of the uncertainties on the AP1000 PRA results and insights associated with (1) the assumed frequencies of large loss-of-coolant-accidents and steam generator tube rupture accidents, and (2) the success criteria assumed for passive containment cooling by air flow.

19.1.10.1-5 Westinghouse should address issues associated with its approach in categorizing success paths for the PRA.

19.1.10.1-6	Westinghouse should identify that the COL applicant should develop procedures for implementing fire-specific operator actions.
19.1.10.2-1	Westinghouse should address issues associated with the analysis of vacuum refill of the RCS from drained conditions (mid-loop).
19.1.10.2-2	Westinghouse should provide the dominant shutdown accident sequences in the AP1000 shutdown PRA.
19.1.10.2-3	Westinghouse should address issues associated with the shutdown risk importance analysis.
19.1.10.2-4	Issues associated with the shutdown PRA sensitivity studies need to be resolved.
19.1.10.2-5	Westinghouse should address issues associated with the documentation of the shutdown focused PRA results.
19.1.10.2-6	Issues associated with the shutdown fire risk evaluation need to be resolved.
19.1.10.3-1	Westinghouse should address documentation issues associated with the representative sequences for assigning source terms for the PRA.
19.1.10.3-2	Westinghouse should provide the major causes of reactor cavity flooding failure and hydrogen igniter failure for the AP1000.
19.2.3.3-1	Westinghouse should address issues associated with the documentation of the ULPU Configuration V testing.
19.2.6-1	For the ultimate capacity of the AP1000 containment analysis, Westinghouse should address why 558 KPa (81 psig) at 149°C (300 °F) is not the limiting severe-accident pressure.
19.2.6-2	Westinghouse should address the question associated with use of a 1.5 multiplier in the analysis for the calculated critical buckling pressure for the equipment hatch covers.
19.2.6-3	Westinghouse should clarify the approach used for the analysis of the contribution to the conditional containment failure probability from the failure of the equipment hatches.
19.3.3-1	Westinghouse should identify that the COL applicant should develop plant specific guidelines that would reduce the potential for loss of reactor coolant system boundary and inventory when using freeze seals.

19.3.7-1	Westinghouse should identify that the COL applicant should develop an outage planning and control program.
19.3.10-1	Westinghouse should address the apparent errors in the calculated core damage frequency associated with the evaluation of plant risk for internal floods at shutdown.
19.4-1	Issues associated with the severe accident mitigation design alternatives analysis need to be resolved.
19A.2-1	Westinghouse should explain how the deterministic strength factor was used in the probabilistic fragility analysis.
19A.2-2	Westinghouse should explain how the variable strength factors are accounted for in the probabilistic fragility analysis.
19A.2-3	Westinghouse should explain how actual material properties that are derived from the yield strength or crushing strength are used in its probabilistic fragility analysis.
19A.2-4	Westinghouse should justify the use of a high variability factor for the natural frequency calculations when using detailed finite element models.
19A.2-5	Because the AP1000 is to be located on hard rock sites Westinghouse should not discuss soil-structure interaction related variability in Chapter 55 of the PRA report for AP1000.
19A.2-6	Westinghouse should validate the high confidence in low probability of a failure (HCLPF) values calculated by the conservative deterministic failure margin method approach.
19A.2-7	Although Westinghouse appears to have used a conservative approach to obtain the equipment HCLPF value from test results, Westinghouse should clarify how the use of known natural frequency values for equipment within the standard design scope will be implemented.
19A.2-8	Westinghouse should address the necessity for a review of the core makeup tank HCLPF value if there is any increase in seismic response of the containment internal structure due to lift off of the internal structure or the nuclear island structure.
19A.2-9	Because of the potential for amplification, Westinghouse should justify the HCLPF values in the range of 0.53 g and 0.73 g that were obtained

from the generic data developed by a joint industry group in the Utility Requirements Document.

19A.3-1	Westinghouse should address the validity of using plant seismic event trees derived from the AP600 model for the seismic margin model.
19A.3-2	Westinghouse should justify why a specific item on plant walkdown verification of seismic interaction between the nuclear island and adjacent structures is not included in the COL interface requirement.
19A.3-3	Westinghouse should explain how service related degradation of steam generator tubes was considered in the development of the HCLPF value for the EQ-SLOCA group.
20.7-1	Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." Westinghouse needs to provide information to describe the extent to which the insulation of all Alloy 600/690 components and welds in the reactor coolant pressure boundary (not just upper reactor vessel head penetrations) will be designed to readily facilitate bare metal visual inspection during refueling outage conditions.
20.7-2	GL-85-06, "Quality Assurance Guidance for ATWS Equipment That is Not Safety Related." Westinghouse should clearly state the quality assurance requirements that are applicable to the DAS and non-class 1E and UPS systems for the purposes of satisfying the requirements of GL 85-06.
21.1-1	Chapter 21 currently contains references to NUREG-1512, which provides the basis for accepting the AP600 testing and computer codes. Prior to issuing the final safety evaluation report for the AP1000, the staff will remove these references and replace the references with the basis for its conclusion that the testing and computer codes are acceptable for the AP1000.
21.5-1	The applicant's submittals and responses to RAIs concerning hot leg phase separation were not sufficient to demonstrate that the codes used in AP1000 safety analysis model the hot leg phase separation process correctly. This issue is considered open until the applicant confirms the sensitivity studies performed by the staff using the code(s) the applicant intends to use to model SBLOCAs in AP1000.
21.5-2	Given the lack of well scaled experimental data on upper plenum entrainment phenomena and the importance of predicting this process in an advanced plant SBLOCA transient, the staff has requested new

	experimental data to support the use of the upper plenum entrainment models in the AP1000.
21.5-3	Additional justification that the AP1000 core will remain covered as predicted by the codes should be provided since high void fractions were predicted.

1.7 Summary of Confirmatory Items

The NRC staff's review of Westinghouse's application for certification of the AP1000 design, including the additional information provided to the NRC through April 21, 2003, identified the following confirmatory items at the time this report was prepared. An item is identified as confirmatory if the staff and Westinghouse have agreed on a resolution of a particular item, but the resolution has not yet been formally documented in the DCD. Each confirmatory item was assigned a unique identifying number. The number identifies the section in this report where the confirmatory item is described. For example, Confirmatory Item 7.2.3-1 is discussed in Section 7.2.3 of this report.

Item	Description
3.7.2.1-1	The applicant agreed to provide a description of analysis procedures presented in the April 2 through 5, 2003, meeting, including how the resulting seismic responses are to be utilized for the equivalent static analyses and design of the subsystems and components, in the DCD.
3.7.2.1-2	The applicant agreed to include the basis for placing the fixed base of the FE model at the middle of the basemat in the future DCD revision.
3.7.2.1-3	The applicant agreed to provide a detailed description of seismic analysis procedures and procedures for applying seismic responses to the design of the NI structures, systems and components in a future revision to the DCD.
3.7.2.2-1	As a result of the meeting discussion on April 3, 2003, the applicant agreed to include all modal properties up to 200 modes in DCD Tier 2 Tables 3.7.2-1 through 3.7.2-4.
3.7.3.2-1	The applicant agreed to revise DCD Tier 2 Section 3.7.3.5.1 to address the comment that subsystems to be analyzed and designed based on this guideline cannot be properly categorized as single mode dominant.
3.8.2.1-1	The applicant will incorporate a design change in the cylinder embedment transition region in a future revision and designate the "inhibit corrosion" function as "Safety" for coatings on the outside surface of the containment vessel in a future revision of DCD Tier 2 Table 6.1-2.

3.8.2.2-1	The applicant will revise its response to RAI 220.003 to include the applicable test data compiled by the Chicago Bridge & Iron Company.
3.8.2.4-1	The applicant acknowledged that “SA537, Class 2" in the second paragraph of DCD Tier 2 Section 3.8.2.4.2.6 should be “SA738, Grade B”, and agreed to correct it in the next DCD revision.
3.8.2.4-2	The applicant agreed to correct the conflicting pressure capacities for the 4.87m (16ft) diameter equipment hatches in the next DCD revision.
3.8.3.2-1	The applicant committed to designate ACI 349-01 and AISC-N690-1984 as Tier 2*
3.8.3.5-1	During the April 2 through 5, 2003, audit, the applicant agreed to revise DCD Tier 2 Figure 3.8.3-8 (sheet 1 of 3) and Figure 3.8.3-15 (sheets 1 and 2).
3.8.3.5-2	The applicant agreed to update final design results for the critical sections of the structural wall modules, IRWST steel wall, and the columns supporting the operating floor
3.8.4.1-1	The applicant will designate the shield building roof and tank dimensions and Elevations in DCD Tier 2 Figure 3.8.4-2 as Tier 2*.
3.8.4.5-1	The design summaries of critical sections documented in Appendix 3H are to be updated in the next DCD revision.
3.8.4.5-2	The applicant agreed to finalize the calculation to document the final design of the shield building roof.
3.8.5.4-1	The applicant plans to update Table 3.8.5-3, to document the final design of the critical sections.
3.8.5.4-2	The applicant agreed to identify pertinent Tier 2* information in DCD Tier 2 Figure 3.8.5-3.
3.11.3-1	The applicant agreed to include Figures 3D.5.6 and 3D.5.7 in a subsequent revision to the DCD.
6.4-1	Westinghouse committed to revise the DCD to include a statement regarding verification that certain chemicals were evaluated to conclude that these chemicals do not represent a toxic hazard to control room operators.

14.2.10-1(2)	The applicant agreed to add an appropriate test abstract to DCD Tier 2 Section 14.2.10.4, “Power Ascension Tests,” to perform the testing recommended by RG 1.68, Item 5.c.c. A Revision of the DCD to address performance of this testing during power ascension testing will be verified.
14.3.2-1	Section 2.7.1, “Nuclear Island Nonradioactive Ventilation System.” Components are listed in Table 2.7.1-1 but not shown in Tier 1 Figure 2.7.1-1 or Tier 2 Figure 9.4.1-1 (Reference RAI 410.022).
14.3.2-2	Section 2.7.3, “Annex/Auxiliary Building Nonradioactive Ventilation System.” Components are listed in Table 2.7.3-1 but not shown in Tier 1 Figure 2.7.3-1 or Tier 2 Figure 9.4.2-1 (Reference RAI 410.022).
14.3.2-3	Section 2.7.5, “Radiologically Controlled Area Ventilation System.” Components are listed in Table 2.7.5-1 but not shown in Tier 2 Figure 9.4.3-1 (Reference RAI 410.022).
14.3.3-1	There are typographical errors throughout the ITAAC: the abbreviation, **"HIS"** should be replaced with **"HSI."**
19.1.10.2-1	A possible math error in the RAI response 720.38 related to flooding scenario number 6 needs to be clarified.
21.5-1	The applicant has committed to update WCAP-15833 to include the final upper plenum entrainment RAI responses.
21.7-1	Regarding the effect of hot leg phase separation on minimum reactor vessel inventory during recovery from a SBLOCA, the applicant has agreed to provide demonstrations that its codes show the same sensitivity as staff calculations and that this conclusion applies to other small break scenarios.

1.8 Index of Exemptions

In accordance with 10 CFR Section 52.48, the staff used the current regulations in 10 CFR Parts 20, 50, 73, and 100 in reviewing Westinghouse’s application for certification of the AP1000 design. During this review the staff recognized that the application of certain regulations to the AP1000 design would not serve the underlying purpose of the rule, or would not be necessary to achieve the underlying purpose of the rule.

In a letter dated December 3, 2002, Westinghouse submitted a list of exemption requests. These exemptions are discussed in the sections of this report listed below.

Section	Exemption
8.2.3.2	Exemption from General Design Criteria 17 requirement For physically independent circuit (second off-site electrical power source)
15.2.9	Exemption from 10 CFR 50.62 requirement for automatic Start up of auxiliary feedwater system
18.8.2.3	Exemption from 10 CFR 50.34(f)(2)(iv) for safety parameter display console

1.9 Index of Tier 2* Information

The NRC staff has determined that changes to or departures from information in the DCD that are proposed by an applicant or licensee who references the certified AP1000 design will require NRC approval before implementation of the change in accordance with the design certification rule. This information will be referred to as Tier 2* in the proposed design certification rule.

The staff has not yet identified all of the Tier 2* information pertaining to the AP1000 design. This effort will be completed to support the final safety evaluation report. This is Open Item 1.9-1.

1.10 COL Action Items

COL applicants and licensees who reference the certified AP1000 standard design will be required to satisfy the requirements and commitments in the DCD, which is the controlling document used in the certification of the AP1000 design. Also, certain introduction and general discussion commitments are identified in the AP1000 DCD as “Combined License Information Items,” and in this report as “COL Action Items.” These COL action items relate to programs, procedures, and issues that are outside of the scope of the certified design review. These COL action items do not establish requirements; rather, they identify an acceptable set of information for inclusion in a plant-specific safety report. An applicant for a COL must address each of these items in its application. It may deviate from or omit these items, provided that the deviation or omission is identified and justified in the plant-specific safety report. Additional items may be identified by either the staff or Westinghouse as the review is completed.

Westinghouse included a summary of COL action items in DCD Tier 2 Table 1.8-2, and provided an explanation of the items in the applicable sections of the DCD. The staff identified a number of COL action items that resulted from its review throughout this report. A cross-reference of the COL action items will be provided in Appendix F of the final safety evaluation report. The staff has not yet completed the cross-reference of the COL action items. This is Open Item 1.10-1.

Figure 1.2-1 AP1000 Reactor Coolant System

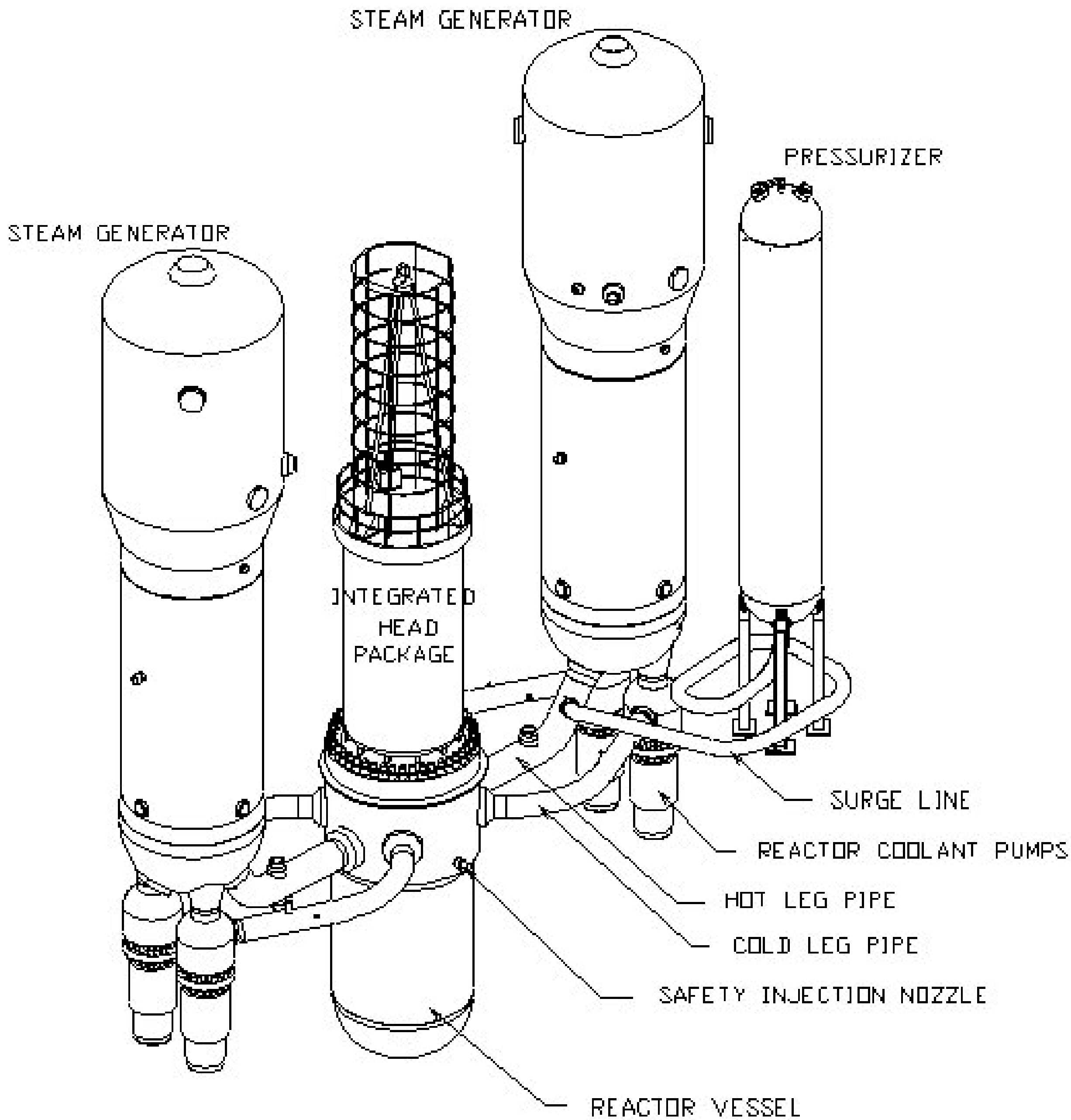

Figure 1.2-2 AP1000 Passive Safety Injection System Post-LOCA, Long Term Cooling

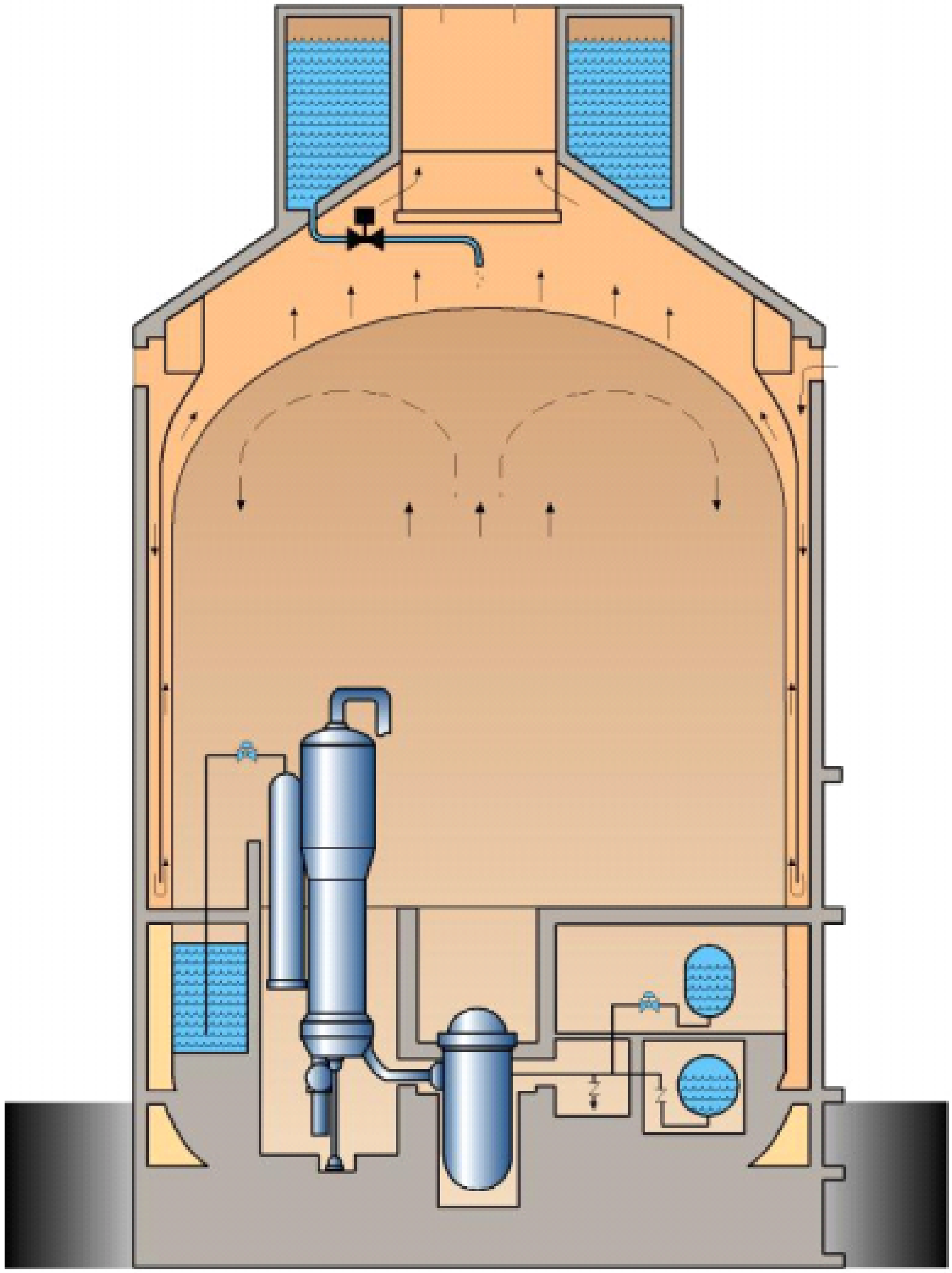

Figure 1.2-3 AP1000 Passive Containment Cooling System

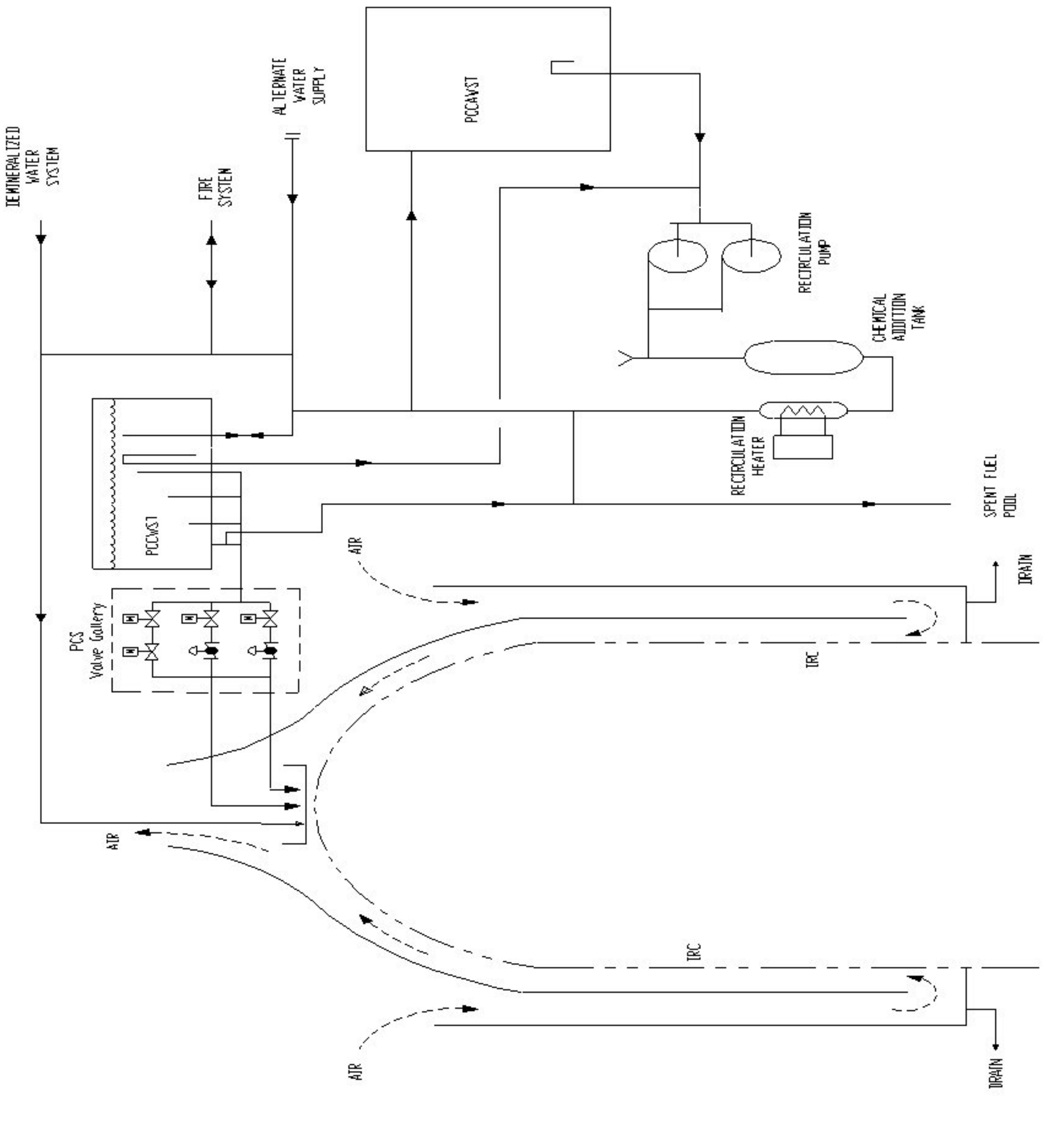

Figure 1.2-4 AP1000 Safety Injection Systems

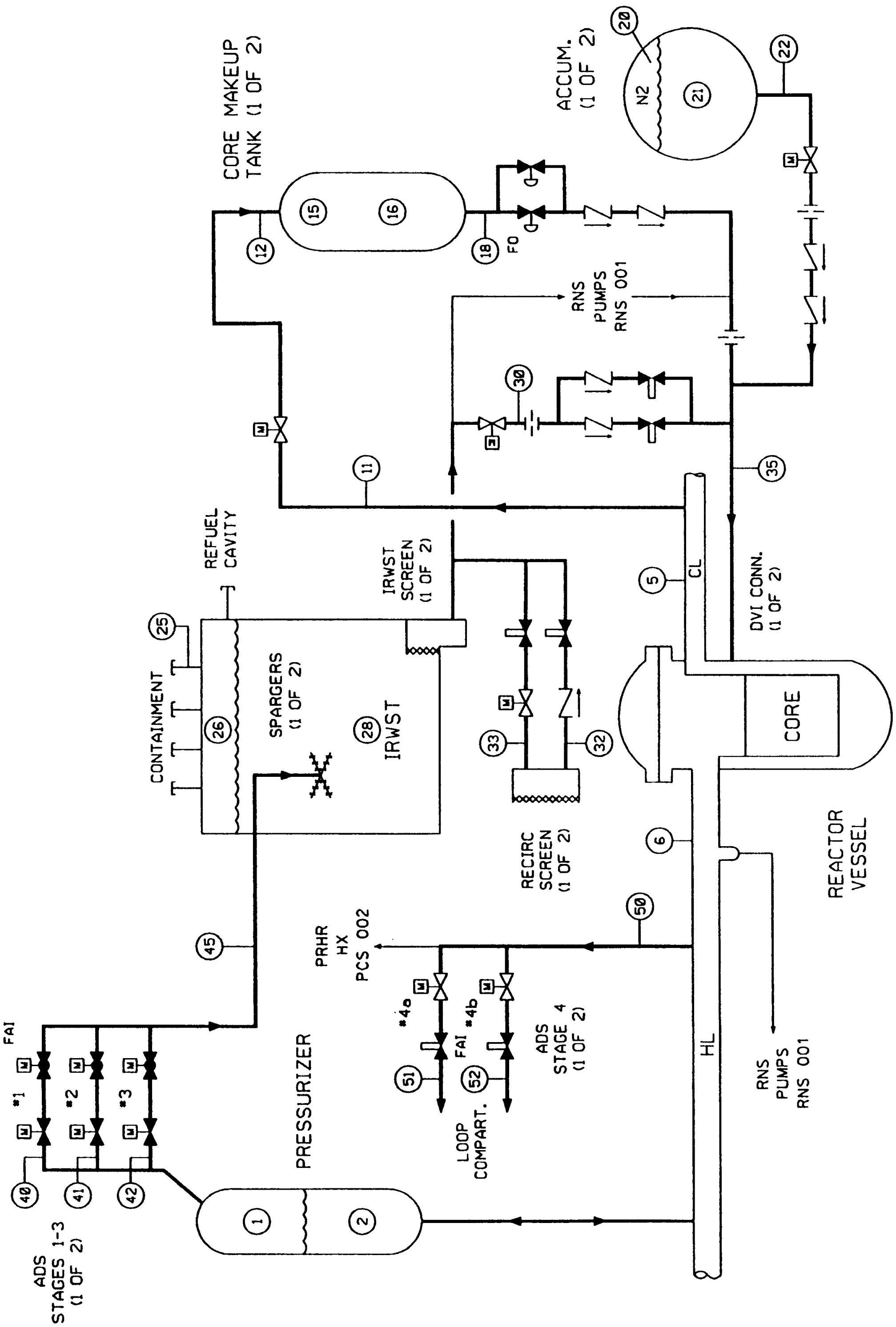

Figure 1.2-5 AP1000 Plant Layout
(Sheet 1 of 2)

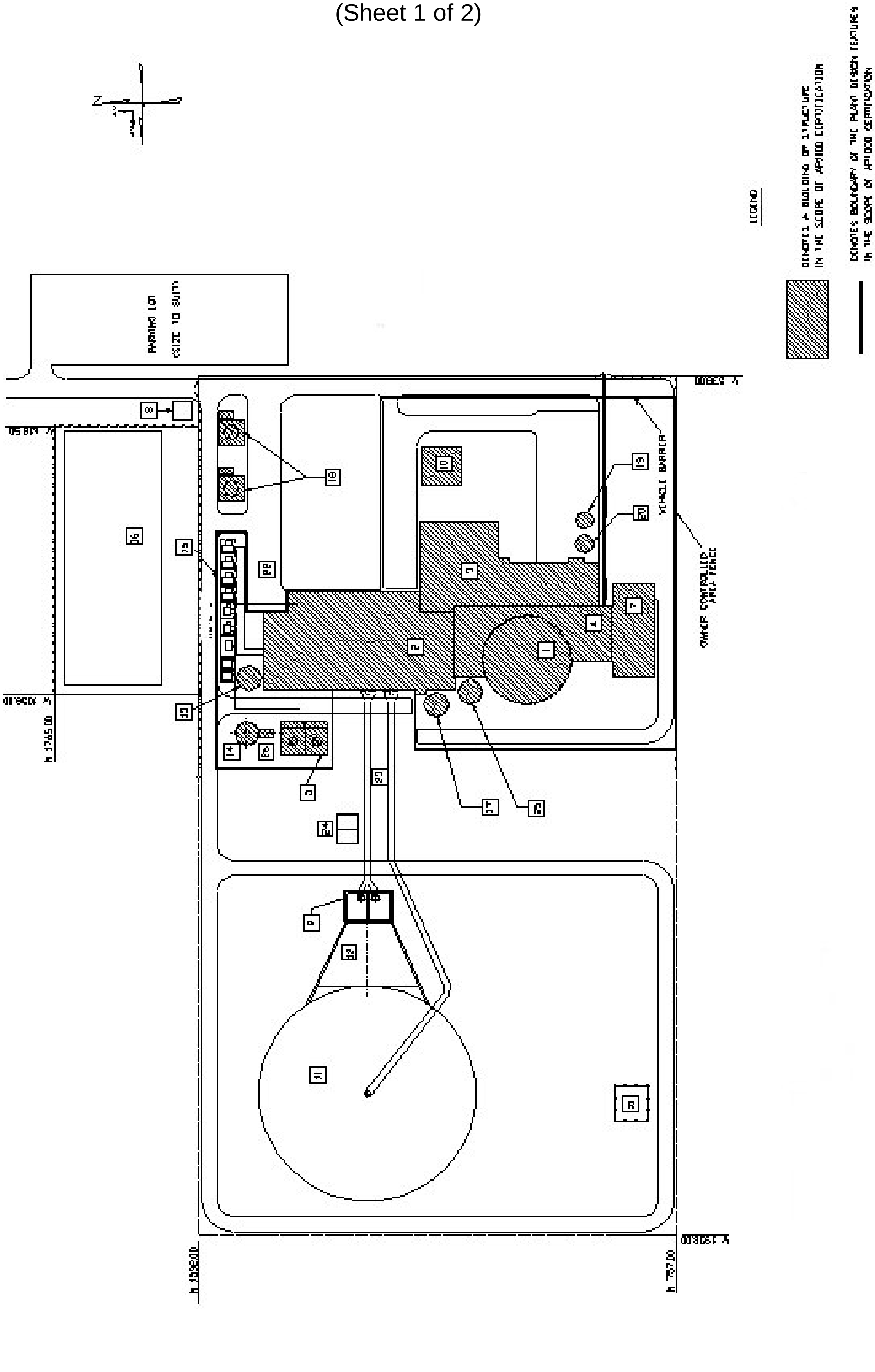

Figure 1.2-5 AP1000 Plant Layout
(Sheet 2 of 2)

1. Containment/Shield Building
2. Turbine Building
3. Annex Building
4. Auxiliary Building
5. SWS Cooling Towers
7. Radwaste Building
8. Plant Entrance
9. Circulating Water Pump Intake Structure
10. Diesel Generator Building
11. CWS Cooling Tower
12. CWS Intake Canal
13. Fire Water/Clearwell Storage Tank
14. Fire Water Storage Tank
15. Transformer Area
16. Switchyard
17. Condensate Storage Tank
18. Diesel Generator Fuel Oil Storage Tank
19. Demineralized Water Storage Tank
20. Boric Acid Storage Tank
21. Hydrogen Storage Tank Area
22. Turbine Building Laydown Area
23. Circulating Water Pipe
24. Waste Water Retention Basin
25. Passive Containment Cooling Ancillary Water Storage Tank
26. Diesel-Driven Fire Pump