AP1000 DOCUMENT COVER SHEET

TDC: ________ Permanent File: ________

AP1000 DOCUMENT NO.	REVISION	PAGE	ASSIGNED TO	OPEN ITEMS (Y/N)
APP-GW-GER-005	1	1 of 74	W-Winters	N

ALTERNATE DOCUMENT NUMBER: WORK BREAKDOWN #: GW

ORIGINATING ORGANIZATION: Westinghouse Electric Company

TITLE: **Safe and Simple : The Genesis and Process of the AP1000 Design**

ATTACHMENTS: List of AP600 and AP1000 DCPs	DCP #/REV. INCORPORATED IN THIS DOCUMENT REVISION: ~~None~~ JF 8/13/08 See record of revisions page
CALCULATION/ANALYSIS REFERENCE:	

ELECTRONIC FILENAME	ELECTRONIC FILE FORMAT	ELECTRONIC FILE DESCRIPTION
APP-GW-GER-005-Rev-1	Word	

LEGAL REVIEW J. C. VALENTINE	SIGNATURE / DATE (If processing electronic approval select option) J. C. Valentine 8/14/2008
PATENT REVIEW M. M. Corletti	SIGNATURE / DATE M.M. Corletti 8/14/08

ORIGINATOR(S) J. A. Speer	SIGNATURE / DATE (If processing electronic approval select option) J. A. Speer 8/13/08	
REVIEWER(S)	SIGNATURE / DATE	
	SIGNATURE / DATE	
	SIGNATURE / DATE	
VERIFIER(S) P. A. Mathewson	SIGNATURE / DATE PA Mathewson 8/13/08	Verification Method: Independent Review PAGE BY PAGE REVIEW

****Plant Applicability**: ☒ All AP1000 plants except: None
☐ Only the following plants:

APPLICABILITY REVIEWER** J. A. Speer	SIGNATURE / DATE J. A. Speer 8/13/08
RESPONSIBLE MANAGER* J. W. Winters	SIGNATURE / DATE J. W. Winters 8/13/08

* Approval of the responsible manager signifies that the document and all required reviews are complete, the appropriate proprietary class has been assigned, electronic file has been provided to the EDMS, and the document is released for use.

****** Electronically approved records are authenticated in the electronic document management system.***

F-AP-3.2-2 Rev. 0

AP1000 RECORD OF CHANGES

AP1000 DOCUMENT NO. APP-GW-GER-005 REVISION 1

ALTERNATE DOC. NO.

DESIGN AGENT ORGANIZATION Westinghouse Electric Company

TITLE SAFE AND SIMPLE: The Genesis and Process of the AP1000 Design

REVISION DESCRIPTIONS		
Rev. No.	**Date**	**Description**
0	**05/02/08**	**Original Issue**
1	**08/13/08**	**The classification has been changed from Westinghouse Proprietary Class 2 to Westinghouse Non-Proprietary Class 3.**

SAFE AND SIMPLE: THE GENESIS AND PROCESS OF THE AP1000 DESIGN

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SAFE AND SIMPLE:
The Genesis and Process of the AP1000 Design

Executive Summary

The current design of the AP1000 is the result of a design philosophy and design process that emphasized safety and simplicity. Since the mid-1980's, Westinghouse and its design partners and collaborators have worked together to establish simple, proven design solutions to a robust set of design criteria. The actual design of AP1000 evolved from the design of the Secure Military Power Plant (SMPP) for the United States Air Force. The design process used throughout the development of SMPP/AP600/AP1000 is to create a safe nuclear power plant with costs, radiation exposures and radioactive discharges as low as reasonably practicable.

Through the 1970s and 1980s nuclear power plant operators in the United States and elsewhere (for example France and South Korea) recognized the benefits of creating a standard design to be deployed on multiple sites. To that end, they worked with the United States Nuclear Regulatory Commission (USNRC) and power plant vendors to create a regulatory scheme based upon licensing and deployment of standard designs (10CFR52). This effort resulted in a large body of NRC regulatory requirements and guidance in addition a body of industry codes and standards. This led to a recognized set of design goals and solutions. In addition, these nuclear utilities created a plant specification for new nuclear plants based upon their collective operating experience and their vision for the future of nuclear power. The specification was called the Utility Requirements Document (URD) and included three parts: Generic Plant Requirements, Requirements for Evolutionary Designs using active safety systems, and Requirements for Plants with Passive Safety Systems using natural circulation for safety related heat removal. Thereafter European utilities created a similar requirements document tailored for Europe and called it the European Utility Requirements (EUR).

In the meantime, the United States had developed a number of natural circulation cooled reactors for military applications. Westinghouse recognized that if one could develop a central station power reactor, based upon proven components, which could perform its safety functions without the need for ac power, a quantum change could be made in the evolution of safe, yet simple nuclear power. This leap resulted in the AP600. The driving design philosophy was to keep it safe, proven and simple. This report outlines the process used for AP600/AP1000 and details many of the discrete design decisions made in its development. By keeping the design safe, proven, and simple, without relying on ac power for safety functions, the design adhered to the principles of as low as reasonably practicable throughout. Probabilistic Risk Assessments (PRA) evaluations played a vital role as a design tool throughout the development of the AP600/AP1000.

When completed, as evidenced by a Design Certification granted by the United States Nuclear Regulatory Commission, the AP600 was the simplest, least expensive nuclear power plant available. It could not, however, compete with natural gas power plants at the time the AP600 received Design Certification. To compete, the AP600 was required to lower its cost per megawatt of installed capacity. The design was as simple as it could get and still provide safe,

reliable power; the cost could not be significantly lowered by additional detailed design modifications. With the substantial safety and operating margins in the AP600, it became obvious that the power output of the design must be raised without negating any safety margins and without raising the cost of the plant except where necessary to raise power. The result is AP1000. This report also outlines the process used for AP1000 and details many of the discrete design decisions made in its development. Elimination of the reliance on ac for safety functions is a key overriding concept for design simplification, enhanced safety and greatly improved PRA results. By keeping the design safe, proven, and simple, without relying on ac power for safety functions, the design continues to adhere to the principles of as low as reasonably practicable throughout.

1.0 Introduction

This report describes the design genesis of the AP1000 nuclear power plant by Westinghouse. It is a study in how experienced, dedicated nuclear designers combined innovations and lessons learned into a pressurized water reactor with safety features unlike any plants before it. The AP1000 design is the result of taking proven designs and design concepts and applying them to a defined set of functional requirements in the most simple, effective way practical. The design is founded upon rigorously holding to a few inviolate principles. First, NO ac power would be required to perform any safety function. This includes performing the big three of: stop the nuclear reaction, remove the decay heat and maintain reactor coolant water inventory without the need for ac power. It also includes all other safety functions such as: spent fuel pit cooling, main control room habitability, seismically qualified fire protection system for safety related equipment, and beyond design basis security related mitigation features. Second, maintain the fission product barriers of the fuel clad, the reactor vessel and coolant system, and the containment vessel. The containment vessel is an ideal barrier against radioactive releases to the environment. Transfer decay heat out of the core using natural, unpumped mechanisms like natural circulation, evaporation , conduction, convection and condensation. Third, minimize core damage frequency and large release frequency as calculated by a robust probabilistic risk assessment (PRA), by designing out failure modes in lieu of designing in mitigation features. This approach ultimately results in a plant design that is safe, because it is simple and the objectives of lowest hazard to the public and operators, lowest risk and lowest cost are achieved as by products of the process.

Through the 1970s and 1980s nuclear power plant operators and executives in the United States and elsewhere (for example France and South Korea) recognized the benefits of creating a standard design to be deployed on multiple sites. To that end, they worked with the United States Nuclear Regulatory Commission (USNRC) and power plant vendors to create a regulatory scheme based upon licensing and deployment of standard designs (10CFR52). This effort resulted in a large body of NRC regulatory requirements and guidance in addition to a body of industry codes and standards. This led to a recognized set of design goals and solutions. These nuclear utilities also created a plant specification for new nuclear plants based upon their collective operating experience and their vision for the future of nuclear power. The specification was called the Utility Requirements Document (URD) and included three parts: Generic Plant Requirements, Requirements for Evolutionary Designs using active safety systems, and Requirements for Plants with Passive Safety Systems using natural circulation for safety related heat removal. Thereafter European utilities created a similar requirements document tailored for Europe and called it the European Utility Requirements (EUR).

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When completed, as evidenced by a Design Certification granted by the United States Nuclear Regulatory Commission, the AP600 was the simplest, least expensive nuclear power plant available. It could not, however, compete in the United States with natural gas power plants at the time the AP600 was certified. To be competitive, the AP600 was required to lower its cost per megawatt of installed capacity. The design was as simple as it could get and still provide safe, reliable power; the cost could not be significantly lowered by additional detailed design modifications. With the substantial safety and operating margins in the AP600, it became obvious that the power output of the design could be raised without negating any safety margins and without raising the cost of the plant except where necessary to raise power. The result is AP1000.

This report also outlines the process used for AP1000 and details many of the discrete design decisions made in its development. Elimination of the reliance on ac for safety functions is a key overriding concept for design simplification, improved safety and greatly improved PRA results. This report starts by describing the United States' nuclear industry desires for change in the early 1980s (Section 2). It then discusses the principle of safety through simplicity and its application in military and commercial designs (Section 3). With this solid basis, the evolution of AP1000 from a military request to AP600 (Section 4) and from AP600 to AP1000 (Section 5) is described. We describe the process of achieving as low as reasonably practicable for every design decision in the AP1000; Section 6 presents many examples of the development of discrete decisions and features. Section 6 contains only a sampling of the design decisions made for AP600 and AP1000. Many more decisions were made to the same "make it safe and simple" process. Design decisions made for the initial or conceptual stage were not documented explicitly. Changes made after the design was placed under change control were documented using Design Change Packages (DCPs). DCPs are listed in Attachment 1.

By using the proven AP1000 design philosophy and process for every design decision, the AP1000 design has been kept safe, proven, and simple, without relying on ac power for safety functions. The design continues to adhere to the principles of as low as reasonably practicable throughout.

2.0 Industry and Standardization

The drive for standardization in the United States was born from its antitheses in the 1960s and 1970s. It was led by the executives of nuclear utilities and reflected the understanding that each different design for a nuclear power plant brought with it the cost of unique design features, initial problems, unique operating scenarios, maintenance and spare parts. Within the allowance that each nuclear power plant vendor and architect engineer/constructor will have its own design preferences, a set of desires, regulations, codes, standards and expectations could be developed that would promote a small set of standardized plant designs. This would lead to increased licensing certainty, relevant operational experience, overall safety and reduced cost.

2.1 Desire

The executives of nuclear utilities in the United States banded together and formed a group dedicated to establish an American nuclear renaissance. It was obvious that new nuclear could not compete with other forms of central station energy in the United States because of the relatively low cost of fossil fuels and the high risk of having a nuclear project delayed due to intervener interaction within the licensing process. Essentially complete nuclear power plants were abandoned because of the cost required to maintain loans while challenges to the plants were settled. These delays were inherent in the process for plant licensing embodied in Part 50 of Title 10 of the Code of Federal Regulations (10CFR50).

The executives wanted to establish a new form of plant licensing and delivery that created licensing and cost certainty while maintaining:

a. Adequate, open and proper safety review of the design and site by the United States Nuclear Regulatory Commission (NRC)
b. Appropriate public access to and comment on the entire licensing process
c. Closure of licensing decisions on a given design, even if that design were deployed on more than one acceptable site
d. Application of lessons learned and design centered rigor in the costing process
e. Obtain the plant Operating License prior to the start of major construction activities.

In summary, they wanted a process that promoted ease and certainty of licensing through one-time certification of a design that could be replicated with certainty of costing based upon a completed design and experience.

2.2 Implementation

Two major outcomes resulted from the utility executives' initiative. One was a new licensing process embodied in 10CFR52 and the other was a single whole plant specification, the Utility Requirements Document (URD).

The new licensing process embodied in 10CFR52 has many elements that lead to licensing certainty through design standardization. It allows for the one-time certification of a reactor plant design, with public interaction, without requiring the attachment of that design to a specific site. It also allows the certification of a site, with public interaction, without reference to a unique technology (plant design) selection. It requires sufficient information to be provided to NRC and found acceptable, with public interaction, for operation before safety related construction can begin. This process provides licensing certainty for the large capital investment required with the establishment of an NRC certified standard power plant design.

The URD established another element of standardization. Since it is a single plant specification, it established requirements and goals for the plant designer. These requirements were comprehensive, definitive, attainable and represented what the ultimate owners of the plants felt were important. First and foremost, the URD requires safety, and then it requires the lowest attainable risk to the public, the operators, and the investors. It covers most plant systems and structures, with requirements from layout to operations and maintenance based on operating experience of the utilities.

To reinforce their desire for new nuclear in the United States, utility executives helped create a program in the United States Department of Energy (DOE) and a legal consortium of nuclear utilities, the Advanced Reactors Corporation (ARC). Federal legislation was passed and funds were obtained to finance the design and certification of new nuclear with a cost sharing process. The government, utilities, reactor vendors, architect/engineer firms, constructors, and component designers and manufacturers shared the costs and effort to develop competing reactor plant designs. ARC took the lead in technical direction to ensure that vendors were adhering to the letter and spirit of the URD. At the same time the NRC was reviewing certification applications and certifying standard design solutions for each vendor's approach.

2.3 NRC Regulatory Requirements and Guidance

The NRC fully recognized the benefits of standard design solutions to standard regulatory guidance. It too had experienced the frustrations that came with every licensed plant in the United States being different. The formation of 10CFR52 created defacto standard design for the United States. 10CFR52 made deviations from a certified standard cumbersome to process through the licensing process. It supports the utilities and vendors objective of standardization. The more that could be standardized the better the 10CFR52 licensing process works. NRC also had a wealth of experience that spanned all the various reactor types and applications. While standard designs were being developed under the ARC sponsored programs, the NRC proceeded to update and revise its guidance for new nuclear within a framework of standard, replica plants. NRC requirements and guidance became more definitive without being necessarily tied to a single vendor or technology. This revision period was also a time of documenting the lowering of the allowable hazards and risks to the public and the operators to an unprecedented low level.

In summary, NRC reinforced a process that promoted ease and certainty of licensing through one-time certification of designs that could be replicated. This process also provides strong incentives for standardization.

2.4 Industry Codes and Standards

Consistent with the desires of utility executives and the efforts of the NRC, industry consensus code and standard organizations recognized the need to focus on developing or revising requirements and guidance. They worked on codes and standards that were comprehensive, concise, based upon experience, and that provided increased levels of safety and decreased levels of risk to the public and the operators. Throughout the nuclear industry, it was recognized that as the fossil fuel prices continued to rise, the need for safe, economic nuclear power also grew. It was clear that reduced risk could be achieved using known technologies and proven techniques. Industry codes and standards could be and were revised to provide increased safety through standardized simplicity. For example, codes and standards were expanded to use results from tests relying on forces like natural circulation and convection. Algorithms were validated to predict natural force response to design and beyond design transients. Alternate solutions to basic functional requirements were encompassed into the overall framework of the code or standard. Experience with new materials was included in the code "allowables". In summary, the industry supported the process that promoted ease and certainty of licensing while lowering the hazards and risk to the public and the operators.

3.0 <u>Principle of Simplicity</u>

Another underlying philosophy of the AP600/AP1000 design process, as well as the URD, is that the best path to safety is through simplicity. For AP600/AP1000 the process pushed the designers to eliminate failure modes in lieu of adding mitigation features. For example, in operating plants today the reactor coolant pumps use a controlled coolant leakage system for establishing a seal on the reactor coolant pump shaft. This shaft seal is a potential source of excessive leakage of reactor coolant. Shaft seal failure mitigation features and safety related responses to excessive leakage must be provided for these plants. In AP1000 the shaft seals are eliminated all together through use of canned motor pumps. Another example is the methods of post accident core decay heat removal. Operating plants today use a variety of systems to take reactor coolant out of containment, cool it down and return it to the core. This creates a large number of potential reactor coolant release scenarios, each requiring a mitigation strategy. In AP1000, reactor coolant remains within containment and only decay heat energy is transferred out of containment. The only remaining containment bypass, reactor coolant release scenarios are the highly unlikely leak in containment itself and the unlikely steam generator tube leakage event.

3.1 Military (SMPP)

The very beginnings of AP600 were influenced by two sources of experience with military nuclear plants, one indirect and one direct. Even though Westinghouse operated a design laboratory for the United States nuclear navy, there was very little interaction between it and Westinghouse's commercial nuclear business. This was for both security and technical reasons. The functional design requirements for a naval nuclear reactor are very different from a commercial one. There were a number of employees in the Westinghouse commercial nuclear business unit that had served in the United States nuclear navy. Although the details of their experience were classified, they knew that core heat removal by natural circulation methods could be very powerful. This background and knowledge had an indirect influence on the AP1000 design process since there was no philosophical need for pumps for decay heat removal.

The direct influence was an unclassified design study prepared by Westinghouse. A branch on the United States military was investigating ways to provide electrical power to advanced bases that did not rely on liquid fossil fuels. The power requirements for a small military power plant (SMPP) were relatively small (about 10 Mw), but the deployment requirements were restrictive. The power plant had to be air transportable to advanced sites. It had to be "walk-away" safe. Setup had to be rapid without extensive site preparation or construction equipment using preconstructed modules. This led to the requirement that the ultimate heat sink during power operations had to be ambient air.

Westinghouse developed an acceptable design with a number of features that were safe, but simple. Complicated solutions are not easy to deploy or set up. The concept of ambient air being the ultimate heat sink may not work for a large commercial station at power, but decay heat power levels for these same plants (less than 6% of full power) were well within the range of the full design power of the military plant. Here was a design demonstration of simple is safe and achievable passive safety.

3.2 Commercial

At the same time that utility executives were striving for a commercial nuclear power renaissance based upon standardization, they also were striving for power plants that were simple. Many of the safety systems they had to operate and maintain were based upon adding features to mitigate the effects of design basis accidents. Many of their unplanned outages were caused by unavailability of mitigation systems rather than by loss of power production systems. Many of these executives had military nuclear experience. Although their experience was not directly applicable to new central station plant designs, the executives knew that there were simpler ways to satisfy a plant's functional requirements than those embodied in their operating plants.

These executives had a bias for simplicity of design and use of natural forces. To satisfy this bias, Westinghouse developed AP600 as a combination of its commercial experience to design power systems to make power and its military experience to design simple safety systems, driven by natural forces to perform the safety functions of shut down the reactor, keep it cool and contain its coolant.

This integration resulted in a radical departure from the Westinghouse commercial power plants that came before AP600. Risk and cost were made as low as practicable by making the safety systems simple, automatic, driven by natural forces and diverse from the systems that make power. Now the power production systems became the non-safety related first line defense in depth systems backed up by simple safety systems. The need for mitigation systems were eliminated with the elimination of rotating equipment and elimination of other potential hazards. AP1000 is an evolutionary extension from AP600. AP600 is a revolutionary combination of simple safety systems with proven power production systems.

4.0 From SMPP to AP600

The beginnings of AP600 were based upon the revolutionary combination of simple safety systems with proven power production systems. The safety systems were called passive in that they did not rely on any ac power for their safety functions. Implementation of the concept of removing decay heat with naturally driven processes was taken from the military design experience noted above. The pressurized water reactor power production processes were taken from years of commercial nuclear power experience. This bold melding would remain a dream unless it could be shown to meet realistic safety, economic

and development goals. Early on it was recognized that if proven components were used throughout the design, component development would not be required. The proper melding of proven components and proof of Westinghouse's ability to analyze this melding became the challenge. The URD requires the use of proven components and proven technology.

4.1 URD followed by EUR

As noted in Section 2 above, the United States utility executives had a desire to establish and enforce standardization in new nuclear plants. They recognized that this could not be effective unless a consistent, consensus set of plant design requirements and guidelines were established. They took on the task of developing a plant level specification for new nuclear in the United States. This plant specification would combine and document the experience and desires of the member utilities as well as those of the nuclear plant vendors, the nuclear architect/engineers and industry consultants such as EPRI. With the help of EPRI as editor, the utilities developed a detailed nuclear plant specification called the Utility Requirements Document (URD).

With 10CFR52, associated rules and regulations of the NRC and the URD as a basis, the United States Department of Energy supported efforts to obtain a Design certification for AP600. It was recognized that First of a Kind Engineering (FOAKE) in addition to that required for a Design Certification would be required to establish realistic cost and schedule estimates for AP600. To support these additional FOAKE effort sixteen utilities (mostly American) joined together in an organization called the Advanced Reactors Corporation (ARC). The URD is a multi-volume specification that covers the four combinations of pressurized or boiling water reactors with evolutionary or passive technology. One volume is dedicated to overall requirements that apply to any technology selection. These include safety, hazard, cost and schedule requirements and goals. More detailed and extensive requirements and goals are contained in subsequent volumes. Each topic area is covered for all technologies with requirements and goals appropriately tailored. Most requirements are the same for all technologies since they cover items related to safety, support systems, reliability and maintainability goals, access, releases, personnel and investment protection and other similar topics. Of course, any design must be licensable in the United States in addition to being highly compliant with the URD.

With the establishment of the URD as the minimum acceptable standard for plant design a number of benefits resulted. The plant designers clearly had the voice of the customer and that voice was a standard, single voice for all participating utilities. This put the plant designers, including AP600, on the same basis for comparison. Technology selection in the United States would be based upon the delivery of the best value meeting the minimum acceptable standard in the URD. This naturally instilled into the design process a need to adhere to the concept of "as low as reasonably practicable". Another benefit of the URD, followed by the cost sharing of ARC into the AP600 design process was that the utilities became involved in the day to day design process. Over many years, up to four utility representatives were installed in the Westinghouse design offices taking an active role in ensuring that the utility definitions of "low" and "best" and "practical" were included in the design details. Utility involvement in the design process ensured that the design of AP600 included not only utility experience, but also the safety and robustness necessary to be a viable nuclear power plant.

A few years later, European utilities formed a steering group for new nuclear in Europe under the banner of European Passive Plant (EPP). The concept of standard was already proven in Europe and the design solution of AP600 was attractive. Again, to have a common specification for all member utilities for EPP, a derivative of the URD was developed. This derivative (the EUR) made changes to the URD only where necessary to reflect different requirements for Europe from the United States. Again, utility involvement became part of the development of AP600 and then AP1000. Westinghouse had the desire to make AP600 an international standard product to the greatest extent possible. This would allow incorporation of best practices from around the world to be part of the AP600 design. This reinforced a worldwide standardization objective of Westinghouse and our potential customers.

4.2 Design Objectives

As for any design, the development of AP600 involved trade offs and implementation decisions. The URD had many requirements, each of which could be satisfied in a number of ways. The development of AP600 stood on the shoulders of extensive design and operating experience in both commercial and military nuclear power. This experience included many design solutions for each challenge based upon design requirements or designer preference or both. To focus on design solutions for AP600 a single set of overall design objectives were established and enforced.

Safety was, and is, always first. Design decisions were made in favor of the more safe solution, even if it were more expensive. As discussed above, the process for AP600 was founded on designing out the potentially unsafe condition, if possible, rather than designing in a complicated mitigation strategy. As shown in many of the examples below, the safest design is also the simplest. Design decisions were carried through a number of reviews, both formal and informal, to ensure that the design was as safe as possible by making it as simple as possible. These reviews included those by the design group itself, the AP600 design team, independent design review teams and teams of utility employees provided by ARC. If any of these reviews reveled that the design implementation of design change reduced safety, those responsible for the design were told to go back and try again.

AP600 had the luxury of starting from a set of functional requirements, a large inventory of experience and a clean sheet of paper. The AP600 design was not constrained by an inviolate set of design solutions. As a result the design team could proceed to solve each functional requirement with the tempering of industry experience using proven components in innovative ways. The AP600 approach to all safety systems and safety related functions is to perform the function without reliance on ac power. No active (requiring ac power) design solutions were allowed. Passive systems using natural forces (gravity) are more reliable, simpler and thus safer than active systems. This provides the lowest potential for core damage and radiation release at the lowest price.

The AP600 safety systems were thoroughly tested. Component level and system level tests were performed in a variety of scales from full to sub-scale. Tests were performed in the United States, Italy, and Canada. Results proved that the design team and the analysis computer codes had the capability to predict the performance of the passive safety features of AP600. The NRC also ran independent tests in the United States and Japan and proved that the AP600 results were conservative. This reinforced the validity of a safety design that is safe, simple, and reliable.

In addition to the design objectives of safety first and no ac power for safety related functions, the AP600 design process included making constructability, reliability, operability and maintainability part of the deign. With the help of utility reviews and daily involvement, these design features were embedded in the design, not added on. By embedding these features, the design team created an AP600 with as low as practicable construction risk, operation risk, operator dose, and maintenance dose.

Constructability was "designed in" by the use of extensive modularization. Modularization is a technique where portions of the plant are assembled and tested before they are placed in their permanent plant location. These modules can be built at an off site facility (factory or shipyard) or on site, but not on the building foundations. Modules provide the benefits of greater quality control in the shops than on site. It promotes greater standardization and ability to use lessons learned from previous builds. It provides for more predictable site construction with less field deviations. In summary, it provides for predictability of the lowest construction risk for the lowest overall price in the shortest possible construction schedule.

Reliability was "designed in" by using the PRA as a design tool in lieu of simply for design verification. PRA considerations were included in many design decisions. It is important to ensure the lowest public risk. Since safety is always the first, consideration of lower risk, as evidenced by PRA results, carried more weight than lower cost. Applying the concept of simplicity in a rigorous manner most often improves safety and reduces cost.

Operability was “designed in” by designing out the operator for response to Design Basis Accidents. This eliminates operator reliability from the response to accidents. Using natural forces promotes the elimination of the operator from safety related responses. Operability was also “designed in” by the active involvement of utility representatives. Through daily involvement of the utility representatives in the AP600 design team and the periodic and ad hoc design reviews by utility review boards, operability concerns were addressed and resolved before designs were finalized.

Maintainability was “designed in” by extensive layout reviews by both the AP600 design team and utility representatives. From the beginning of AP600 development, the layout was generated in 3D CAE software. As each item (structure, equipment, pipeline, duct, and tray) was added to the design, it was checked for interferences, inspection access and maintenance access. These evaluations were performed with utility involvement. Design decisions were made to minimize maintenance time and accumulated dose. For the digital instrumentation and control systems in AP600 self diagnostics are included.

This fresh approach and the design objectives of having larger margins than operating plants while using proven components in a passive way resulted in the robustness of the AP600 design.

4.3 Economics

Deployment of nuclear power is dependent upon it being the most economical total cost solution for central station power for the customer. Deployment of AP600 was dependent upon it being the most economical power generation solution. Three main contributors to cost are complexity, development and time. The AP600 design process and design objectives attacked all of these contributors to drive the cost as low as reasonable practicable.

Complexity was attacked by the design team’s commitment to simplicity. Not only is simple the most reliable and the safest, it is the least cost. Realization of the design commitment to simplicity reaps the result of a commitment to low cost. Development was attacked by the design team’s commitment to only using proven components. Using proven components eliminates the need for development all together. The remaining task is to justify use of the component in the AP600 instance. Justification of use is much easier than development of technology, and takes less time. Time was also attacked by minimizing construction time using modules. Using modules allows construction activities to proceed in parallel. Major portions of the plant can be constructed in different locations and the critical path can be determined by placement of concrete in lieu of equipment.

The design process for AP600 used economics as another driver to make the design as safe, simple and proven as possible.

4.4 Licensing

Licensing in the United States using 10CFR52 allows for the granting of a Design Certification of a design by NRC. A Design Certification can be generic and not site specific. It creates licensing certainty of the design since, once certified, changes to the design and questions on it are limited by law. This pre-licensing of the plant design prior to the start of construction requires completion of significant portions of the design. Pre-engineering creates a design with implementation risk as low as possible.

The NRC conducted an extensive review of AP600. It asked over 7,500 formal questions and conducted over 400 review meetings. This resulted in a design that maximized safety while minimizing risk and hazard to the public and operators. It maintained the design reliance on passive safety features using natural forces for protecting the core and the public. The AP600 design that received a Design Certification was safe, simple and in NRC words, the "most thoroughly tested plant licensed".

A pre-licensed, pre-engineered design contributes to the certainty of meeting cost estimates and construction schedules, thus achieving a "low a reasonably practical" cost and schedule.

5.0 From AP600 to AP1000

When AP600 received its Design Certification, it was the safest, simplest, least expensive nuclear power plant on the world market. As indicated above, other nuclear plants were not AP600's competition, other central station power sources were. In particular, natural gas plants were the economic plants of choice in the United States. In order to compete against natural gas plant at the time, the AP600 would have to lower its cost per megawatt by over 30%. As indicated above, the AP600 cost per megawatt was already as low as reasonable practicable because of its inherent simplicity. To lower its cost by eliminating any more systems, structures, or components would lessen its safety margins and increase its risk to the public. Obviously this approach was rejected. Instead, it was decided to raise the power level of the design without raising the overall plant price an equivalent amount to drive the cost per megawatt down so that the cost of electricity generated by a nuclear plant could compete with natural gas plants.

This design power increase needed to be constrained to reap the benefits of the $450,000,000 worth of design and licensing effort already invested in the AP600 design. The constraints included:

a. Safety first – maintain large margins to safety limits
b. Maintain passive nature of all safety functions
c. Maintain no operator actions for safety functions
d. Maintain use of proven components and technology
e. Do not change the plant footprint and lose layout and analysis already completed
f. No design impacts unrelated to power
g. Minimize design impacts on the Design Control Document (Design Certification)

The resulting AP1000 design met cost goals while changing only those features necessary to increase power and maintain safety margins. The nuclear island footprint remained unchanged by adding height to the reactor vessel and containment vessel while maintaining their diameters. Large margins to safety limits were kept. No departures from proven components were introduced. The testing data obtained for AP600 were shown to be applicable to AP1000. The AP600 design process and decisions were retained. The design improvements admitted into the AP1000 design were to implement the higher power with the same dedication to safety and simplicity as for AP600. The resultant AP1000 design became at least as low as reasonable practicable as the AP600 and, in some areas, more so. URD compliance was retained since design features included to satisfy URC requirements were generally not modified in the transition from AP600 to AP1000.

6.0 Examples of Passive or Simple or Both

This section has a selection of design decisions made for AP600 and AP1000. Since none of the design decisions described for AP600 were reversed in AP1000, the entire section ultimately applies to AP1000. These decisions occurred over the design life of AP600 and AP1000, some 15 years, and many occurred concurrently. The selection of these example decisions was such that they are mostly independent of each other. The reason for including them in this report is to demonstrate the comprehensive nature of the AP600/AP1000 design process. In all aspects of the plant design, the process reinforced a rigorous, disciplined approach to achieving safety through simplicity and developing a design that is as low as reasonably practicable.

6.1 AP600

The decisions discussed below are examples of design choices made to develop the AP600. These decisions were carried on into AP1000. They demonstrate that the design process for AP600 proceeded in a manner that continuously mandated and reinforced the concept of safety through simplicity and as low as reasonably practicable.

6.1.1 Reactor Coolant Pump Selection

The selection of canned motor pumps for AP1000 epitomizes the benefits of selecting safety through simplicity. The function of the reactor coolant pump is to deliver adequate cooling water for power operations and for accident shutdown situations. The classic reactor coolant pump style in the United States is a shaft seal pump. It can be made large and can have high hydraulic and electrical efficiencies. A basic premise of AP600 was to maintain safety and respond to accidents without reliance on ac. For core cooling this meant natural circulation through the core to the reactor coolant heat sink. The tradeoffs here could be numerous, dc powered safety pumps, ac powered shaft seal pumps, canned motor pumps, no pumps (natural circulation), and others.

A basic premise of AP600 was to maintain safety and respond to accidents without reliance on ac. For post reactor trip core cooling this meant natural circulation through the core to the reactor coolant heat sink. Clearly the no pump solution is the simplest choice if natural circulation can be shown to provide adequate post shutdown core cooling. Many members of the AP600 design team had experience with canned motor pumps in the United States nuclear navy. They knew that these types of pumps were highly reliable and represented a reactor cooling pump solution without coolant leakage. They recognized that canned motor pumps could never be made to be as efficient as shaft seal pumps.

The basic tradeoff here for power operations, for the same level of safety, is pump efficiency versus simplicity and reliability. Since the motor and the pump bearings are within the coolant boundary, the canned motor pump also allows the designer to eliminate shaft seal pump support systems such as seal injection, seal leak off, lube oil and fire protection systems. Other aspects of the tradeoff leading to the lowest hazard and risk to the operators were investigated. Unlike shaft seal pumps, canned motor pumps cannot be repaired in situ. Designs were required for quick removal and replacement of entire pumps. Unlike shaft seal pumps, canned motor pumps of the size required for AP600 had not been built before. This led to the decision to use two canned motor pumps of modest extrapolation for each steam generator loop. This allowed the attachment of both pumps directly to their steam generator, eliminating the cross over leg required for shaft seal pumps. This also eliminates the high/low stagnation portion of the cross over thus promoting natural circulation for post accident cooling. Selection of the canned motor pump did add a complication. Relying on natural circulation core cooling in the long term is fine if the core/heat sink thermal centers are far enough apart. Natural circulation does not supply sufficient cooling flow at the very beginning of a shut down transient. The passive solution to this challenge is the addition of rotating inertia to the pump in the form of a heavy flywheel. A true trade off between shaft seal and canned motor pumps can now be made.

The selection was made for the canned motor pump based on simplicity and reliability. Sacrificing efficiency for higher inherent reliability and the elimination of all pump support systems and attendant reactor coolant leakage was an easy decision. Canned motor pumps are proven commodities that had to be enlarged for AP600. The new design features for additional rotating inertia were tested and proven. The pump is not expected to function post accident and its pressure boundary is continuous without any planned or unplanned leakage.

In summary, the canned motor pump was chosen over the shaft seal pump for reactor coolant service in a process that promoted satisfying its design requirements with lowest radioactive effluent, lowest risk for accidental loss of coolant, high reliance on proven technology, lowest risk for public or operator radiation exposure and lowest overall plant cost. It exemplifies safety through simplicity.

6.1.2 Reactor Coolant Post-LOCA Injection and Cooling

Following a loss of coolant accident (LOCA), a reactor plant's safety systems must provide makeup for the water lost in such a way as to maintain reactor core cooling. Many pressurized water reactors today rely on pumped systems and large sources of water from outside containment to provide this make up and cooling. These types of systems require safety grade and seismic 1 sources of ac power and water. In the case of AP600, this must be done without reliance on ac power.

Many tradeoffs exist for meeting the basic functional requirement here. The required motive power could be dc, gravity, stored (static) energy in the form of pressurized gas or a combination of all of these. The required water could be stored in containment, on containment, outside containment or a combination of all of these. Different LOCA scenarios require different amounts of water at different times. Sufficient inventory with sufficient delivery capacity throughout the transient must be maintained. In addition, it is very desirable to deliver the right amount of water, at the right time without operator involvement. The true tradeoff here is then between the current complicated, ac powered, outside containment delivery and cooling system, and one that simply relies on total pressure balances and natural circulation.

There was little large scale experience with total pressure balance/natural circulation LOCA response systems. As part of the AP600 development program, extensive testing was performed to validate system functionality and analysis capability for candidate pressure balance/natural circulation LOCA response systems. In addition, probabilistic risk assessment studies were performed to assess the safety value of various system alternatives.

The decision process for system element selection included design basis analysis and PRA results. It focused on developing the simplest set of systems that could maintain core cooling with all safety related water inventory contained within containment. Resulting implementation strategies included redundant and diverse systems and components within those systems. Safe shutdown conditions with margin and without operator actions and no requirement for ac power was achieved.

The resultant selected set of passive core cooing components are wholly contained within containment. They include the passive residual heat removal heat exchanger, the core makeup tanks, the accumulators, the in containment refueling water storage tank and the passive core cooling long term recirculation system.

The benefit of this solution is a very safe, simple set of core cooling features, driven by natural forces, extensively tested and analyzed. It provides safety through simplicity by satisfying its design requirements with no potential radioactive effluent, no risk for accidental loss of coolant outside containment, high reliance on proven technology, lowest risk for public or operator radiation exposure and lowest overall plant cost.

6.1.3 Load Follow w/Rods, Elimination of Boron/Water Recycle

Most central station nuclear power plants today are operated as base load plants. The United States utilities, in the URD, required that new nuclear plants must be designed for a defined level of load follow. To provide some level of load follow today, many plants have systems that manage boron concentrations in and recycle boron in and out of the reactor coolant water. This requires elaborate and complicated boron and water handling systems and results in restrictions on the rate of load follow available. Based upon military experience, AP600 designers recognized that there are alternatives to reactivity control other than boron concentration in the reactor coolant. The tradeoff for AP600 came to boron recycle versus shim control in the control rods. Shim control is the use of moveable control rods with low density neutron absorber (gray rods) that can be moved to provide reactivity controls in addition to normal reactivity feedbacks. The materials for shim rods are well known and their effectiveness for partial reactivity control is easily analyzed. Note that shim rods are used in addition to safety rods and are not needed for reactor shutdown. The decision process for load follow control chose the proven, safe and simple method of shim rods over the complex method of boron recycle. The benefit of this solution is that it provides safety through simplicity by satisfying its design requirements with no potential radioactive effluent, no risk for accidental loss of coolant outside containment, high reliance on proven technology, lowest risk for public or operator radiation exposure and lowest overall plant cost while maintaining complete shutdown margin in the shutdown rods.

6.1.4 Use of Demineralizers in Lieu of Evaporators for Liquid Waste Processing

Radioactive isotopes accumulate in the reactor coolant and spent fuel pool cooling water during operation. Some of these isotopes are gaseous or volatile; most are soluble or suspended in reactor or spent fuel pool coolant water. During plant heatup or coolant boron concentration adjustments by feed and bleed, volumes of this potentially radioactive water accumulate as waste water. In addition volumes accumulate as a result of sampling operations or as leakage. These sources will accumulate to the point where they must be discharged from the plant. Unlike many plants, AP1000 has no planned leakage of reactor coolant from pump shaft seal leak off systems. See Section 6.1.1. In addition, AP1000 has no plans to recycle dissolved boron in reactor coolant for load follow changes. See Section 6.1.3. By these design decisions, AP1000's major radioactive water source is from let down during heatup. Reuse of this water will normally be required after many months of operation during cooldown.

The major design decisions here are two. First, should the potentially radioactive waste water be stored and recycled and second, how should the plant process potentially radioactive water to concentrate the radioactivity to levels that minimize discharge volumes without creating a radiological hazard to the public or operators. For storage, the tradeoff is between having the equipment to store, monitor, process and recycle relatively small amounts of water and not requiring any equipment for this capability at all. For concentration processing, the tradeoff is between the use of evaporators and concentrating the radionuclides in liquid radioactive waste or the use of demineralizers to concentrate the radionuclides in resin as a form of solid radioactive waste. Evaporators are complicated, involve a number of fluid systems and use plant energy that could be used as net electrical output. Ion exchangers or demineralizers use disposable resin to capture radionuclides in a highly concentrated solid form.

The tradeoff then is simplicity versus complication. For storage, it is between additional equipment and operational requirements for storage and the no additional equipment by making a small amount of additional demineralized water between shutdowns. For concentration, it is between complicated evaporators with liquid radioactive discharge and simple demineralizers with solid radioactive discharge. The design decisions were for simplicity, reduction of equipment, operations, potential failure modes and energy loss.

In summary, simple capture of radioactive isotopes in ion exchange resins was chosen over more complicated methods in a process that promoted satisfying its design requirements with lowest risk for accidental loss of radionuclides, high reliance on proven technology, and lowest cost.

6.1.5 Chemical and Volume Control

The functional requirements for the Chemical and Volume Control System (CVS) are to fill, make up, let down, drain and maintain the proper chemistry of reactor coolant water. In operating plants today, these functions are performed by a variety of safety related subsystems that are outside containment. The basic design philosophy of passive systems eliminates the need for safety related coolant charging or letdown. Other tradeoffs to simplify other portions of the AP600 CVS (see 6.1.1, 6.1.3, 6.1.4 and 6.1.20) have eliminated the requirement to continuously pump borated makeup water into the reactor coolant system or to include complicated water processing systems in the design. This allowed additional obvious simplifications to the CVS. The functions of reactor coolant makeup, boron injection, letdown, purification, and others are non-safety related making most of the system non-safety related. Redundancies and potential safety related failure modes associated with these functions were eliminated. Boric acid transfer is gravity fed from the boric acid tank to the reactor coolant makeup pump.

In summary, the CVS functional requirements were satisfied by simple designs using a design process that promoted the lowest radioactive effluent, lowest risk for accidental loss of coolant, high reliance on proven technology, and lowest risk for public or operator radiation exposure.

6.1.6 Post Accident Isotope Control

Radioactive isotopes accumulate in the reactor coolant during operation. During a loss of coolant accident (LOCA) these accumulated isotopes are released into containment. Some of these isotopes are gaseous or volatile, most are soluble or suspended in reactor coolant water. During a LOCA these soluble and suspended isotopes are dispersed throughout upper containment creating a radiation source. This source can be strong enough to be a hazard to those outside containment. Operating plants today use a containment spray system to "wash" these soluble and suspended isotopes out of the containment atmosphere and off the containment walls. These containment spray systems include a water source outside containment, containment penetrations, pumps, valves, nozzles, and other equipment that must be redundant, qualified, controlled, tested, maintained and repaired.

AP600 already relied on natural forces like buoyancy, condensation and conduction to move decay heat energy from lower regions of containment to the containment walls. The steam/water mix that condenses on the containment wall returns to the Incontainment Refueling Water Storage Tank (IRWST) or the containment sump by gravity. Through analysis and test it has been shown that soluble and suspended isotopes move through upper containment with the water and thus move to the lower portions of containment. The tradeoff then is simplicity versus complication. It is between no water passing through containment and a system of equipment outside containment with its associated risk of not working or creating a containment to atmosphere leak. Using the natural isotope movement process does create a slightly higher general accident dose rate outside containment that is still within allowable limits. Following analysis and tests of the mechanisms for movement of isotopes within containment, the designers chose to use the simple natural removal process and not require a safety related containment spray system.

In summary, natural movement of LOCA related isotopes without containment spray was chosen over a containment spray system in a process that promoted satisfying its design requirements with lowest risk for accidental loss of coolant, high reliance on proven technology (natural forces), and lowest cost.

6.1.7 Beyond Design Basis Features

In addition to showing that the design of AP600 could pass the acceptance criteria for design basis accidents on a worst case basis, the design team had to show that AP600 could acceptably deal with beyond design basis events on a best estimate basis. Beyond design basis events are severe accident scenarios selected based upon NRC criteria. Beyond design basis features are included in the design to maintain the impact of selected severe accidents to as low as reasonably practicable. The postulated impact of selected severe accidents is calculated using an expanded version of PRA (Level 3). The use of PRA techniques allows for making design decisions in a disciplined way that provides for the lowest risk at the lowest overall incremental cost.

The best example of an AP600 beyond design basis feature is In Vessel Retention (IVR). Although numerous PRA techniques are involved in the selection and analyzing beyond design basis events, a common one is core melt. The common design feature for core melt is some form of "core catcher" outside the reactor vessel. A core catcher would have features that precluded recriticality of the corium and cooled it to slow its reaction with materials around the reactor vessel. This could have been the design solution for AP600. The design team recognized that implicit in its passive core cooling approach is the introduction of vast amounts of water into the lower portions of the containment. The design level of water in containment after an accident is above the nozzles of the reactor vessel. So another design solution for the core melt scenario is to take credit for this water and cool the corium with it.

To realize this solution, the design team had to establish two major things. First, prototypic testing had to be performed to establish the capability of water outside, but on, the reactor vessel to cool it with the heat fluxes expected during a core melt. Second, a mechanical design of the reactor vessel insulation had to be completed that allowed for water to get next to the reactor vessel in a severe accident while not allowing air to flow next to the reactor vessel during normal operation. Both of these were established. Testing was performed at the University of California – Santa Barbara to establish design parameters for cooling the vessel during core melt. A unique design of the lower portion of the reactor vessel insulation design was developed to use buoyancy to allow water in when present, but not allow air. This solution of IVR was selected and implemented in the design.

Implementing IVR this way provides a safe, simple, natural cooling mechanism for the reactor vessel that keeps vessel integrity and obviates the need for an external core catcher. Again, the need for mitigating features (core catcher) was eliminated by eliminating the failure mechanism (reactor vessel melt through). In the event of an extremely unlikely severe accident leading to a core melt, the incontainment water sources from the IRWST and passive core cooling components is collected in the lower portions of containment. It is allowed to flow into the reactor vessel insulation structure and next to the reactor vessel. It then cools the reactor vessel by convection and evaporation. The steam rises into the upper containment carrying core heat with it. This steam condenses on the containment vessel inner surface and returns to the lower portion of containment completing the cycle.

In summary, natural movement of incontainment water over the reactor vessel was chosen over a core catcher outside the reactor vessel in a process that promoted satisfying severe accident design requirements with lowest risk for accidental loss of the cooling function, high reliance on proven technology (natural forces), and lowest cost.

6.1.8 Selection of Squib Valves (customer driven, leak tight, reliable actuation, simple In service testing)

Introduction

The use of squib valves in AP600 design was originally suggested by the utilities. The nature of squib valve body design makes it valve virtually leak free (valve is not subject to internal leakage as with standard valve designs-globe, butterfly, gate, check, etc.)

Squib valves are used for the following thee (3) applications in the AP1000 passive safety systems:

1. Stage 4 Automatic Depressurization System (ADS) valves
2. IRWST injection line isolation
3. Containment recirculation line isolation

The design of the passive safety systems include sufficient redundancy in the design to accommodate limiting single failures, which includes consideration for the failure of a squib valve or their associated power supply or actuation signal.

There are four Stage 4 ADS depressurization flow Paths, two on each HL. Each of these valves is powered/actuated from three different I&C systems. Two of these are PMS divisions and the other is DAS. Any one of the three I&C systems can actuate the valve.

For IRWST injection, four squib valves are provided; two parallel squib valves are located in each of the two injection lines from the IRWST to the reactor vessel. Each IRWST injection squib valve is powered/actuated from two different I&C systems. One of the systems is a PMS division and the other is DAS. Either system can actuate the valve.

For containment recirculation, four squib valves are provided; two parallel squib valves are located in each PXS recirculation subsystem. Each PXS recirculation subsystem connects to one of the two redundant IRWST recirculation lines. Each recirculation squib valve is powered/actuated from two different I&C systems. One of these systems is a PMS division and the other is DAS. Either system can actuate the valve.

Reliable Squib Valve Actuation

The squib valve actuation equipment includes numerous design features that contribute to reliable squib valve actuation during an event.

There are four safety-related divisions of digital I&C equipment used for squib valve actuation. The safety-related I&C design helps to provide reliable automatic or manual squib valve actuation and also help to prevent inadvertent squib valve operation. The digital I&C features include:

a. Redundant power supplies for each instrumentation division
b. Redundant component actuation circuits within each I&C division, which, for example, would include redundant integrated logic cards that perform the two-of-four comparison for the actuation logic and associated automatic actuation permissive and interlocks.
c. Redundant signal transmission paths on the digital data highway such that a digital cable failure will not disable either the input/output cabinets or main control room operator workstations.
d. Continuous, automatic self-checking features that monitor the performance of the individual instrumentation channels and perform channel comparisons.

All squib valves receive firing signals from one safety-related instrumentation division (energized by separate Class 1E batteries for each division) except for Stage 4 ADS valves that each has two safety-related igniter firing circuits for improved reliability.

The safety-related actuation circuitry includes four divisions of actuation sensors that input the actuation signals and perform the comparator logic fire the squib valves automatically when two of the four actuation signals reach the set point. The comparison is performed by all four divisions of the actuation circuitry. Then each division provides the actuation signal to fire the igniters for the specific squib valve assigned to that division.

The two-of-four automatic actuation logic fires the squib valve when plant conditions are satisfied, even with a failure of a single instrumentation channel since three channels remain to satisfy the actuation logic. In the same sense, the failure of a single instrumentation channel will NOT cause spurious squib valve actuation.

In addition, each squib valve has a separate nonsafety-related (DAS) igniter firing circuit (energized from a non-Class 1E battery) that is actuated by manual operator actions in the event of a failure of the safety-related actuation circuitry.

Squib Valve Design Reliability Comparisons

Experience has shown the squib valves are more reliable than both AOVs and MOVs, because of the reliability of the actuating propellants and also because of the simplicity of the squib valve mechanical design, as compared to other types of valves in the same process application.

For comparison, the following probabilities for the failure of each type to open on demand are used in the AP1000 PRA:

AOVs	8.76E-03
MOVs	1.41E-02
Squib Valves	5.80E-04

In-Service Testing

The in-service testing for each squib valve includes both a test for remote position indication and test firing of the igniter/propellant outside the valve without the valve being actuated. ASME code requires that 20 % of the charges be tested every two years. AP1000 performs these tests during refueling outages when the squib valves can be accessed for propellant charge removal.

The squib valve charge assembly is removed and test fired outside of the valve in a test rig that can monitor explosive charge performance. Any failures would result in the removal of all charges from the same production lot and replacement with new charges from a different lot.

It is also important to carefully track the shelf life and service life for the explosive charges and igniters that are both stored on site and installed in the plant valves.

In summary, squib valve design provides a simple, reliable, and leak free solution to our AP600/AP1000 passive system needs.

6.1.9 Fire Protection Function for PCS Tank

The regulations for nuclear power plants in the United States include extensive requirements for fire protection. Included within these requirements is one that specifies that the fire protection water delivery system for fires affecting safety related equipment must be classified as seismic category 1. Operating plants today have seismically qualified fire pumps, pump power sources, ground loops, storage tanks, and delivery systems. AP600 has the self imposed requirement that safety functions must be performed without ac power. In addition, AP600 only has one building that houses safety related equipment and is seismic category 1

(Nuclear Island). This building is divided at each level by a concrete wall without doors. On one side are systems with potentially radioactive fluids. On the other are the plant control and protection equipment and control room operators (clean side). The lowest level of the Auxiliary Building is below grade, so there can be no drainage of fire water without ac power. As a result, AP600 has a restriction of how much water can be put into the clean side of the Auxiliary Building.

The tradeoff here is that a conventional seismically qualified fire system be installed with a diesel driven fire pump and limitations on water delivery to the clean side of the Auxiliary Building, or that some other, much simpler, design solution be developed. For either solution, standard National Fire Protection Association (NFPA) approved equipment must be used. It is obvious that the simplest, least expensive fire protection system is one that requires no fire fighting fluids at all. The AP600 design process was directed at creating a layout and defining fire areas and zones so that all equipment in a given fire area could be lost to the fire without loss of overall plant safety functions. Within containment additional spatial separation requirements were enforced for redundant equipment to ensure safety functions could be performed in the event of a fire in containment. This eliminated the requirement for a pumped fire protection system for reactor safety except for beyond design basis fire events in the clean side of the Auxiliary Building. Pumped fire protection delivery systems are throughout the design for investment protection, but are not required for protection of safety equipment.

The AP600 design process led to a very simple solution for providing a seismically qualified fire water delivery system that does not require ac power and can deliver only a limited amount of water to the clean side of the seismic category 1 Auxiliary Building. This solution is to dedicate the amount of water required to satisfy the NRC requirement for 2 hose streams of 75 gallons per minute for 2 hours within the seismic category 1 Passive Containment Cooling Water Tank (PCCWT) on top of the Shield Building portion of the Auxiliary Building. This fire water delivery system and the building housing it is seismic category 1. This fire water delivery system delivers water to fire hose stations by gravity. This eliminates the need for any active fire pumps. Standpipes in the PCCWT limit the available amount of water for fire fighting to less than that used for flood up protection of the equipment in the clean side of the Auxiliary Building.

The benefit of this solution is a very safe, simple set of fire protection features, driven by natural forces. It provides safety through simplicity by satisfying its design requirements without the need for ac or diesel power or the need for seismically qualified fire piping and tanks outside the Auxiliary Building.

6.1.10 Low Leakage Containment (passive, dose reduction)

Containment is the required last boundary between uncontrolled release of radioactive fission products and the environment. There have been many design solutions for the design of containment. These have included steel containments, concrete containments and steel lined concrete containments. Included in the design requirements for containment are that it must retain gases inside containment up to the containment design pressure and the design pressure must exceed the maximum expected pressure during a design basis event such as a steam line break or large break loss of coolant accident (LOCA). Another is that the pressure inside containment must be reduced to one half the peak event pressure in 24 hours. There are a number of ways this second requirement has been met including containment spray and controlled containment leakage or release. Added on to these requirements are the fundamental AP600 requirements that the design solution must be passive and simple.

The designers knew that the simple solution for post accident isotope control was the elimination of containment spray (see Section 6.1.6) and that the safest way to control accident releases from containment is not to have any. They designed a containment that is a free standing steel pressure vessel in accordance with the requirements of the ASME Code. This vessel has a high enough design pressure, a large enough free volume and a large enough heat transfer area to accommodate the worst design basis pressure challenge without the requirement to vent. Pressure vessel design requirements extend to all penetrations and attachments. The addition of passive containment cooling by distributing water over the exterior of the vessel provides a passive means of aiding heat removal and reducing internal pressure.

This solution was chosen using a process that promoted satisfying design requirements with the simplest possible design goal, eliminating the likelihood of containment leakage or the need for containment venting. This results in a solution with high reliance on proven technology that reduces the risk for public or operator radiation exposure.

6.1.11 Startup Feedwater Cavitating Venture (passive, avoid SG overfill)

The startup feedwater pumps and their associated flow paths perform a defense-in-depth function of decay heat removal from the reactor coolant system to prevent unnecessary actuation of the passive safety-related decay heat removal system. During a transient at least one out of two startup feedwater pumps take suction from the condensate storage tank and deliver feedwater to the steam generators. Although it mitigates loss of feedwater events, this function of the startup feedwater system is nonsafety related. The passive core cooling system is the safety-related system that provides safety-related protection for loss of feedwater.

The potential exist for excessive cooldown or steam generator overfill if startup feedwater flow increases too high. The design of AP1000 employs a cavitating venturi at the discharge of the startup feedwater pumps to limit pump flow. The venturi flow elements provide a passive mean to choke startup feedwater flow and avoid further flow increase as pump flow reaches flow limits. The cavitating venturi also provides a flow measurement signal at normal flow rates.

In summary, the startup feedwater system provides defense-in-depth function of decay heat removal which prevents actuation of the passive safety systems. Each startup feedwater pump is equipped with a cavitating venturi which provides protection against steam generator overfill or excessive cooldown by limiting pump flow. The venturi was chosen using a process that promoted satisfying the design requirements with lowest risk for excessive startup feedwater flow, high reliance on proven technology (passive design), and lower cost.

6.1.12 Catalytic Hydrogen Recombiner (passive)

There are a variety of mechanisms in a nuclear power plant that can generate free hydrogen gas. Most of these generate very small amounts, while some related to beyond the design base severe accidents can generate large amounts. Regardless of the source, accumulations of hydrogen can rise to a potentially explosive level. To ensure continuous, simple, hydrogen removal capability that does not rely on ac power and that can be environmentally qualified for post accident service, catalytic hydrogen recombiners were chosen for incontainment hydrogen control. These recombiners are in addition to the hydrogen igniters placed throughout containment.

In summary, the catalytic hydrogen recombiners were chosen over more complicated hydrogen removal schemes in a process that promoted satisfying design requirements with lowest radioactive effluent, lowest risk for hydrogen burning, high reliance on proven technology, and lowest risk for public or operator radiation exposure.

6.1.13 TSP Baskets – passive

Post LOCA conditions within containment require that the free water in containment be treated to maintain its pH within prescribed limits. This is done to ensure that the chemistry of fission products is proper. In many operating plants this pH control is established by the chemistry of the containment recirculation water brought in from tanks outside containment. As stated earlier, the AP600 safety related response to LOCA without ac power is performed without any water entering or leaving containment. A variety of solutions for pH control inside containment during a LOCA were investigated. These included incontainment tanks with buffer solution and safety related controls and baskets with solid tri-sodium phosphate TSP in containment. TSP is safe, stable, readily soluble in water and easy to inspect. The solution chosen was to install baskets low in containment that contain TSP. In the event of a LOCA, the water accumulating in lower region of containment would self buffer by dissolving the TSP.

The benefit of this solution is very safe, simple post LOCA incontainment pH buffering, driven by natural forces, extensively tested and analyzed. It provides safety through simplicity by satisfying its design requirements with no potential radioactive effluent, no risk for accidental loss of coolant outside containment, high reliance on proven technology, lowest risk for public or operator radiation exposure and lowest overall plant cost.

6.1.14 Ancillary Equipment – Extend Coping Period with No Off-Site Power from 72 Hours to 7 Days

NRC mandated that passive system accident mitigation features must be operable for 72 hours after the start of the accident. This time is clearly sufficient for AP600 response to design basis accidents. The only requirement for motive power after an accident starts is for the one time realignment of safety valves. Heat and fluids then move by natural forces. These one time valve movements are few (<20) and happen relatively quickly (<30 minutes). Having motive power for 72 hours provides for significant margin to the need.

After 72 hours, worst case analyses predicted that additional monitoring or containment cooling on its outside may be required. One solution is to rely on support from outside, offsite sources after 3 days. These outside sources could bring additional generators or pumping fire trucks to allow fluid movements outside containment if required. Some, including NRC, concluded based upon the experiences of hurricanes in the United States, that 3 days may not be sufficient time to allow external resources to arrive on site. They also concluded that other potential onsite sources of water may not be available if they were non-seismic. Site fire water is an example of a water source that is assumed not available after 3 days. The NRC then imposed the requirement that although safety related power sources must be available for 72 hours, some sort of on site seismically qualified capability to replenish the passive containment cooling water storage tank (PCCWST) must be available for 7 days assuming loss of all off-site power.

One option to satisfy this requirement is to extend the seismically qualified structures for AP600 off the Auxiliary Building basemat to include the current non-seismic fire protection system. Another is to add a new seismically qualified system to provide makeup water to the PCCWST. It was obvious that the simplest solution is to add a seismically qualified tank and diesel driven pump to the auxiliary building for a 7 day supply of water to the PCCWST. Since this solution is non-safety and to be used after the 3 day passive capability requirement, the use of a diesel powered pump is acceptable.

In summary, the addition of this ancillary equipment was chosen over making the entire fire protection system seismically qualified in a process that promoted satisfying its design requirements with lowest risk for accidental loss of cooling water, high reliance on proven technology, and lowest cost.

6.1.15 In Containment CVS System

One of the functional requirements for the Chemical and Volume Control System (CVS) is to maintain the proper chemistry of reactor coolant water. This includes removal of impurities (both radioactive and nonradioactive) from the reactor coolant system. In operating plants today, this function is performed by taking a portion of the reactor coolant out of containment, reducing its pressure and temperature, purifying it and forcing it back into containment and the reactor coolant system with a high pressure pumping system. This process introduces potential reactor coolant leak sites outside containment, as well as imposing additional reactor coolant inventory control requirements. Since the tradeoffs to simplicity in other portions of the AP600 CVS have been made (see 6.1.1, 6.1.3, 6.1.4 and 6.1.5 above) there is no other reason to continuously pump makeup water into the reactor coolant system. A simple approach to coolant purification was developed that performed continuous purification of a portion of the reactor coolant at reactor coolant pressure using reactor coolant pump head as a motive force and keeping all the purification equipment and reactor coolant within the containment vessel. High pressure water purification using ion exchangers is an industry proven process.
In summary, the incontainment, high pressure coolant purification was chosen over out of containment, pumped, low pressure purification. This created a process that promoted satisfying design requirements with lowest radioactive effluent, lowest risk for accidental loss of coolant, high reliance on proven technology, and lowest risk for public or operator radiation exposure.

6.1.16 Containment Spray from Fire System (NRC demand)

As indicated in Section 6.1.6 above, the AP600 design does not require containment spray for post LOCA isotope migration or removal. This position was endorsed by the United States Advisory Committee on Reactor Safety, but not entirely endorsed by NRC staff. NRC staff agreed that containment spray need not be safety related, but AP600 must be designed to have the ability to provide containment spray as a beyond design basis, manual action. One design solution to this requirement is to include a dedicated containment spray system with its own pumps, valves, water source and containment penetration(s). An alternate, simpler solution is to feed containment spray headers and nozzles from a system in containment that already had pumps, valves, water source and containment penetration(s). The selected system was the incontainment portion of the fire protection system. This provides for the containment spray function without the addition of the equipment and risk associated with a dedicated spray system.

The benefit of this solution is a very safe, simple set of incontainment equipment driven by a required system that is extensively tested and maintained. It provides safety through simplicity by satisfying NRC staff requirements with no potential radioactive effluent, no risk for accidental loss of coolant outside containment, high reliance on proven technology, lowest risk for public or operator radiation exposure and lowest overall plant cost.

6.1.17 Fire System as Alternative to PCS

The Passive Containment Cooling System (PCS) provides evaporative cooling to the outside of containment during accidents that pressurize containment by passively draining water onto it. The system is sized and designed to deliver water onto containment with margin both in delivery time and delivery flowrate over time. In the very unlikely event that PCS would need to deliver water onto containment in excess of its design requirements, defense in depth methods are required. These methods can be non-safety related, but must be designed for seismic Category 1 and not rely on station or offsite ac power. The primary defense in depth system consists of the ancillary PCS water tank and its diesel powered pump. Another alternative was desired.

Since the fire protection system already had a gravity driven, seismic Category 1 standpipe from the PCS tank into the Auxiliary Building (see Section 6.1.13), designers chose to use it as an additional method for replenishing PCS water inventory. Included in the fire protection system is a fire supply hose connection on the outside of the Auxiliary Building in addition to the non-seismic balance of the fire protection system from the fire protection water tanks. This simple use of the existing fire protection connection allows defense in depth capability to replenish PCS water from the fire water storage tanks or from any external source using a pumper truck. This solution was chosen using a process that promoted satisfying design requirements with lowest change to the current design, high reliance on proven technology, and lowest risk for public or operator radiation exposure.

6.1.18 Reduction of Containment Penetrations

Penetrations through the containment are designed to be leak tight assemblies allowing pipes and cables to pass through the leak tight containment vessel boundary. Very often they are the sites of small leak paths. The penetrations themselves and their enclosed piping up to the first isolation valves are safety related and must be periodically inspected and tested. One of the fundamental design objectives for passive cooling of the AP1000 is to isolate containment during a Design Basis Accident with no ac so that only energy passes through the containment boundary, not fluids. This minimizes the number of penetrations and reduces design, inspection and maintenance burdens.

Designers further reduced penetrations by implementation of a variety of innovative techniques. Service systems in containment like component cooling water or compressed air are split and routed inside containment resulting in only one supply or return penetration for each service. Some intermittent services with common fluids share common penetrations. For example both chilled water and hot water heating services to HVAC in containment share common penetrations since they won't be used at the same time. The fire protection water and containment spray supply systems also share a common penetration. Instrumentation and control penetrations are reduced by taking advantage of digital data highway technology. Multiplexing cabinets are located such that instrumentation and control signals share a common highway penetration in lieu of multiple individual signal penetrations.

This solution was chosen using a process that promoted satisfying design requirements with lowest number of containment penetrations, high reliance on proven technology, and lowest risk for containment leakage and public or operator radiation exposure.

6.1.19 Once Through FPS Cooling of RNS Heat Exchanger (diverse cooldown source)

The Normal Residual Heat Removal System (RNS) provides low pressure cooling of the core during shutdown conditions from outside of containment. The system is sized and designed to deliver water and cooling into and out of containment with margin. In the very unlikely event that RNS would need a cooling source other than its design basis and backup source, another defense in depth method is required.

Since the fire protection system already has lots of water at atmospheric temperatures designers chose to use it as an additional method for cooling the RNS. This simple use of a fire protection connection in a once through cooling mode allows defense in depth capability to cool RNS from the fire water storage tanks or from any external source using a pumper truck. This solution was chosen using a process that promoted satisfying design requirements with lowest change to the current design, high reliance on proven technology, and lowest risk for public or operator radiation exposure.

6.1.20 Elimination of Water Sources from Clean Aux (avoid flooding/safety sump pumps)

The AP1000 Auxiliary Building is designed so that on each floor there is a solid concrete wall separating the spaces that are potentially radioactive and those that are not. This mandates the requirement that personnel moving into and out of potentially radioactive spaces must pass through security and health physics. Using this feature, the equipment allowed to be in the "clean" Auxiliary Building (clean Aux) was reduced to the Main Control Room and the rooms housing the plant control and protection hardware and their battery rooms. Since the lowest level of the clean Aux is two floors below grade and since no ac can be used for safety related functions, a solution had to be developed for what to do with any water that might collect in the lowest level of the clean Aux.

The tradeoff here is either establish a large sump and active means for clearing the sump, not using ac, or to eliminate the flooding initiator (water sources) altogether. Except for potable water for the control room, fire protection water for safety related equipment in the clean Aux and pipes carrying water through the clean Aux from containment to the turbine building there should be no water in the clean Aux. The selected solution is to eliminate the flooding initiators and not rely on flooding mitigation features.

Layout design and pipe routing in the clean Aux effectively eliminates water sources from the clean Aux. Potable water is only required in the continually manned Main Control Room spaces. The potable water piping is routed in these spaces so that any leakage will be detected. The pipes are sized so that even in the event of a pipe rupture, the sump system can discharge the leakage with a very large margin. Potable water leaks are not a Design Basis Accident so that ac is available for sump discharging. In addition, potable water in the clean Aux is not a continuously functioning system. There is a potable water day tank above the Main Control Room area that is filled as necessary and potable water is isolated so that leakage is limited to the day tank volume. The available volume of fire protection water in the Aux building is limited by the design solution discussed in Section 6.1.13. Even in the event that ac power is not available and that all the available fire protection water floods the lowest level of the clean Aux, the water level will be below that of the safety related batteries (the lowest safety related equipment in the clean Aux). All pass through piping containing water are routed through two rooms from the containment to the turbine building. Both of these rooms are enclosed in concrete and the only paths for water to escape from them, including through the doors is into drain paths to the turbine building.

This solution was chosen using a process that promoted satisfying design requirements with the simplest possible design goal, eliminate the initiating event. This results in a solution with high reliance on proven technology (gravity drains to out of the clean Aux), elimination of the flooding threat to safety related hardware, thus, lowest risk for public or operator radiation exposure.

6.1.21 Air Diaphragm Waste Pumps (cheap, no NPSH required, operating experience)

Liquid waste water (oily, radioactive, non-radioactive) must be transferred within the plant from tank to tank or for processing and must be transferred out of the plant. In plants today this transfer is powered by a wide variety of pump types (centrifugal, positive displacement, air operated and others). The tradeoff here was to continue with this variety approach or to pick a standard pump type for all AP600 waste pump services. After consideration of the available types the decision was made to use inexpensive, simple, air operated, fully contained pumps for waste water service. In these types of pumps the working fluid remains inside its pressure boundary. This eliminates any chance of seal leakage since there are no seals, especially no rotating seals.

The benefit of this solution is a very safe, simple set of pumps, common for common service. It provides safety through simplicity by satisfying its design requirements with no potential radioactive or oily effluent, no risk for accidental loss of radioactive fluid outside containment, high reliance on proven technology, lowest risk for public or operator radiation exposure and lowest overall plant cost.

6.1.22 High Pressure RNS System (avoid interfacing LOCA)

The NRC staff concluded that the core damage frequency caused by inter-system loss-of-coolant accidents (ISLOCAs) could be substantially greater than previous Probabilistic Risk Assessment (PRA) estimates. An ISLOCA is defined by the NRC as a class of events in which a break occurs outside containment in a system connected to the RCS. This is interpreted as a beyond design basis event. The staff indicated that these earlier PRAs have typically been limited to modeling ISLOCA sequences that include only the catastrophic failures of check valves that isolate the Reactor Coolant System (RCS) from low pressure systems. Also, the PRAs included little consideration of human errors leading to ISLOCA and the effects of the accident-caused harsh environment or flooding on plant equipment and recovery activities. Based on this concern, the traditional design pressure of the Normal Residual Heat Removal System (RNS) was re-evaluated.

The RNS is a nonsafety-related system that provides shutdown cooling for the RCS. During normal shutdown operations, the RCS is cooled and depressurized to the RNS cut-in temperature and pressure. Once RCS pressure has been reduced, the RNS suction line isolation valves are opened, and the RNS pumps are started to provide shutdown cooling. The RNS takes suction from the RCS hot leg and discharges to the reactor vessel through the direct vessel injection lines. The RNS suction contains three normally closed isolation valves in series with a design pressure equal to RCS design pressure. The valves are interlocked so that they cannot be opened unless RCS pressure is reduced to a pressure within the design pressure of the RNS (450 psig). The third normally closed isolation valve is a containment isolation valve and is designed to full RCS pressure. Overpressurization would only occur if either all three motor-operated gate isolation valves leaked excessively, or if the valves were inadvertently opened with the RCS pressure above the design pressure of the low-pressure portion of the RNS.

The second potential overpressurization pathway for the RNS is via the discharge branch lines, which each connect to a DVI line. Each line contains two normally closed check valves that are reactor coolant pressure boundary valves and are designed to the RCS design pressure. The branch line connects to a common header that penetrates containment. The header contains two containment isolation valves. Overpressurization would occur only if three check valves and the motor operated gate isolation valves all leaked excessively.
The portions of the RNS from the RCS to the containment isolation valves outside containment are designed to the operation pressure of the RCS. Traditionally, the portion of the RNS outside containment was designed to 600 psig. In operating plants today, ISLOCAs are discredited based on the suction valves interlock with RCS pressure and the power lock out of these valves at the valve motor control centers. This design provided multiple redundant system isolation and a system design pressure that is 27 percent of RCS design pressure and 150 psig higher than normal RNS operating pressure. NRC guidance has suggested that a design pressure of 40 percent of RNS normal operating pressure and a minimum wall thickness enhances the survivability of piping above 90 percent when pressurized to full RCS pressure. As a result the design pressure of the AP1000 RNS outside of containment was increased to 900 psig to decrease the likelihood of ISLOCAs in the RNS.

In summary, the design pressure of the RNS was increased to 900 psig to reduce the likelihood of an ISLOCA. This increased system design pressure is a safe, simple solution that was extensively studied and satisfies NRC staff requirements that lower risk for public or operator radiation exposure.

6.1.23 Use of Digital I&C

With digital I&C, opportunities exist to incorporate advanced control system concepts, allowing tighter control, online component testing at higher power levels in reduced time duration, improvements in availability achieved through comprehensive system redundancy and elimination of single points of failure, advanced diagnostics, asset management tools, and other functional improvements. Digital I&C systems offer additional opportunities to implement advanced control concepts such as soft control, advanced/integrated alarm systems, and advanced human interface resources. The Westinghouse integrated digital system approach minimizes the number of control room/operator interfaces and platforms that are required, takes advantage of shared system resources, and results in a more highly integrated solution.

The benefits of modern digital I&C systems are numerous. They have reliability and availability improvements through redundancy, advanced diagnostics, and system design. Modern digital I&C also has significantly reduced cost for operation and maintenance, ease of installation, improved process control, enhanced human-machine interface features to reduce operator burden, advanced system capabilities resulting in reduced power deratings, plant trips related to I&C, and opportunities to support power upratings by operating closer to setpoints, as well as asset management capabilities providing significant opportunities to reduce preventative maintenance and transition to predictive maintenance.

Digital I&C using a data highway eliminates large quantities of mechanical I&C components, cabling, cable tray, cable spreading areas, containment penetrations and other equipment. It provides a safe, simple platform for plant protection and control.

In summary, the use of digital I&C has many advantages. The benefits listed above result in a more reliable, efficient, and modern plant.

6.1.24 Design of DAS (PRA based functions)

The protection and safety monitoring system (PMS) is designed to prevent common mode failures. However, in the low probability case where a common mode failure does occur, the diverse actuation system (DAS) provides diverse protection. The DAS is a nonsafety-related system that provides a diverse backup to the protection system.

The DAS is included in the instrumentation and control architecture to support the aggressive reliability goals in the AP600 Probabilistic Risk Assessment for analyzed events. The DAS reduces the probability of a severe accident that potentially results from the unlikely coincidence of postulated transients and postulated common-mode failures in the protection and control systems. Common-mode failure between the PMS (Protection and Safety Monitoring System) and the DAS is unlikely since it has no software and has no shared sensors with the protection system.

The DAS is not required to be safety-related. In addition, it is not required to have redundancy: two out of two voting logic is used to prevent spurious actuation. Although the DAS is a nonsafety system, it is designed to higher quality standards than normal non safety systems. It is designed for simplicity and reliability. It is a nonsafety-related, diverse system that provides an alternate means of initiating reactor trip (both automatic and manual) and selected engineered safety features (again, both automatic and manual). In other words, the DAS serves as a backup to the PMS. In addition, the DAS also provides plant information needed by the operator for a manual actuation of critical safety functions. In summary, the DAS has three functions: diverse automatic actuation, diverse manual actuation, and diverse indication. These functions help to support the PRA reliability goals through the use of a simple, reliable system that is separate and diverse from the PMS and safety systems.

6.1.25 Use of Advanced Control Room

The control room is the main focal point for the safe monitoring and control of both the AP600 and AP1000 plant design. It is required to provide all the facilities that the operations personnel need in order to safely operate and maintain the plant in a safe state, deal with any potential abnormal conditions, and to produce electricity. Over the past few decades, there have been numerous substantial advances in the technology available to provide the operator-machine interfaces in modern control rooms. The success of modern control rooms has been proven in other comparable industries.

The design of the AP600 and the subsequent enhancements in AP1000 take advantage of new operator-interface technology and the resultant main control room (MCR) represents a move away from the traditional 'control board' control room design. The amount of fixed controls and displays has been minimized, to the extent practical. The main operator-machine interface is via computer based monitors, mice and keyboards. The VDU-based operator-interface integrates a number of systems into one flexible interface technology. This includes the use of large screen displays that enable plant overview and alarm status information to be visible from any likely operator location in the MCR, thus facilitating crew group plant status awareness and decision-making.

The current technology has been proven to improve operator performance, increase productivity and reduce the likelihood of human errors using safe, simple technology. Furthermore, it enables the number of operations personnel required to be located in the control room to be decreased and assists in reducing electric generation costs.

6.1.26 Use of Two Equipment Hatches through Containment

Throughout the development of the layout of AP600, significant attention was placed upon establishing open areas and access paths for outage setup and performance. In addition, designers needed to develop a design that could be built and could pass seismic evaluations. This meant that the foundation level of the nuclear island could not be too far below grade. The resulting set of requirements or utility desires created a set of conflicting design objectives. Make the plant as small, seismically robust, accessible and outage friendly as practicably achievable.

The solution was a comprehensive solution incorporating at least partial satisfaction of all the seismic and outage layout design goals. A single basemat elevation for the entire Auxiliary/Shield Building was chosen to ensure maximum basemat reinforcement from building walls and floors. With plant grade being defined as 100', the bottom of the nuclear island basemat was established at 60'6" or 40' below grade. This promoted acceptable seismic results. Floor levels were established to promote adequate overhead space on each floor and reasonable construction access. This placed the open operating deck inside containment at 135'3" and the maintenance deck inside containment at 107'2". To support efficient outage setup and laydown, large open spaces in the Auxiliary and Annex Buildings were required outside the selected equipment hatch locations. To service the operating deck in containment, this open space was placed at the 135'3" elevation with a containment equipment hatch for access. To accommodate the need for access at grade open space is provided at the 107'2" elevation with another equipment hatch. Outside the Annex Building a local, low slope ramp is provided for truck access form grade (elevation 100'). Floor hatches in containment just inside the 107'2" hatch provide access to the operating deck using a crane. The two equipment hatches are at different azimuthal angles and each is paired with a personnel access airlock.

This solution was chosen using a process that promoted satisfying design requirements with lowest overall cost, highest capability to support seismic evaluations and outage setup and access, and lowest risk for public or operator radiation exposure.

6.2 AP1000

The decisions discussed below are examples of design choices made to develop the AP1000 from the AP600 as discussed in Section 5.0 above. They demonstrate that the design process for AP1000 proceeded in a manner that continuously mandated and reinforced the concept of safety through simplicity and as low as reasonably practicable.

6.2.1 Containment Height Increase (plate size, retain AP600 layout)

The move from AP600 to AP1000 required the increase in size of components necessary for power. Inside containment, these components were the reactor vessel, steam generators, and pressurizer. In addition, the increase in power inherently increases the mass and energy releases to containment as a result of a LOCA or main steam line break. As indicated in Section 5 above, design objectives for the move from AP600 to AP1000 included no change to the plant footprint to retain design effort in AP600 for AP1000, and maintain large margins to safety limits.

One of the functions of containment in a passive plant like AP600 or AP1000 is to provide sufficient free volume to accept the mass and energy release from a LOCA or main steam line break without challenging the containment design limits. Not only were the mass and energy releases for AP1000 greater than those for AP600, the limiting event changed from a LOCA to a main steam line break. Making containment larger by increasing its diameter was not an option because this type of redesign affects completed layouts, pipe routings, building structural calculations, building seismic responses, component seismic responses, system flow calculations, accident response calculations, containment free volume, containment floodup volumes and more. So the containment was made taller.

The design decision here is to make it taller as little as possible. A taller containment means a taller shield building forcing the passive containment cooling water storage tank (PCCWST) higher. A higher PCCWST makes satisfactory seismic results for the Auxiliary Building more challenging. Another design constraint (self imposed for simplicity) is to have containment vessel plate thin enough that post weld heat treatment in the field is not required.

Since the diameter is fixed, strength of the containment vessel is determined by material type and thickness. The selection process was first to investigate alternate plate material to maximize vessel strength. This would minimize height by minimizing the volume increase required for the increased mass and energy release. Once plate material and thickness were selected, the height was changed by integral increments of plate width to maximize simplicity of fabrication. Volume increases for each plate width, resulting in accident pressure decreases were compared to containment vessel allowable pressures. The least number of additional plate widths was chosen resulting in increased margin to plate allowables compared to AP600.

In summary, the containment vessel plate material and additional vessel height were chosen a process that promoted satisfying its design requirements with lowest risk for accidental containment breach, high reliance on proven technology (natural forces), and lowest cost.

6.2.2 PXS Line Size Increases (retains safety margin, increase flow, retain layout)

Following a Design Basis Event without ac power available the safety related Passive Core Cooling System (PXS) activates to keep the core shut down and cool. These functions were performed using fluid stored in vessels within containment and natural circulation. For AP600 the analysis was complete for system performance, for pipe size, routing and stresses, and for building structural response. The design challenge was to increase the capacity of PXS while changing as little of the AP600 physical design as possible. Alternatives include adding safety related pumps, increasing the thermal head differences from the core to heat sinks, rerouting pipe to reduce pressure drop or increasing pipe sizes.

Adding safety related pumps would defeat the passive nature of AP600/1000 and was rejected at the outset. Increasing thermal head differences would require a redesign of structures inside containment. This type of redesign affects completed layouts, pipe routings, building structural calculations, building seismic responses, component seismic responses, system flow calculations, accident response calculations, containment free volume, containment floodup volumes and more. Rerouting piping for pressure reduction would yield very little since the piping was already routed for minimum resistance while maintaining structural adequacy. The remaining alternative, increasing pipe sizes was selected because it has the lowest impact on the completed design as practicable.

Rough calculations were performed to determine a goal pipe size. Then the next larger standard pipe size was selected and placed on the same centerlines as for AP600. This approach created pipe routings that had margin in the pressure drop due to slightly larger pipe than required and that could probably pass structural evaluations since the smaller sizes already passed. These assumptions proved valid and the PXS design was resized for AP1000 with as little extra impact on the fully certified AP600 design as reasonably practicable.

6.2.3 CMT Increase – (reduce injection gap for small break LOCA space constrains)

Following a Design Basis Event without ac power available the safety related Passive Core Cooling System (PXS) activates to keep the core shut down and cool. Part of these functions are performed using fluid stored within Core makeup Tanks (CMT) within containment by natural circulation. For AP600 the analysis was complete for system performance, for tank size, timing and stresses, and for building structural response. The design challenge was to increase the capacity of the CMTs while changing as little of the AP600 physical design as possible.

Rough calculations were performed to determine a goal tank size. Then the largest tank size that could fit into the rooms assigned to the CMTs without changing the incontainment structural layout was selected and placed on the same centerlines as for AP600. This approach created CMT volumes that still had margin for small break LOCAs while maintaining and incontainment structural, floodup and high head injection capability that had already passed minimum criteria. The CMTs were resized for AP1000 with as little extra impact on the fully certified AP600 design as reasonably practicable.

6.2.4 PRHR Heat Exchanger – (meet NRC/URD cooldown time limit)

Following a Design Basis Event without ac power available, the safety related Passive Core Cooling System (PXS) activates to keep the core shut down and cool. Part of these functions are performed using heat transfer from the passive residual heat removal heat exchanger (PRHR HX) within containment by natural circulation. For AP600 the analysis was complete for system performance, for heat exchanger size, timing and stresses, and for building structural response. The design challenge was to increase the capacity of the PRHR HX while changing as little of the AP600 physical design as possible.

Rough calculations were performed to determine a goal heat exchanger size. Additional tubes were added to the AP600 PRHR HX sufficient to satisfy the NRC and the URD requirements for cooldown time. Attachment details of the PRHR HX to the incontainment refueling water storage tank (IRWST) were developed without changing the incontainment structural layout and the PRHR HX was placed on the same centerlines as for AP600. This approach created a PRHR HX that still had margin for AP1000 design basis events while maintaining and incontainment structural, and IRWST volume that had already passed minimum criteria. The PRHR HX was resized for AP1000 with as little extra impact on the fully certified AP600 design as reasonably practicable.

6.2.5 ADS 4 Increase More Than Proportional

Automatic Depressurization System (ADS) piping and valves had to increase in their size to reflect the higher power and reactor coolant volume of AP1000 over AP600. This increase had to ensure that the DCD Chapter 15 Design Basis Accident Analysis was satisfied without question. Preliminary calculations indicated a required pipe size for ADS 4 piping that fell between standard pipe sizes. The next larger pipe size (14 inches) was chosen and the pipe centerlines were not changed to simplify incorporation of the larger pipe into the layout. This resulted in an ADS 4 capability that increased more an increase proportional to power. In summary, ADS Stage 4 piping size increased from 10 inches to 14 inches with as little extra impact on the fully certified AP600 design as reasonably practicable and with additional capability for dealing with design basis accidents.

6.2.6 ADS 1,2,3 No Increase

Automatic Depressurization System (ADS) Stage 1, 2, 3 valves were sized for AP600 to provide satisfactory pressure relief capability to ensure long term core protection without use of ADS Stage 4 valves. The ADS 1, 2, 3 valves would reduce pressure to below 100 PSI to allow starting of the RNS system. Long term core make up and cooling would continue even if Stage 4 ADS was disabled for some reason. Early PRA analysis conducted for AP1000 showed that AP600 ADS stage 1, 2, and 3 were sufficiently sized so that no additional size increase was required. CMT capacity was increased some to allow for additional make up until system pressure was reduced below required operation levels or actuation setpoints.

In summary, ADS pipe sizes remained at 4 inches for Stage 1, and 8 inches for Stages 2 and 3 between AP600 and AP1000. This selection process maintained safety while imposing as little impact on the already completed AP600 detail design a reasonably practicable.

6.2.7 IRWST Ultrasonic Level (increase injection head)

It is important to be able to measure Incontainment Refueling Water Storage Tank (IRWST) level during normal operating conditions. The previous instrumentation used for the IRWST level measurement was wide-range DP level sensors. This sensor application results in relatively large errors in measuring the normal water level (when the tank is full). This approach was adequate for AP600. However, for AP1000 it was desirable to increase the post LOCA containment flood up level in order to maintain / increase the long term core cooling safety margins.

One of the changes made to the AP1000 to accomplish this was to increase the normal water level in the IRWST without changing structures within containment. In order to maintain the operating margin in the tank, a more accurate narrow range ultrasonic sensor was added. It is possible to use this type of sensor for monitoring normal water level because it does not have to function in the post accident environment. In the post accident environment, the safety related wide range level sensors are sufficient to monitor the drain down of the IRWST.

In summary, the ultrasonic level sensor is a simple device that is wall-mounted inside the IRWST, above the maximum water level. By adding this narrow range instrumentation, much of the error is eliminated, which allows the normal water level in the IRWST to be raised while maintaining the previous operating margin. This allows for increased water volume capacity, and thus, increased flood up level post LOCA.

6.2.8 Spent Fuel Sprays/Water Tight Compartments (reduce fuel pool vulnerability)

The design of the spent fuel pool cooling system (SFS) for AP1000 utilizes safety-related passive means to provide long term pool cooling in the event that normal SFS cooling is lost. Beyond design basis accidents have been postulated that could potentially drain the entire contents of the spent fuel pool. An event of this nature could lead to the overheating of freshly discharged spent fuel sufficiently that the zirconium cladding could ignite. The resulting fire from such an event could release significant radiation.

A potential solution to provide beyond design basis cooling would be to equip the area surrounding the spent fuel pool with hose stations capable of spraying the pool to provide continued cooling. However, even in a beyond design basis scenario, this method would be difficult to quantify and could potential place personnel at high exposure risks. A simple alternative solution was developed to enhance design features to help ensure that the spent fuel would not become uncovered if the pool were to drain into either of the rooms located below. The bottom elevation of the AP1000 spent fuel pool is 92.7' with a 3.2' thick concrete base. Located below the spent fuel pool are two separate waste holdup tank rooms. The waste holdup tank rooms were made leak tight by placing water tight doors on the room. Analysis shows that if the pool drained into only one of the waste holdup tank rooms the water level in the spent fuel would not drain below the top of the spent fuel racks. Additionally, a redundant spray system has been embedded in the East and West walls of the spent fuel pool that each are capable of providing emergency cooling to the pool should it drain. The cooling water will be delivered to the spray nozzles from either the PCCWST using gravity driven flow or the FPS using either the motor driven or diesel driven fire pumps.

This solution was chosen using a process that promoted satisfying design requirements with the simplest possible design goal: eliminate the initiating event by stopping the spent fuel pool water level from draining below the top of spent fuel. This results in a solution with high reliance on proven technology (water tight doors and gravity spray flow), which decreases the likelihood that the zirconium cladding could ignite, thus, lowering the risk for public or operator radiation exposure.

6.2.9 Move DAS – (mitigate consequence of large fire)

The four divisions (A, B, C, or D) of the Protection and Safety Monitoring System (PMS) and Class 1E DC and Uninterruptible Power Supply System (IDS), electrical containment penetrations, portions of Plant Control System (PLS) instrumentation and control, the main control room, and the remote shutdown workstation are concentrated within the northern portion of the auxiliary building. A beyond design basis large fire or explosion affecting the northern section of the auxiliary building could render these features unusable for a significant amount of time. Such an event would most likely result in an immediate plant scram due to the loss of power and control functions; however, most of the plant instrumentation and control (I&C) would be unavailable.

By relocating selected portions of the AP1000 Diverse Actuation System (DAS) to the southern portion of the auxiliary building, the plant I&C capabilities are decentralized. Thus, there is an increased probability that at least the relocated DAS I&C capability would remain intact after a beyond design basis large fire or explosion in the northern portion of the auxiliary building. The relocated DAS control cabinet penetrates containment through a separate containment penetration in the southern portion of the auxiliary building. An internal battery-backed uninterruptible power supply is included within this cabinet so that DAS instrumentation can be accessed from this location without the need for an external power source.

This solution was chosen using a process that promoted satisfying design requirements with the simplest possible design goal: eliminate the initiating event by increasing separation of the control system. This resulted in a solution with high reliance on proven technology, which decreases the likelihood that a beyond design basis large fire or explosion could debilitate the plant control system, thus, lowering the risk for public or operator radiation exposure.

6.2.10 Single Failure Cask Loading Crane (international requirements)

The cask loading crane is a crane in the fuel handling portion of the Auxiliary Building that handles fuel assemblies into the out of plant fuel storage and transportation cask. Various analyses were performed for AP600 to establish public safety in the event of a dropped fuel assembly from this crane. As a result, there was no requirement in the United States for this crane to be single failure proof for AP1000.

It was discovered that for some non-US countries this type of crane was required to be single failure proof by rule. Although there was some impact on building arrangement and the cost of the crane, it was decided to specify that this crane be single failure proof for all AP1000s, regardless of country of deployment. This makes the crane as safe as reasonably practicable with as low an impact on plant design and deployment as reasonably practicable. This approach also promotes standardization of the AP1000 design.

6.2.11 Zinc Addition (reduce fuel corrosion)

Chemical build up in the Reactor Coolant System (RCS) has the potential to cause water stress corrosion cracking (PWSCC) and Crud Induced Power Shift (CIPS). The design of AP1000's Chemical and Volume Control System (CVS) incorporates a zinc addition subsystem. Operation with chemical zinc in the coolant has been demonstrated to change the oxide film on primary piping and components which significantly reduces the potential for PWSCC and CIPS. Zinc addition has also been found to significantly reduce occupational radiation exposure by as much as 50% when incorporated as early as hot functional testing.

Zinc addition to the RCS has been implemented at numerous Pressurized Water Reactors to date. Zinc concentrations ranging from 5 to 40 parts per billion (ppb) in the RCS have been used for the purposes of reducing the potential for CIPS and PWSCC and lowering occupational radiation exposure. The reactor coolant water chemistry specifications for AP1000 specify a maximum zinc concentration of 40 ppb in order to maximize benefits associated with zinc addition.

This solution was chosen using a process that promoted satisfying design requirements with the simplest possible design goal, reducing the likelihood of PWSCC or CIPS. This results in a solution with high reliance on proven technology (a zinc injection skid) that reduces occupational radiation, thus, lowering the risk for public or operator radiation exposure.

6.2.12 AC Power Fast Bus Transfer (avoid trip on generator bus failure)

Previously in AP1000 design, single contingency loss of any one of five large oil-filled transformers, loss of 26kV isophase bus duct, or associated protective relay malfunction for this electrical equipment would have caused total loss of plant AC power for up to approximately 2 minutes until Onsite Standby Diesel Generators started, warmed up, and were loaded or plant operators performed manual transfer of selected 6.9 kV busses to the Reserve Auxiliary Transformer (RAT). This condition would result in a reactor trip due to a loss of all 4 Reactor Coolant Pumps (RCPs) powered from the Unit Auxiliary Transformers (UAT) buses. To prevent a reactor trip from the electrical faults mentioned above, fast bus transfer capability was incorporated into the AP1000 design.

Adding the fast bus transfer capability to the AP1000 design required an additional Reserve Auxiliary Transformer (RAT) to allow complete bus transfer from Unit Auxiliary Transformers to Reserve Auxiliary Transformers. The Reserve Auxiliary Transformers and Unit Auxiliary Transformers are of identical capacity rating, secondary voltage rating and impedance. Both Reserve Auxiliary Transformers are powered from the maintenance power source, which is site specific.

In summary, the addition of another Reserve Auxiliary Transformer was chosen to allow for the complete bus transfer from UATs to RATs. This resulted in a simple, safe fast bus transfer capability which prevents a reactor trip in case of any of the electrical faults mentioned above.

6.2.13 IVR Design Improvements (IVR performance and testing)

The additional power in AP1000 from AP600 increased the severe accident (core melt) demands on the In Vessel Retention design solution. See Section 6.1.7. To maintain the passive response to severe accidents, the AP1000 IVR design required modification. The result of the testing performed for AP600 to establish the thermal hydraulic parameters associated with core melt did not bound the calculated parameters for AP1000. The University of California, Santa Barbara performed additional testing to expand their results to envelop AP1000 parameters. These new results required additional structure and a shaped internal boundary for the reactor vessel insulation design. In addition, the insulation design had to be able to pass additional fluid and energy flows required by the change from AP600 to AP1000.

The reactor vessel insulation design was modified only as necessary to incorporate the additional structure and the internal shaping required. The water inlet devices and steam outlet devices were modified to increase their flow areas, make their operation by natural forces simpler and more reliable, to create a new flow path to provide additional shielding during normal operations and to make the design easier to fabricate and erect in the field.

This solution was chosen using a process that promoted satisfying design requirements with lowest change to the current design, high reliance on proven technology (natural forces), and lowest risk for public or operator radiation exposure.

6.2.14 Shield Building Air Inlet (airplane crash, reduce sky shine)

The passive containment cooling system (PCS) is a safety-related system which is capable of transferring heat directly from the steel containment vessel to the environment. The transfer of heat prevents the containment from exceeding its design pressure and temperature. One feature of the PCS is air inlets located near the top shield building. The air inlets provide a pathway for air cooling of containment following an accident. It is prudent that the design of the air inlets be structurally robust since they are part of the shield building. It is necessary that the design of AP1000 take into account the potential effects of the impact of a large commercial aircraft. The design of the shield building should incorporate design features that provide additional inherent protection to withstand the effects of a beyond design basis aircraft impact.

The design of AP600 air inlets consist of 15 large discrete openings in the top of the shield building. The openings penetrate the three foot thick reinforced concrete shield building. The design of the air inlets are sufficient to aid containment cooling such that the peak containment pressure does not surpass containment design pressure during a design basis accident. Air inlets of this type are also sufficient to support containment cooling for AP1000. However, it has been questioned whether this design is robust enough to withstand an aircraft impact. Additionally, the large openings provide a pathway for debris or fuel from an aircraft impact to penetrate containment potentially resulting in a fire that might impact the steel containment vessel.

An alternative design was explored for AP1000 which maintained the original air inlet cooling capabilities and strengthened the building's ability to withstand a beyond design basis large aircraft impact. One solution would be to increase the thickness of the shield building at the air inlet elevation. Impact testing has also shown that high strength concrete contained within steel liner plates on both faces significantly increase impact resistance. Analysis showed that modifying the portion of the shield building containing the air inlets to 4.5 foot thick high strength concrete contained within steel liners on both faces significantly increased that portion of the shield building's ability to withstand a beyond design basis aircraft impact.

The air inlet portion of the shield building could be further enhanced by reducing the size of the air inlets to restrict debris or fuel from entering the building. Containment cooling requires a minimum inlet area to provide adequate air cooling for the containment vessel. However, each air inlet does not have to be as large as 15 air inlets for AP600 as long as the total required area is still met. The flow area could be divided amongst many smaller air inlets. The design was optimized to include 384 small inlet ducts to take the place of the 15 large discrete openings. The smaller inlets consist of square steel tubes inclined upward form outside face to inside face. The new air inlets present no significant change to the design basis pressure response for cases when PCS operates or the beyond design basis air only cooling assumed by PRA. The redesigned air inlets also provide a significant increase in the shield building's resistance to restrict debris or fuel from entering the building due to their small size and orientation. This design provides an addition safety benefit by reducing radiation sky shine.

In summary, the redesign of the air inlets provide inherent protection against aircraft impact while maintaining its design functions associated with provided passive containment cooling.

6.2.15 Shield Building Structure (airplane crash)

In response to the September 11, 2001 attack on the United States by terrorists using commercial aircraft, the NRC developed a proposed Rule to require that new nuclear power plants be evaluated against the unlikely event of a targeted crash of a large commercial aircraft. This evaluation would be considered beyond the design base and it, as well as its input parameters and acceptance criteria would be considered Safeguards Information (SGI). Because of timing, his new requirement for assessment will not apply to AP1000. AP1000 has voluntarily performed an assessment in accordance with the proposed rule anyway. The AP1000 design objective is to acceptably withstand an airplane crash in lieu of adding on mitigation measures. For AP1000 this required a change from the shield building design of AP600. The AP1000 seismic acceptability analysis was already complete and seismic input curves were already in use for analysis of safety related equipment.

The tradeoff here is to add reinforced concrete thickness to withstand the airplane and rework all seismic analyses completed to date or to develop some other design solution that does not impact the completed seismic analysis results and equipment inputs. The second path was chosen. The construction techniques for the shield building was changed from reinforced concrete to a plate and concrete sandwich structure similar to that used inside the Auxiliary Building and already approved by NRC. Details of the impact analysis are classified SGI, but results are acceptable and the seismic design of AP1000 is essentially unchanged.

This solution was chosen using a process that promoted satisfying design requirements with lowest change to the current design, high reliance on proven technology, and lowest risk for public or operator radiation exposure.

6.2.16 High Density Fuel Racks (optimize spent fuel storage)

The AP600 spent fuel pool had the capacity to store 619 spent fuel assemblies. It was desired to increase the storage capacity of the AP1000 spent fuel pool while maintaining the same safety basis for pool makeup. The AP600 and AP1000 spent fuel pool cooling systems are designed to provide means of cooling for the spent fuel pool passively for 72 hours and then for the balance of 7 days using on site sources if normal forced flow cooling is lost.

Increasing the size of the spent fuel pool would create additional room to add storage spaces. However, this was not an option since one of the design objectives for the move from AP600 to AP1000 included no change to the nuclear island footprint and building design. Changing the building design would necessitate a change to the seismic design that was substantially complete. Alternatively, the design of the spent fuel pool storage racks could be changed to create additional storage cells without altering the dimensions of the spent fuel pool. The AP600 spent fuel storage racks consisted of only Region 1 racks with a 10.9 inch center to center spacing. Using high density storage racks that contain a combination of Region 1 and Region 2 racks allow for increased storage capability within the same footprint. The center to center spacing of the Region 2 racks is 9.028 inches. The total spent fuel pool capacity was capable of being increased to 889 spent fuel assemblies using a combination of Region 1 and Region 2 storage racks. The increased storage capacity of the spent fuel pool increased the maximum decay heat in the pool. The existing makeup water sources were found to be adequate to provide safety related cooling to the pool in the event forced flow cooling would be lost with the new higher heat loads.

In summary, the capacity of the spent fuel pool was increased from 619 storage spaces to 889 storages spaces while maintaining the same pool footprint by switching to high density storage racks and maintaining the same safety related makeup sources.

6.2.17 Diversity in Squib Valve Design (PRA)

The squib valve provides diversity against common failure mode by using three different valve designs: one for Stage 4 ADS valves, a second high-pressure valve design for the IRWST injection line and check valve recirculation line squib valves, and a third low-pressure recirculation line squib valve for the lines with the normally-open motor-operated isolation valves in series with the squib valves.

By employing three different valve designs, common mode failure within the squib valves themselves is minimized. The squib valves are significantly more reliable than conventional valves due to the relative simplicity of the design and the very high reliability of the igniters and explosive charges.

Design diversity is achieved through differences in the design details of the key valve actuation components, which requires differences in their physical configurations (and design tolerances) for the following:

- Valve body (inside surface forms shearing piston walls)
- Valve bonnet and retaining hardware (cylinder head which also houses the propellant cartridge)
- Propellant cartridges (volume/arrangement excluding propellant material/igniters)
- Actuation plug (shearing piston)
- Actuation plug piston tensioning (and shearing) bolt
- Shear caps (shearing wall thickness which is pressure dependent)
- Valve latching mechanism (hold shearing piston in place after actuation)
- Metal foam (compression dampening upon actuation)
- Metal foam retainer plate and retaining hardware
- Various valve body bolts and compression chamber metal o-rings

In summary, design variations and design tolerances between the various designs provide adequate design diversity to protect against squib valve common failure modes.

6.2.18 Control Room Added Operator Panel (Human Factors Assessment)

The AP1000 Main Control Room (MCR) is, in essence, an evolution of the AP600 design. Similar to AP600, the MCR takes full advantage of the latest control room operator-interface technology. A comprehensive detailed human factors engineering program supports the development of the MCR and operator-interface design. This program includes task analysis, operating experience reviews, engineering tests, the application of human factors design guidelines and verification and validation assessments.

The overall purpose of the MCR is to provide a seismically qualified, habitable, good and comfortable environment for the operators and supervisors to safely, efficiently, and reliably monitor and control plant process during normal, abnormal, and accident conditions. The MCR provides an area that enables the operations personnel to focus their attention on the safe and efficient operation of the plant. This is especially important in potential abnormal or emergency conditions. It supports good operator performance by supplying the facilities for the MCR operators to interact with other plant personnel, while preventing distractions by non-operations personnel. It provides a facility that supports the intended 'concept of operations' in terms of supporting the operations personnel in the effective and timely execution of their assigned tasks and responsibilities. Alarms are provided to draw the operator's attention to key indications that may require operator action; displays are provided to enable the operators to determine the plant status; and control facilities are provided to allow the operators to execute control actions.

The MCR accommodates an operator console, supervisor's console, safety consoles, the Wall Panel Information System (WPIS) large screen displays and the DAS panel. The operator console provides the displays and controls to start up, maneuver, and shut down the plant, and is designed to be staffed by one to six operators. The operator-interfaces are the non-safety control system displays, soft controls, alarm presentation system displays, computerized procedures displays, as well as the VDU monitors, keyboards, and mice. The supervisor's console is a smaller version of the operators' console, and designed to be staffed by one or two personnel. The primary dedicated safety panel and VDU-based safety system workstations are located at the center of the operator console, with a secondary safety panel located in close proximity to the supervisor's console. The DAS panel is located at a side wall in the MCR. The MCR also includes communication devices, document lay down areas, printers and storage space. A meeting table is provided and equipped with a VDU-based workstation to allow access to the non-safety control system by, for example, a technical advisor or shift manager, without disrupting control room operations. In close proximity to the MCR are the shift supervisor's office, the operations staff area, an operations work area, restrooms, and kitchen facilities.

In summary, the major benefit of the design of the MCR is that it provides a focal point for all AP1000 operations. From a human factor, operations and safety perspective, the integrated design of the operator-interfaces has many advantages in terms of successful operator performance in both normal and abnormal or fault conditions.

7.0 Conclusion

The driving forces behind the design of the AP1000 are "Safe" and "Simple". By adhering to these two principles, the AP1000 design also resulted in being "Economical". To be clear - economics did not drive the design, rather the principles of safety and simplicity drove the design.

To maximize the benefits of safety and simplicity, these two principles were, and had to be, taken together - not separately. Inevitably, tradeoffs were considered as described in this document. From the outset of the AP1000, careful consideration was given to tradeoffs and alternative design features, but neither one of the principles of safety or simplicity dominated to the detriment of the other. And, importantly, Westinghouse learned during the course of the many years of design, analysis, and testing of the AP1000, that these principles were neither in competition with nor subservient to one another. In fact, Westinghouse learned that maintaining a balance between safety and simplicity had a synergistic, if not symbolic, effect. For example, if a new reactor design were being developed, from a clean slate, the designer would ensure safety through components, systems, training, etc. It is easy to observe that adding safety components and systems can add to the overall safety of the design. However, as these features are added, more complexity must be taken into consideration. The more complex, the greater chance of some failure of a device or system to operate properly. Thus adding safety components and systems can increase safety, but in the extreme, increases safety only in small increments at best, because with more complexity comes a greater likelihood of mal-performance of integrated systems that interact with one another. The AP1000 is unique in the elegance of its simpler safety systems that actually scored higher in reliability with the US NRC than today's already safe operating nuclear plants.

A corollary to the principles of "Safe" and "Simple" was the use of proven components. The development of a new reactor design carries with it the question of "will it work as designed?" Westinghouse recognized the innovative nature of the safety systems. Westinghouse did not want to also develop new components as part of the process. The combination of innovative systems and the need to develop first of a kind components would detract from likelihood of success for this design. Therefore, all of the components chosen were those that had already been developed and had a history of successful operation, though not necessarily in the nuclear industry. This decision and direction were taken at the outset of the design and checked periodically during the course of the design. Ultimately, as the AP1000 design matured, the proveness of components, and the rejection of new component development, provided a robust underpinning for the reliability of the AP1000.

A key example of the tradeoffs for reliability considered for proven components is that of the reactor coolant pump. Conventional PWR's employ a shaft seal reactor coolant pump, a proven device with very high pump efficiency. Westinghouse asked itself - "Can we do better?" This led to considerations of a canned motor pump, used in navy applications. The canned motor pump was also a proven component, but its efficiency is lower than that of the shaft seal pump. Recognizing that electric utilities do not limit their objective to generating simply megawatts, but rather have as their objective to generate megawatt-hours, resulted in a study of pump reliability. The conclusion was that what the canned motor pump sacrificed in efficiency was more than compensated for by its superior reliability compared to the shaft seal pump, as well as protecting against excessive leakage of reactor coolant since shaft seals were eliminated. In the longer run, the canned motor pump would result in a more robust design, generating more megawatt-hours than its counterpart, the shaft seal pump.

Westinghouse fully recognized the developmental nature of the safety systems. These were an elegant combination of proven components, resulting in simple, robust systems employing natural forces described in this document. To demonstrate the reliable operation of the systems, Westinghouse not only performed extensive and rigorous analyses, but also embarked upon an extensive testing program. Westinghouse performed tests at multiple facilities in different locations - tests that overlapped so that Westinghouse could demonstrate that different test facilities delivered the same result - which the safety systems performed as predicted by sophisticated computer programs. The US NRC had their own independent testing performed in Japan. Once again, results were consistent with previously run Westinghouse tests, and underscored the fact that passive safety systems were robust, operated predictably, and were orders of magnitude safer than today's operating nuclear units. The US NRC hailed the development of the AP1000 (and its predecessor, the AP600) as the most thoroughly tested design that they had reviewed and licensed.

As noted in Section 4.0, in the mid 1980's, U.S. utilities and EPRI worked together to generate a set of requirements for the next generation of nuclear reactor designs. This effort resulted in the issuance of the ALWR URD. These requirements not only utilized operator experience, but also insisted on designs that were robust in both safety and operation. The requirements for safety and operational margins were implemented throughout the AP600 and AP1000 selection of components and design of systems. These are described at length in Section 6.0, along with alternatives considered. In all cases, the route of robustness was selected. Not only was this in keeping with the letter of requirements of the URD, but was also a key dimension of Westinghouse's philosophy of designing an innovative reactor that would be a highly reliable, safe, and simple design.

Lastly, the principles of simplicity were extended to the constructability of the AP1000. Constructability was part of the design process from day one. Nuclear utility personnel and plant constructors were invited to review, in detail, the design and layout ensuring attention to both sound and predictable construction processes/schedules during the process itself, not as an afterthought. To improve the quality of construction, the AP1000 was designed such that it could be largely constructed in a modular fashion in offsite factories. Factory fabricated modules enjoy the benefits of better quality control and ease of standardization. The finished modules are shipped to the site via rail and/or barge, and provide for a more predictable site construction process than a "stick built" plant.

Employing the principles of safety and simplicity, and the corollary of component proveness, the AP1000 is a prime example of a design and design process that continually addressed and adhered to the principles of as-low-as-reasonably-practicable throughout. The safety of the AP1000 was confirmed through the granting of a Design Certification by the US NRC. It joins the Westinghouse AP600 as the only two Generation III+ designs granted Design Certification in the U.S.

ATTACHMENT 1

LISTS OF AP600 AND AP1000 DCPs

AP600 Design Change Packages

AP600 Document	Rev.	Title
GW-GEE-001	0	Revised AP600 Plant Parameters Changing Power Rating and Steam Generator Design
GW-GEE-002	0	Split up of Spent Fuel Cooling System
GW-GEE-002	1	Split of AP600 SFCS into Independent Systems of Spent Fuel Pit Cooling and Normal Residual Heat Removal
GW-GEE-003	0	AP600 Fuel Assembly Design Changing Fuel Assembly from 17x17 OFA to Modified 17x17 (V5-H) with IFAs
GW-GEE-004	1	Reactor Internals Change-Longer Fuel Assembly Added CRDMs
GW-GEE-005	0	Vessel Head to 3-Loop Hemispherical
GW-GEE-006	1	Offset Cold Leg 17.5" Above Hot Leg on Reactor Vessel
GW-GEE-007	0	Fuel Assembly Length Increase
GW-GEE-008	0	RC Pump Reference Casing Material and Configuration Change
GW-GEE-009	0	Shield Building Roof and PCCS Tank Configuration
GW-GEE-010	0	Automatic Depressurization Sparger and Discharge Lines
GW-GEE-011	0	Containment Vessel Diameter
GW-GEE-012	0	Two On-Site Standby Power Sources
GW-GEE-013	0	Component Cooling Water System Modifications
GW-GEE-015	0	Hydroball ICIS to Top Mounted Fixed Incore
GW-GEE-016	0	Spring Mounted Turbine Support
GW-GEE-017	0	WGS System Modification
GW-GEE-018	0	Reactor Vessel Lower Plenum
GW-GEE-019	1	PRHR Heat Exchanger Change
GW-GEE-020	0	AP600 Nuclear Island General Arrangement Revision 0
GW-GEE-021	1	AP600 Primary Sampling System (PSS)
GW-GEE-022	0	Conceptual Design of Liquid Radwaste System
GW-GEE-023	0	Seismic Response Spectra
GW-GEE-024	0	Material for Refueling Canal and IRWST
GW-GEE-025	0	Stainless Steel Liners to Ease Cleanup
GW-GEE-026	0	Reactor Coolant System (RCS) Changes for P&ID Revision 5
GW-GEE-027	0	Reduction in the Number of Main Steam Safety Valves
GW-GEE-028	0	Elimination of Operating Basis Earthquake (OBE) as a Design Requirement
GW-GEE-029	1	AP600 Reactor Coolant Pump and Steam Generator Channel Head Changes
GW-GEE-030	0	Additional RCS Changes for P&ID Revision 5
GW-GEE-031	0	Chemical & Volume Control System Modifications
GW-GEE-032	0	Normal RHR System Modification

AP600 Document	Rev.	Title
GW-GEE-033	0	Octagonal Flatsided Interior for Primary Shield
GW-GEE-034	0	Shield Building Roof Line Modification
GW-GEE-036	0	Containment Vessel Embedment
GW-GEE-038	0	External Recombiner System
GW-GEE-039	0	DELTA 75 Steam Generator
GW-GEE-040	0	AP600 Systems List Revision
GW-GEE-041	0	Nuclear Island General Arrangements
GW-GEE-042	0	Chemical & Volume Control System (CVS) Modifications with Regard to Inadvertent Boron Dilution Accidents
GW-GEE-043	0	Pressurizer Spray Block Valve Addition
GW-GEE-044	0	Revised AP600 Plant Parameters Revision 2
GW-GEE-045	0	PXS Design Changes
GW-GEE-046	0	Cooling Water Systems (RNS & CCS) Resizing Due to Increased SW Temperature
GW-GEE-048	0	SPS Water Storage
GW-GEE-049	1	PCCS/FPSI
GW-GEE-050	0	Pressurizer Upper Support
GW-GEE-051	0	Control Room Ceiling
GW-GEE-052	0	Water Distribution Weir System for the Containment Vessel
GW-GEE-053	0	Internal Recombiner
GW-GEE-054	0	Changes to AMSAC to Increase Reliability and Diversity
GW-GEE-055	0	SGS Isolation Provisions
GW-GEE-056	0	PXS Design Change for CMT
GW-GEE-058	0	Optimizing Diesel Generator Loading/Design Change to Diesel Generator Backed Pressurizer Heater Banks
GW-GEE-059	0	Cooling Water Source for Compressed Air System Equipment/Component Cooling Water System as the Cooling Water Source
GW-GEE-060	0	Containment Spray System Removal
GW-GEE-061	0	Implementation of Non-Safety Hydrogen Igniters
GW-GEE-062	0	Revision to Tornado Design Parameters
GW-GEE-063	0	Air Temperature Site Interface
GW-GEE-064	0	Nuclear Island General Arrangements Revision 2
GW-GEE-065	0	Spent Fuel Handling System
GW-GEE-066	0	SFS Spent Fuel Pump Available NPSH
GW-GEE-067	0	Chemical & Volume Control System (CVS) Modifications for Quality Group D
GW-GEE-068	0	72-Hour Battery Configuration for AP600
GW-GEE-069	0	CVCS Shutdown Purification Line
GW-GEE-070	0	Chemical & Volume Control System (CVS) Modifications to Revise Boric Acid Tank Capacity & Makeup Pump Suction Line Size
GW-GEE-071	0	Increase in Containment External Design Pressure
GW-GEE-072	0	Containment Radiation Signal Modification
GW-GEE-073	1	SGS Configuration Revisions (Feedwater Material,Valves and SG taps)

AP600 Document	Rev.	Title
GW-GEE-074	1	Seismic Design Criteria
GW-GEE-075	1	Piping Standardization
GW-GEE-075	2	Piping Standardization
GW-GEE-076	0	Microprocessor Based Rod Position Indication System
GW-GEE-077	1	IHP Head and Cable Bridge Maintenance Modifications
GW-GEE-078	0	Reactor Coolant System Changes and Clarifications
GW-GEE-079	0	AP600 Main Control Room (MCR) Vestibule Volume
GW-GEE-080	0	P&ID VFS M6001/002 Revision 2 Changes
GW-GEE-081	0	SGS/MSS Pressure Category Reduction
GW-GEE-082	0	VBS P&ID Changes
GW-GEE-083	0	Due to clerical error, this change control number will remain in the system and will reference DCP 93-84/0.
GW-GEE-084	0	Nuclear Island General Arrangement Drawings, Revision 4
GW-GEE-085	1	Steam Generator Head Layout Change
GW-GEE-086	0	VWS P&ID Changes
GW-GEE-087	0	P&IDs FPS-M6-001, 002, 004, 005 Revision 2 and FPS-M6-003, 006 Revision 1 Changes
GW-GEE-088	0	P&IDs WRS-M6-001 through -003, Rev. 1 Changes
GW-GEE-089	0	System Specification Document for Class 1E DC and UPS System IDS-E8-001, Rev. 1
GW-GEE-090	0	System Specification Document for Main AC Power System, ECS-E8-001, Rev. 1
GW-GEE-091	0	AC Power System, Station One Line Diagram ECS-E3-001, Rev. 1
GW-GEE-092	0	One Line Diagrams for Class 1E DC and UPS System, IDS-E3-001, 2, and 3 Rev. 1
GW-GEE-093	0	AP600 Electrical Systems Design Criteria GW-E1-001, Rev. 1
GW-GEE-095	0	Radiological Zones-Normal Operation/Shutdown, 1010-N5-001through 1070-N5-001 Revision 4 Changes
GW-GEE-096	0	Radiological Zones-Post Accident 1010-N5-101 through 1070-N5-101 Revision 3 Changes
GW-GEE-098	0	Piping Class Sheets Specification PL02-ZO-001 Rev. 1
GW-GEE-099	0	AC Power System, Station One Line Diagram ECS-E3-002, Rev. 1
GW-GEE-100	0	Steam Generator System Piping Material Change to SA333 Grade 6
GW-GEE-102	0	Nuclear Island Fire Area Drawing Revision 1 Changes
GW-GEE-104	0	Steam Generator Secondary Side Dimensional Changes
GW-GEE-105	0	Steam Generator Support Column Base Change
GW-GEE-106	1	Nuclear Island General Arrangement Drawings, Rev. 5
GW-GEE-107	0	GW-M1-002 Revision 1 Changes
GW-GEE-108	0	Revision to Design Criteria Document GW-N1-004
GW-GEE-110	0	GW-N1-006, Revision 1 Changes
GW-GEE-112	0	P&ID VAS-M6-007 Revision 3 Changes
GW-GEE-115	0	Site Interface Design Inputs
GW-GEE-115	1	Site Interface Design Inputs
GW-GEE-116	0	Passive Core Cooling System Changes

AP600 Document	Rev.	Title
GW-GEE-117	0	System Specification Document (SSD) VWS-M3-001 Revision 1 Changes
GW-GEE-120	0	Revision to AP600 Power Capability Parameters
GW-GEE-121	0	Chemical and Volume Control System and Reactor Coolant System Changes
GW-GEE-123	0	Primary Sampling System (PSS) P&ID Update
GW-GEE-128	0P	P&IDs FPS-M6-001, 002, 003 & 006 Proposed Rev. 4 Changes
GW-GEE-130	0	Elimination of Pressurizer to CMT Line
GW-GEE-131	0	Gaseous Radwaste System Design Update
GW-GEE-133	0	Revision 1 of Radioactive Waste Drain System SSD, WRS-M3-001
GW-GEE-134	0	Revision 1 of Radiologically Controlled Area Ventilation System SSD, VAS-M3-001
GW-GEE-135	0	Nuclear Island Non-Radioactive Ventilation System SSD VBS-M3-001 Revision 1
GW-GEE-136	1P	Liquid Radwaste System Design Update
GW-GEE-136	0	Liquid Radwaste System Design Update
GW-GEE-137	0	Component Cooling System Design Update
GW-GEE-138	0	Revision to AP600 Fuel Handling Machine
GW-GEE-140	0	CVS and RNS P&ID Revisions
GW-GEE-141	0P	P&ID FPS-M6-004 Proposed Revision 4 Changes
GW-GEE-142	1	Pressurizer Lower Support Change
GW-GEE-144	0	Pressurizer Spray Line Size Reduction
GW-GEE-145	0	Primary Coolant Loop Piping Outline Drawing Changes
GW-GEE-146	0	VES Air Storage Tank Design Modifications
GW-GEE-147	0	Reactor Vessel Head Vent Modification
GW-GEE-148	0	P&ID Presentation of I&C Information
GW-GEE-149	0P	P&ID FPS-M6-005 Proposed Revision 4 Changes
GW-GEE-151	0P	Revision 4 of Containment Air Filtration System P&ID,VFS-M6-001 & 002
GW-GEE-152	0	Revision 2 of ECS-E3-001and Revision 2 of ECS-E3-002
GW-GEE-153	0	Station One Line Diagrams for Class 1E DC and UPS System, IDS-E3-001, 2, & 3, Rev. 2
GW-GEE-154	0	Legend for Electrical Power, Lighting, and Communication Drawing GW-E9-001
GW-GEE-156	0P	Nuclear Island Fire Area Drawings 1020, 1030, 1040-AF-001 Revision 2 Changes
GW-GEE-157	0	Seismic Reclassification of Annex and Turbine Buildings
GW-GEE-158	0P	Central Chilled Water System P&IDs - VWS-M6-001 thru VWS-M6-007, Rev. 5
GW-GEE-159	0	Central Chilled Water System SSD, VWS-M3-001, Rev. 2
GW-GEE-161	0	PCS - Containment Cooling Recirculation Piping
GW-GEE-162	0	Startup Feedwater Control Logic Revision
GW-GEE-163	0	Revision 1 of Containment Air Filtration System SSD, VFS-M3-001
GW-GEE-164	0	Radwaste Building Change

AP600 Document	Rev.	Title
GW-GEE-168	0	Nuclear Island General Arrangement Drawings, Rev. 6
GW-GEE-169	0	Type Change for Valve CVS-PL-V025
GW-GEE-170	0	Core Makeup Tank
GW-GEE-171	0	Passive RHR Hx Design Changes
GW-GEE-172	0P	Radioactive Waste Drain System Changes
GW-GEE-173	0	Change of AP600 Design Basis Fuel Cycle Length from 18 to 24 Months & Reactor Internals Bypass Flow Change
GW-GEE-174	0	Eliminating the Use of Shared Sensors in the AP600 Diverse Actuation System Design
GW-GEE-175	0	Steam Generator Manways
GW-GEE-177	0	RCS Hot Leg Level Instrumentation Modification
GW-GEE-179	0	Ion Exchange Vessel Arrangement Change
GW-GEE-181	1	Turbine building and control room arrange. and seismic design
GW-GEE-182	0	Nuclear Island Roof Slopes
GW-GEE-183	0	VES System Modifications
GW-GEE-184	0	Reactor Coolant Drain Tank and Pumps
GW-GEE-185	0	Normal Residual Heat Removal Heat Exchanger Cooling Water Flow Control
GW-GEE-186	0	Change to Squib Valves in PXS IRWST Lines
GW-GEE-186	1	Change to Squib Valves in PXS IRWST Injection Lines
GW-GEE-187	0	Separate Startup Feedwater Line to Steam Generator
GW-GEE-188	0P	Fire Protection System Changes
GW-GEE-189	0	AP600 Fuel Handling Machine Simplification
GW-GEE-190	0	Passive Autocatalytic Hydrogen Recombiners for DBA Hydrogen Control
GW-GEE-192	0	Nuclear Island Seismic Analysis: Revised Damping and Soil Case
GW-GEE-193	0	PCWG Parameters for the AP600 Stretch Rating
GW-GEE-194	1	Pressurizer Volume Increase to Provide Increased Design Margin
GW-GEE-195	0	VES System Modifications
GW-GEE-196	1	Revise SGBD Isolation Logic
GW-GEE-197	0	Revise SG Drain Line Size
GW-GEE-198	0	Eliminate One PRHR HX
GW-GEE-199	0	Squib Valves for ADS Stage Four
GW-GEE-200	0	SGS Design Changes to Provide Additional Design Margin
GW-GEE-201	0	Reactor Vessel Minor Design Changes
GW-GEE-202	0	Seismic Classification of Annex Building
GW-GEE-203	0	pH Tank Size / Arrangement Change
GW-GEE-204	0	Accumulator Tank Minor Design Changes
GW-GEE-205	1	Removal of the CVS Boric Acid Batching Tank Seam Jacket and Miscellaneous CVS Changes
GW-GEE-207	0	Replacement of Diaphragm Valves in SFS
GW-GEE-208	0	General Arrangement Drawing Revision 7
GW-GEE-208	1	General Arrangement Drawing Revision 7

AP600 Document	Rev.	Title
GW-GEE-209	0	Piping Class Sheets and Standard Detail Changes
GW-GEE-209	1P	PL02-ZO-001, Rev. 2 Piping Class Sheets and Standard Details
GW-GEE-214	0P	Miscellaneous PXS Changes
GW-GEE-215	1	Removal of WLS Radiation Monitor
GW-GEE-216	1	WLS Degasifier Arrangement
GW-GEE-217	0P	HVAC Simplification Changes
GW-GEE-218	0P	Class 1E Battery Room Fire Barrier Changes
GW-GEE-219	0P	Charcoal Filter Water Spray Deletion
GW-GEE-220	1P	Spent Fuel Shipping Cask Crane Design Classification
GW-GEE-221	0	WGS High Oxygen Control Functions
GW-GEE-222	0P	Auxiliary Cooling Water Systems Interface Parameters
GW-GEE-223	0	R.I. Reflector Top Flange Pressure Relief Holes
GW-GEE-224	0	Change 4" PXS Check Valves and Piping to 6"
GW-GEE-225	0	Single Moisture Separator Reheater (MSR)
GW-GEE-226	0P	RNS Layout Changes
GW-GEE-228	0	WGS Gas Flow Control
GW-GEE-229	0P	CVS Equipment Classification and Protection Logic Change
GW-GEE-230	0	RCS Wide Range Pressure Transmitter Re-location and other RCS Modifications
GW-GEE-231	0	Protection and Safety Monitoring System Logic Changes
GW-GEE-232	0	CCS Design Update
GW-GEE-233	0P	Primary Sample System - AC Valves, Piping Specification, Safety Class
GW-GEE-234	0	In-service Testing Requirement Changes
GW-GEE-235	0P	Site Plot Plan Changes for Revision 1
GW-GEE-236	0	Modification of Valve CVS V043
GW-GEE-237	0P	Additional Margin for T/G and Associated Systems
GW-GEE-238	0P	Elimination of Reactor Cavity Fans
GW-GEE-239	0	WGS Drain Pot Elimination
GW-GEE-240	0P	RNS Relief Valve Discharge Piping Modification
GW-GEE-241	0	WGS Design Update
GW-GEE-242	0	RNS Piping Layout Changes
GW-GEE-243	1	DWS ISLOCA Modifications
GW-GEE-244	0P	Delete Suction Supply from DST to Startup Feedwater Pumps
GW-GEE-245	0P	PL02-Z0-001, Rev. 3 Piping Class Sheets and Standard Details
GW-GEE-246	0	Incore Instrumentation Thimble Tubings Material Change
GW-GEE-247	0P	AP600 P&ID Legend Revision 5
GW-GEE-248	0	VES System Modifications Resulting From SSD Review
GW-GEE-249	0	VES Air Flowrate Modification
GW-GEE-251	1	CVS-WLS Containment Penetration Deletions
GW-GEE-252	0	WLS Pumps Suction Isolation Valves and Tanks Nozzles
GW-GEE-253	0	Design Methodology of Structural Modules GW-SUP-001 Rev.0
GW-GEE-254	0	Revised Reactor Cavity/Insulation/Shielding
GW-GEE-255	3P	RMS Radiation Monitors

AP600 Document	Rev.	Title
GW-GEE-256	0P	Security Design Report Changes for Revision 1
GW-GEE-257	0P	Add Autostart Capability to the Standby GSS Exhauster and Incorporated Miscellaneous GSS Modifications
GW-GEE-258	1P	VBS Supplemental Air Filtration Unit Changes
GW-GEE-259	0	Change of RTndt Value in the Reactor Vessel Core Belt Region
GW-GEE-260	0	Pressurizer Spray Nozzle Size Change
GW-GEE-261	0P	VYS P&ID Revision 2 Changes
GW-GEE-262	0	PL02-Z0-001, Rev.4, Piping Class Sheets and Standard Details
GW-GEE-263	0	Steam Generator Channel Head Configuration
GW-GEE-264	0	Change Post Accident pH Adjustment from NaOH to TSP
GW-GEE-265	0	Non-Structural Gypsum Partitions
GW-GEE-266	0	SFS Purification Flow Control and Pipe Size Change
GW-GEE-267	0P	CDS P&ID Revision 1 Changes
GW-GEE-268	0	CVS System Changes to Address High Energy Line Break Layout Issues
GW-GEE-269	0P	Connect BDS S/G Recirculation Line to Nonsafety-Related SFW Piping
GW-GEE-270	0P	VBS MCR/TSC HVAC Subsystem Changes
GW-GEE-271	0P	PCS Revisions Rev. 8 of P&ID
GW-GEE-272	0P	BDS Pump Suction Piping Changes
GW-GEE-273	1	Small Diameter Penetrations Through Containment Vessel
GW-GEE-274	0	Valve Leakoff Elimination
GW-GEE-275	0	CCS Chemical Addition Tank Changes
GW-GEE-276	0	Miscellaneous Changes to PXS
GW-GEE-277	0	Revised Backup (Startup) Feedwater Pump Parameters
GW-GEE-278	0	Fuel Oil Storage Tanks and Transfer Modules Arrangement Change
GW-GEE-279	0P	WRS Radioactive Waste Drain System Changes
GW-GEE-280	0P	Expand Annex Building Clean South AHU Equipment Room & Relocate Air Handling Units (AHU)
GW-GEE-281	1	Secondary Side Design Pressure Increase
GW-GEE-281	0	Secondary Side Design Pressure Increase (For Additional Margin)
GW-GEE-282	1P	MSS P&ID Rev. 4 & 5 Changes
GW-GEE-282	1	MSS P&ID Revision 4 & 5 Changes
GW-GEE-283	1P	Service Air Subsystem Modifications
GW-GEE-284	0P	VYS P&ID Revision 3 Changes
GW-GEE-285	1	WLS Leak Detection
GW-GEE-286	0	RNS Relief Valve Discharge Piping Changes
GW-GEE-287	0	SFS Piping Entry to CWP and Fuel Transfer Canal
GW-GEE-288	0P	Allow MFW Pumps to Supply SFW Header
GW-GEE-289	0P	FWS P&ID Revision 2 Changes
GW-GEE-290	0	RCS Leakoff Line Safety Classification Change
GW-GEE-291	0P	Instrument Air Subsystem Modifications
GW-GEE-292	0P	High Pressure Air Subsystem Modifications
GW-GEE-293	0	Demineralized Water Transfer & Storage Sys.Mod.
GW-GEE-293	1	Demineralized Water Transfer & Storage System Modifications

AP600 Document	Rev.	Title
GW-GEE-294	0P	Potable Water System
GW-GEE-295	1	AP600 Orifice Flange Set Arrangement Changes
GW-GEE-296	0	Rev.8 of Nuclear Island General Arrangement Drawings.
GW-GEE-296	1P	Revision 8 Nuclear Island General Arrangement Drawings
GW-GEE-297	0	VES System Upgrade to 4000 psi Air Storage Tanks
GW-GEE-297	1	VES System Upgrade to 4000 psi Air Storage Tanks
GW-GEE-298	0	Elimination of One WLS Monitor Tank
GW-GEE-298	1	Elimination of One WLS Monitor Tank
GW-GEE-299	0P	VBS MCR/TSC HVAC Subsystem/MCR Envelope Isolation Damper Changes
GW-GEE-300	0	WLS Oil Separator
GW-GEE-301	1	Miscellaneous Changes to RCS P&ID
GW-GEE-302	0	ADS Valve Selection
GW-GEE-302	1	ADS Valve Type and Piping Configuration
GW-GEE-303	0	1E Battery Chargers Undervoltage
GW-GEE-304	0	Valve Standardization
GW-GEE-305	0	Solid Radwaste System P&ID Update
GW-GEE-306	0	Selected Changes to Safety Classifications
GW-GEE-307	1	Containment Penetration Standardization
GW-GEE-308	0P	Fire Protection System Layout & Piping Spec Changes
GW-GEE-309	0	IRWST Vent/Overflow Changes
GW-GEE-310	0	Demineralized Water Treatment System (DTS) Revision 0 Changes
GW-GEE-311	0	CVS Radiation Monitor Elimination
GW-GEE-312	0P	Steam Generator Reference Leg Temperature Elimination
GW-GEE-315	0	WGS Pump Changes
GW-GEE-316	0	Changes to Valves in SFW Flowpath
GW-GEE-317	1P	Miscellaneous HVAC Changes (1/23/96)
GW-GEE-318	0P	Optimization of the Central Chilled Water System
GW-GEE-319	1	CCS Cooling Water Piping Changes
GW-GEE-319	0	CCS Cooling Water Piping Changes
GW-GEE-320	0	Turbine Island Chemical Feed CFS Update
GW-GEE-321	0	PCS Valve Power Assignments
GW-GEE-322	0P	Deletion of VBS Class 1E Battery Room Duct Heaters
GW-GEE-323	0	Deletion of Security Lighting (DCS) as Separate AP600 System
GW-GEE-324	0P	Annex Building Final Design Review Changes
GW-GEE-325	0	CCS Cooling of RNS Design Changes
GW-GEE-326	0P	Use CCS vs. TCS for Cooling Water to Condensate Pump Motors
GW-GEE-327	0P	ASS P&ID Revision 2 Changes
GW-GEE-328	1	CVS Modifications
GW-GEE-330	0P	SWS P&ID Revision 1 Changes
GW-GEE-331	0P	TCS P&ID Revision 1 Changes
GW-GEE-332	0P	CDS P&ID Revision 2 Changes
GW-GEE-333	0P	Circulating Water System Modifications
GW-GEE-334	0P	Gland Seal System Modifications

AP600 Document	Rev.	Title
GW-GEE-335	0P	Waste Water System Modifications
GW-GEE-336	0	Turbine Building General Arrangements Changes for Rev. 2
GW-GEE-337	0P	Site Plot Plan Changes for Revision 3
GW-GEE-338	3	Primary Sampling System Modification Rev. 3
GW-GEE-339	0P	Reconfigure Condensate Cooling to BDS Heat Exchangers
GW-GEE-340	0	Revision of Missile Design Criteria
GW-GEE-341	0	Startup Feedwater Line Size Increase and Automatic Isolation of Main Feedwater Crossover Line
GW-GEE-342	0	PCS Delivery Line Size Reduction
GW-GEE-343	1	Sanitary Drain System (SDS) Changes to Revision 1
GW-GEE-344	0P	Secondary Sampling System Changes to Rev.0
GW-GEE-344	1P	Secondary Sampling System (SSS) Changes to Revision 0 P&IDs
GW-GEE-345	0P	RCP Drain Connection Simplification
GW-GEE-346	0	Change PXS Valves V118A/B
GW-GEE-347	0	RCP Hydraulics Modification
GW-GEE-348	0	Revised Electrical Feed for Non-1E Battery Chargers
GW-GEE-349	0	PL02-Z0-001, Rev. 5 Piping Class Sheets and Standard Details
GW-GEE-350	1	PMS Changes
GW-GEE-351	0	Addition of MCCs 133 and 233 & Reassignment of ECS System Loads
GW-GEE-352	0	Revised Location of Startup Feedwater Check Valve
GW-GEE-354	0	Steam Generator Blowdown System Revision 1 Changes
GW-GEE-355	1	MSS P&ID Steam Isolation Signal and Turbine Bypass Isolation Valves
GW-GEE-356	0	Demineralized Water Transfer Pump Capacity Modification
GW-GEE-358	1P	Electrical Penetration Test Connections/ILRT Instrumentation
GW-GEE-359	0P	Addition of Distribution System P&ID Sheets
GW-GEE-360	0P	Wastewater System Modifications
GW-GEE-361	0P	PCS Non-Safety Related System Changes
GW-GEE-362	0P	VBS Duct Layout and Optimization Changes
GW-GEE-362	1P	VBS Duct Layout and Optimization Changes
GW-GEE-363	0	Heater Drain Modification (HDS) P&ID Revision 2 Changes
GW-GEE-364	1	CVS Makeup Pump
GW-GEE-365	1	IRWST Injection Throttling
GW-GEE-366	0	CVS Tank Drains
GW-GEE-367	0	Seismic Classification of Emergency Lighting in Main Control Room and Remote Shutdown Area
GW-GEE-368	0	PAMS Modifications
GW-GEE-369	0	Revision 2 of Annex Building General Arrangement Drawings
GW-GEE-370	0	CVS/DWS Interface & Boron Dilution Protection Logic
GW-GEE-370	1	CVS/DWS Interface and Boron Dilution Protection Logic
GW-GEE-371	0	DWS Storage Tank Piping Modification
GW-GEE-372	0	Generic Level Instrumentation for Tanks
GW-GEE-374	0	Remove Temperature Elements and Pressure Transmitters in MSS
GW-GEE-375	0P	FPS Tank/Pump Configuration and Miscellaneous Changes

AP600 Document	Rev.	Title
GW-GEE-376	0P	VBS Duct Layout and Optimization Changes
GW-GEE-377	0	CVS Stop Check Valve Changes
GW-GEE-378	0	PRHR HX Changes in Extended Flange/Channel Head Assembly Arrangement
GW-GEE-379	0P	Remove Demineralized Water Supply from ASS
GW-GEE-380	0	WGS & WLS Pipe Specification Changes
GW-GEE-381	1	RNS Test Connection Addition
GW-GEE-382	1P	N.I. Room Number and Name Changes
GW-GEE-382	0P	Nuclear Island Room Number and Name Changes
GW-GEE-383	0	SFS Refueling Cavity Drain Connection
GW-GEE-384	0	WLS Relief Valve and Drain Valve Additions
GW-GEE-385	2	VES Modifications - Rev.2
GW-GEE-386	0P	Addition of PWS Distribution Lines
GW-GEE-388	0P	Deletion of VXS Battery Room Duct Heaters
GW-GEE-389	0	Hot Water Heating System (VYS) Steam Supply Piping Modifications and Changes to Revision 4 of the P&IDs
GW-GEE-390	0P	General Arrangement Changes to Waste Monitor Tank Room Access
GW-GEE-390	0	General Arrangement Change for Waste Monitor Tank Room Access
GW-GEE-392	0	WLS Anti-Siphon Feature Change
GW-GEE-393	0	Power Division Assignments
GW-GEE-394	0P	Compressed & Instrument Air System (CAS) Distribution System Additions
GW-GEE-395	0P	DWS Distribution System Addition
GW-GEE-395	0	DWS Distribution System Additions
GW-GEE-396	0	Security Boundary Drawing Rev.1
GW-GEE-397	0	Security Design Report Rev. 2
GW-GEE-398	0	Turbine Building Fire Area Drawing Revision
GW-GEE-399	0	WLS Degasifier Vessel and Piping Change
GW-GEE-400	0	FWS P&ID Rev. 4 Changes
GW-GEE-401	0	PL02-Z0-001, Rev 6 Piping Class Sheets and Standard Details
GW-GEE-402	0	Increased PCS Flow Rates
GW-GEE-403	0p	Fire Area Drawing Changes
GW-GEE-403	0	Fire Area Drawing Changes
GW-GEE-404	1	Show Vibration and Temperature Monitoring for MFW Pumps
GW-GEE-405	0	Diverse Actuation System Manual Containment Isolation Logic Correction
GW-GEE-408	0	Reactor Vessel Support Dwg. Changes
GW-GEE-409	0	Pressurizer Support Drawing Changes
GW-GEE-410	0	Steam Generator Support Drawings Changes
GW-GEE-411	0	Elimination of Stop-Check Valves in the VES
GW-GEE-412	0	Changes to Numbering System
GW-GEE-413	1P	Revisions to Power Division Assignments
GW-GEE-413	1	Revisions to Power Division Assignments
GW-GEE-414	0	Delete Pumphouse Ventilation System, VPS

AP600 Document	Rev.	Title
GW-GEE-415	0	Main Control Room Panel Lighting
GW-GEE-416	1	Letdown Purification Line Isol; CVCS Isol. and Neutron Flux Instrument Design Changes
GW-GEE-416	0	Letdown purification lined isol,cvcs,isol, and neutron flux.
GW-GEE-417	1	CONDENSATE STORAGE TANK HEADER MODIFICATION
GW-GEE-417	0	Condensate Storage Tank Header Modification
GW-GEE-418	0	Reactor System Drawing Changes
GW-GEE-419	0	Misc. Changes to PMS
GW-GEE-419	1	Miscellaneous Changes to Protection and Safety Monitoring System
GW-GEE-420	1	SFS Post-72 Hours Changes
GW-GEE-421	0	Annex Building Flooding Mitigation Provisions
GW-GEE-421	1	Annex Building Flooding Mitigation Provisions
GW-GEE-422	0	MSIV Exterior Wall
GW-GEE-423	0	CVS Pipe Tunnel Design Pressure
GW-GEE-424	1	Battery-Backed Electrical Power for Hydrogen Igniters
GW-GEE-425	0	Update Heat Balance Diagrams
GW-GEE-427	0	WGS System Simplification
GW-GEE-428	0	FWS P&ID Rev. 5 Changes
GW-GEE-429	0	Reactor Coolant Pump - Redesign of Gasket Seal
GW-GEE-430	0	WLS Chemical Addition Tank Changes
GW-GEE-431	0	DAS Functional Requirements
GW-GEE-432	0	Design Requirements for PCS Components
GW-GEE-434	0	PAMS Changes Based on NRC Review of ERGs & SSAR Chapter 7.5
GW-GEE-435	0	Fuel Handling System Drawing Changes
GW-GEE-436	0	Revise CCS P&ID to Support Pipe Routing in Turbine & Auxiliary Buildings
GW-GEE-438	1	Electrical Power & Containment for 72 Hours to 72 Days
GW-GEE-439	0	Change to PXS Drain/Make-Up Lines
GW-GEE-440	3	Addition of Non-Safety Containment Spray System
GW-GEE-441	0	Editorial Change to Fire Zone Designations
GW-GEE-442	0	Live Loads for Seismic Analysis
GW-GEE-443	1	MSS Changes (Post-FOAKE)
GW-GEE-444	1	Steam Generator Design & Drawing Changes
GW-GEE-445	0	Elimination of Leak-before-Break for SGS-Main Feedwater & RCS-Pressurizer Spray
GW-GEE-446	1	Change CMS & GSS AP600 Classification from "E" to "D"
GW-GEE-447	0	Deletion of Rad Chem Lab HVAC Subsystem
GW-GEE-448	0	Cable Tray Damping
GW-GEE-450	1	CVS Purification Loop Equipment Classification Change, Misc. P&ID Changes & Cavitating Venturi Location Change
GW-GEE-451	1	System Changes to Address Human Factors Minimum Inventory & the Shutdown Evaluation Report
GW-GEE-452	0	Reactor Internals Safety Classification
GW-GEE-453	0	Turbine Building Eccentric Bracing Configuration
GW-GEE-454	0	Reactor Vessel Support Design Change

AP600 Document	Rev.	Title
GW-GEE-455	0	Pressurizer Upper Support Design Change
GW-GEE-457	0	Turbine Building Ventilation System (VTS) Changes for P&ID Rev 0
GW-GEE-458	0	PMS/ Logic Change
GW-GEE-458	1	PMS Logic Changes
GW-GEE-460	0	Accumulator Tank Assembly
GW-GEE-461	0	Change to ADS Stage 1/2/3 Line Resistance
GW-GEE-463	2	Revisions to VAS, VBS, VES, & PCS for Post-72 Hours
GW-GEE-464	2	VAS Duct Layout and Damper Changes
GW-GEE-465	0	Design Changes Pursuant to AP600 Shutdown Evaluation Report
GW-GEE-466	0	PXS-B Compartment Vent & Access Opening
GW-GEE-467	1	FPS & ECS Fire Pump Arrangement, VPS Fire Dampers, FPS Underground Valves
GW-GEE-468	2	RNS Heat Exchanger Bypass Line Addition
GW-GEE-469	1	SGS Revisions
GW-GEE-470	1	RNS Containment Isolation Valve Actuation Logic
GW-GEE-471	1	Reconfiguration of Some Walls and Floors in Nuclear Island
GW-GEE-472	1	Additional PRHR Actuation Logic Change
GW-GEE-473	0	Containment Vessel Stiffener at Elevation 131'9"
GW-GEE-474	0	SFS Level Instrument Names
GW-GEE-475	0	Chilled Water System/Containment Recirculation Cooling System Changes for Compliance with GL 9606
GW-GEE-476	1	VFS Debris Screen Assemblies
GW-GEE-477	0	Fire Protection System Design Changes
GW-GEE-478	0	Addition of One Radiation Monitor to Fuel Storage Area
GW-GEE-480	0	Cooling Water Flow Control to RNS Heat Exchangers
GW-GEE-481	1	Change IRWST Gutter to Safety-Related
GW-GEE-482	0	Change TSP Volume
GW-GEE-483	0	Revisions to Turbine-Generator Trip Logic
GW-GEE-484	0	Change in N.I. Basement Reinforcement for Soft Soil Construction
GW-GEE-484	1	Change in N.I. Basement Reinforcement for Soft Soil Construction
GW-GEE-485	1	Changes to Containment Recirculation System
GW-GEE-486	0	Automatic Actuation of Fourth Stage ADS on Low Hot Leg Level
GW-GEE-487	0	REVISION OF AP600 LOCA SOURCE TERM
GW-GEE-489	0	Core Makeup Tank Dwg. Changes
GW-GEE-490	0	Accumulator Tank Dwg. Changes
GW-GEE-492	2	Chilled Water System Design Changes
GW-GEE-493	1	VLS Additional Igniters in Dome and Safety-Related PAR in IRWST
GW-GEE-494	1	Add Mechanical Overspeed Device to Main Turbine
GW-GEE-495	1	Change to ADS Stage 4 Permissive
GW-GEE-496	1	CDS P&ID Revision 3 Changes
GW-GEE-497	1	Additional Thickness and Rebar Requirements for Nuclear Island and Annex Building Walls, Floors, & Ceilings
GW-GEE-498	1	Compressed Air System Revisions
GW-GEE-498	0	Compressed Air System Revisions
GW-GEE-500	2	Central Chilled Water System Design Changes

AP600 Document	Rev.	Title
GW-GEE-502	2	Cooling of RNS Heat Exchangers Using Fire Protection Water
GW-GEE-503	1	FPS Changes Including Additional Turbine Building Sprinklers
GW-GEE-504	1	Curb Heights for CVS and PXS Compartments and Additional CVS Compartment PAR
GW-GEE-505	1	Spent Fuel Pool Boiling Issue Design Changes
GW-GEE-506	2	VES and VBS Design Changes
GW-GEE-507	0	MCC Renumbering
GW-GEE-508	0	Waste Water System Modifications and Additions
GW-GEE-508	1	Waste Water System Modifications and Additions
GW-GEE-509	2	VBS MCR Isolation Valves and Div. B Trip Switchgear Room Fire Dampers
GW-GEE-510	0	Radioactive Waste Drain System Design Changes
GW-GEE-511	0	Pressurizer Heater Block Implementation
GW-GEE-513	0	Reconfiguration of PCS Makeup Lines to PCCWST and Spent Fuel Pool
GW-GEE-514	1	FPS Changes to Seismic Standpipe Supply
GW-GEE-515	0	CCS Heat Exchanger Changes
GW-GEE-516	1	WLS Chemical Waste Tank Size Change
GW-GEE-517	1	Change CMT Level Instrument Type
GW-GEE-518	0	Change of ADS Valve Test Frequency
GW-GEE-519	0	Reactor Vessel Drawing Changes
GW-GEE-520	0	Pressurizer Drawing Changes
GW-GEE-521	1	Relocation of Steam Line Pressure Transmitters Outside of Steam Line Compartment
GW-GEE-522	1	Reactor Internals Drawing Changes
GW-GEE-523	1	Basemat and Exterior Wall Shear Reinforcement
GW-GEE-523	2	Basemat and Exterior Wall Shear Reinforcement
GW-GEE-524	2	Reduction of Fire-Induced Spurious Actuations
GW-GEE-525	0	PCS Recirculation Pump Resizing for Post 72 Hr. Operation
GW-GEE-526	1	Changes to Valve Response Times for ITAAC's
GW-GEE-527	1	In-Containment Nonsafety Paint & Insulation Changes
GW-GEE-528	2	Changes in Containment Interior Structure
GW-GEE-529	0	Sample System Isolation Valve Survivability
GW-GEE-530	0	CVS Changes to Improve RCS Pressure Boundary
GW-GEE-531	0	Missile Design Criteria
GW-GEE-533	1	Fire Protection Related Changes to Satisfy NRC
GW-GEE-534	0	Correct PMS Containment Pressure Setpoints
GW-GEE-535	0	Sample System Isolation Valve Survivability & Modifications to Incorporate Sentry Grab Sample Panel
GW-GEE-536	0	Diesel Generator Temperature Control Valves
GW-GEE-537	0	Containment Leak Rate and Source Term Changes
GW-GEE-538	1	Main Control Room Layout
GW-GEE-539	0	Add Pressure Instrument in VES Header
GW-GEE-540	1	Limitation on Use of Fibrous Insulation
GW-GEE-541	2	Upgrade of 2" Containment Isolation Butterfly Valves to 3" Valves

AP600 Document	Rev.	Title
GW-GEE-543	0	36" Containment Penetration Modification
GW-GEE-548	0	Add CVS Leak Test Condition
GW-GEE-552	1	Door Between RCDT Room and Reactor Cavity Compartment
GW-GEE-553	0	Fire Protection Changes to Satisfy NRC
GW-GEE-554	1	PXS and RCS P&ID Note Changes
GW-GEE-555	1	Civil, Seismic, and Wind Design Criteria Changes
GW-GEE-556	1	Reinforcement Details in Shield Building Roof
GW-GEE-558	1	Fix PMS-J1-200 and Incorporate PAMS Requirements into Sods
GW-GEE-559	1	Relocation of Diesel-Driven Fire Pump
GW-GEE-560	1	PMS Logic Changes
GW-GEE-561	0	Elimination of Line VES 1" DAC L012A
GW-GEE-562	1	Modification of Spent Fuel (SFS) P&ID
GW-GEE-564	0	Lateral Support for Exterior Wall on Column Line 1
GW-GEE-565	1	Openings Through Shear Walls on Column Lines 4 and 5 and Floors Around Tanks
GW-GEE-566	1	Security Controlled Access Enclosures
GW-GEE-569	0	Modification of Lube Oil System (LOS) P&ID
GW-GEE-570	0	Fire Protection Changes for PXS Valves
GW-GEE-571	0	List of AP600 Modules
GW-GEE-572	0	GSS P&ID Change to Meet URD Chapter 2, Requirement 4.3.8
GW-GEE-573	0	Modification of Site Location Ids
GW-GEE-574	1	Cleanup of Component Cooling Water P&ID
GW-GEE-575	0	Cleanup of Auxiliary Steam P&ID
GW-GEE-576	0	Cleanup of PCS P&ID
GW-GEE-577	1	Structural Modules Inside Containment: Constructability Changes
GW-GEE-578	0	CB liner modules inside containment
GW-GEE-579	1	Changes to HCS to Establish Consistency with Interfacing Systems
GW-GEE-580	0	Primary Shield and CVS Structural Modules
GW-GEE-581	2	Update Plant Design Criteria
GW-GEE-582	2	Nuclear Island General Arrangement Drawings Rev. 9
GW-GEE-583	0	Openings Through Shear Wall on Column Line 7.3
GW-GEE-584	1	Modify Paving West of Turbine Building
GW-GEE-585	0	PXS-B Maintenance Hatch Cover
GW-GEE-587	0	Correction of Site Location ID Boundaries
GW-GEE-588	0	Correction of Demineralized Water Storage Tank Size
GW-GEE-589	0	CA20 Module Boundaries
GW-GEE-591	0	Requirements for Doorway between RCDT Room and Reactor Cavity Compartment
GW-GEE-592	1	Auxiliary Building, Annex Building, & Containment Elevator
GW-GEE-593	0	Surge Line Penetration for Steam Generator Compartment
GW-GEE-594	2	Dual Refueling Canal Overflow Pipes
GW-GEE-595	0	Legs for Waste Hold-Up Tanks in Auxiliary Building Area 6
GW-GEE-596	0	Connection of IRWST Steel Wall to Operating Floor
GW-GEE-597	0	Struc. Mod. CA01 Misc. Rev. *Given to R. Vijuk for Project Mgr Approval as CI 1/2/2001*
GW-GEE-598	1	Clarification of Piping Component Orientations in Piping & Instrumentation Diagrams
GW-GEE-599	1	CVS Letdown Modifications for Improved Deborating Demineralizer & Shutdown Purification

AP1000 Design Change Packages

AP1000 Document	Rev.	Title
APP-GW-GEE-001	0	Reactor Vessel & Internals Design Changes
APP-GW-GEE-002	0	Containment Vessel Crane Girder and External Pressure
APP-GW-GEE-003	1	Steam Generator System Instrumentation Changes for AP1000
APP-GW-GEE-004	0	AP1000 Nuclear Island Fire Area Drawings
APP-GW-GEE-005	0	Containment Vessel Design Temperature
APP-GW-GEE-006	1	AP1000 Plant Parameters Update to Revision 1
APP-GW-GEE-007	2	Higher Curbs & Configuration Changes for PXS-A, PXS-B, and the DVS Compartment Openings
APP-GW-GEE-008	1	AP1000 Reactor Coolant Pump Revisions
APP-GW-GEE-009	1	Revisions to Civil Design Criteria - MSIV Subcompartment Pressures and IRWST Transient
APP-GW-GEE-010	0	Logic Changes to Improve ATWS
APP-GW-GEE-012	1	Change from Remote Shutdown Room
APP-GW-GEE-013	1	Reactor Vessel Incore Instrumentation Head Penetrations & Upper Support Column Extension Modification Required for Head Inspection
APP-GW-GEE-014	0	IRWST Steel Wall Horizontal Stiffeners and Concrete Embedment
APP-GW-GEE-015	1	Containment Vessel Corrosion Allowance and External Pressure/SSE Load Combinations
APP-GW-GEE-016	0	Auxiliary Building Radiation Zone Drawing Revision
APP-GW-GEE-018	0	Flow Holes in Core Shroud Bottom Plate
APP-GW-GEE-019	0	Revised Seismic Analyses Considering Concrete Cracking
APP-GW-GEE-020	0	Radiation Zone Map - Change to Note on Post-Accident Legend
APP-GW-GEE-021	0	AP1000 Auxiliary Building, Turbine Building, & Annex Building Stairwell Enclosures
APP-GW-GEE-022	1	AP1000 RCP Design Analyses and Water Volumes
APP-GW-GEE-023	1	Addition of Shear Studs on Containment Vessel
APP-GW-GEE-025	0	AP1000 Core Shroud
APP-GW-GEE-026	1	AP1000 Elimination of Lower Annulus Room 12243
APP-GW-GEE-027	1	AP1000 New Core Makeup Tank Configuration
APP-GW-GEE-028	0	AP1000 Inconsistent Direct Vessel Injection Nozzle Pipe Size
APP-GW-GEE-030	1	Location of AP1000 Turbine Building Room 20303 for FPS Motor Driven Pump
APP-GW-GEE-035	2	Incorporation of Zinc Addition Subsystem and Hydrogen Makeup Enhancements
APP-GW-GEE-037	0	Steam Generators Upper Support Snubber Stiffness
APP-GW-GEE-038	1	AP1000 Design Change Proposal for PRHR Hx Design Optimization
APP-GW-GEE-039	1	Replacement of Reactor Coolant Flow Probe
APP-GW-GEE-040	3	AP1000 Pressurizer Design
APP-GW-GEE-041	0	AP1000 Reactor Vessel Design Change Proposal for Reactor Vessel Configuration
APP-GW-GEE-043	0	AP1000 Pressurizer Safety Valve Discharge Piping Pressure Rating Reduction

AP1000 Document	Rev.	Title
APP-GW-GEE-044	2	Spent Fuel Pool Cooling System Modifications
APP-GW-GEE-045	2	PMS Logic Changes
APP-GW-GEE-046	3	AP1000 Accumulator Tank Design Drawing Revision DOOSAN
APP-GW-GEE-049	2	Stainless Steel Surfaces for CA Modules
APP-GW-GEE-050	1	Spent Fuel Shipping Cask Handling Crane Upgrade
APP-GW-GEE-053	1	Normal Residual Heat Removal System (RNS) P&ID Changes
APP-GW-GEE-054	0	QDPS Display Variable Modifications
APP-GW-GEE-056	1	Reactor Vessel Fluence Reduction Addition of Neutron Pads and Increase in Reactor Vessel Diameter
APP-GW-GEE-057	0	AP1000 WLS P&ID's - Rev 0
APP-GW-GEE-059	2	AP1000 Core Make-Up Tank Design
APP-GW-GEE-061	1	AP1000 Accumulator Design Specification Revision
APP-GW-GEE-062	2	AP1000 Steam Generator Primary Side Configuration
APP-GW-GEE-063	2	AP1000 Steam Generator Secondary Pressure Boundary Comfit
APP-GW-GEE-064	1	Design Change Proposal for Secondary Internals
APP-GW-GEE-065	1	Various CVS P&ID Changes
APP-GW-GEE-066	2	Component Cooling System P&ID
APP-GW-GEE-067	1	PMS Functional Design Changes
APP-GW-GEE-068	3	Spent Fuel Racks Design
APP-GW-GEE-069	1	Change in Material of ICI Sheath IHP Dose Reduction
APP-GW-GEE-072	0	Steam Dump Control Logic Changes
APP-GW-GEE-073	0	New Fuel Rack Design
APP-GW-GEE-074	1	AP1000 Reactor Coolant System Design Transients Update to Revision 1
APP-GW-GEE-075	1	Steam Generator Channel Head Nozzle Dimension Changes
APP-GW-GEE-076	1	Rev 2 to Control Room Emergency Habitability System and P&ID
APP-GW-GEE-077	0	Lower Core Support Plate Flow Hole Size
APP-GW-GEE-078	2	AP1000 Flow Orifice Flange Set Standardization
APP-GW-GEE-079	2	AP1000 Air Operated (Pneumatic) Double Diaphragm Pumps
APP-GW-GEE-080	1	Safety/Non-Safety Communications
APP-GW-GEE-082	2	DAS Functional Diagram Changes
APP-GW-GEE-083	2	AP1000 Pressurizer Manway and Instrumentation Nozzle
APP-GW-GEE-084	1	AP1000 Design Change Proposal for APP-PL02-Z0-001 Rev 1 Piping Class Sheets and Standard Details
APP-GW-GEE-086	0	Revision 1 to Emergency Operating Procedure (EOP) E-1
APP-GW-GEE-087	1	Addition of Insulation in the Vicinity of Containment Vessel (CV) Equipment Hatch and Airlock and Documentation of the Clarifications to be Incorporated in the CV Design Specification
APP-GW-GEE-088	0	Protection and Safety Monitoring System PMS Multiplexer Elim
APP-GW-GEE-089	2	AP1000 Fuel Management
APP-GW-GEE-090	1	Modify Applicability of Operating Procedures and Test Specifications
APP-GW-GEE-091	1	AP1000 Containment Vessel Material Replacement of Supplementary Requirement S17 by S1
APP-GW-GEE-093	0	CVS Control System Corrections
APP-GW-GEE-094	1	Reactor Internals Materials

AP1000 Document	Rev.	Title
APP-GW-GEE-096	0	Revision of Reactor Vessel & Reactor Vessel Internals Design Class 3
APP-GW-GEE-098	0	Spent Fuel Pool Spray
APP-GW-GEE-099	0	Pressurizer Spray Nozzle Configuration Change
APP-GW-GEE-100	0	Changes to DRAP Component List
APP-GW-GEE-101	2	Design Changes for Vacuum Refill Operation
APP-GW-GEE-103	1	Access Control Modifications Auxiliary and Annex Building
APP-GW-GEE-104	1	Leak Before Break Evaluation DCD Modification
APP-GW-GEE-106	0	Replacement of Tongue & Groove Flanges with Raised Face Flanges
APP-GW-GEE-108	1	Relocation and Redesign of Specimen Baskets
APP-GW-GEE-109	1	Pressure Boundary Material
APP-GW-GEE-110	1	Design Change Proposal for Reactor Coolant Pump Design
APP-GW-GEE-111	1	Addition of a Flow Skirt to the Reactor Vessel Lower Head
APP-GW-GEE-112	0	Spent Fuel Pool Layout
APP-GW-GEE-113	1	AP1000 DCD Change Request Associated with a Revised Definition of Core Flow Area (DCP 085 was incorporated into this DCP)
APP-GW-GEE-114	0	Pressurizer Nozzle Design
APP-GW-GEE-116	1	Change of Sensor Type for Containment Flood up Level PXW-050, 051, 052
APP-GW-GEE-117	1	Change PXS Level Measurement Sensor Types and Accumulator Tank Tapping Points
APP-GW-GEE-118	0	Heater Sheath Material
APP-GW-GEE-119	0	Relocation of Radial Support Keys
APP-GW-GEE-121	1	Remote DAS Indication and Squib Valve Capability
APP-GW-GEE-122	1	Changes to Startup Feedwater P&ID
APP-GW-GEE-123	2	Temporary Strainers Added to Major Auxiliary Pump Suction Piping
APP-GW-GEE-124	1	PCS P&ID Changes
APP-GW-GEE-125	0	Add 4 x 4 Lateral to Class JCD/JCE
APP-GW-GEE-126	1	Changes Associated with the Main Steam Safety Valve
APP-GW-GEE-128	0	Add Threaded Pipe Caps to Piping Class Sheets
APP-GW-GEE-129	0	Tech Spec Modification to Permit Improved RCS Flow Trending
APP-GW-GEE-130	1	Rod Control System General Arrangement
APP-GW-GEE-131	0	Modifying Piping Class Sheet Documentation to Add Fittings Already in 3D Model
APP-GW-GEE-137	0	Modifications of ISS Penetration Length
APP-GW-GEE-139	0	Reactor Vessel Insulation System RVIS Intermediate Design
APP-GW-GEE-140	0	CMT& PRHR H/Ex Safe End and Bolting Material Change
APP-GW-GEE-141	0	Add Line Numbers and Names to Instrument Tubing on P&IDS
APP-GW-GEE-142	0	Condenser Design for AP1000 Standard Plant
APP-GW-GEE-143	3	Pressurizer Upper Instrumentation Level Tap
APP-GW-GEE-145	0	Correction of HVAC Humidifier Reference
APP-GW-GEE-146	1	Document Numbering Procedure Pipe Hanger Drawing Number Format

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APP-GW-GEE-147	1	Rod Control System Double Hold
APP-GW-GEE-148	1	Grey Rod Swap and Operational Requirements Consistent with use of the On-Line Power Distribution
APP-GW-GEE-149	1	Rod Control System General Arrangement
APP-GW-GEE-150	0	PRHR HX Man way Cover and Bolting Material Change
APP-GW-GEE-151	0	Correction of VBS P&ID Chilled Water Control
APP-GW-GEE-152	0	Revision 2 to Service Water System P&ID and Associated SWS Parameters
APP-GW-GEE-153	0	BDS Piping and Instrumentation Diagram Changes
APP-GW-GEE-155	1	Elimination of RVIS Steam Vent Ducts
APP-GW-GEE-156	1	DCD Changes Related to Functional Design of PLS
APP-GW-GEE-157	2	Addition of VBS Heaters and Controls
APP-GW-GEE-158	1	Annex Building Expanded Office Area
APP-GW-GEE-159	0	Variable Frequency Drive Equipment Relocation Turbine Building
APP-GW-GEE-160	1	Water Tight Features for Waste Holdup Tanks Rooms 12166 and 12167
APP-GW-GEE-162	0	Turbine Building Observation Posts and Personnel Enclosures
APP-GW-GEE-163	0	Remove Reference to Eagle Platform and Remove Reference to WCAP-15927 in DCD Chapter 7
APP-GW-GEE-165	1	DCD changes for Digital Metal Impact Monitoring System for Transition from AP600 to AP1000
APP-GW-GEE-167	1	Radwaste Building Extension Observation Post and Roof Parapet
APP-GW-GEE-168	1	Annex Building East Stairwell Elevator Enclosure Observation Post Personnel Enclosure and Roof
APP-GW-GEE-170	0	Polar Crane Design
APP-GW-GEE-171	0	Re-Evaluation of AP1000 Equipment Qualification and Severe Accident Radiation Dose
APP-GW-GEE-172	1	P&ID Changes for Tag Numbers RCS-TE125A, B, C, D
APP-GW-GEE-173	3	Electrical Changes
APP-GW-GEE-174	1	Diesel Generator Building East Stairwell Enclosure Observation Posts and Roof Parapet
APP-GW-GEE-175	0	Dry Storage Cask Transportation Related Modifications Auxiliary and Radwaste Buildings
APP-GW-GEE-177	2	Reactor Coolant System ADS Stage 1, 2, 3 Pressure Class Correction
APP-GW-GEE-179	0	Design of Auxiliary Building Walls and Slabs Critical Sections Rebars Reconfiguration
APP-GW-GEE-181	1	Revised Transducers for the Reactor Internals Preoperational Vibration Measurement Program
APP-GW-GEE-182	1	Spent Fuel Pool Water Level and Dose
APP-GW-GEE-183	0	Elimination of Pipe Class TGY
APP-GW-GEE-184	1	PMS Functional Changes II
APP-GW-GEE-185	1	IHP Design
APP-GW-GEE-187	0	Document Numbering Procedure Structural Sub Module Drawing Number Format
APP-GW-GEE-188	3	Re-Evaluation of MSIV Compartment Temperature Response to Main Steam Line Break
APP-GW-GEE-189	0	Steam Generator Feedwater Piping Change

AP1000 Document	Rev.	Title
APP-GW-GEE-190	0	Main Control Room Layout
APP-GW-GEE-191	1	Differential Pressure Flowmeter for PRHR HX Flow
APP-GW-GEE-192	1	CMT Upper and Lower Range Level Switch
APP-GW-GEE-193	0	Ultrasonic Level Measurement for PCCWST Narrow Range
APP-GW-GEE-196	0	Source of Cooling Water for TCS and CMS Heat Exchangers
APP-GW-GEE-198	2	Removal of PWS Source and WWS Retention Basins from Westinghouse AP1000 Scope of Certification
APP-GW-GEE-199	0	PCS Changes
APP-GW-GEE-200	1	Electrical Penetration Description Change
APP-GW-GEE-201	2	Fast Bus Transfer
APP-GW-GEE-202	0	Auxiliary Building Critical Sections Rebar Changes
APP-GW-GEE-203	0	Revision of NQA-1 for AP1000
APP-GW-GEE-204	0	Revised Site Temperature Limits
APP-GW-GEE-205	1	Correction of Dose Reduction Feature of O-Ring
APP-GW-GEE-206	0	New Fuel Storage Pit Fuel Assembly Drop
APP-GW-GEE-207	0	AP1000 PXS Containment Recirculation and IRWST Screen Configurations
APP-GW-GEE-208	1	Liquid Radwaste Discharge Pipe and Hold-up Tank Capacity
APP-GW-GEE-209	0	Miscellaneous CVS P&ID Changes
APP-GW-GEE-211	2	CVS PAMS Instrument Range Corrections
APP-GW-GEE-212	1	Removal of Smart Valves
APP-GW-GEE-213	1	Selection and Sizing of Auxiliary Steam Boiler for AP1000
APP-GW-GEE-214	1	Black Poly Piping
APP-GW-GEE-215	2	Design Change Proposal to Steam Generator Channel Head
APP-GW-GEE-216	0	Toshiba Turbine Generator and Steam Cycle Design
APP-GW-GEE-218	2	Main Steam Line Condensate Drain Changes
APP-GW-GEE-220	1	Primary Pressure Boundary Materials Code Cases
APP-GW-GEE-221	0	Capacity for New Fuel Handling Crane
APP-GW-GEE-224	2	Changes to DCD Section 3.11 based on Westinghouse and NuStart Comments.
APP-GW-GEE-226	0	RV ICI Guide Tube Diameter
APP-GW-GEE-228	0	Outside Air Inlet Structure
APP-GW-GEE-229	1	PMS CVS Letdown Isolation Logic Change
APP-GW-GEE-230	2	CRDM Material Manufacturing Changes
APP-GW-GEE-231	0	Spent Resin Transfer Pump Description in DCD
APP-GW-GEE-232	0	Radwaste Building Extension WLS Monitor Tank Addition
APP-GW-GEE-233	1	MCR Layout Rooms Outside of Main Control Area
APP-GW-GEE-234	1	MCR Vestibule Air Purge Addition
APP-GW-GEE-235	0	AP1000 Steam Generator Tubesheet Primary Side Stand-off Length
APP-GW-GEE-236	0	AP1000 Steam Generator Nozzle Dam Support Design
APP-GW-GEE-238	0	AP1000 Technical Support Center TSC Rename to Control Support Area CSA for DCD
APP-GW-GEE-239	0	Design Change Proposal for RV IHP Lug Configuration
APP-GW-GEE-241	0	Spent Fuel Shipping Cask Handling Crane Change

AP1000 Document	Rev.	Title
APP-GW-GEE-242	0	In-service Testing of IRWST Injection Check Valves V122A, B and V124A, B Modification
APP-GW-GEE-243	1	RNS Pump Rooms Air Handling Units
APP-GW-GEE-244	1	Addition of Cyber Security Level 3 Computer Rooms in Annex Building Expanded Office Area
APP-GW-GEE-245	0	Reactor Coolant Pump Flywheel Material Change
APP-GW-GEE-246	1	Containment Air Cooling Diffuser Grating Platforms
APP-GW-GEE-247	1	Miscellaneous RCS P&ID Errors
APP-GW-GEE-248	2	AP1000 Shield Building Structure Design Enhancement
APP-GW-GEE-249	0	AP1000 Steam Generator Channel Head Cladding Thickness
APP-GW-GEE-250	0	AP1000 Steam Generator CVS Nozzle Relocation
APP-GW-GEE-251	0	Steam Generator Channel Head Bottom Redesign
APP-GW-GEE-252	0	AP1000 Steam Generator Tubesheet Periphery Groove Redesign
APP-GW-GEE-253	0	Steam Generator Blowdown Passage Nozzle Redesign
APP-GW-GEE-254	0	AP1000 Steam Generator Main Feedwater Nozzle Projection Length
APP-GW-GEE-255	0	AP1000 Steam Generator Startup Feedwater Nozzle
APP-GW-GEE-256	0	Maintenance Hatch Hoist Design
APP-GW-GEE-257	2	RV Coating Before Shipment
APP-GW-GEE-258	0	AP1000 Steam Generator Design Responsibilities Class 3
APP-GW-GEE-259	1	Revision of Load Follow Design Transient
APP-GW-GEE-260	0	Revision of 1-Line to Allow for Raw Water Pumps Auxiliaries
APP-GW-GEE-261	1	AP1000 Steam Generator Nozzle Design Loads
APP-GW-GEE-262	1	Design Change Proposal for Non Safety Related Classification for AP1000 Fuel Handling Equipment
APP-GW-GEE-265	0	Add Instrument Root Valve for APP-RCS-PI-310
APP-GW-GEE-270	0	Polar Crane and Cask Handling Crane Design References
APP-GW-GEE-275	0	IHP Lug Design on the Reactor Vessel Closure Head
APP-GW-GEE-278	1	DRAP Component List Changes
APP-GW-GEE-279	0	AP1000 Plant Identifier Codes
APP-GW-GEE-286	1	Next Revision of the AP1000 Emergency Operating Procedures
APP-GW-GEE-299	0	AP1000 Plant Facility Codes
APP-GW-GEE-325	0	Update of Code Affectivity Date for Pipe Supports
APP-GW-GEE-353	0	Update to Eight Inch Squib Valve Calculations