AP1000 Safety Case Overview—Roadmap
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ORIGINATOR(S) SIGNATURE / DATE (If processing electronic approval select option)
D.Popp 12/11/08

REVIEWER(S) SIGNATURE / DATE
J.A.OCilka 12-10-08

VERIFIER(S) SIGNATURE / DATE
R.P. Vijuk 12/11/08

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APPLICABILITY REVIEWER** SIGNATURE / DATE
J. A. Speer 12/11/08

RESPONSIBLE MANAGER* SIGNATURE / DATE
W.E. Cummins 12/11/08

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AP1000 Safety Case Overview – Roadmap

UKP-GW-GL-740, Revision 0
## REVISION HISTORY

<table>
<thead>
<tr>
<th>Report</th>
<th>Description of Change</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td></td>
</tr>
</tbody>
</table>

UKP-GW-GL-740     iii       Revision 0
# TABLE OF CONTENTS

REVISED HISTORY ........................................................................................................................................ iii

1.0 INTRODUCTION ........................................................................................................................................ 1

2.0 AP1000 EUROPEAN DESIGN CONTROL DOCUMENT ......................................................................... 1

   2.1 Chapter Overview ................................................................................................................................. 1
   2.2 Technical Areas ................................................................................................................................... 4

3.0 UK AP1000 ENVIRONMENT REPORT ................................................................................................. 11

4.0 AP1000 PRE-CONSTRUCTION SAFETY REPORT ............................................................................... 13

   4.1 Objectives ........................................................................................................................................ 13
   4.2 Scope ............................................................................................................................................... 14
1.0 INTRODUCTION

The “AP1000 Safety Case Overview – Roadmap” is comprised of three primary documents: the “AP1000 European Design Control Document” (DCD) (EPP-GW-GL-731), the “UK AP1000 Environment Report” (UKP-GW-GL-790), and the “AP1000 Pre-Construction Safety Report” (UKP-GW-GL-732). Provided in the following sections are an overview and roadmap for the content and purpose of each of these documents.

2.0 AP1000 EUROPEAN DESIGN CONTROL DOCUMENT

The DCD serves as the regulatory baseline of the AP1000™ design in regard to the Generic Design Assessment (GDA) application. This document was developed using a format and content guide established in the United States by the United States Nuclear Regulatory Commission (NRC) Regulatory Guide 1.70, “Format and Content Guide for Nuclear Power Plants.” It should be noted that the information provided relates only to the scope of a reactor vendor. References are provided in areas of the DCD where plant Owner information would be required to complete the format and content guide.

The content guide is organized in the following manner: the beginning chapters provide general regulatory and design criteria. This is followed by chapters that provide structure, system, and component design descriptions, and the related regulatory criteria. Chapters 12 to 14 describe radiation protection, Owner programs, and startup testing. Deterministic design bases analyses and probabilistic risk analyses are discussed separately in the DCD format.

2.1 Chapter Overview

The paragraphs below provide a brief discussion of each of the DCD chapters.

Chapter 1, “Introduction and General Description of the Plant,” provides general information about the AP1000 and a list of all reference reports used in the DCD. It also provides a discussion on how the AP1000 complies with U.S. Regulatory Guides as well as how the AP1000 design has addressed safety issues that have been raised over the last 30 years by U.S. utilities and the NRC.

Chapter 2, “Site Characteristics,” contains the bounding site-related information used in the AP1000 design. The site data presented in this chapter provides the Owner/utility with the interface requirements when considering a site for the AP1000. The data includes maximum wind, temperatures, seismic, and cooling water.

Chapter 3, “Design of Structures, Components, Equipment and Systems,” describes the safety and design criteria used as design input for the AP1000 design. This chapter contains a discussion of the general design criteria, and the classification of structures, systems, and components (SSCs) used in design of the AP1000 passive plant. Each SSC is classified according to the function that is being performed. The classification will determine what

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studies and programs need to be applied. For example, if a component is required to shut down the reactor, that component will need to be qualified and tested to perform its function after an earthquake. This chapter also contains criteria for building and piping designs and a discussion on the seismic and environmental programs that will be applied to SSC that performs a safety function.

Chapter 4, “Reactor,” describes the AP1000 reactor fuel and its fuel rods and assemblies, and their placement into the reactor core. A description of the computer programs used to evaluate the nuclear properties of the reactor core and fuel assemblies is also included.

Chapter 5, “Reactor Coolant System and Connected Systems,” describes the reactor coolant system, including details of the reactor system piping, and primary components, including the reactor vessel, reactor coolant pumps, steam generators, and pressurizer.

Chapter 6, “Engineered Safety Features,” describes the containment passive core cooling system and its components. The design bases for the analyses of the containment is also provided.

Chapter 7, “Instrumentation and Controls,” discusses the digital safety and control systems, including the reactor trip and engineered safety features. In addition, discussions are provided on reactor display systems, reactor control system, and diverse actuation system, which is a backup system to the primary safety systems.

Chapter 8, “Electric Power,” describes the offsite and the ac/dc onsite electrical power distribution system.

Chapter 9, “Auxiliary Systems,” provides a discussion of the plant support systems. This includes new and spent fuel storage systems, load handling systems, water systems, along with descriptions of service and other water systems, instrument air, and gas systems, HVAC systems, and reactor auxiliary process systems. Appendix 9A provides a detailed analysis and methodology of fire protection.

Chapter 10, “Steam and Power Conversion,” provides a description of the steam generator feedwater systems, the turbine generator, and the main steam supply systems.

Chapter 11, “Radioactive Waste Management,” contains design descriptions of the solid, liquid, and gaseous waste handling systems as well as the design approach for waste management, radioactive source terms assumed to be present in the reactor coolant, and radiation monitoring system.

Chapter 12, “Radiation Protection,” contains design details on the AP1000, as-low-as-reasonable allowable (ALARA) radiation, radiation sources, radiation zones, and health physics facilities.

Chapter 13, “Conduct of Operations,” discusses the commitments for programs required for an operating plant to be addressed by the site license application.
Chapter 14, “Initial Test Program,” provides a detailed discussion of the AP1000 test program, including the initial test, preoperational tests, startup tests, test procedure, and Owner organization and staffing.

Chapter 15, “Accident Analyses,” discusses the design bases and accident analyses performed for the AP1000 design, including the initial conditions, computer programs used, and general criteria applied. Appendix 15A provides for the evaluation of radiological consequences of accidents, and Appendix 15B analyzes the removal of airborne activity following a loss-of-coolant accident (LOCA).

Chapter 16, “Technical Specifications,” contains the plant operating safety levels and administrative controls for the reactor plant systems.

Chapter 17, “Quality Assurance,” is an overview of the Westinghouse quality programs for the AP1000 design construction and operation, including the Design, Reliability Assurance Program (D-RAP). D-RAP identifies risk-significant (SSCs) in the plant for inclusion in the Maintenance Rule (MR) Program using probabilistic, deterministic, and other methods.

Chapter 18, “Human Factors Engineering,” describes the programs, engineering processes, and analyses that compare the human factors process.

Chapter 19, “Probabilistic Risk Assessment (PRA),” is a summary of the PRA Level 1, 2, and 3 programs that were formulated for the AP1000 design.

Level 1 analysis includes the following:

- Internal initiating events evaluation
- Event tree and success criteria analyses
- Plant systems analysis using fault tree models
- Common cause failure and human reliability analyses
- Data analysis
- Fault tree and event tree quantification to calculate the core damage frequency

Level 2 analysis includes the following:

- An evaluation of severe accident phenomena and fission product source terms
- Modeling of the containment event tree and associated success criteria
- Analysis of hydrogen burning and mixing

Level 3 analysis is an offsite dose evaluation.

The low power and shutdown analysis includes Level 1 shutdown assessment.

External events analyses include the following:

- Internal fire assessment
- Internal flooding assessment
- Seismic margin assessment
• High winds assessment
• External flooding assessment
• Transportation and nearby facility accident assessment

Chapter 20, “Decommissioning,” describes the AP1000 preferred strategy and features of AP1000 that can allow a more efficient decommissioning process.

Chapter 21, “Construction Verification Process,” discusses the construction verification program required by the NRC to be implemented on all plants constructed in the United States. The chapter describes the Inspections, Tests, Analyses, and Acceptance Criteria (ITACC) that are to be verified by the plant Owner in the United States. The information on the Verification Process is provided to allow confidence that standard plants constructed in the UK will be verified during construction in the same degree as the program specified for the U.S. constructed plants.

2.2 Technical Areas

The following paragraphs describe some specific technical areas discussed in the DCD.

Primary Circuit and Its Main Characteristics – DCD Section 5.2

The primary circuit of the AP1000 reactor retains most of the general design features of current Westinghouse designs with added evolutionary features to enhance the safety and maintainability of the system. The system consists of two heat transfer circuits, each with a single hot leg and two cold legs, a steam generator, and two reactor coolant pumps installed directly onto the steam generator to eliminate the primary piping between the pumps and steam generator. A simplified support structure for the primary systems reduces inservice inspections and improves accessibility for maintenance.

Reactor Core and Fuel Design – DCD Chapter 4

The core, reactor vessel, and reactor internals of the AP1000 are similar to those of conventional Westinghouse pressurized water reactor (PWR) designs. Several important enhancements, all based on existing technology, have been used to improve the performance characteristics of the design. Fuel performance improvements include ZIRLO™ grids, removable top nozzles, and longer burnup features. The AP1000 core incorporates the Westinghouse ROBUST fuel assembly design compared to the Vantage 5-H design of the AP600. The reactor core is comprised of 157, 14-foot (426.7 mm), 17x17 fuel assemblies. The AP1000 core design provides a robust design with at least 15 percent in departure from nucleate boiling margin.

The materials that serve to attenuate neutrons originating in the core, and gamma rays from both the core and structural components, consist of the core shroud, core barrel, and associated water annuli. These are within the region between the core and the pressure vessel.

The core consists of three radial regions with different enrichments; the enrichment of the fuel ranges from 2.35 to 4.95 percent. The temperature coefficient of the reactivity of the core is
negative. The core is designed for a fuel cycle of 18 months with a 93-percent capacity factor, and region average discharge burnups as high as 60,000 MWd/t.

The AP1000 uses reduced-worth control rods (termed “grey” rods) to achieve daily load follow without requiring changes in the soluble boron concentration. The use of grey rods has both operational and environmental benefits. The use of grey rods, in conjunction with an automated load follow control strategy, eliminates the need for processing thousands of gallons of water per day to change the soluble boron concentration. As a result, systems are simplified through the elimination of boron processing equipment (such as evaporator, pumps, valves, and piping). With the exception of the neutron absorber materials used, the design of the grey rod assembly is identical to that of a normal control rod assembly.

**New Fuel Storage – DCD Section 9.1.1**

New fuel is stored in a high-density rack, which includes integral neutron absorbing material to maintain the required degree of subcriticality. The rack is designed to store fuel of the maximum design basis enrichment. The new fuel rack includes storage locations for 72 fuel assemblies. Minimum separation between adjacent fuel assemblies is sufficient to maintain a subcritical array even in the event the building is flooded with unborated water and fire extinguishing aerosols, or during any design basis event.

**Spent Fuel Storage – DCD Section 9.1.2**

Each AP1000 plant has its own spent fuel pond and associated handling systems (fuel handling machine and 150-ton (136.078-metric ton) single failure proof cask handling crane). The capacity is 889 storage locations; five of those locations can handle defective fuel cells. These five cells are oversized to handle special debris containers able to store unconsolidated fuel-bare fuel rods. The storage pond is good for 18+ years of operation, considering an 18-month fuel cycle with 64 assemblies discharged. A criticality calculation has been completed that provides the burnup limits for this two discrete zone spent fuel rack system (Region I racks for new fuel and freshly discharged fuel, and Region II for storage of less reactive fuel). Long-term storage beyond 18 years is to be addressed by the site license application. Spent fuel storage is both site- and utility-dependent.

**Steam Generators – DCD Section 5.4.2**

Two model Delta-125 steam generators are incorporated in the AP1000 design. The high reliability of the steam generator design is based on an enhanced and a proven design. The steam generator design is based on the following proven operating models: Delta-75 replacement steam generators for V. C. Summer and other plants; Delta-94 replacement steam generator for the South Texas plant; replacement steam generators (1500 MWt per steam generator) for Arkansas (AN0); and San Onofre and Waterford steam generator designs with capacities similar to the AP1000 steam generators. The steam generators operate on all volatile treatment secondary side water chemistry. Steam generator design enhancements include full-depth hydraulic expansion of the tubes in the tube sheets, nickel-chromium-iron Alloy 690 thermally treated tubes on a triangular pitch, broached tube support plates, improved antivibration bars, upgraded primary and secondary moisture separators, enhanced maintenance features, and a primary-side channel head design that allows for easy access and
maintenance by robotic tooling. All tubes in the steam generator are accessible for sleeving, if
necessary.

**Pressurizer – DCD Section 5.4.5**

The AP1000 pressurizer is a conventional design, based on proven technology. The
pressurizer volume is 2100 ft³ (59.5 m³). The large pressurizer avoids challenges to the plant
and operator during transients, which increases transient operation margins and results in a
more reliable plant with fewer reactor trips. It also eliminates the need for fast-acting
power-operated relief valves, a possible source of reactor coolant system leakage and
maintenance.

**Reactor Coolant Pumps – DCD Section 5.4**

The high-inertia reactor coolant pumps are highly-reliable, low-maintenance, hermetically
sealed canned-motor pumps, which circulate the reactor coolant through the reactor core, loop
piping, and steam generators. The AP1000 pump is based on the AP600 canned-motor pump
design with provisions to provide more flow and a longer flow coastdown.

The motor size is minimized through the use of a variable speed controller to reduce motor
power requirements during cold coolant conditions. Two pumps are mounted directly in the
channel head of each steam generator. This configuration eliminates the cross-over leg of
coolant loop piping; reduces the loop pressure drop; simplifies the foundation and support
system for the steam generator, pumps, and piping; and reduces the potential for uncovering of
the core by eliminating the need to clear the loop seal during a small LOCA. The reactor
coolant pumps have no seals. This eliminates the potential for seal failure LOCA, which
significantly enhances safety and reduces pump maintenance. The pumps use flywheels to
increase the pump rotating inertia. The increased inertia provides a slower rate-of-flow
coastdown to improve core thermal margins following the loss of electric power. Testing has
validated the manufacturability and operability of the pump flywheel assembly.

**Reactor Coolant Piping – Section 5.2**

The reactor coolant system loop layout contains several important features that provide for a
significantly simplified and safer design. The reactor coolant pumps mount directly on the
channel head of each steam generator, which allows the pumps and steam generator to use the
same structural support. This greatly simplifies the support system and provides more space
for pump and steam generator maintenance. The combined steam generator/pump vertical
support is a single pinned column extending from the floor to the bottom of the channel head.
The integration of the pump suction into the bottom of the steam generator channel head
eliminates the crossover leg of coolant loop piping. This avoids the potential for core
uncovery due to loop seal venting during a small LOCA.

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

**Passive Core Cooling System – DCD Section 6.2**

The passive core cooling system protects the plant against reactor coolant system leaks and ruptures of various sizes and locations. The passive core cooling system provides the safety functions of core residual heat removal, safety injection, and depressurization. Safety analyses (using approved codes) demonstrate the effectiveness of the passive core cooling system in protecting the core following reactor coolant system break events up to a full double-ended rupture of a reactor coolant system main loop pipe.

**Passive Residual Heat Removal (PRHR) – DCD Section 5.4.14**

The passive core cooling system includes a PRHR heat exchanger (HX). The PRHR HX is connected through inlet and outlet lines to reactor coolant system loop 1. The PRHR HX protects the plant against transients that upset the normal steam generator feedwater and steam systems. The PRHR HX satisfies the safety criteria for loss of feedwater, feedwater line breaks, and steam line breaks. The in-containment refueling water storage tank (IRWST) provides the heat sink for the PRHR HX. The IRWST water volume is sufficient to absorb decay heat for more than one hour before the water begins to boil. Once boiling starts, steam passes to the containment. This steam condenses on the steel containment vessel and, after collection, drains by gravity back into the IRWST. The PRHR HX and the passive containment cooling system provide indefinite decay heat removal capability with no operator action required.

**Passive Containment Cooling – DCD Section 6.2**

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

The steel containment vessel provides the heat transfer surface that removes heat from inside the containment and rejects it to the atmosphere. Heat is removed from the containment vessel by continuous natural circulation flow of air. During an accident, the air cooling is supplemented by evaporation of water. The water drains by gravity from a tank, sized for 3 days of storage, located on top of the containment shield building. If additional water is necessary, water can be provided from the ancillary water storage tank, which is part of the fire protection system, or offsite water sources.
Main Control Room Emergency Habitability – DCD Section 6.4

The main control room emergency habitability system provides fresh air, cooling, and pressurization to the main control room following a plant accident. Operation of the main control room emergency habitability system is automatically initiated upon receipt of a high main control room radiation signal, which isolates the normal control room ventilation path and initiates pressurization. Following system actuation, all functions are completely passive. The main control room emergency habitability system air supply is contained in a set of compressed air storage tanks. The main control room emergency habitability system also maintains the main control room at a slight positive pressure to minimize the infiltration of airborne contaminants from the surrounding areas.

Containment Isolation – DCD Section 6.2.4

AP1000 containment isolation is significantly improved over that of conventional PWRs. One major improvement is the large reduction in the number of penetrations. Furthermore, the number of normally open penetrations is reduced by 60 percent. There are no penetrations required to support post-accident mitigation functions (the canned motor reactor coolant pumps do not require seal injection, and the passive residual heat removal and passive safety injection features are located entirely inside containment).

Chemical and Volume Control System – DCD Section 9.3.4

The chemical and volume control system consists of regenerative and letdown heat exchangers; demineralizers and filters; makeup pumps; tanks; and associated valves, piping, and instrumentation. The chemical and volume control system is designed to perform the following major functions:

- Purification – maintain reactor coolant purity and activity level within acceptable limits.
- Reactor coolant system inventory control and makeup – maintain the required coolant inventory in the reactor coolant system; maintain the programmed pressurizer water level during normal plant operations.
- Chemical shim and chemical control – maintain reactor coolant chemistry during plant startups, normal dilution to compensate for fuel depletion, and shutdown boration; and provide the means for controlling the reactor coolant system pH by maintaining the proper level of lithium hydroxide.
- Oxygen control – provide the means for maintaining the proper level of dissolved hydrogen in the reactor coolant during power operation and for achieving the proper oxygen level before startup after each shutdown.

Normal Residual Heat Removal System – Section 5.4.7

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

- **Shutdown heat removal** – remove residual and sensible heat from the core and the reactor coolant system during plant cooldown and shutdown operations. The normal residual heat removal system provides reactor coolant system cooldown from 350° to 120°F (177° to 48.9°C) within 96 hours after shutdown. The system maintains the reactor coolant temperature at or below 120°F (48.9°C) during plant shutdown.

- **Shutdown purification** – provide reactor coolant system and refueling cavity purification flow to the chemical and volume control system during refueling operations.

- **IRWST cooling** – provide cooling to the IRWST to limit the IRWST water temperature to less than 212°F (100°C) during extended operation of the passive residual heat removal system and to not greater than 120°F (48.9°C) during normal operation.

- **Low pressure reactor coolant system makeup and cooling** – provide low pressure makeup from the cask loading pit and then the IRWST to the reactor coolant system and provide additional margin for core cooling.

- **Low temperature overpressure protection** – provide low temperature overpressure protection for the reactor coolant system during refueling, startup, and shutdown operations.

- **Spent fuel pool cooling** – provide backup for cooling the spent fuel pool.

### Spent Fuel Pool Cooling System – DCD 9.1.3

The spent fuel pool cooling system removes decay heat generated by stored fuel assemblies from the water in the spent fuel pool. This is done by pumping the high temperature water from within the fuel pool through a heat exchanger, and then returning the water to the pool. A secondary function of the spent fuel pool cooling system is clarification and purification of the water as follows:

- **Spent fuel pool cooling** – remove heat from the water in the spent fuel pool during operation to maintain the pool water temperature within acceptable limits.

- **Spent fuel pool purification** – provide purification and clarification of the spent fuel pool water during operation.

- **Refueling cavity purification** – provide purification of the refueling cavity during refueling operations.

- **Water transfers** – transfer water between the IRWST and the refueling cavity during refueling operations.

- **IRWST purification** – provide purification and cooling of the IRWST during normal operation.
Turbine Generator Plant – DCD Section 10.2

The AP1000 turbine is a power conversion system designed to change the thermal energy of the steam flowing through the turbine into rotational mechanical work, which rotates a generator to provide electrical power. The turbine generator is intended not only for base load operation, but also for load follow capability. Mechanical design of the turbine root and rotor steeple attachments uses optimized contour to significantly reduce operational stresses. Steam flow to the high pressure turbine is controlled by two floor-mounted steam chests. Each contains two throttle/stop valve assemblies and two load-governing valves.

Condensate and Feedwater Systems – DCD Section 10.4.7

The condensate and feedwater system supplies the steam generators with heated feedwater in a closed steam cycle using regenerative feedwater heating. The condensate and feedwater system is composed of the condensate system, the main and startup feedwater system, and portions of the steam generator system. The condensate system collects condensed steam from the condenser and pumps condensate forward to the deaerator. The feedwater system takes suction from the deaerator and pumps feedwater forward to the steam generator system using high pressure main feedwater pumps. The steam generator system contains the safety-related piping and valves that deliver feedwater to the steam generators. The condensate and feedwater systems are located within the turbine building and the steam generator system is located in the auxiliary building and containment.

The main feedwater system includes three 33-1/3 percent single speed motor-driven feedwater pumps, which operate in parallel and take suction from the associated feedwater booster pumps. The discharge from the main feedwater pumps is supplied to the high-pressure feedwater heaters and then to the steam generator system.

The feedwater cycle consists of seven stages of feedwater heating with three parallel string, stages 1 and 2 low pressure feedwater heaters located in the condenser neck, with the next two parallel string, stages 3 and 4 low pressure heaters, deaerator, and two string sixth and seventh high pressure heaters located within the turbine building. The condenser hotwell and deaerator storage capacity allows margin in the design. This margin, coupled with three 50-percent condensate pumps, provides greater flexibility and the ability for an operator to control feedwater and condensate transients.

Reactor Protection System – DCD Sections 7.2 and 7.3

The AP1000 provides instrumentation and controls to sense accident situations and initiate engineered safeguards features. The occurrence of a limiting fault, such as a LOCA or a secondary system break, requires a reactor trip plus actuation of one or more of the engineered safeguards features. This combination of events prevents or mitigates damage to the core and reactor coolant system components, and provides containment integrity.

The protection and safety monitoring system provides the safety-related functions necessary to shut down the plant, and to maintain the plant in a safe shutdown condition. The protection and safety monitoring system controls safety-related components in the plant that may be operated from the main control room or remote shutdown workstation.
3.0 UK AP1000 ENVIRONMENT REPORT

The UK Environment Agency (EA) has issued its “Process and Information Document for Generic Assessment of Candidate Nuclear Power Plant Designs” (P&I Document) that describes the information on waste management and environmental issues that the EA needs to perform a generic assessment of new nuclear power plants. Specifically, this document addresses requirements for waste management of new plants for England and Wales. The EA reviewed previous Westinghouse submittals, including the “AP1000 European Design Control Document” (formerly known as the UK AP1000 Safety, Security, and Environment Report), and issued RI-AP1000-001 to provide guidance on the additional information needed. The UK AP1000 Environment Report provides that additional information. The Environment Report is a supplement of the information in DCD Chapter 11. The Environment Report presents an option for managing waste at an AP1000 on a generic basis.

The section numbers in this report as shown below are from the reference numbers in Table 1, “Information to Be Provided” of the EA’s P&I Document.

Section 1, GENERAL

Section 1.1, “Management System” – The Management System for the development of the design and production of the submission is discussed.

Section 1.2, “General Facility Information” – General information relating to the facility, including identification of plants, systems, and processes that have a bearing on conventional and radioactive waste (solid, liquid, and gaseous) generation, treatment, measurement, assessment, and disposal is presented.

Section 1.3, “Generic Site Characteristics” – The generic site characteristics that the EA is to take into account when assessing the environmental impact of the AP1000 design are presented.

Section 1.4, “Waste and Spent Fuel Strategy” – A proposed waste and spent fuel strategy based on the expected waste generation and management practices throughout the facility lifecycle is discussed, including the use of cement encapsulation of resin intermediate level waste and compaction of compactable low-level waste.

Section 1.5, “Best Available Techniques Assessment” – An analysis of the AP1000 design that included an evaluation of options considered to minimize the production and discharge or disposal of waste showed that Best Available Techniques is used. This includes reviewing the standard AP1000 design and analyzing waste management options included in the “UK AP1000 Environment Report.”

Section 2, RADIOACTIVE WASTE AND SPENT FUEL ISSUES

Section 2.1, “Radioactive Waste Arisings, Management, and Disposal Limits” – A design for an option of how low-level and intermediate-level wastes can be treated and stored at an AP1000 is developed. This discusses an option for encapsulating resin intermediate level waste and onsite storage of the resulting waste packages.
Section 2.2, “Design Basis Estimates for Monthly Discharges of Gaseous and Liquid Radioactive Waste” – Monthly radioactive discharges over an 18-month fuel cycle for an AP1000 were estimated. The variability considered the results of a fault analysis and included events such as startup, shutdown, and maintenance. Of the plants compared to AP1000, when the values were normalized to an annual basis and 1000 MW output, AP1000 generally had lower discharges than similar plants, and essentially the same as the remaining one.

Section 2.3, “Proposed Annual Limits of Radioactive Gaseous and Liquid Discharges” – Proposed annual limits of radioactive gaseous and liquid discharges from a generic AP1000 site are identified.

Section 2.4, “Design Basis Estimates of Annual Arisings of Solid Radioactive Waste” – Annual and life-of-plant estimates of solid radioactive wastes generated during the operational phase of an AP1000 and estimates of solid radioactive wastes generated during decommissioning of an AP1000 are developed.

Section 2.5, “Spent Fuel Management” – How spent fuel will be managed, including storage in the spent fuel pool and storage in an optional dry underground storage facility, is described.

Section 2.6, “Measurement and Assessment of Discharges and Disposals of Radioactive Waste” – AP1000 sampling arrangements, techniques, and systems proposed for measurement and assessment of discharges and disposals of radioactive waste, and supporting reasoning are analyzed.

Section 2.7, “Prospective Dose Assessment for the Generic Site” – Prospective dose for the generic site is assessed at the proposed limits for levels of discharge. The total dose to the AP1000 design critical group is well below the UK dose constraint.

Section 2.8, “Collective Dose Assessments for Discharges at 500 Years” – Collective dose assessments for discharges from the facility truncated at 500 years to the UK, European, and world populations are provided.

Section 2.9, “Assumed Data for Others to Perform Dose Assessments” – Data required for AP1000 dose assessments by using assumed data on the generic site characteristics, AP1000 design information, and critical group habit data are collated.

Section 2.10, “Assessment of Likely Impact of Radioactive Discharges on Non-Human Species” – The likely impact of the radioactive discharges on non-human species is assessed. The results indicate that the atmospheric emissions from the AP1000 nuclear power plant will cause negligible impact on the reference organisms beyond the site boundary, 200 m distant from the emission point. The results also predict negligible risk to the reference marine organisms beyond 100 m from the point of discharge (150 m offshore).

Section 3, ENVIRONMENTAL MATTERS OTHER THAN RADIOACTIVITY

Section 3.1, “Environmental Impact of Cooling,” – Potential generic environmental impacts on the marine and coastal environment are identified and assessed in relation to the AP1000 cooling water system.
Section 3.2, “Analysis of Chemical Liquid Waste Streams,” – How liquid waste streams will arise, be managed, and be disposed of throughout the facility’s lifecycle is analyzed.

Section 3.3, “Pollution Prevention and Control,” – The applicability of Environmental Permitting Regulations to the AP1000 nuclear power plant combustion activities is addressed. The standby diesel generators are the only non-nuclear activity at the AP1000 nuclear power plant potentially falling within the regulations, but they fall below the threshold for the regulations and do not need to be permitted as fossil fuel fired combustion devices.

Section 3.4, “Storage of COMAH Substances,” – The need for onsite storage of substances above the qualifying thresholds in “The Control of Major Accident Hazards Regulations” was identified. The generic AP1000 site is a lower tier COMAH site due to the onsite storage of hydrazine and sulfuric acid in quantities above the lower tier thresholds.

4.0 AP1000 PRE-CONSTRUCTION SAFETY REPORT

The Westinghouse AP1000 is an advanced and passively safe PWR with an output capability of 1117MWe and an expected service life of 60 years. Its design includes passive safety features not present on the Generation 2 plants in service today, and extensive plant simplifications to enhance nuclear safety and facilitate the construction, operation, and maintenance of the plant.

4.1 Objectives

The aim of this generic “AP1000 Pre-Construction Safety Report” (PCSR) is to demonstrate to UK regulatory organizations and potential operating organizations that an AP1000 constructed on a generic UK site can, through plant design and through-life management, satisfy the legal requirements associated with plant safety throughout the plant life cycle. These requirements are defined as the following:

- Risk associated all radiological aspects of the plant should be as low as reasonably practicable (ALARP)
- Risk and radiological dose associated with the plant meets all targets defined by the UK regulatory organizations

The AP1000 PCSR presents claims and arguments to show that the AP1000 meets these UK safety requirements, referring to the assessments and technical analysis presented Westinghouse documentation for the detailed evidence. Those documents providing the detailed information are also presented to the UK regulator, as part of the GDA process. The PCSR also demonstrates that the AP1000 safety assessment embodies significant elements of relevant UK and international good practice.
4.2 Scope

The PCSR scope encompasses all periods of the plant operational life cycle, the principal of these being the following:

- Commissioning
- Operation
- Refueling (and associated handling of radioactive materials on the site)
- Maintenance
- Radioactive waste management
- Decommissioning

The level of detail associated with each of these stages is a variable, dependent upon the level of design and process definition completed at this time. Areas subject to further development are as follows:

- Site-specific risk and dose information, which will be further developed in the Site-Specific AP1000 PCSR
- Provision of a detailed program and process definition for commissioning, which will be developed in more detail as in the Pre-Commissioning Safety Report (PCMSR)