1. INTRODUCTION AND GENERAL DISCUSSION

1.1 Introduction

On August 24, 2005, General Electric (GE)-Hitachi Nuclear Energy (hereinafter referred to as GEH or the applicant) tendered its application for certification of the economic simplified boiling-water reactor (ESBWR) standard nuclear reactor design with the U.S. Nuclear Regulatory Commission (the NRC or Commission). The applicant submitted this application in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) Part 52, "Licenses, Certifications, and Approvals for Nuclear Power Plants," Subpart B, "Standard Design Certifications." The application included the ESBWR design control document (DCD) and the ESBWR probabilistic risk assessment (PRA). The NRC formally accepted the application as a docketed application for design certification (Docket No. 52-010) on December 1, 2005.

The DCD information is divided into two categories, denoted as Tier 1 and Tier 2. Tier 1 means the portion of the generic design-related information that is proposed for approval and certification, including, among other things, the inspections, tests, analyses, and acceptance criteria (ITAAC). Tier 2 means the portion of the generic design-related information proposed for approval but not certification. Tier 2 information includes, among other things, a description of the design of the facility required for a final safety analysis report by 10 CFR 50.34, "Contents of Applications; Technical Information." Subsequently, the applicant supplemented the information in the DCD by providing revisions to that document. The applicant submitted the most recent version, DCD Revision 7, to the Commission on March 29, 2010.

Throughout the review, the NRC staff (staff) requested that the applicant submit additional information to clarify the description of the ESBWR design. This report discusses some of the applicant's responses to these requests for additional information (RAIs). Appendix E to this report provides a list of the issuance and response dates for each RAI the staff submitted to the applicant. The DCD, PRA, Tier 1 information, and all other pertinent information and materials are available for public inspection at the NRC Public Document Room and the Agencywide Documents Access and Management System, Public Electronic Reading Room.

This final safety evaluation report (FSER) summarizes the staff's safety review of the ESBWR design against the requirements of 10 CFR Part 52, Subpart B, and delineates the scope of the technical details considered in evaluating the proposed design. In addition, this FSER documents the resolution of the open and confirmatory items identified in the safety evaluation report (SER) with open items for the ESBWR design. Appendix G to this report includes a copy of the report by the Advisory Committee on Reactor Safeguards (ACRS) required by 10 CFR 52.53, "Referral to the ACRS."

As described above, the applicant supplemented the information in the DCD by providing revisions to the document. The staff has completed its review of the most recent version of the DCD, as documented throughout this report, and, for the reasons set forth herein, finds it to be acceptable.

Sections 1.2 and 1.3 of this chapter summarize the ESBWR design. Section 1.4 identifies the agents and contractors who provided design services to the applicant or other support for the design. Section 1.5 describes the performance of new safety features included in the ESBWR design. Section 1.6 includes the material referenced from topical and technical reports. Section 1.7 contains drawings and other detailed information for the ESBWR design.

Section 1.8 provides ESBWR design interfaces with standard designs. Section 1.9 describes the ESBWR design conformance with regulatory guidance. Section 1.10 provides an index of exemptions for the ESBWR design. Section 1.11 discusses Tier 2* information. Section 1.12 discusses combined license (COL) information items. Section 1.13 discusses RAIs. Section 1.14 provides a list of references.

1.1.1 Metrication

This report conforms to the Commission's policy statement on metrication published in the *Federal Register* on June 19, 1996. Therefore, all measures are expressed as metric units, followed by English units in parentheses. The unit of air volume flow was converted from standard cubic feet per minute at 14.7 pounds-force per square inch absolute and 68 degrees Fahrenheit (F) to standard cubic meters per hour at 760 millimeters of mercury and 0 degrees Celsius (C).

1.1.2 Proprietary Information

This report references several GEH reports. Some of these reports contain information that the applicant requested be held exempt from public disclosure, as provided by 10 CFR 2.390, "Public Inspections, Exemptions, Requests for Withholding." For each report, the applicant provided a nonproprietary version, similar in content except for the omission of the proprietary information. The staff predicated its findings on the proprietary versions of these documents, which are those primarily referenced throughout this report.

1.1.3 Combined License Applicants Referencing the ESBWR Design

Future applicants who reference the ESBWR standard design for specific facilities will retain architect-engineers, constructors, and consultants, as needed. As part of its review of an application for a COL referencing the ESBWR design, the staff will evaluate, for each plant-specific application, the technical competence of the COL applicant and its contractors to manage, design, construct, and operate a nuclear power plant. COL applicants will also be subject to the requirements of 10 CFR Part 52, Subpart C, "Combined Licenses," and any requirements resulting from the staff's review of this standard design. Throughout the DCD, the applicant identified matters to be addressed by plant-specific applicants as "COL information items." This report also refers to such matters as "COL information items" throughout. Appendix F to this report provides a list of COL information items identified in the DCD and this report.

1.1.4 Additional Information

Appendix A to this report provides a chronology of the principal actions, submittals, and amendments related to the processing of the ESBWR application. Appendix B provides a list of references identified in this report. Appendix C contains definitions of the acronyms and abbreviations used throughout this report. Appendix D lists the principal technical reviewers who evaluated the ESBWR design. Appendix E provides an index of the staff's RAIs and the applicant's responses. Appendix F contains a list of the COL information items contained in the DCD. Appendix G includes a copy of the letter from the Advisory Committee on Reactor Safeguards providing the results of its review of the ESBWR design.

The NRC licensing project managers assigned to the ESBWR standard design review are Amy Cubbage, Dennis Galvin, Bruce Bavol, and David Misenhimer. They may be reached by calling (301) 415-7000, or by writing to the U.S. Nuclear Regulatory Commission, Office of New Reactors, Washington, DC 20555-0001.

1.2 General Design Description

1.2.1 Scope of the ESBWR Design

The requirement that governs the scope of the ESBWR design can be found in 10 CFR 52.47 which requires that an applicant for certification provide a complete design scope, except for site-specific elements. Therefore, the scope of the ESBWR design must include all of the plant structures, systems, and components that can affect the safe operation of the plant, except for its site-specific elements. The applicant described the ESBWR standard design scope in DCD, Tier 2, Revision 7, Section 1.8, including the site-specific elements that are either partially or wholly outside the standard design scope. The applicant also described interface requirements (see DCD, Tier 2, Revision 7, Tables 1.8-1 and 1.8-2) and representative conceptual designs, as required by 10 CFR 52.47(a)(25) and 10 CFR 52.47(a)(26), respectively.

1.2.2 Summary of the ESBWR Design

The plant designed by GEH includes a boiling-water reactor (BWR) nuclear steam supply system (NSSS). The plant would be constructed at any location that meets the parameters identified in Chapter 2 of the DCD, Tier 2, Revision 7. The ESBWR design provides a low-leakage containment vessel, which comprises the drywell and wetwell. The containment vessel is a cylindrical steel-lined reinforced concrete structure integrated with the reactor building. The information presented herein pertains to one reactor unit with an NSSS thermal power rating of up to 4,500 megawatt thermal (MWt). Based on the reference design, the plant operates at an estimated gross electrical power output at a rated power of 1,594 megawatt electric (MWe) and a net estimated electrical power output of approximately 1,535 MWe. These electrical output numbers can vary by as much as 50 MWe, depending on the turbine island design and site-specific conditions. The COL applicant and its equipment suppliers will establish the rated electrical power output based on the turbine island design selected and site-specific conditions and may base the COL application on a lower rated thermal power output to satisfy site-specific environmental parameters.

The goal for the overall plant availability is projected to be 95 percent, considering all forced and planned outages, with a rate of less than one unplanned reactor trip per year. The applicant stated that the plant has a design objective of 60 years without a planned replacement of the reactor vessel.

The ESBWR uses a direct-cycle, natural circulation BWR for normal operation and has passive safety features. Within the containment structure are the reactor, elevated gravity-driven cooling system (GDCS) water pools to passively provide an emergency core cooling system (ECCS), and a raised suppression pool. The ESBWR standard plant includes buildings dedicated exclusively or primarily to housing systems and equipment related to the nuclear system or controlled access to these systems and equipment. Figure 1.1-1 illustrates a conceptual layout showing the approximate relative locations of these buildings, but an individual COL may be arranged differently. A containment boundary is illustrated in Figure 1.1-2, which also shows key features of the safety system configuration. A reactor building design surrounds the containment. The following is a general description of the ESBWR design. Subsequent chapters of this report provide detailed descriptions of the individual systems that make up the ESBWR design.

1.2.2.1 *Principal Design Criteria*

The design includes the following principal plant structures:

- reactor building—houses all safety-related structures, systems, and components, except the main control room (MCR), safety-related distributed control and information system equipment rooms, and spent fuel storage pool; includes the reactor, containment, equipment rooms and compartments outside containment, the refueling area with the fuel buffer pool, and the auxiliary equipment area

- control building—houses the MCR and all safety-related controls outside the reactor building

- fuel building—houses the spent fuel storage pool, its auxiliary equipment, and the lower end of the fuel transfer machine

- turbine building—houses equipment associated with the main turbine and generator and their auxiliary systems and equipment, including the condensate purification system and the process offgas treatment system

- radwaste building—houses equipment associated with the collection and processing of solid and liquid radioactive waste generated by the plant

- electrical building—houses the two nonsafety-related standby diesel generators (DGs) and their associated auxiliary equipment

- service building—houses the equipment and control facilities associated with personnel entry into the reactor building and turbine building, eating areas, radiation protection areas, changing rooms, shops, and offices

- ancillary diesel building—houses the ancillary DGs and their associated support systems

- firewater service complex—consists of two firewater storage tanks and a fire pump enclosure that share a common basemat

The following summarizes the principal plant structures and design criteria provided in the DCD, Tier 2, Revision 7, Section 1.2.1.

<u>General Power Generation (Nonsafety) Design Criteria</u>

The plant is designed to produce electricity from a turbine generator (TG) unit using steam generated in the reactor. Heat removal systems have sufficient capacity and operational adequacy to remove heat generated in the reactor core for the full range of normal operational conditions and anticipated operational occurrences (AOOs). Backup heat removal systems remove decay heat generated in the core when the normal operational heat removal systems become inoperative. The capacity of such systems is adequate to prevent fuel cladding damage. The fuel cladding, in conjunction with other plant systems, is designed to retain integrity, so that the consequences of any failures are within acceptable limits throughout the range of normal operational conditions and AOOs for the design life of the fuel. Control equipment allows the reactor to respond automatically to load changes and AOOs. The reactor power level can be controlled manually.

General Safety Design Criteria

The reactor core is designed so its nuclear characteristics do not contribute to a divergent power transient. The reactor is designed so there is no tendency for divergent oscillation of any operating characteristic, considering the interaction of the reactor with other appropriate plant systems. The design alerts plant operators when limits on the release of radioactive material are approached and provides sufficient indications to determine that the reactor is operating within the envelope of conditions considered safe by plant analysis. Those portions of the nuclear system that form part of the reactor coolant pressure boundary (RCPB) are designed to retain integrity as a radioactive material containment barrier following AOOs and to ensure cooling of the reactor core following accidents. Where positive, precise action is immediately required in response to AOOs and accidents, such action is automatic and requires no decision or manipulation of controls by plant operations personnel.

Safety-related functions rely on equipment of sufficient redundancy and independence so that no single failure of active components, or of passive components, in certain cases, in the long term, prevents performance of the safety-related functions. For systems or components to which Institute of Electrical and Electronics Engineers document 603, revised 1991 (with corrections through January 30, 1995), applies, single failures of either active or passive electrical components are considered, in recognition of the higher anticipated failure rates of passive electrical components, compared to passive mechanical components. Provisions are made for the control of active components of safety-related systems from the control room. The design of safety-related structures, systems, and components includes allowances for natural environmental disturbances, such as earthquakes, floods, and storms, at the station site. Standby electrical direct current (dc) power sources have sufficient capacity to power those safety-related systems requiring electrical power concurrently. Standby electrical power sources allow prompt reactor shutdown and removal of decay heat, even if normal auxiliary power is not available.

The boundary of the containment completely encloses the reactor systems, drywell, and wetwell (or suppression chamber). The containment employs the pressure suppression concept. ECC systems limit the fuel cladding temperature to less than the limit in 10 CFR 50.46, ‘‘Acceptance Criteria for Emergency Core Cooling Systems for Light-Water Nuclear Power Reactors,” in the event of a design-basis loss-of-coolant accident (LOCA). The ECC systems provide for continuity of core cooling over the complete range of postulated break sizes in the RCPB piping. Emergency core cooling initiates automatically when required, regardless of the availability of offsite power supplies and the normal generating system of the station. The control room is shielded against radiation, so that continued occupancy under design-basis accident (DBA) conditions is possible.

Nuclear System Criteria

The fuel cladding is a fission product barrier designed to retain integrity, so that any fuel failures occurring during normal operation do not result in dose consequences that exceed acceptable limits. The fuel cladding, in conjunction with other plant systems, is designed to retain integrity to ensure that dose consequences, as a result of any fuel failures occurring during any AOO, are within acceptable limits. Those portions of the nuclear system that form part of the RCPB are designed to retain integrity as a fission product barrier, during normal operation and following AOOs, and to retain sufficient integrity to ensure core cooling following accidents. The capacity of the heat removal systems provided to remove heat generated in the reactor core for the full range of normal operational transients, as well as for AOOs, is adequate to prevent fuel cladding damage that results in dose consequences exceeding acceptable limits. The reactor is capable of being shut down automatically in sufficient time to prevent fuel cladding damage during AOOs.

Sufficient normal, auxiliary, and standby sources of electrical power allow prompt shutdown and continued maintenance of the station in a safe condition under all credible circumstances. The dc power sources are adequate to accomplish the required safety-related functions under all postulated accident conditions.

Electrical Power System Process Control Criteria

The safety-related dc power systems are designed with four divisions. During AOOs, operation of any three mechanical trains is adequate to safely place the unit in the safe-shutdown condition and meet all other design requirements associated with these events. For LOCAs, operation of any three mechanical trains is adequate to safely place the unit in a safe-shutdown condition. Operation of any two safety-related electrical divisions ensures that three mechanical trains will remain operational. Two nonsafety-related standby DGs start and connect to both safety-related and nonsafety-related loads if other alternating current (ac) power sources are lost. If these nonsafety-related DGs are also inoperable, all safety-related loads are powered by the safety-related divisional batteries. The control room personnel monitor the function of key safety-related electrical systems and components.

1.2.2.2 *Plant Description*

The following summarizes the plant description provided in DCD, Tier 2, Revision 7, Section 1.2.2.

Nuclear Steam Supply

The reactor pressure vessel (RPV) assembly consists of the pressure vessel and its appurtenances, supports, and insulation, and the reactor internals, enclosed by the vessel (excluding the core, incore nuclear instrumentation, neutron sources, control rods, and control rod drives (CRDs)). The RCPB of the RPV retains integrity as a radioactive material barrier during normal operation and following AOOs and retains integrity to contain coolant during DBAs. Certain RPV internals support the core and instrumentation used during a DBA. Other RPV internals direct coolant flow, separate steam from the steam and water mixture leaving the core, hold material surveillance specimens, and support instrumentation used for normal operation.

The RPV, together with its internals, provides guidance and support for the fine-motion control rod drives (FMCRDs). Reactor internals associated with the standby liquid control (SLC) system are used to distribute sodium pentaborate solution when necessary to achieve core subcriticality by means other than inserting control rods. The RPV restrains the FMCRDs to prevent ejection of a control rod connected with a drive, in the event of a postulated failure of drive housing.

The overall RPV height permits natural circulation driving forces to produce abundant core coolant flow. To increase the internal flow-path length, relative to most prior BWRs, a long “chimney” extends from the top of the core to the entrance to the steam separator assembly. A shroud assembly, which extends to the top of the core, supports the chimney and steam separator assembly. The large RPV volume provides a substantial reserve of water above the core, which translates directly into a much longer period of time (compared to prior BWRs) before core uncovery can occur as a result of a feedwater (FW) flow interruption or a LOCA. This gives an extended period of time during which automatic systems or plant operators can reestablish reactor inventory control using any of several normal, nonsafety-related systems capable of injecting water into the reactor. Timely initiation of these systems precludes the need for activation of emergency safety-related equipment. The large RPV volume also reduces the reactor pressurization rates that develop and can eventually lead to actuation of the safety relief valves (SRVs) when the reactor is suddenly isolated from the normal heat sink.

Nuclear Boiler System

The nuclear boiler system (NBS) has the following primary functions:

- to deliver steam from the RPV to the turbine main steam system (TMSS)
- to deliver FW from the condensate and FW system (CFS) to the RPV
- to provide overpressure protection of the RCPB
- to provide automatic depressurization of the RPV in the event of a LOCA, where the RPV does not depressurize rapidly
- with the exception of monitoring the neutron flux, to provide the instrumentation necessary for monitoring conditions in the RPV, such as RPV pressure, metal temperature, and water level

The main steamlines (MSLs) direct steam from the RPV to the TMSS; the FW lines direct FW from the CFS to the RPV; and the RPV instrumentation monitors the conditions within the RPV over the full range of reactor power operation.

Operating Characteristics

The ESBWR uses natural circulation to provide core flow. Natural circulation in the ESBWR occurs because of the density differences between the water in the vessel annulus (outside the shroud and chimney) and the steam and water mixture inside the shroud and chimney. The colder, higher density water in the annulus creates a higher pressure or a driving head when compared to the hotter, lower density fluid (steam and water) in the core and chimney. The energy produced in the core of the reactor heats the water entering at the bottom of the core and begins converting it to a steam and water mixture. In the core, the subcooled water is first heated to the saturation temperature, and then, as more heat is added, the core coolant starts boiling. As the coolant travels upward through the core, the percent of saturated steam increases until, at the exit of the core, the average percentage of saturated steam is approximately 25 weight percent. This steam and water mixture travels upward through the chimney to the steam separators, where centrifugal force separates the steam from the water. The separated, saturated water returns to the volume around the separators, while the slightly “wet” steam travels upward to the steam dryer and eventually out the main steamline nozzles and piping to the turbine.

Cooler FW reenters the vessel at the top of the annulus, where it mixes with the saturated water around the separators and subcools this water. The resulting mixture is subcooled only a few degrees below the saturation temperature. The cooler mixture then travels downward through the annulus to reenter the core. The water therefore forms a recirculation loop within the vessel. The mass of steam leaving the vessel is matched by the mass of FW entering.

The chimney adds height to this density difference; in effect, it provides additional driving head to the circulation process. A forced circulation BWR acts in the same basic manner but uses the internal or external pumps to add driving head to this recirculation flow instead of the elevation head provided by the chimney.

1.2.2.3 *Safety Considerations for the Facility*

Process Radiation Monitoring System

The process radiation monitoring system (PRMS) measures and displays radioactivity levels in process and effluent gaseous and liquid streams, initiates protective actions, and activates alarms in the MCR on high radiation signals. The PRMS provides radiological monitoring during plant operation and following an accident. The PRMS safety-related channel trip signals provide inputs for the generation of protective action signals.

The PRMS has the following primary functions:

- to monitor the various gaseous and liquid process streams and effluent releases and provide MCR display, recording, and alarm capability
- to initiate alarms in the MCR to warn operating personnel of high-radiation activity
- to initiate the appropriate actions and controls to prevent further radioactivity releases to the environment

Area Radiation Monitoring System

The area radiation monitoring system (ARMS) continuously monitors the gamma radiation levels in various key areas throughout the plant and provides an early warning to operating personnel when it detects high radiation levels, so the appropriate action can be taken to minimize occupational exposure.

1.2.2.4 *Engineered Safety Features and Emergency Systems*

Engineered safety features (ESFs) are systems that mitigate the consequences of postulated accidents. The ESFs can be divided into three general groups: (1) fission product containment and containment cooling systems, (2) ECC systems, and (3) control room habitability systems (CRHS).

Each general group has the following systems:

(1) fission product containment and containment cooling systems

- containment system
- passive containment cooling system (PCCS)

(2) ECCS

- GDCS
- automatic depressurization system (ADS)
- isolation condenser (IC) system (ICS)
- SLC system

(3) CRHS: control room habitability area (CRHA) heating, ventilation, and air conditioning (HVAC) subsystem (CRHAVS)

ESFs comprise the systems described below.

<u>Containment System</u>

The ESBWR containment, centrally located in the reactor building, features a pressure suppression design concept. The containment consists of a steel-lined, reinforced concrete containment structure in order to fulfill its design basis as a fission product barrier under the pressure conditions associated with a postulated pipe rupture.

The main features include the upper and lower drywell surrounding the RPV and a wetwell containing the suppression pool that serves as a heat sink during abnormal operations and accidents.

<u>Containment Vessel</u>

The containment structure is a reinforced concrete containment vessel (RCCV). The RCCV supports the upper pools, whose walls are integrated into the top slab of the containment to provide structural capability for LOCA and testing pressures.

<u>Passive Containment Cooling System</u>

The PCCS maintains the containment within its pressure limits for DBAs such as LOCAs. The system is passive and requires no moving components for initiation or operation.

The PCCS consists of six low-pressure, independent steam condenser modules (passive containment cooling condensers) that condense steam on the tube side and transfer heat from the drywell to water in a large cooling pool (IC/PCCS pool), which is vented to the atmosphere. Each PCCS condenser is located in a subcompartment of the IC/PCCS pools. The IC/PCCS pool subcompartments on each side of the reactor building communicate at their lower ends, to enable full use of the collective water inventory, independent of the operational status of any given PCCS condenser.

Each condenser, which is an integral part of the containment, contains a drain line to the GDCS pool and a vent discharge line, the end of which is submerged in the pressure suppression pool.

The PCCS condensers are driven by the pressure difference created between the containment drywell and the wetwell during a LOCA. Consequently, they require no sensing, control, logic, or power-actuated devices for operation.

PCCS vent fans are teed off of each PCCS vent line and exhausted to the GDCS pools. The fans aid in the long-term removal of noncondensable gas from the PCCS for continued condenser efficiency.

The PCCS is classified as safety-related and seismic Category I.

Gravity-Driven Cooling System

The GDCS provides emergency core cooling, in conjunction with the ADS in the event of a LOCA. When an initiation signal is received, the ADS depressurizes the reactor vessel and the GDCS injects sufficient cooling water to maintain the fuel cladding temperatures below the limits defined in 10 CFR 50.46. In the event of a severe accident that results in a core melt, with the molten core in the lower drywell region, GDCS floods the lower drywell cavity region with the water inventory of the three GDCS pools and the suppression pool.

The GDCS is classified as safety-related and seismic Category I. The GDCS instrumentation and the dc power supply are safety-related.

Automatic Depressurization System

The ADS function of the NBS depressurizes the RPV in sufficient time for the GDCS injection flow to replenish the core coolant to maintain the core temperature below design limits in the event of a LOCA. After an accident, it also maintains the reactor depressurized for continued operation of the GDCS, without the need for power.

The ADS consists of SRVs and depressurization valves and their associated instrumentation and controls.

Isolation Condenser System

The ICS removes decay heat after any reactor isolation during power operations. Decay heat removal limits any further pressure rise and keeps the RPV pressure below the SRV pressure setpoint. It consists of four independent trains, each containing a heat exchanger that condenses steam on the tube side and transfers heat by heating and evaporating water in the IC/PCCS pools, which are vented to the atmosphere.

The ICS is initiated automatically on high reactor pressure, a main steam isolation valve (MSIV) closure, or a low water level signal. To start an IC, a condensate return valve and condensate return bypass valve are opened, whereupon the standing condensate drains into the reactor and the steam-water interface in the IC tube bundle moves downward below the lower headers to a point in the main condensate return line. The ICS can also be initiated manually from the MCR. Each IC has a fail-open nitrogen piston-operated condensate return bypass valve that opens if power is lost, or on a low reactor water level signal.

An in-line vessel is located on the condensate return line, downstream of the nitrogen motor-operated valve. The in-line vessel is located on each ICS train to provide additional condensate volume for the RPV.

The ICS is isolated automatically when either a high radiation level or excess flow is detected in the steam supply line or condensate return line. The ICS is also isolated after the depressurization valves have been opened.

The equipment storage pool and reactor well are designed to have sufficient water volume to provide makeup water to the IC/PCCS expansion pools for the initial 72 hours of a LOCA. The ICS pool cross-connect valves open to allow the water in the equipment storage pool and reactor well to flow into the IC/PCCS inner expansion pools.

Standby Liquid Control System

The SLC system provides an alternative method of reactor shutdown (i.e., without control rods) from full power to cold subcritical by injecting a neutron absorbing solution into the RPV.

The SLC system has two independent 50-percent capacity trains, which include piping, valves, accumulator, and instrumentation that can inject a neutron absorber solution into the reactor. The system is designed to operate over the range of reactor pressure conditions, up to the elevated pressures of an anticipated transient without scram (ATWS), and to inject sufficient neutron absorber solution to reach hot subcritical conditions after system initiation. The system is also credited with providing makeup water to the RPV during a LOCA.

Control Room Habitability Area Heating, Ventilation, and Air Conditioning Subsystem

The CBVS includes the CRHAVS and the CBGAVS. The CBGAVS is nonsafety-related and performs no safety-related functions. Portions of the CRHAVS that are safety-related include the CRHA envelope, the emergency filter units, related ductwork, dampers, instrumentation, and controls. The remaining portion of the CRHAVS is nonsafety-related. The CRHAVS serves the CRHA (MCR and associated areas) during normal plant operation, emergency operation, plant startup, and plant shutdown. The CBGAVS serves the general areas of the control building during normal plant operation, plant startup, and plant shutdown. The CRHAVS contains a redundant set of emergency filter units that are capable of being powered by safety-related batteries for the 72-hour passive duration and that supply breathing and pressurization air to the CRHA during a potential radiological release concurrent with a station blackout.

1.2.2.5 *Instrumentation, Control, and Electrical Systems*

The following summarizes the plant description provided in DCD, Tier 2, Revision 7, Sections 1.2.2.2 and 1.2.2.13.

Rod Control and Information System

The purpose of the rod control and information system (RC&IS) is to safely and reliably provide the following:

- control of the reactor power level by controlling the movement of control rods in the reactor core in manual, semiautomatic, and automated modes of plant operations

- display of summary information about control rod positions and status in the MCR

- transmission of FMCRD status and control rod positions and status data to other plant systems (e.g., the nonsafety-related distributed control and information system (N-DCIS))

- automatic control rod run-in function of all operable control rods following a scram (scram follow function)

- automatic enforcement of rod movement blocks (which do not have an effect on the scram insertion function) to prevent potentially undesirable rod movements

- manual and automatic insertion of all control rods by an alternative and diverse method (alternate rod insertion and motor run-in function, respectively).

- enforcement of a preestablished sequence for control rod movement when reactor power is below the low-power setpoint

- enforcement of fuel operating thermal limits when reactor power is above the low-power setpoint

- selected control rod run-in function for mitigating a loss of FW heating event or for reducing power after a load rejection event or a turbine trip (that does not result in scram)

The RC&IS is classified as a nonsafety-related system, has only a nonsafety-related control design basis, and is not required for the safe shutdown of the plant. A failure of the RC&IS does not result in gross fuel damage. However, the rod block function of RC&IS is used to limit the effects of a rod withdrawal error and to prevent violations of local fuel operating thermal limits during normal plant operations. Therefore, the RC&IS is designed to be single-failure proof and highly reliable.

<u>Control Rod Drive System</u>

The CRD system comprises three major elements:

(1) FMCRD mechanisms
(2) HCU assemblies
(3) CRD hydraulic subsystem

The FMCRDs provide electric-motor-driven positioning for normal insertion and withdrawal of the control rods and hydraulic-powered rapid control rod insertion (scram) for abnormal operating conditions. In conjunction with a scram, the FMCRDs also provide an electric-motor-driven run-in of all control rods, as a path to rod insertion that is separate from the hydraulic-powered scram. The hydraulic power required for scram uses high-pressure water stored in the scram accumulator within the individual HCUs. Each HCU is designed to scram up to two FMCRDs. The HCUs also provide the flow path for purge water to the associated drives during normal operation. The CRD hydraulic subsystem supplies high-pressure demineralized water, which is regulated and distributed to provide charging of the scram accumulators, purge water flow to the FMCRDs, and backup makeup water to the RPV when the FW flow is not available.

During power operation, the CRD system controls changes in core reactivity by movement and positioning of the neutron absorbing control rods within the core in fine increments using the FMCRD electric motors, which are operated in response to control signals from the RC&IS.

The CRD system provides rapid control rod insertion (scram) in response to manual or automatic signals from the reactor protection system (RPS), so that no fuel damage results from any plant transient.

<u>Feedwater Control System</u>

The FW control system accomplishes both RPV water level control and FW temperature control. The RPV water level is controlled by manipulating the speed of the FW pumps. The FW temperature is controlled by manipulating the heating steam flow to the seventh-stage FW heaters or by directing a portion of the FW flow around the high-pressure FW heaters. The two functions are performed by two sets of triple-redundant controllers located in separate cabinets.

<u>Neutron Monitoring System</u>

The neutron monitoring system (NMS) (described in DCD Tier 2, Revision 7, Section 7.2.2) provides an indication of neutron flux in the core in all modes of reactor operation. The safety-related NMS functions are the startup range neutron monitor (SRNM), the local power range monitor (LPRM), the average power range monitor (APRM), and the oscillation power range monitor (OPRM), the logic for which resides in the same hardware and software of the APRM. The nonsafety-related subsystems are the automated fixed incore probe (AFIP) and the multichannel rod block monitor (MRBM). The LPRMs and APRMs make up the power range neutron monitor (PRNM) subsystem. The safety-related portions of the NMS are classified as seismic Category I.

The NMS provides signals to the RPS, the RC&IS, the safety system logic and control, the N-DCIS, and the plant automation system. The NMS provides trip signals to the RPS for a reactor scram on rising excessive neutron flux or too short a period for flux generation. The safety-related NMS subsystems consist of four divisions that correspond and interface with those of the RPS. This independence and redundancy ensure that no single failure interferes with the system operation.

<u>Remote Shutdown System</u>

The remote shutdown system (RSS) provides the means to safely shut down the reactor from outside the MCR. The RSS provides remote manual control of the systems necessary to do the following:

- Achieve and maintain a safe (hot) shutdown of the reactor after a scram.
- Achieve a subsequent stable shutdown of the reactor.
- Achieve a subsequent cold shutdown of the reactor.
- Maintain safe conditions during shutdown.

The RSS is classified as a safety-related system. The RSS includes control interfaces with safety-related equipment.

Reactor Protection System

The RPS initiates an automatic and prompt reactor trip (scram) by means of a rapid hydraulic insertion of all control rods whenever selected plant variables exceed preset limits. The primary function is to achieve a reactor shutdown before fuel damage occurs. The RPS also provides reactor status information to other systems and causes one or more alarms in the MCR whenever selected plant variables exceed the preset limits. The RPS is a four-division safety-related protection system, differing from a reactor control system or a power generation system. The RPS and its components are safety-related. The RPS and its electrical equipment are classified as seismic Category I. DCD Tier 2, Revision 7, Section 7.2.1, contains RPS descriptions. The RPS initiates reactor trip signals within individual sensor channels when any one or more of the conditions listed below exists during reactor operation. A reactor scram results from any of these conditions, in accordance with the system logic described below.

- drywell pressure high
- reactor power (neutron flux or simulated thermal power) exceeds operating mode limit
- reactor power rapid increase (short period)
- reactor vessel pressure high
- reactor water level low (Level 3)
- reactor water level high (Level 8)
- MSIVs closed (Run mode only)
- scram accumulator charging water header pressure—low-low
- suppression pool temperature high
- turbine stop valve closure and insufficient turbine bypass available
- turbine control valve fast closure and insufficient turbine bypass available
- main condenser vacuum low
- loss of power to FW pumps (Run mode only)
- operator-initiated manual scram
- reactor mode switch in “Shutdown” position

Electrical Power Distribution System

Depending on the plant operating status, either the plant TG or an offsite power source supplies onsite power. During normal operation, the main generator supplies plant loads through the main and unit auxiliary transformers. A generator breaker allows the unit auxiliary transformers to stay connected to the grid to supply loads by backfeeding from the switchyard when the turbine is not online.

Direct Current Power Supply

The plant’s dc power supply system consists of four independent 250 volt (V) dc safety-related power supply subsystems and five independent nonsafety-related power supply subsystems consisting of three 250 V dc power supply subsystems and two 125 V dc power supply subsystems.

The safety-related dc power supply subsystem provides power to the safety-related uninterruptible ac buses through inverters and to the loads required for safe shutdown.

During a total loss of offsite power, the safety-related system is powered automatically from two nonsafety-related standby onsite ac power supplies. If these are not available, each safety-related division isolates itself from the nonsafety-related system, and safety-related batteries provide uninterrupted power to safety-related loads. In all divisions, the safety-related batteries are divided into two groups that are sized to power various safety-related loads for a period of 72 hours.

The nonsafety-related dc power supply subsystem is normally supplied through nonsafety-related battery chargers from the nonsafety-related power centers. In the event that this power supply is lost, the nonsafety-related batteries supply power. The nonsafety-related batteries are sized for a 2-hour duty cycle. The nonsafety-related dc buses also supply power to the nonsafety-related inverters.

Standby Onsite AC Power Supply

Two separate nonsafety-related ancillary DGs are located onsite to provide separate sources of onsite power for certain regulatory treatment of nonsafety system (RTNSS) functions after the first 72 hours of a DBA. Section 8.3.1.1.9 and Appendix 19A provide additional details. These ancillary DGs are seismic Category II and are located in the ancillary DG building, which is also seismic Category II.

Two separate nonsafety-related standby onsite DGs provide separate sources of onsite power for various load groups when the normal and alternate preferred power supplies are not available. The standby onsite ac power supply system is configured to provide power to the permanent nonsafety-related buses.

Either the main generator or the normal preferred offsite power source normally energizes the plant buses. Transfer to the onsite standby DGs is automatic when no other power supply capable of feeding the buses is available. Should these power supplies fail, their supply breakers trip and the standby onsite power supply (DGs) is automatically signaled to start. After the standby voltage and frequency reach normal values, the standby supply breakers close. After bus voltage is reestablished, large motor loads are sequentially started.

On a defense-in-depth basis, the standby onsite ac power supply system can provide power to important safety-related loads. However, these loads are powered by uninterruptible power supplies (UPSs) (for ac loads) or safety-related dc power from safety-related station batteries, if the preferred power supply or the standby onsite ac power supply is not available.

Uninterruptible AC Power Supply

The safety-related UPS provides redundant, reliable power to the safety-related logic and control functions during normal, upset, and accident conditions.

Each of the four divisions of this safety-related uninterruptible power is separate and independent. Each division is powered from an inverter supplied from the divisional isolation power center and the safety-related dc bus. The dc bus receives its power from a divisional battery charger and battery.

The nonsafety-related UPS system for the two power-distribution load groups in the plant is supplied from the 480 V ac power center in the same group. In addition, another UPS system supplies the N-DCIS loads.

Two dedicated UPS systems supply the Technical Support Center.

Lighting Power Supply

The lighting systems include the normal, standby, emergency, security, and MCR emergency lighting systems. The normal lighting system provides illumination under all normal plant conditions, including maintenance, testing, and refueling operations. It is powered from the nonsafety-related buses. The standby lighting system supplements both the normal lighting system and the emergency lighting system in selected areas of the plant. The standby lighting system normally receives power from the main generator or the offsite power system, or alternatively, from the standby onsite ac power supply system. The normal, emergency, and standby lighting systems are nonsafety-related. See Section 9.5.3 for a detailed description.

Upon loss of the normal lighting system, the emergency lighting system provides illumination in areas where emergency operations are performed (e.g., MCR, remote shutdown station, battery rooms, local control stations, ingress and egress routes). It includes self-contained dc battery-operated units for exit and stair lighting. The illumination ranges of lighting systems in all areas of the plant comply with the standards of the Illuminating Engineering Society of North America.

The emergency lighting system is supplied from the four divisions of the safety-related ac UPS. The emergency lighting fixtures and the raceways carrying cables to the fixtures inside the MCR have seismic Category I support.

1.2.2.6 *Power Conversion System*

Turbine Main Steam System

The TMSS supplies steam, generated in the reactor, to the turbine, moisture separator reheaters, steam auxiliaries, and turbine bypass valves. The TMSS does not include the seismic-interface restraint or main turbine stop or bypass valves. The TMSS does the following:

- It accommodates operational stresses, such as internal pressure and dynamic loads, without failures.
- It provides a seismically analyzed fission product leakage path to the main condenser.
- It includes suitable access and remote functions to permit in-service testing and inspections.
- It closes the steam auxiliary isolation valve(s) on branch lines between the MSIVs and the main turbine stop valves (excluding the fission product leakage path to the condenser) on an MSIV isolation signal. These valves fail closed on a loss of electrical power to the valve actuating solenoid or on loss of pneumatic pressure.

The TMSS main steam piping consists of four lines from the seismic-interface restraint to the main turbine stop valves. The header arrangement upstream of the turbine stop valves allows the valves to be tested online and supplies steam to the power cycle auxiliaries, as needed.

The TMSS is located in the steam tunnel and the turbine building.

Condensate and Feedwater System

The CFS consists of the piping, valves, pumps, heat exchangers, controls, and instrumentation, and the associated equipment and subsystems, that supply the reactor with heated FW in a closed steam cycle using regenerative FW heating. The CFS extends from the main condenser outlet up to, but not including, the seismic-interface restraint outside containment.

The CFS provides a dependable supply of high-quality FW to the reactor at the required flow, pressure, and temperature. The condensate pumps take the deaerated condensate from the condenser hotwell and deliver it through the steam jet-air ejector (SJAE) condenser, the gland steam condenser, the offgas condenser, and the condensate filters and demineralizers, and through three strings of low-pressure FW heaters to the open FW tank.

The CFS does not serve or support any safety function and has no safety design basis. Failure of this system cannot compromise any safety-related systems or prevent safe shutdown.

Condensate Purification System

The condensate purification system (CPS) continuously purifies and treats the condensate, as required, to maintain reactor FW purity, using filtration to remove solid corrosion products and ion exchange to remove condenser leakage and other dissolved impurities. The CPS does not perform or support any safety-related function and, thus, has no safety design basis. No failure within the CPS could prevent safe shutdown. Wastes from the CPS are collected in controlled areas and sent to the radwaste system for treatment or disposal. The CPS is located in the turbine building.

Main Turbine

The main turbine for the ESBWR standard plant has one high-pressure turbine and three low-pressure turbines. Other turbine configurations may be selected for plant-specific applications to obtain the optimal thermal performance of the turbine plant under site-specific conditions. The steam passes through moisture separator reheaters before entering the low-pressure turbines. Steam exhausted from the low-pressure turbines is condensed and degassed in the condenser. Steam is extracted from each turbine and is used to heat the FW.

The control system for the main turbine provides control and monitoring of turbine speed, load, and steam flow for startup, normal operation, and shutdown by operating the main steam turbine stop valves, control valves, and intermediate stop and intercept valves. The main turbine system includes supervisory instrumentation that is provided for startup and shutdown monitoring, operational analysis, and malfunction diagnosis.

The main turbine is equipped with a single-speed, electric-motor-driven turning gear, which is used to rotate the TG shafts slowly and continuously, if needed, when the main turbine is not in service, and especially during startup and shutdown periods, when turbine rotor temperature changes occur.

The TG system is enclosed within the turbine building. The TG is oriented within the turbine building to be in line with the reactor building to minimize the potential for any high energy TG-system-generated missiles to damage any safety-related equipment or structures.

Turbine Gland Seal System

The turbine gland seal system (TGSS) provides seal steam to prevent the escape of radioactive steam from the turbine shaft, casing penetrations, and valve stems and to prevent air in-leakage through subatmospheric turbine glands. The TGSS consists of a sealing steam pressure regulator, a sealing steam header, a gland steam condenser, two full-capacity exhaust blowers, and associated piping, valves, and instrumentation. The TGSS is a nonsafety-related system.

Turbine Bypass System

The turbine bypass system (TBS) can pass steam directly to the main condenser under the control of the steam bypass and pressure control (SB&PC) system. Steam is bypassed to the condenser whenever the reactor steaming rate exceeds the load permitted to pass to the TG. The TBS in the ESBWR standard plant is designed to shed 110 percent of rated steam flow, which facilitates the shedding of 100 percent of the TG rated load without a reactor trip or operation of the SRVs. The SB&PC system provides main turbine control valve and bypass valve flow demands, to maintain a nearly constant reactor pressure during normal plant operation.

The TBS, which does not perform or ensure any safety-related function, is classified as nonsafety-related. No failure within the TBS could prevent safe shutdown. However, the TBS is used to mitigate AOOs (which are defined as part of normal operations in Appendix A, “General Design Criteria for Nuclear Power Plants,” to 10 CFR Part 50, “Domestic Licensing of Production and Utilization Facilities”) and is analyzed to demonstrate structural integrity under the safe-shutdown earthquake loading conditions.

Main Condenser

The main condenser is designed to condense and deaerate the exhaust steam from the main turbine and provide a heat sink for the TBS. The main condenser does not perform, ensure, or support any safety-related function and, thus, has no safety design basis. It is, however, designed with necessary shielding and controlled access to protect plant personnel from radiation.

The main condenser for the ESBWR standard plant is a multipressure, triple-shell unit. However, nothing precludes the use of a single-pressure and parallel (instead of series) circulating water (CIRC) system, because these features have no impact on the nuclear island. Circulating water flows through each of the single-pass tube bundles as cooling water to remove waste heat rejected by the TG cycle.

Circulating Water System

The CIRC system provides cooling water to remove the power cycle waste heat from the main condensers and transfers this heat to the normal power heat sink. The CIRC system does not perform, ensure, or support any safety-related function and, thus, has no safety design basis.

1.2.2.7 *Fuel Handling and Storage Systems*

<u>Refueling Equipment</u>

The reactor building contains a refueling machine to move fuel and to service the RPV. The refueling machine is a gantry-type crane that spans the reactor vessel cavity and the buffer pool to handle fuel and perform other ancillary tasks in the reactor building. It is equipped with a traversing trolley on which is mounted a telescoping mast and integral fuel grapple. The refueling machine is classified as nonsafety-related but is designed as seismic Category I.

The fuel handling platform is only used for fuel servicing and transporting tasks in the fuel building. It is equipped with a traversing trolley on which is mounted a telescoping mast and integral fuel grapple, as well as an auxiliary hoist. The fuel handling machine is classified as nonsafety-related but is designed as seismic Category I.

<u>Fuel Storage Facility</u>

Storage facilities exist for new and spent fuel and associated equipment. New fuel is stored in the new fuel storage racks in the buffer pool of the reactor building. These are side-loading racks of stainless steel construction with neutron absorbing material. This ensures that a full array of loaded fuel remains 5 percent below criticality under all conditions. Adequate water shielding is always maintained in storage pools by the use of level sensors. All storage pools are constructed with stainless steel liners to form a leak-tight barrier. A leak detection system monitors liner integrity.

The thermal-hydraulic design of the rack provides sufficient natural convection cooling flow to remove decay heat without exceeding 100 degrees C (212 degrees F).

1.2.2.8 *Cooling Water and Other Auxiliary Systems*

<u>Reactor Water Cleanup/Shutdown Cooling System</u>

The reactor water cleanup/shutdown cooling (RWCU/SDC) system has the following primary functions:

- Purify the reactor coolant during normal operation and shutdown.
- Transfer sensible and core decay heat produced when the reactor is being shut down or is in the shutdown condition.
- Provide decay heat removal and high-pressure cooling of the primary coolant during periods of reactor isolation (hot standby).
- Remove excess reactor coolant during startup and hot standby.
- Maintain coolant flow from the reactor vessel bottom head to reduce thermal stratification.
- Warm the reactor coolant before startup and vessel hydro testing.

The system consists of two independent trains. Each train includes the following:

- one nonregenerative heat exchanger
- one regenerative heat exchanger
- one low-capacity cleanup (function) pump
- one high-capacity shutdown cooling pump
- one demineralizer
- associated valves and pipes

The RWCU/SDC system is classified as a nonsafety-related system. However, its RCPB and containment isolation functions are safety-related and, thus, those functions are seismic Category I. The electrical power supplies to the two trains are from separate nonsafety-related, diesel-backed electrical buses.

During normal plant operation, the system operates at reduced flow in the cleanup mode, continuously withdrawing water from the RPV. The water is cooled through the heat exchangers and is circulated by the cleanup pump to the demineralizer for removal of impurities. Purified water returns to the regenerative heat exchanger, where it is reheated, and then flows into the FW lines and is returned to the RPV. One train is in operation while the other is on standby.

Fuel and Auxiliary Pools Cooling System

The fuel and auxiliary pools cooling system (FAPCS) consists of two physically separated cooling and cleaning trains, each with 100-percent capacity during normal operation. Each train contains a pump, a heat exchanger, and a water treatment unit for cooling and cleaning pools, except the IC/PCCS pools. A separate subsystem with its own pump, heat exchanger, and water treatment unit is dedicated to cooling and cleaning the IC/PCCS pools, independent of the FAPCS cooling and cleaning train operation during normal plant operation.

A four-valve bridge of motor-operated valves attaches to each end of the FAPCS cooling and cleaning trains. With proper alignment of the motor-operated valves of these bridges, the cooling and cleaning train connects to one of the two pairs of suction and discharge piping loops to establish a flow path for cooling and cleaning the desired pool. One loop provides the flow path for serving the spent fuel pool and auxiliary pools, and the other loop serves the GDCS pools and suppression pool.

The primary design function of FAPCS is to cool and clean pools located in the containment, reactor building, and fuel building, during normal plant operation. Through its piping system, FAPCS provides flow paths for filling and makeup of these pools during normal plant operation and under postaccident conditions, as necessary.

FAPCS is also designed to provide the following accident recovery functions, in addition to the spent fuel pool cooling function:

- suppression pool cooling
- drywell spray
- low-pressure coolant injection of suppression pool water into the RPV
- alternate shutdown cooling

At least one FAPCS cooling and cleaning train is available for continuous operation to cool and clean the water of the spent fuel pool during normal plant operation. The other train can be placed in standby mode or another operating mode. During refueling outages, both trains may be used to provide maximum cooling capacity for cooling the spent fuel pool, if needed.

Each FAPCS cooling and cleaning train has sufficient flow and cooling capacity to maintain the bulk water temperature of the spent fuel pool below the limit under normal heat load conditions. Under the maximum spent fuel pool heat load conditions associated with keeping a full core offload and irradiated fuel in the spent fuel pool for 20 years of plant operations, both trains are needed to maintain the bulk temperature below the limit.

The FAPCS is a nonsafety-related system, with the exception of the piping and components required for the following:
:

- isolating the containment

- refilling the IC/PCCS pools and the spent fuel pool with emergency water supplies from the fire protection system or another onsite or offsite source

- providing the high-pressure interface with the RWCU/SDC system used for low-pressure coolant injection

The FAPCS piping and components that are required to provide safety-related and accident recovery functions have a Quality Group B or C and seismic Category I or II classification.

1.2.2.9 ***Radioactive Waste Management System***

<u>Liquid Waste Management System</u>

The liquid waste management system (LWMS) collects, monitors, and treats liquid radioactive waste for plant reuse whenever practicable. The LWMS consists of the following four subsystems:

(1) equipment (low conductivity) drain subsystem
(2) floor (high conductivity) drain subsystem
(3) chemical drain subsystem
(4) detergent drain subsystem

The LWMS processing equipment is located in the radwaste building. Any discharge is such that concentrations and quantities of radioactive material and other contaminants are in accordance with applicable local, State, and Federal regulations.

All potentially radioactive liquid wastes are collected in sumps or drain tanks at various locations in the plant. These wastes are transferred to collection tanks in the radwaste building.

Waste processing occurs on a batch basis. Each batch is sampled, as necessary, in the collection tanks to determine concentrations of suspended solids and chemical contaminants. Equipment drains and other low-conductivity wastes are treated by filtration, demineralization, or both, and are transferred to the condensate storage tank for reuse. Floor drains and other high-conductivity wastes are treated by filtration, reverse osmosis process, and ion exchange before being either discharged or recycled for reuse. Laundry drain wastes and other detergent wastes of low activity are treated by filtration, sampled, and released through the liquid discharge pathway. Chemical wastes are preconditioned by adding a chemical solution in the chemical drain collector tank and are transferred to floor drain collection tanks for further processing. Design redundancy, instrumentation for the detection and alarm of abnormal conditions, automatic isolation, and administrative controls protect against the inadvertent release of liquid radioactive waste. Processing equipment, such as filtration, demineralization, and the reverse osmosis unit, and cross-connections with each subsystem, augment the waste processing capability and flexibility.

If the liquid is returned to the plant, it meets the purity requirements for condensate makeup. If the liquid is discharged, the activity concentration is consistent with the discharge criteria in 10 CFR Part 20, "Standards for Protection Against Radiation," and the dose commitment in 10 CFR Part 50, Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low as is Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents."

<u>Solid Waste Management System</u>

The solid waste management system (SWMS) is designed to control, collect, handle, process, package, and temporarily store, before shipment, solid radioactive waste generated as a result of normal operation, including AOOs. This includes filter backwash sludge, bead resins generated by the LWMS, FAPCS, RWCU/SDC and condensate systems, and concentrated wastes generated by the LWMS. Contaminated solids, such as high-efficiency particulate air and cartridge filters, rags, plastic, paper, clothing, tools, and equipment, are sorted and packaged into several kinds of waste containers for offsite disposal. There is no liquid plant discharge from the SWMS. The SWMS consists of the following four subsystems:

(1) a waste collection subsystem
(2) a waste processing subsystem
(3) a dry solid waste accumulation and conditioning subsystem
(4) a container storage subsystem

Spent bead resin sluiced from the LWMS, FAPCS, RWCU/SDC and condensate systems is transferred by the waste collection subsystem to one of three spent resin tanks for decay and storage. Filter backwash sludge from the condensate system and LWMS is transferred to one of two phase separators. Concentrated wastes from the LWMS are collected into a concentrated-waste tank.

The waste processing subsystem consists of built-in dewatering stations. High-integrity containers (HICs) are filled with sludge from the phase separator, bead resin from the spent-resin tanks, and concentrated wastes from the concentrated-waste tank. Spent cartridge filters may also be placed in the HIC.

Dry wastes consist of such items as air filters, miscellaneous paper, and rags from contaminated areas; contaminated clothing, tools, and equipment parts that cannot be effectively decontaminated; solid laboratory wastes; and wastes that may not be contaminated. The activity of much of this waste is low enough to permit handling by contact. These wastes are collected in containers located in appropriate areas throughout the plant. The filled containers are sealed and moved to a controlled-access enclosed area for temporary storage. Connections are provided for processing systems to augment the waste processing capability and flexibility.

The radwaste building can provide temporary storage for more than one month's volume of packaged waste. Packaged waste includes HICs, compactor boxes, shielded filter containers, and 208-liter (55-gallon) drums, as necessary.

The SWMS is designed to package the radioactive solid waste for offsite shipment and burial, in accordance with the requirements of the applicable NRC and U.S. Department of Transportation regulations, including Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," 10 CFR Part 61, "Licensing Requirements for Land Disposal of Radioactive Waste," 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," and 49 CFR Part 170 through 49 CFR Part 178.

Gaseous Waste Management System

The gaseous waste management system minimizes and controls the release of gaseous radioactive effluents by delaying, filtering, or diluting various offgas process and leakage gaseous releases, which may contain the radioactive isotopes of krypton, xenon, iodine, and nitrogen. The offgas system (OGS) is the principal gaseous waste management subsystem. The various building HVAC systems perform other gaseous waste functions.

The OGS provides for holdup and decay of radioactive gases in the offgas from the SJAEs and consists of process equipment, along with monitoring instrumentation and control components.

The OGS design minimizes the explosion potential in the offgas process stream by recombining radiolytic hydrogen and oxygen under controlled conditions. Although the OGS is nonsafety-related, it is capable of withstanding an internal hydrogen explosion.

The OGS includes redundant hydrogen/oxygen catalytic recombiners and ambient temperature charcoal beds to provide for process gas volume reduction and radionuclide retention and decay. The system processes the SJAE discharge during plant startup and normal operation before discharging the airflow to the turbine building stack.

A manually operated, three-way switch allows the charcoal adsorbers to operate in (1) AUTO, (2) TREAT or (3) BYPASS mode:

(1) OGS startups are normally made in the AUTO mode, which provides valve alignment to send the offgas only through the first (guard bed) charcoal adsorber.

(2) Normal OGS operation is in the TREAT mode, which provides valve alignment to send the offgas through both the guard bed and the main charcoal adsorber beds.

(3) OGS operation in the BYPASS mode provides valve alignment to allow offgas flow to completely bypass the charcoal adsorbers. However, this mode of operation requires simultaneous actuation of two manual switches by the plant operator from the MCR.

1.3 Comparison with Similar Facility Designs

The ESBWR standard design contains many features that are not found in currently operating reactors. For example, various engineering and operational improvements provide additional safety margins and address Commission policy statements regarding severe accidents, safety goals, and standardization. The most significant improvement to the design is the use of safety systems that rely on passive means, such as gravity, natural circulation, condensation and evaporation, and stored energy, for accident prevention and mitigation.

DCD Tier 2, Revision 7, Section 1.3, highlights the principal design features of the ESBWR and compares its major features with those of other BWR facilities. The ESBWR design is based on proven technology obtained during the development, design, construction, and operation of BWRs of similar types. Comparison tables provided in DCD Tier 2, Revision 7, Section 1.3, include the following:

- reactor system design characteristics, listed in Table 1.3-1
- ECC systems and safety-related containment cooling systems, listed in Table 1.3-2
- containment design characteristics, listed in Table 1.3-3
- structural design characteristics, listed in Table 1.3-4

The COL applicant will update the ESBWR design characteristic values identified in DCD Tier 2, Revision 7, Section 1.4, Table 1.3-1, based on these chapters of the COL application, as required: Chapter 10, turbine heat balance and associated Chapter 1 reactor heat balance; Chapter 4, initial core design and analysis; and Chapter 15, safety analysis.

1.4 Identification of Agents and Contractors

GEH is the principal ESBWR designer. In 2007, GE and Hitachi formed an alliance and GE-Hitachi Nuclear Energy (GEH) became the applicant for the ESBWR. GEH is responsible for the overall design and design certification of the ESBWR nuclear power plant. In addition, GE Nuclear Energy (predecessor to GEH) was the applicant for the first design certified under 10 CFR Part 52, when the NRC issued the advanced BWR design certification rule as Appendix A, “Design Certification Rule for the U.S. Advanced Boiling Water Reactor,” to 10 CFR Part 52, in May 1997 (effective June 11, 1997).

DCD Tier 2, Revision 7, Section 1.0, describes the following major contractors that provided input to the ESBWR DCD under the direction of GEH:

- Shimizu Corporation
- Empresarios Agrupados
- BVZ (Black & Veatch Zachry)
- URS
- ANATECH
- EXCEL Services Corporation
- WorleyParsons
- Panlyon
- Equipos Nucleares SA
- Adecco
- Granite Services International, Inc.

The staff evaluated GEH’s technical qualifications based on the information provided in DCD Tier 2, Revision 7, Chapter 1, and the NRC’s previous experience with GE Nuclear Energy. The staff determined that GEH is technically qualified to supply the ESBWR design certification represented in Revision 7 of the GEH DCD.

1.5 <u>Performance of New Safety Features</u>

The ESBWR, as described in the DCD Tier 2, Revison 7, has major simplifying improvements drawn from predecessor designs, such as pressure-suppression containment, natural circulation, IC handling of waste heat, and gravity-driven makeup water systems. The incorporation of these features from predecessor designs has been accomplished with safety in mind and has emphasized the employment of passive means of dealing with operational transients and hypothetical LOCAs. The result of this particular design assemblage of previously licensed plant features is a simplified operator response to these events. Most plant upset conditions are dealt with in essentially the same manner as is typical for the hypothetical steamline break. In addition, operator response times for all hypothetical events have been extended from minutes for previously licensed reactors to days for the ESBWR. Most features of the ESBWR have been taken directly from licensed commercial BWRs and reviewed and redesigned, as appropriate, for the ESBWR. The ESBWR draws together the best of previously licensed plant features to continue the simplification process.

As described in DCD Tier 2, Revision 7, Section 1.5.2, the applicant used a database of feature performance in licensed reactors, combined with the recent thorough licensing review of the advanced BWR, as a foundation from which to make extrapolations to the ESBWR. To make that extrapolation, GEH developed one computer code (TRACG) to use for design and for three out of the four most limiting licensing analyses. GEH developed the TRACG code, validated by operating plant experience and appropriate testing, to analyze the challenges to the fuel (10 CFR 50.46 and Section 6.3 of Appendix K, "ECCS Evaluation Models," to 10 CFR Part 50), the challenges to the containment (Section 6.2), and many of the AOOs ("Minimum Critical Power Ratio," Chapter 15). The radiological responses to hypothetical accidents (LOCAs) are also presented in Chapter 15 but do not use TRACG for analysis. Thus, TRACG draws from the very large database of licensed BWRs, which includes all features of the ESBWR (albeit in various configurations) and appropriate testing, and allows direct application to the ESBWR design and analysis.

TRACG is used to perform safety analyses of the AOOs, described in DCD Tier 2, Revision 7, Chapter 15, and the American Society of Mechanical Engineers (ASME) reactor vessel overpressure protection event, as described in Section 5.2 and Section 15.5.1.

<u>ATWS Analysis</u>

TRACG is used to evaluate the ATWS events in DCD Tier 2, Revision 7, Chapter 15. The analysis determines the most limiting ATWS events in terms of reactor vessel pressure, heat flux, neutron flux, peak cladding temperature, suppression pool temperature, and containment pressure. The results are used to demonstrate the capability of the ESBWR mitigation design features to comply with the ATWS licensing criteria.

<u>ECCS-LOCA Analysis</u>

TRACG evaluates the complete spectrum of postulated break sizes and locations, together with possible single active failures, in Section 6.3. This evaluation determines the worst-case break and single failure combinations. The results demonstrate the ESBWR ECCS capability to comply with the licensing acceptance criteria.

Sensitivity analyses of important parameters affecting LOCA results are performed using TRACG. For the ESBWR, the LOCA analysis results show no core uncovery for any LOCA.

Based on the sensitivity studies, a bounding calculation is performed for the minimum water level inside the shroud for use as the licensing basis. The ESBWR LOCA results have a large margin with respect to the licensing acceptance criteria.

Containment Analysis

TRACG also evaluates containment response during a LOCA. The analysis determines the most limiting LOCA for the containment (or a DBA), in terms of containment pressure and temperature responses. The DBA is determined from the consideration of a full spectrum of postulated LOCAs. The results demonstrate compliance with the ESBWR containment design limits, as defined in DCD Tier 2, Revision 7, Table 6.2-1. TRACG evaluates the sensitivity of the containment response to parameters identified as important, to assess the effect of uncertainties of these parameters on the containment responses. Based on the sensitivity studies, a bounding calculation is performed for the containment pressure and temperature response for use as the licensing basis.

Testing

In DCD Tier 2, Revision 7, Section 1.5.3, GEH describes the testing programs for performing the ESBWR safety analysis. These programs include: testing the heat transfer correlation for steam condensation in tubes in the presence of noncondensable gases, using the Single Tube Condensation Test Program to investigate steam condensation inside tubes in the presence of noncondensable gases, and testing to provide a basis for extrapolations of the PCCS and the GDCS. Major certification and confirmatory tests that are unique to the ESBWR and applicable to its design are listed below.

GIST (Confirmatory)

GIST was an experimental program conducted by GE to demonstrate the GDCS concept and to collect data to qualify the TRACG computer code for ESBWR applications. Simulations included DBA LOCAs representing an MSL break, a bottom drainline break, a GDCS line break, and a non-LOCA loss of inventory. GE used test data to qualify TRACG for ESBWR applications. GE completed the tests in 1988 and documented them in 1989. GIST data validated certain features of TRACG.

GIRAFFE (Certification)

GIRAFFE is an experimental program conducted by the Toshiba Corporation to investigate thermal-hydraulic aspects of the PCCS. The program included fundamental steady-state tests on condensation phenomena in the PCCS tubes. It also ran simulations of DBA LOCAs; specifically, the MSL break. GIRAFFE data substantiate PANDA and PANTHERS data at a different scale and support validation of certain features of TRACG. Two additional series of tests in the GIRAFFE facility were as follows: The first (GIRAFFE/Helium) demonstrated the operation of the PCCS in the presence of lighter-than-steam noncondensable gas, and the second (GIRAFFE/SIT) provided additional information regarding potential system interaction effects in the late blowdown and early GDCS period.

PANDA (Certification)

PANDA is an experimental program run by the Paul Scherrer Institut in Switzerland. PANDA is a full-vertical-scale, 1/25 volume, model of the simplified BWR (SBWR) system designed to model the thermal-hydraulic performance and post-LOCA decay heat removal of the PCCS. It includes both steady-state and transient performance simulations. Testing under the same thermal-hydraulic conditions as previously tested in GIRAFFE and PANTHERS allows scale-specific effects to be quantified. Before testing started, the NRC received blind pretest analyses using TRACG. PANDA data directly validated certain features of TRACG.

PANTHERS (Certification)

PANTHERS is an experimental program performed by SIET in Italy, with the dual purpose of providing data for TRACG qualification and demonstration testing of the prototype PCCS and IC heat exchangers. Steam and noncondensable gases were supplied to prototype heat exchangers over the complete range of SBWR conditions to demonstrate the capability of the equipment to handle post-LOCA heat removal. Testing occurred under the same thermal-hydraulic conditions as in GIRAFFE and PANDA. Before testing started, the NRC received blind pretest analyses of selected test conditions using TRACG. PANTHERS data directly validate certain features of TRACG.

In addition to thermal-hydraulic testing, an objective of PANTHERS was to demonstrate the structural adequacy of the heat exchangers to exceed the SBWR/ESBWR expected lifetime requirement. The program accomplished this by pre- and post-test nondestructive examination, following cycling of the equipment in excess of requirements.

Additional PANDA Tests (Confirmatory)

The PANDA test facility also ran a supplementary program known as TEPSS to test an earlier ESBWR configuration with the GDCS pool connected to the wetwell gas space, rather than the drywell. These tests confirm the expected increased margin to the containment design pressure for this ESBWR configuration. This series of tests also included injection of helium, providing data on PCCS performance with light noncondensable gases at an additional scale.

Scaling of Tests

References 1.5-13 and 1.5-14 in DCD Tier 2, discuss scaling of the major SBWR and ESBWR tests. These reports contain a complete discussion of the features and behavior of the SBWR and ESBWR during challenging events. The analysis includes the general (top-down approach) scaling considerations, the scaling of specific (bottom-up approach) phenomena, and the scaling approach for the specific tests discussed above. The scaling analysis shows that the SBWR and ESBWR tests represent the ESBWR response, without significant distortions, and can be used to qualify the TRACG code for ESBWR applications.

1.6 Material Referenced

DCD Tier 2, Revision 7, Tables 1.6-1 and 1.6-2, contain all topical reports and technical reports that are incorporated by reference as part of the application. The staff reviewed these reports.

1.7 Drawings and Other Detailed Information

DCD Tier 2, Revision 7, Chapter 1, Table 1.7-1, lists the standard piping designations and specifications used in the drawings; Table 1.7-2 provides a summary of the electrical, instrumentation, and control system configuration drawings; and Table 1.7-3 provides a summary of the mechanical system configuration drawings.

1.8 Interfaces with Standard Designs

For a design certification application, the NRC requires the applicant to address interface requirements for those design features that are outside the scope of the certified design, as identified by the applicant, and a representative conceptual design for those portions of the plant for which the application does not seek certification; 10 CFR 52.47(a)(24) requires a conceptual design and 10 CFR 52.47(a)(25) sets forth interface requirements for out-of-scope portions of the design. ITAAC, required by 10 CFR 52.47(b)(1), apply only to in-scope portions of the design and are not related to 10 CFR 52.47(a)(24) and (25).

DCD Tier 2, Revision 7, Table 1.8-1, cross-references the NSSS safety-related systems and supporting interface areas with the matching portions of the plant and the associated section(s) where they are described. The DCD addresses all interface requirements for safety-related systems.

DCD Tier 2, Revision 7, Table 1.8-2, cross-references the balance of plant (BOP) systems and supporting interface areas with the matching portions of the plant and the associated section(s) where they are described. Except for postaccident MCR atmosphere control, the ESBWR has no safety-related BOP system (i.e., all service, cooling, and makeup water and all other HVAC systems are nonsafety-related). Therefore, the applicant did not intend Table 1.8-2 to address all of the BOP systems; however, Table 1.8-2 does address the major BOP systems.

The ESBWR DCD Tier 2, Revision 7, includes designs for the following BOP systems: CIRC system, plant service water system (PSWS), offsite power transmission system, makeup water system, potable and sanitary water systems, station water system, and independent spent fuel storage installation. The staff evaluated the overall acceptability of these designs.

An applicant for a COL that references the ESBWR certified design must provide design features or characteristics that comply with the interface requirements for the plant design and ITAAC for the site-specific portion of the facility design, in accordance with 10 CFR 52.80(a).

DCD Tier 1, Revision 7, Chapter 4, identifies these interfaces for the conceptual design portion of the PSWS and offsite power for the certified design.

Plant Service Water System

The PSWS is the heat sink for the reactor component cooling water system (RCCWS). The PSWS does not perform any safety-related function, and there is no interface with any safety-related component. The PSWS provides the nonsafety-related functions to support the post-72-hour cooling for RCCWS. The PSWS must have the volume of water necessary to accommodate losses caused by such things as evaporation and drift, without makeup water, for 7 days, using the most limiting condition of operation as defined by the PRA model. The volume maintained must also ensure that the PSWS pumps have sufficient available net positive suction head at the pump suction location for the lowest probable water level of the heat sink. The most limiting condition equates to 2.02×10^7 megajoules (MJ) (1.92×10^{10} British thermal units (BTU)) over a period of 7 days.

The PSWS cooling towers and basins are not within the scope of the certified design. A specific design for this portion of the PSWS shall be selected for any facility that has adopted the certified design. The plant-specific portion of the PSWS shall meet the interface requirements defined below.

The interface requirements are necessary to support the post-72-hour cooling function of the PSWS. The volume of water shall be sufficient such that no active makeup shall be necessary to remove 2.02×10^{7} MJ (1.92×10^{10} BTU) over a period of 7 days. Additionally, the PSWS pumps must have sufficient available net positive suction head at the pump suction location for the lowest probable water level of the heat sink. Consequently, verification of compliance with the interface requirements shall be achieved by inspections, tests, and analyses that are similar to those provided for the certified design. The COL applicant referencing the certified design shall develop these inspections, tests, and analyses, together with their associated acceptance criteria.

Offsite Power

The offsite portion of the preferred power supply (PPS) consists of at least two electrical circuits and associated equipment that interconnect the offsite transmission system with the plant's main generator and the onsite portions of the PPS. The PPS consists of the normal preferred and alternate preferred power sources and includes those portions of the offsite and onsite power systems required for power flow from the offsite transmission system to the safety-related incoming line breakers of the isolation power centers.

The interface between the normal preferred ESBWR certified plant onsite portion of the PPS and the site-specific offsite portion of the PPS is at the switchyard side terminals of the high-side motor-operated disconnect of the unit auxiliary transformer circuit breaker and main generator circuit breaker. The interface between the alternate preferred ESBWR certified plant onsite portion of the PPS and the site-specific offsite portion of the PPS is at the switchyard side terminals of the reserve auxiliary transformer high-side motor-operated disconnects.

A COL applicant referencing the ESBWR certified design shall develop an ITAAC to verify that the as-built offsite portion of the PPS from the transmission network to the interface with the onsite portions of the PPS satisfy the applicable provisions of General Design Criterion 17, "Electric Power Systems," in Appendix A to 10 CFR Part 50. Specifically, the ITAAC shall verify the following:

- At least two independent circuits supply electric power from the transmission network to the interface with the onsite portions of the PPS.
- Each offsite circuit interfacing with the onsite portions of the PPS is adequately rated to supply the load requirements during design-basis operating modes
- During steady-state operation, the offsite portion of the PPS is capable of supplying voltage at the interface with the onsite portions of the PPS that will support operation of safety-related loads during design-basis operating modes.
- During steady-state operation, the offsite portion of the PPS is capable of supplying the required frequency at the interface with the onsite portions of the PPS that will support the operation of safety-related loads during design-basis operating modes.

- The fault current contribution of the offsite portion of the PPS is compatible with the interrupting capability of the onsite fault current interrupting devices.

1.9 <u>Conformance with Regulatory Guidance</u>

<u>Regulatory Guides</u>

DCD Tier 2, Revision 7, Tables 1.9-20, 1.9-21, and 1.9-22, contain Standard Review Plans (SRPs), branch technical positions, regulatory guides, and industrial codes and standards that are applicable to the ESBWR design. Applicable revisions are also shown. The applicability column of Tables 1.9-20 and 1.9-21 refers to whether the requirement is applicable during design certification of the ESBWR. SRPs, branch technical positions, and regulatory guides that apply only during detailed design, construction, fabrication, and erection are indicated by a dash in the applicability column and a comment.

To simplify the licensing process for future COL applicants, the ESBWR incorporates ASME Code Case N-782 by reference. This ASME Code case endorses the use of the edition and addenda of ASME Boiler and Pressure Vessel Code Section III, Division 1, that was applied during design certification as an alternative to the requirements of ASME Code paragraphs NCA-1140(a)(2)(a) and NCA-1140(a)(2)(b).

<u>Conformance with the Review Guidance</u>

DCD Tier 2, Revision 7, Section 1.9.1, provides the information required by 10 CFR 52.47(a)(9), showing conformance with the SRP. Tables 1.9-1 through 1.9-19 summarize the differences from requirements in each SRP section on a section-by-section basis. If no difference is indicated, the ESBWR design does not deviate from the requirements in the SRP section. SRP sections with deviations provide a reference location for additional information.

<u>Generic Issues and Three Mile Island Requirements</u>

DCD Tier 2, Revision 7, Table 1.9-23, lists NUREGs that the ESBWR DCD included as references. Appendix 1C addresses the applicability of NRC generic letters and bulletins. Chapter 20 of this report addresses unresolved safety issues and medium- and high-priority generic safety issues identified in NUREG-0933, “Resolution of Generic Safety Issues,” issued July 2008. Chapter 20 also discusses the Three Mile Island requirements set forth in 10 CFR 50.34(f).

<u>Operational Experience (Generic Communications)</u>

Chapter 20 of this report discusses how the ESBWR design incorporates operating experience insights from generic letters and bulletins.

1.10 Index of Exemptions

In accordance with 10 CFR 52.4B, the staff used the current regulations in 10 CFR Part 20,; 10 CFR Part 50; 10 CFR Part 73, "Physical Protection of Plants and Materials," and 10 CFR Part 100, "Reactor Site Criteria," in reviewing the GEH application for certification of the ESBWR design. During this review, the staff recognized that the application of certain regulations to the ESBWR design would not serve the underlying purpose of the rule or would not be necessary to achieve the underlying purpose of the rule.

GEH submitted two exemption requests in letters dated March 10, 2009, and March 12, 2010. The following sections of this report discuss these exemptions:

Section	Exemption
5.2.1.1.3	Request for alternative to ASME Code, Section III, NCA-1140 requirements
18.8.3.2	Request for exemption from regulations in 10 CFR 52.47(a)(8) and 10 CFR 50.34(f)(2)(iv)

1.11 Index of Tier 2* Information

The NRC staff has determined that certain changes to, or departures from, information in the DCD that are proposed by an applicant or licensee who references the certified ESBWR design will require NRC approval before it can implement the change, in accordance with the design certification rule. The proposed rule will refer to this information as Tier 2*.

In 2010, GEH submitted proposed DCD Tier 2, Revision 8, Table 1D-1, listing the items designated as Tier 2* information. The staff has completed its review of the Tier 2* information pertaining to the ESBWR design. For the reasons set forth throughout this report regarding Tier 2* information, the staff finds such information acceptable.

1.12 COL Information Items

COL applicants and licensees referencing the certified ESBWR standard design must satisfy the requirements and commitments identified in the DCD. The ESBWR DCD Tier 2, Revision 7, identifies certain general commitments as "COL information items." These items relate to programs, procedures, and issues that are outside the scope of the certified design review. They do not establish requirements; rather, they identify an acceptable set of information to be included in a plant-specific safety analysis report. An applicant for a COL must address each of these items in its application. It may deviate from or omit these items, provided that the deviation or omission is identified and justified in the plant-specific safety analysis report.

1.13 Requests for Additional Information

RAIs are questions asked of GEH by the NRC staff concerning the application. The NRC sent questions to GEH in letters and GEH responded to the NRC staff in letters. Appendix E of this report lists these letters.

The nomenclature for RAIs concerning the DCD took one of the following two forms:

(1) XX.Y-Z, where XX was the DCD Chapter number, Y was the section number, and Z was the RAI sequence number.

(2) XX.Y.A-Z, where XX was the DCD Chapter number, Y was the section number, A was the subsection number, and Z was the RAI sequence number.

1.14 References

U.S. Nuclear Regulatory Commission, "Safety Evaluation Report with Open Items for the ESBWR Design,"

GE-Hitachi Nuclear Energy Americas LLC (GEH)

— — — — —, "General Electric Company Application for Final Design Approval and Design Certification of ESBWR Standard Plant Design," August 2005, (ADAMS Accession No. ML052490334).

— — — — —, "ESBWR Standard Plant Design Certification Application Design Control Document, Revision 7, Tier 1 and Tier 2," March 2010, (ADAMS Accession No. ML101960315).

— — — — —, "Request for Exemption from 10 CFR 52.47(a)(8) and 10 CFR 50.34(f)(2)(iv)," March 10, 2009, (ADAMS Accession No. ML090690565).

— — — — —, "Request for Addition of ASME Boiler and Pressure Vessel Code Case N-782 to ESBWR DCD," March 12, 2010, (ADAMS Accession No. ML100710759).

Institute for Electrical and Electronics Engineers (IEEE)

— — — — —, IEEE Std 603-1991, IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations,"

U.S. Code of Federal Regulations

— — — — —, *Title 10, Energy*, Part 2, ""Rules of Practice For Domestic Licensing Proceedings and Issuance of Orders."

— — — — —, *Title 10, Energy*, Part 20, "Standards for Protection Against Radiation."

— — — — —, *Title 10, Energy*, Part 50, "Domestic Licensing of Production and Utilization Facilities."

— — — — —, *Title 10, Energy*, Part 52, "Energy Site Permits; Standard Design Certifications; and Combined Licenses for Nuclear Power Plants."

— — — — —, *Title 10, Energy*, Part 61,"Licensing Requirements for Land Disposal of Radioactive Waste."

— — — — —, *Title 10, Energy*, Part 71, "Packaging and Transportation of Radioactive Material."

— — — — —, *Title 10, Energy*, Part 73, "Physical Protection of Plants and Materials."

— — — — —, *Title 10, Energy*, Part 100, "Reactor Site Criteria."

— — — — —, *Title 49,* 49 CFR Parts 170-178

U.S. Nuclear Regulatory Commission

Commission Papers

— — — — —, SECY-93-087, "Policy, Technical and Licensing Issues Pertaining to Evolutionary and Advanced Light-Water Rector (AWLR) Designs," April 1993 (ADAMS Accession No. ML003708021).

NUREG-Series Reports

— — — — —, NUREG-0933 "Resolution of Generic Safety Issues," Main Report with Supplements 1-32, July 2008 (ADAMS Accession No. ML082410719).

Regulatory Guides

— — — — —, Regulatory Guide 1.143, Design Guidance for Radioactive Waste Management Systems; Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," November 2001 (ADAMS Accession No. 013100305).

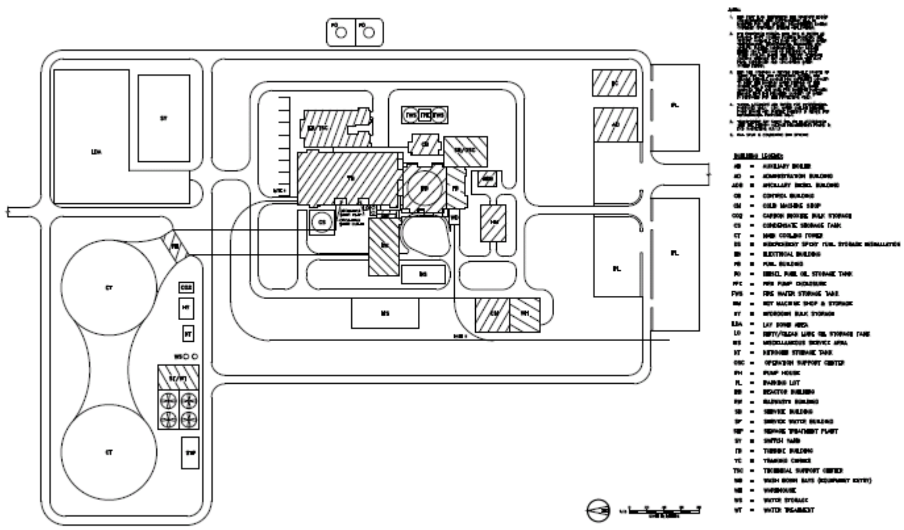

Figure 1.1-1 ESBWR Standard Plant General Site Plan

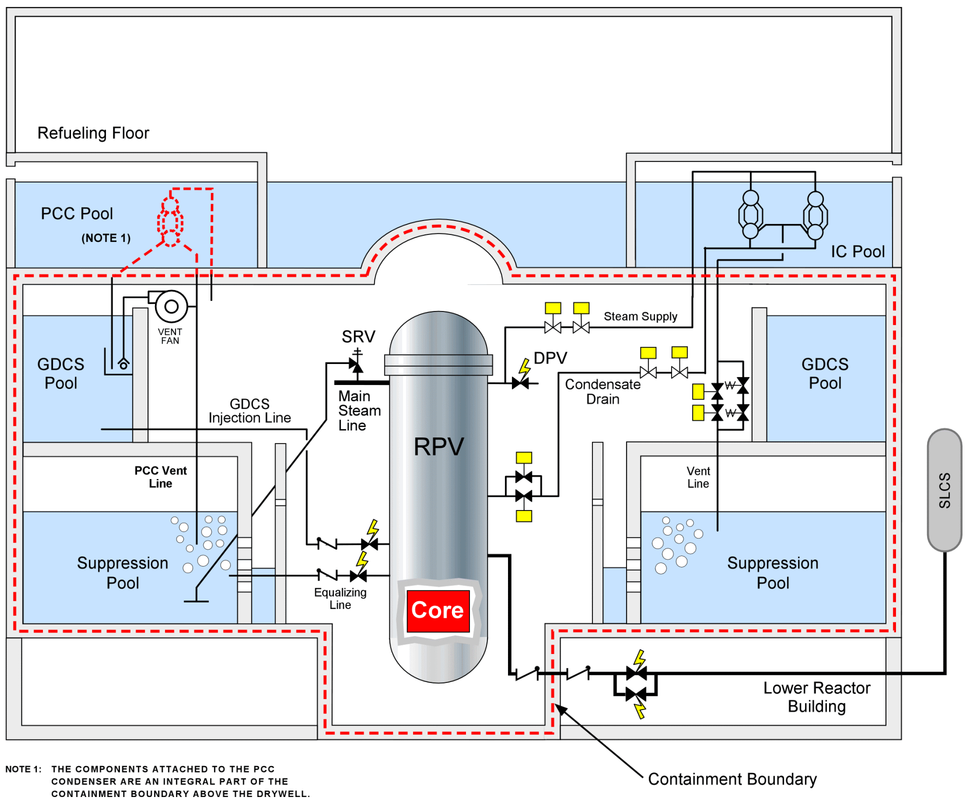

Figure 1.1-2 Safety System Configuration (not to scale)