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1.0 Introduction and General Description of Plant

1.1 Introduction

1.1.1 Format and Content

Tier 2 is written in accordance with Regulatory Guide (RG) 1.70. For consistency with NUREG-0800, Tier 2 includes Section 15.8, which addresses anticipated transients without scram (ATWS), and Chapter 18, which addresses human factors. In addition, GE's response to TMI related matters is presented in Appendix 1A. Appendix 1C describes the ABWR station blackout considerations.

GE's response to the severe accident policy statement is provided in Chapter 19 of Tier 2. Chapter 20 is included to provide a Question and Response guide. Chapter 21 provides the engineering drawings.

1.1.2 ABWR Standard Plant Scope

The ABWR Standard Plant includes all buildings which are dedicated exclusively or primarily to housing systems and the equipment related to the nuclear system or controls access to this equipment and systems. There are five such buildings within the scope of the ABWR Standard Plant:

(1) Reactor Building (including containment)

(2) Service Building

(3) Control Building

(4) Turbine Building

(5) Radwaste Building

In addition to these buildings and their contents, the ABWR Standard Plant provides the supporting facilities shown in Figure 1.2-1. A detailed listing of structures and systems for the ABWR Standard Plant scope of design is provided in Table 3.2-1.

The ABWR evolutionary design provides an essentially complete nuclear power plant except for site-specific elements. The site-specific elements are included as representative conceptual designs with interface requirements sufficient for the final safety analysis and design-specific probabilistic risk assessment in accordance with 10CFR52.47(a) (1) (vii) and (b) (1). Unless otherwise noted, the following site-specific elements are outside the scope of the ABWR Standard Plant:

(1) Ultimate heat sink (9.2.5), interfaces with reactor service water (spray pond, conceptual)

(2) Offsite power (8.2.4), transmission (The offsite power transmission network is out of scope starting from the low voltage terminals of the main and reserve transformers, reference conceptual design is provided.)

(3) Makeup water (9.2.8), preparation (well and treatment facilities, conceptual)

(4) Potable and sanitary water systems (9.2.4), partial (Portions inside the buildings of Figure 1.2-1 are in scope. All other portions are conceptual and outside scope of the standard ABWR design.)

(5) Reactor service water (9.2.15), rejects heat to the ultimate heat sink, partial (Portions inside the buildings of Figure 1.2-1 are in scope. All other portions are conceptual [pumps, valves, pipes, strainers and other equipment (Figure 9.2-7)] and are outside the scope of the standard ABWR design.)

(6) Turbine service water (9.2.16), rejects heat to the power cycle heat sink, partial (Portions inside the buildings of Figure 1.2-1 are in scope. All other portions are conceptual [pumps, valves, pipes, strainers and other equipment (Figure 9.2-8)] and are outside the scope of the standard ABWR design.)

(7) Communications (9.5.2), partial (Communication equipment inside the buildings of Figure 1.2-1 are in scope. All other portions, including connections to offsite networks are outside the scope of the standard ABWR design.)

(8) Site security (13.6.2)

(9) Circulating water system (10.4.5), circulates water (Portions inside the buildings of Figure 1.2-1 are in scope. All other portions are conceptual [pumps, valves, pipes, strainers and other equipment (Figure 10.4-3)] and are outside the scope of the standard ABWR design. This system includes the power cycle heat sink which provides a heat sink for the Circulating Water and Turbine Service Water Systems-cooling tower with makeup water and chemical control, conceptual

(10) Heating, ventilating and air conditioning (9.4), partial (Involving potential need for toxic gas monitors).

A detailed listing of the above site-specific elements is also provided in Table 3.2-1.

1.1.3 Engineering Documentation

Engineering documentation for the ABWR Standard Plant is listed on Master Parts List

(MPL) No. 18NS07A03*. This MPL is a controlled list, structured by system, which contains the identification of hardware and software documentation that defines the ABWR Standard Plant.

1.1.4 Design Process

GE and its associates control the review and approval of ABWR Common Engineering design documents with a procedure using the Engineering Review Memorandum (ERM). Evidence of design verification is entered into the design records of the responsible design organization. For engineering documents prepared uniquely by GE for the U.S. ABWR, changes to engineering documents are entered into the GE design record files. A COL applicant will establish the design, including the supporting detailed design documentation, consistent with the design control document referenced in the certified design rule. See Subsection 1.1.11.1 for COL license information requirements.

1.1.5 Type of License Required

Tier 2 is submitted in support of the application for design certification (DC) for the ABWR Standard Plant.

1.1.6 Number of Plant Units

For the purpose of this document, only a single standard plant will be considered.

1.1.7 Description of Location

This plant can be constructed at any location which meets the parameters identified in Chapter 2.

1.1.8 Type of Nuclear Steam Supply

This plant will have a boiling water reactor (BWR) nuclear steam supply system (NSSS) designed and supplied by GE and designated as ABWR.

1.1.9 Type of Containment

The ABWR will have a low-leakage containment vessel which comprises the drywell and pressure suppression chamber. The containment vessel is a cylindrical steel-lined reinforced concrete structure integrated with the Reactor Building. The containment nomenclature is specified in Figure 1.1-1.

* GE Proprietary

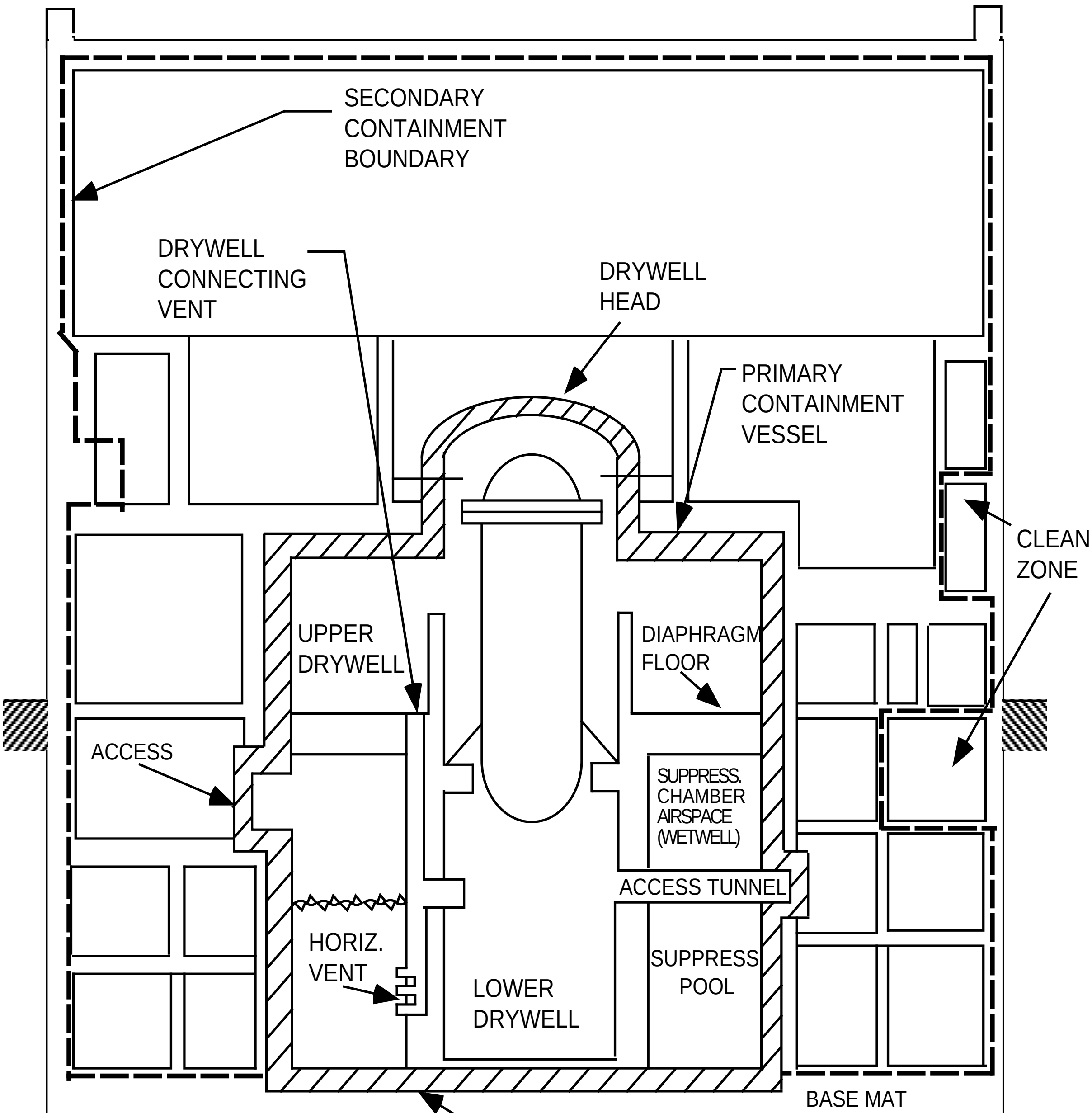
1.1.10 Core Thermal Power Levels

The information presented in Tier 2 pertains to one reactor unit with a rated power level of 3926 MWt and a design power level of 4005 MWt. The station utilizes a single-cycle, forced-circulation BWR. The heat balance for rated power is shown in Figure 1.1-2. The station is designed to operate at a gross electrical power output of approximately 1356 MWe and net electrical power output of approximately 1300 MWe.

1.1.11 COL License Information

1.1.11.1 Design Process to Establish Detailed Design Documentation

The COL applicant will provide the design process required to establish the detailed design documentation (see Subsection 1.1.4).



Note: This drawing is to illustrate the scope and requirements of the text and is not intended to show the final detail of the design.

Figure 1.1-1 ABWR Standard Plant Nomenclature

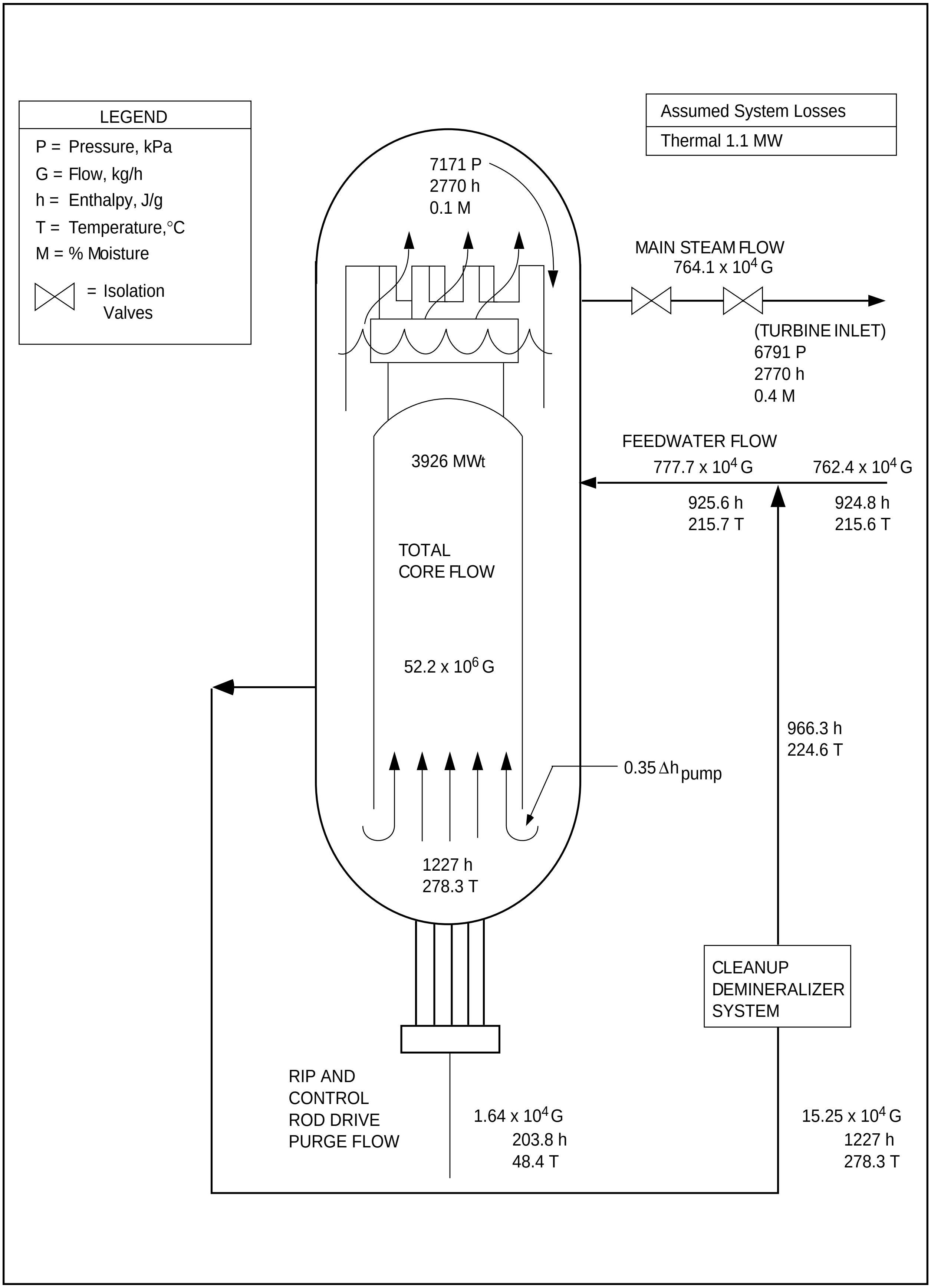


Figure 1.1-2 Heat Balance at Rated Power

1.2 General Plant Description

1.2.1 Principal Design Criteria

The principal design criteria are presented in two ways. First, they are classified as either a power generation function or a safety function. Second, they are grouped according to system. Although the distinctions between power generation or safety functions are not always clear-cut and are sometimes overlapping, the functional classification facilitates safety analyses, while the grouping by system facilitates the understanding of both the system function and design.

1.2.1.1 General Design Criteria

1.2.1.1.1 Power Generation Design Criteria

(1) The plant is designed to produce steam for direct use in a turbine-generator unit.

(2) Heat removal systems are provided with sufficient capacity and operational adequacy to remove heat generated in the reactor core for the full range of normal operational conditions and abnormal operational transients.

(3) Backup heat removal systems are provided to remove decay heat generated in the core under circumstances wherein the normal operational heat removal systems become inoperative. The capacity of such systems is adequate to prevent fuel cladding damage.

(4) 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 abnormal operational transients for the design life of the fuel.

(5) Control equipment is provided to allow the reactor to respond automatically to load changes and abnormal operational transients.

(6) Reactor power level is manually controllable.

(7) Control of the reactor is possible from a single location.

(8) Reactor controls, including alarms, are arranged to allow the operator to rapidly assess the condition of the reactor system and locate system malfunctions.

(9) Interlocks or other automatic equipment are provided as backup to procedural control to avoid conditions requiring the functioning of nuclear safety systems or engineered safety features.

(10) The station is designed for routine continuous operation whereby steam activation products, fission products, corrosion products, and coolant dissociation products are processed to remain within acceptable limits.

1.2.1.1.2 Safety Design Criteria

(1) The station design conforms to applicable codes and standards as described in Subsection 1.8.2.

(2) The station is designed, fabricated, erected, and operated in such a way that the release of radioactive material to the environment does not exceed the limits and guideline values of applicable government regulations pertaining to the release of radioactive materials for normal operations, abnormal transients, and accidents.

(3) The reactor core is designed so its nuclear characteristics do not contribute to a divergent power transient.

(4) 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.

(5) The design provides means by which plant operators are alerted when limits on the release of radioactive material are approached.

(6) Sufficient indications are provided to allow determination that the reactor is operating within the envelope of conditions considered safe by plant analysis.

(7) Radiation shielding is provided and access control patterns are established to allow a properly trained operating staff to control radiation doses within the limits of applicable regulations in any mode of normal plant operations.

(8) 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 abnormal operational transients and accidents.

(9) Nuclear safety systems and engineered safety features function to assure that no damage to the RCPB results from internal pressures caused by abnormal operational transients and accidents.

(10) Where positive, precise action is immediately required in response to abnormal operational transients and accidents, such action is automatic and requires no decision or manipulation of controls by plant operations personnel.

(11) Safety-related actions are provided by equipment of sufficient redundance and independence so that no single failure of active components, or of passive components in certain cases in the long term, will prevent the required actions.

(12) Provisions are made for control of active components of safety-related systems from the control room.

(13) Safety-related systems are designed to permit demonstration of their functional performance requirements.

(14) The design of safety-related systems, components and structures includes allowances for natural environmental disturbances such as earthquakes, floods, and storms at the station site.

(15) Standby electrical power sources have sufficient capacity to power all safety-related systems requiring electrical power concurrently.

(16) Standby electrical power sources are provided to allow prompt reactor shutdown and removal of decay heat under circumstances where normal auxiliary power is not available.

(17) A containment is provided that completely encloses the reactor systems, drywell, and suppression chambers. The containment employs the pressure suppression concept.

(18) It is possible to test primary containment integrity and leaktightness at periodic intervals.

(19) A secondary containment is provided that completely encloses the primary containment above the Reactor Building basemat. This secondary containment provides for a controlled, monitored release of any potential radioactive leakage from the primary containment.

(20) The primary containment and secondary containment, in conjunction with other safety-related features, limit radiological effects of accidents resulting in the release of radioactive material to the containment volumes to less than the prescribed acceptable limits.

(21) Provisions are made for removing energy from the primary containment as necessary to maintain the integrity of the containment system following accidents that release energy to the containment.

(22) Piping that penetrates the primary containment and could serve as a path for the uncontrolled release of radioactive material to the environs is automatically isolated when necessary to limit the radiological impact from an uncontrolled release to less than acceptable limits.

(23) Emergency core cooling systems (ECCS) are provided to limit fuel cladding temperature to less than the limits of 10CFR50.46 in the event of a loss-of-coolant accident (LOCA).

(24) The ECCS provide for continuity of core cooling over the complete range of postulated break sizes in the RCPB.

(25) Operation of the ECCS is initiated automatically when required regardless of the availability of offsite power supplies and the normal generating system of the station.

(26) The control room is shielded against radiation so that continued occupancy under design basis accident conditions is possible.

(27) In the event that the control room becomes inaccessible, it is possible to bring the reactor from power range operation to cold shutdown conditions by utilizing alternative controls and equipment that are available outside the control room.

(28) Backup reactor shutdown capability independent of normal reactivity control is provided. This backup system has the capability to shut down the reactor from any normal operating condition and subsequently to maintain the shutdown condition.

(29) Fuel handling and storage facilities are designed to prevent inadvertent criticality and to maintain shielding and cooling of spent fuel as necessary to meet operating and offsite dose constraints.

(30) Systems that have redundant or backup safety functions are physically separated, and arranged so that credible events causing damage to one region of the Reactor Island complex has minimum prospect for compromising the functional capability of the redundant system.

1.2.1.2 System Criteria

The principal design criteria for particular systems are listed in the following subsections.

1.2.1.2.1 Nuclear System Criteria

(1) The fuel cladding is a radioactive material barrier designed to retain integrity so that failures do not result in dose consequences that exceed acceptable limits throughout the design power range.

(2) 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 any abnormal operational transient.

(3) Those portions of the nuclear system that form part of the RCPB are designed to retain integrity as a radioactive material barrier during normal operation and following abnormal operational transients and accidents.

(4) 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 abnormal operational transients is adequate to prevent fuel cladding damage that results in dose consequences exceeding acceptable limits.

(5) The reactor is capable of being shut down automatically in sufficient time to permit decay heat removal systems to become effective following loss of operation of normal heat removal systems. The capacity of such systems is adequate to prevent fuel cladding damage.

(6) The reactor core and reactivity control system are designed such that control rod action is capable of making the core subcritical and maintaining it even with the rod of highest reactivity worth fully withdrawn and unavailable for insertion.

(7) Backup reactor shutdown capability is provided independent of normal reactivity control provisions. This backup system has the capability to shut down the reactor from any operating condition and subsequently to maintain the shutdown condition.

(8) The nuclear system is designed so there is no tendency for divergent oscillation of any operating characteristic, considering the interaction of the nuclear system with other appropriate plant systems.

1.2.1.2.2 Electrical Power Systems Criteria

Sufficient normal auxiliary and standby sources of electrical power are provided to attain prompt shutdown and continued maintenance of the station in a safe condition under all credible circumstances. The power sources are adequate to accomplish all required essential safety actions under all postulated accident conditions.

1.2.1.2.3 Auxiliary Systems Criteria

(1) Fuel handling and storage facilities are designed to prevent inadvertent criticality and to maintain adequate shielding and cooling for spent fuel.

(2) Other auxiliary systems, such as service water, cooling water, fire protection, heating and ventilating, communications, and lighting, are designed to function as needed, during normal and/or accident conditions.

(3) Auxiliary systems that are not required to effect safe shutdown of the reactor or maintain it in a safe condition are designed so that a failure of these systems shall not prevent the essential auxiliary systems from performing their design functions.

1.2.1.2.4 Shielding and Access Control Criteria

Radiation shielding is provided and access control patterns are established to allow a properly trained operating staff to control radiation doses within the limits of applicable regulations in any normal mode of plant operation.

1.2.1.2.5 Process Control Systems Criteria

The principal design criteria for the process control systems are listed in the following subsections.

1.2.1.2.5.1 Nuclear System Process Control Criteria

(1) Control equipment is provided to allow the reactor to respond automatically to load changes within design limits.

(2) It is possible to control the reactor power level manually.

(3) Nuclear systems process displays, controls and alarms are arranged to allow the operator to rapidly assess the condition of the nuclear system and to locate process system malfunctions.

1.2.1.2.5.2 Electrical Power System Process Control Criteria

(1) The Class 1E power systems are designed with three divisions with any two divisions being adequate to safely place the unit in the hot shutdown condition.

(2) Protective relaying is used to detect and isolate faulted equipment from the system with a minimum of disturbance in the event of equipment failure.

(3) Voltage relays are used on the emergency equipment buses to disconnect the normal source in the event of loss of offsite power and to initiate starting of the standby emergency power system diesel generators.

(4) The standby emergency power diesel generators are started and loaded automatically.

(5) Safety-related electrically operated breakers are controllable from the control room.

(6) Monitoring of essential generators, transformers, and circuits is provided in the main control room.

1.2.1.2.5.3 Power Conversion Systems Process Control Criteria

(1) Control equipment is provided to control the reactor pressure throughout its operating range.

(2) The turbine is able to respond automatically to minor changes in load.

(3) Control equipment in the feedwater system maintains the water level in the reactor vessel at the optimum level required by steam separators.

(4) Control of the power conversion equipment is possible from a central location.

1.2.1.2.6 Power Conversion Systems Criteria

Components of the power conversion systems shall be designed to perform the following basic objectives:

(1) Produce electrical power from the steam coming from the reactor, condense the steam into water, and return the water to the reactor as heated feedwater with a major portion of its gases and particulate impurities removed.

(2) Assure that any fission products or radioactivity associated with the steam and condensate during normal operation are safely contained inside the system or are released under controlled conditions in accordance with waste disposal procedures.

1.2.1.3 Plant Design and Aging Management

The COL applicant shall initiate the life cycle management program early enough in the design process to aid in the application, selection and procurement of components with optimum design life characteristics, and to develop an aging management plan capable of assuring the plant’s original design basis throughout its life.

The aging management plan shall cover containment structures, liner plates, embedded or buried structural components, piping and components. The plan shall consider the potential causes of corrosion which ultimately may be present at the site, including the potential corrosion from copper ground mats. The plan should be initiated early in the design process so that adequate provisions for mitigation measures can be made.

In developing the life cycle management program, the COL applicant shall consider the design life requirements prescribed in the EPRI Utility Requirements Document (URD) and the insights gained from the USNRC Nuclear Plant Aging Research (NPAR) Program. (e.g. NUREG/CRs - 4731 and - 5314)

See Subsection 1.2.3.1 for COL license information.

1.2.2 Plant Description

1.2.2.1 Site Characteristics

1.2.2.1.1 Site Location

The plant is located on a site adjacent or close to a body of water with sufficient capacity for either once-through or recirculated cooling or a combination of both methods.

1.2.2.1.2 Description of Plant Environs

1.2.2.1.2.1 Meteorology

The safety-related structures and equipment are designed to retain required functions for the loads resulting from any tornado with characteristics not exceeding the values provided in Table 2.0-1.

Tornado missiles are discussed in Section 3.5.

1.2.2.1.2.2 Hydrology

The safety design basis of the plant provides that structures of safety significance will be unaffected by the hydrologic parameter envelope defined in Chapter 2.

1.2.2.1.2.3 Geology and Seismology

The structures of safety significance for the plant are designed to withstand a safe shutdown earthquake (SSE) which results in a freefield peak acceleration of 0.3g.

1.2.2.1.2.4 Shielding

Shielding is provided throughout the plant, as required, to maintain radiation levels to operating personnel and to general public within the applicable limits set forth in

10CFR20 and 10CFR100. It is also designed to protect certain plant components from radiation exposure resulting in unacceptable alterations of material properties or activation.

1.2.2.1.3 Site Arrangements

The containment and building arrangements, including equipment locations, are shown in Figures 1.2-2 through 1.2-31. The arrangement of these structures on the plant site is shown in Figure 1.2-1.

1.2.2.2 Nuclear Steam Supply Systems

The Nuclear Steam Supply System (NSSS) includes a direct-cycle forced-circulation BWR that produces steam for direct use in the steam turbine. A heat balance showing the major parameters of the NSSS for the rated power conditions is shown in Figure 1.1-2.

1.2.2.2.1 Reactor Pressure Vessel System

The Reactor Pressure Vessel (RPV) System contains the reactor pressure vessel with the reactor internal pump (RIP) casings; core and supporting structures; the steam separators and dryers; the control rod guide tubes; the spargers for the feedwater, RHR and core flooder system; the control rod drive (CRD) housings; the incore instrumentation guide tubes and housings; and other components. The main connections to the vessel include steamlines, feedwater lines, RIPs, CRDs and incore nuclear instrument detectors, core flooder lines, RHR lines, head spray and vent lines, core plate differential pressure lines, internal pump differential pressure lines, and water level instrumentation.

A venturi-type flow restrictor is a part of the RPV nozzle configuration for each steamline. These restrictors limit the flow of steam from the reactor vessel before the main steamline isolation valves (MSIVs) are closed in case of a main steamline break outside the containment.

The RPV System provides guidance and support for the CRDs. It also distributes boron (sodium pentaborate) solution when injected from the Standby Liquid Control (SLC) System.

The RPV System restrains the CRD to prevent ejection of the control rod connected with the CRD in the event of a failure of the RCPB associated with the CRD housing weld.

CRD blowout restraints are located internal to the reactor vessel and the control rod drive. A restraint system is also provided for each RIP in order to prevent the RIP from becoming a missile in the event of a failure of the RCPB associated with the RIP casing weld.

The reactor vessel is designed and fabricated in accordance with applicable codes for a pressure of 8620 kPaG. The nominal operating pressure in the steam space above the separators is 7170 kPaA. The vessel is fabricated of low alloy steel and is clad internally with stainless steel or Ni-Cr-Fe Alloy (except for the top head, RIP motor casing, nozzles other than the steam outlet nozzle, and nozzle weld zones which are unclad).

The reactor core is cooled by demineralized water that enters the lower portion of the core and boils as it flows upward around the fuel rods. The steam leaving the core is dried by steam separators and dryers located in the upper portion of the reactor vessel. The steam is then directed to the turbine through the main steamlines. Each steamline is provided with two isolation valves in series, one on each side of the containment barrier.

1.2.2.2.2 Nuclear Boiler System

1.2.2.2.2.1 Main Steamline Isolation Valves

All pipelines that both penetrate the containment and offer a potential release path for radioactive material are provided with redundant isolation capabilities. Isolation valves are provided in each main steamline to isolate primary containment upon receiving an automatic or manual closure signal. Each is powered by both pneumatic pressure and spring force. These valves fulfill the following objectives:

(1) Prevent excessive damage to the fuel barrier by limiting the loss of reactor coolant from the reactor vessel resulting from either a major leak from the steam piping outside the containment or a malfunction of the pressure control system resulting in excessive steam flow from the reactor vessel.

(2) Limit the release of radioactive materials by isolating the RCPB in case of the detection of high steamline radiation.

1.2.2.2.2.2 Main Steamline Flow Instrumentation

The steam flow instrumentation is connected to the venturi type steam nozzle of the RPV. The instrumentation provides high nozzle flow isolation signals in case of a main steamline break, flow signals for the feedwater flow control system and indication in the control room.

1.2.2.2.2.3 Nuclear System Pressure Relief System

A pressure relief system consisting of safety/relief valves (SRVs) mounted on the main steamlines is provided to prevent excessive pressure inside the nuclear system as a result of operational transients or accidents.

1.2.2.2.2.4 Automatic Depressurization System

The ADS rapidly reduces reactor vessel pressure in a loss-of-coolant accident, enabling the low-pressure RHR to deliver cooling water to the reactor vessel.

The ADS uses some of the SRVs that are part of the nuclear system pressure relief system. The SRVs used for ADS are set to open on detection of appropriate low reactor water level and high drywell pressure signals. The ADS will not be activated unless either an HPCF or RHR/low-pressure flooder loop pump is operating. This is to ensure that adequate coolant will be available to maintain reactor water level after depressurization.

1.2.2.2.2.5 Reactor Vessel Instrumentation

In addition to instrumentation for the nuclear safety systems and engineered safety features, instrumentation is provided to monitor and transmit information that can be used to assess conditions existing inside the reactor vessel and the physical condition of the vessel itself. This instrumentation monitors reactor vessel pressure, water level, coolant temperature, reactor core differential pressure, coolant flow rates, and reactor vessel head inner seal ring leakage.

1.2.2.2.3 Reactor Recirculation System

The reactor internal pumps (RIPs) are internal pumps which provide a continuous internal circulation path for the core coolant flow. The RIPs are located at the bottom of the vessel. The pump motors are enclosed in casings which are a part of the vessel. A break in the casing as described in Subsection 15B.3.4.3 will result in a leak flow that is less than the ECCS capacity, thus allowing full core coverage. The internal pumps are a wet motor design with no shaft seals, thereby providing increased reliability, reduced maintenance requirements and decreased operational radiation exposure. The RIP has a low rotating inertia. Coupled with the solid-state adjustable speed drives, the RIP can respond quickly to load transients and operator demands.

1.2.2.3 Control and Instrument Systems

1.2.2.3.1 Rod Control and Information System

The Rod Control and Information System (RCIS) provides the means by which control rods are positioned from the control room for power control. The system operates the rod drive motors to change control rod position. For operation in the normal gang movement mode, one gang of control rods can be manipulated at a time. The system includes the logic that restricts control rod movement (rod block) under certain conditions as a backup to procedural controls.

1.2.2.3.2 Control Rod Drive System

When scram is initiated by the RPS, the Control Rod Drive (CRD) System inserts the negative reactivity necessary to shut down the reactor. Each control rod is normally controlled by an electric motor unit. When a scram signal is received, high-pressure water stored in nitrogen charged accumulators forces the control rods into the core and the electric motor drives are signalled to drive the rods into the core. Thus, the hydraulic scram action is backed up by an electrically energized insertion of the control rods.

1.2.2.3.2.1 Control Rod Braking Mechanism

An electromechanical braking mechanism is incorporated in each control rod drive to prevent ejection of the attached control rod in the event of a hydraulic line break. This action limits the rate of reactivity insertion resulting from a rod ejection accident.

1.2.2.3.2.2 Control Rod Ejection

A nuclear excursion is prevented in case of a housing failure and thus the fuel barrier is protected because, as discussed in Subsection 1.2.2.2.1, the housing and the drive are restrained internally to the vessel to prevent the control rod ejection.

1.2.2.3.3 Feedwater Control System

The Feedwater Control System (FCS) automatically controls the flow of feedwater into the reactor pressure vessel to maintain the water within the vessel at predetermined levels. A fault-tolerant triplicated, digital controller using a conventional three-element control scheme is used to accomplish this function.

1.2.2.3.4 Standby Liquid Control System

The Standby Liquid Control System (SLCS) provides an alternate method to bring the nuclear fission reaction to subcriticality and to maintain subcriticality as the reactor cools. The system makes possible an orderly and safe shutdown in the event that not enough control rods can be inserted into the reactor core to accomplish shutdown in the normal manner. The system is sized to counteract the positive reactivity effect from rated power to the cold shutdown condition.

1.2.2.3.5 Neutron Monitoring System

The Neutron Monitoring System (NMS) consists of incore neutron detectors and out-of-core electronic monitoring equipment. The NMS provides indication of neutron flux, which can be correlated to thermal power level for the entire range of flux conditions that can exist in the core. There are fixed incore sensors, the startup range neutron monitors (SRNM), which provide level indications during reactor startup and low power operation. The local power range monitors (LPRM) and average power range monitors (APRM) allow assessment of local and overall flux conditions during power range operation. The automatic traversing incore probe (ATIP) system provides a means to calibrate the local power range monitors. The NMS provides inputs to the RCIS to initiate rod blocks if preset flux limits or period limits for rod block are exceeded, as well as inputs to the RPS if other limits for scram are exceeded.

Those portions of the NMS that input signals to the RPS qualify as a safety-related system. The SRNM and the APRM which monitor neutron flux via incore detectors provide scram logic inputs to the RPS to initiate a scram in time to prevent excessive fuel clad damage as a result of over-power transients. The APRM system also generates a simulated thermal power signal. Both upscale neutron flux and upscale simulated thermal power are conditions which provide scram logic signals.

1.2.2.3.6 Remote Shutdown System

In the event that the control room becomes inaccessible, the reactor can be brought from power range operation to cold shutdown conditions by use of controls and equipment that are available outside the control room.

1.2.2.3.7 Reactor Protection System

The Reactor Protection System (RPS) initiates a rapid, automatic shutdown (scram) of the reactor. It acts in time to prevent fuel cladding damage and any nuclear system process barrier damage following abnormal operational transients. The RPS overrides all operator actions and process controls and is based on a failsafe design philosophy that allows appropriate protective action even if a single failure occurs.

1.2.2.3.8 Recirculation Flow Control System

During normal power operation, the speed of the reactor internal pumps (RIP) is adjusted to control flow. Adjusting RIP speed changes the coolant flow rate through the core and thereby changes the core power level. The system can automatically adjust the reactor power output to the load demand. The solid-state adjustable speed drives (ASD) provide variable-voltage/variable-frequency electrical power to the RIP motors. In response to plant needs, the recirculation flow control system adjusts the ASD power supply output to vary RIP speed, core flow, and core power.

1.2.2.3.9 Automatic Power Regulator System

The Automatic Power Regulator System is summarized in Subsection 7.7.1.7(1).

1.2.2.3.10 Steam Bypass and Pressure Control System

A turbine bypass system is provided which passes steam directly to the main condenser under the control of the pressure regulator. Steam is bypassed to the condenser whenever the reactor steaming rate exceeds the load permitted to pass to the turbine generator. The turbine bypass system has the capability to shed 40% of the turbine-generator rated load without reactor trip or operation of safety/relief valves. The pressure regulation system provides main turbine control valve and bypass valve flow demands so as to maintain a nearly constant reactor pressure during normal plant operation. It also provides demands to the recirculation system to adjust power level by changing reactor recirculation flow rate.

1.2.2.3.11 Process Computer (Includes PMCS, PGCS)

Online process computers are provided to monitor and log process variables and make certain analytical computations. The performance and power generation control systems are included.

1.2.2.3.12 Refueling Platform Control Computer

The refueling platform control computer provides (1) memory of all the fuel and platform positions, (2) directions for the traversable area and traveling paths, (3) directions for the speed functions for all modes of travel, and (4) control of the fuel load. The computer controls automatic or manual refueling between fuel storage and the reactor from the remote control room.

1.2.2.3.13 CRD Removal Machine Control Computer

The CRD handling machine control computer provides automatic positioning, continuous operation and prevention of erroneous operation in the stepwise removal and installation of CRDs from the remote control room.

1.2.2.4 Radiation Monitoring Systems

1.2.2.4.1 Process Radiation Monitoring System

The process radiation monitoring system measures and controls radioactivity in process and effluent streams and activate appropriate alarms and controls.

The process radiation monitoring system measures and records radiation levels associated with selected plant process streams and effluent paths leading to the environment. All effluents from the plant which are potentially radioactive are monitored.

1.2.2.4.2 Area Radiation Monitoring System

The area radiation monitoring system alerts local occupants and the control room personnel of excessive gamma radiation levels at selected locations within the plant.

1.2.2.4.3 Containment Atmospheric Monitoring System

The Containment Atmospheric Monitoring System (CAMS) measures, records and alarms the radiation levels and the oxygen and hydrogen concentration levels in the primary containment under post-accident conditions. It is automatically put in service upon detection of LOCA conditions.

1.2.2.5 Core Cooling System

In the event of a breach in the RCPB that results in a loss of reactor coolant, three independent divisions of ECCS are provided to maintain fuel cladding below the temperature limit as defined by 10CFR50.46. Each division contains one high pressure and one low pressure inventory makeup system.

1.2.2.5.1 Residual Heat Removal System

The Residual Heat Removal (RHR) System is a system of pumps, heat exchangers, and piping that fulfills the following functions:

(1) Removes decay and sensible heat during and after plant shutdown.

(2) Injects water into the reactor vessel following a LOCA to reflood the core in conjunction with other core cooling systems (Subsection 5.5.1).

(3) Removes heat from the containment following a LOCA to limit the increase in containment pressure. This is accomplished by cooling and recirculating the suppression pool water by containment sprays.

1.2.2.5.1.1 Low Pressure Flooder

Low pressure flooding is an operating mode of each RHR system, but is discussed here because the low pressure flooder (LPFL) mode acts in conjunction with other injection systems. LPFL uses the RHR pump loops to inject cooling water into the pressure vessel. LPFL operation provides the capability of core flooding at low vessel pressure following a LOCA in time to maintain the fuel cladding below the prescribed temperature limit.

1.2.2.5.1.2 Residual Heat Removal System Containment Cooling

The RHR System is placed in operation to: (1) limit the temperature of the water in the suppression pool and the atmospheres in the drywell and suppression chamber following a design basis LOCA; (2) control the pool temperature during normal operation of the safety/relief valves and the RCIC System; and (3) reduce the pool temperature following an isolation transient. In the containment cooling mode of operation, the RHR main system pumps take suction from the suppression pool and pump the water through the RHR heat exchangers, where cooling takes place by transferring heat to the service water. The fluid is then discharged back either to the suppression pool, the drywell spray header, the suppression chamber spray header, or the RPV.

1.2.2.5.1.3 Wetwell/Drywell Spray

A spray system is provided for wetwell/drywell cooling in the suppression chamber and drywell air space. The wetwell/drywell spray can be initiated manually if a high containment pressure signal is received. Each subsystem is supplied from a separate redundant RHR subsystem.

1.2.2.5.2 High Pressure Core Flooder System

High pressure core flooder (HPCF) systems are provided in two divisions to maintain an adequate coolant inventory inside the reactor vessel to limit fuel cladding temperatures in the event of breaks in the reactor coolant pressure boundary. The systems are initiated by either high pressure in the drywell or low water level in the vessel. They operate independently of all other systems over the entire range of system operating pressures. The HPCF System pump motors are powered by a diesel generator if auxiliary power is not available. The systems may also be used as a backup for the RCIC System.

1.2.2.5.3 Leak Detection and Isolation System

The leak detection and isolation system (LDS) detects and monitors leakage from the reactor coolant pressure boundary and initiates isolation of the leakage source. The system initiates isolation of the process lines that penetrate the containment by closing the appropriate inboard and outboard isolation valves. LDS monitors leakage inside and outside of the drywell and annunciates excessive leakages in the control room. The following control and isolation functions are automatically performed by LDS:

(1) Isolates the main steamlines

(2) Isolates the reactor water cleanup process lines

(3) Initiates the standby gas treatment system

(4) Isolates the Reactor Building HVAC system

(5) Isolates the containment purge and vent lines

(6) Isolates the cooling water lines in the Reactor Building

(7) Isolates the RHR shutdown cooling system lines

(8) Isolates the steamline to the RCIC turbine

(9) Isolates the suppression pool cleanup system lines

(10) Isolates the flammability control system lines

(11) Isolates the drywell sumps drain lines

(12) Isolates the fission products monitor sampling and return lines

(13) Initiates withdrawal of the automated traversing incore probe

In addition to the above functions, LDS monitors leakage inside the drywell from the following sources and annunciates the abnormal leakage levels in the control room:

(1) Fission products releases

(2) Condensate flow from the drywell air coolers

(3) Drywell sump level changes

(4) Leakages from valve stems equipped with leak-off lines

Other leakages from the FMCRDs, the SRVs and from the reactor vessel head seal flange are monitored by their respective systems.

1.2.2.5.4 Reactor Core Isolation Cooling System

The RCIC System provides makeup water to the reactor vessel when the vessel is isolated and is also part of the emergency core cooling network. The RCIC System uses a steam-driven turbine-pump unit and operates automatically in time and with sufficient coolant flow to maintain adequate water level in the reactor vessel for events defined in Section 5.4.

One division contains the RCIC System, which consists of a steam-driven turbine which drives a pump assembly and the turbine and pump accessories. The system also includes piping, valves, and instrumentation necessary to implement several flow paths. The RCIC steam supply line branches off one of the main steamlines (leaving the RPV) and goes to the RCIC turbine with drainage provision to the main condenser. The turbine exhausts to the suppression pool with vacuum breaking protection. Makeup water is supplied from the condensate storage tank (CST) or the suppression pool with preferred source being the CST. RCIC pump discharge lines include the main discharge line to the feedwater line, a test return line to the suppression pool, a minimum flow bypass line to the suppression pool and a cooling water supply line to auxiliary equipment.

Following a reactor scram, steam generation in the reactor core continues at a reduced rate due to the core fission product delay heat. The turbine condenser and the feedwater system supply the makeup water required to maintain reactor vessel inventory.

In the event the reactor vessel is isolated, and the feedwater supply is unavailable, relief valves are provided to automatically (or remote manually) maintain vessel pressure within desirable limits. The water level in the reactor vessel drops due to continued steam generation by decay heat. Upon reaching a predetermined low level, the RCIC System is initiated automatically. The turbine-driven pump supplies water from the suppression pool or from the CST to the reactor vessel. The turbine is driven with a portion of the decay heat steam from the reactor vessel, and exhausts to the suppression pool.

In the event of a LOCA, the RCIC System, in conjunction with the two HPCF systems, is designed to pump water into the vessel from approximately 1.0 MPaG to full operating pressure. These high pressure systems, combined with the RHR low pressure flooders and ADS, make up the ECCS network which can accommodate any single failure and still shut down the reactor (see Subsection 6.3.1.1 for a detailed description of ECCS redundancy and reliability).

During RCIC operation, the wetwell suppression pool acts as the heat sink for steam generated by reactor decay heat. This results in a rise in pool water temperature. Heat exchangers in the RHR System are used to maintain pool water temperature within acceptable limits by cooling the pool water directly.

1.2.2.6 Reactor Servicing Equipment

1.2.2.6.1 Fuel Servicing Equipment

Fuel servicing equipment is summarized in Subsection 9.1.4.2.3.

1.2.2.6.2 Miscellaneous Servicing Equipment

The servicing aids equipment includes general handling fuel pool tools such as actuating poles with various end configurations. General area underwater lights and support brackets are provided to allow the lights to be positioned over the area being serviced independent of the platform. A general-purpose, plastic viewing aid is provided to float on the water surface to provide better visibility. A portable underwater closed circuit television camera may be lowered into the reactor vessel pool and/or the fuel storage pool to assist in the inspection and/or maintenance of these areas. An underwater vacuum with submersible pump and filter for cleaning.

1.2.2.6.3 Reactor Pressure Vessel Servicing Equipment

Equipment associated with servicing the reactor pressure vessel is used when the reactor is shutdown and the reactor vessel head is being removed or installed. Tools used consist of strongbacks, nut racks, stud tensioners, protectors, wrenches, etc. Lifting tools are designed for a 60-year life and for a safety factor of 10 or better with respect to the ultimate strength of the material used.

1.2.2.6.4 RPV Internal Servicing Equipment

The majority of internal servicing equipment was designed to be attached to the refueling platform auxiliary hoist and used when the reactor is open. A variety of equipment (e.g., grapples, guides, plugs, holders, caps, strongbacks and sampling stations) is used for internal servicing. In addition to these are the RIP handling devices for repair and/or installation. Lifting tools are designed for a safety factor of 10 or better with respect to the ultimate strength of the material used.

1.2.2.6.5 Refueling Equipment

The fuel servicing equipment includes a 1.471 MN Reactor Building crane, refueling machine, and other related tools for reactor servicing.

The Reactor Building crane handles the spent fuel cask from the transport device to the cask loading pit. The refueling machine transfers the fuel assemblies between the storage area and the reactor core. New fuel bundles are handled by the Reactor Building crane. The bundles are stored in the new fuel vault on the reactor refueling floor and are transferred from the vault to the spent fuel pool with the Reactor Building crane auxiliary hook.

The handling of the reactor head, removable internals, reactor insulation, and drywell head during refueling is accomplished using the Reactor Building crane.

1.2.2.6.5.1 Refueling Interlocks

A system of interlocks that restricts movement of refueling equipment and control rods when the reactor is in the refueling and startup modes is provided to prevent an inadvertent criticality during refueling operation. The interlocks backup procedural controls that have the same objective. The interlocks affect movement of the refueling machine, refueling machine hoists, fuel grapple, and control rods.

1.2.2.6.6 Fuel Storage Facility

New and spent fuel storage racks are designed to prevent inadvertent criticality and load buckling. Sufficient cooling and shielding are provided to prevent excessive pool heatup and personnel exposure, respectively. The design of the fuel pool provides for corrosion resistance, adherence to Seismic Category I requirements, and prevention of k_{eff} from reaching 0.95 under dry or flooded conditions.

1.2.2.6.7 Undervessel Servicing Equipment

This equipment is used for the installation and removal work associated with the fine motion control rod drive (FMCRD), RIP, incore monitoring (ICM) and so on. A handling platform provides a working surface for equipment and personnel performing work in the undervessel area. The polar platform is capable of rotating 360 degrees, and has an FMCRD handling trolley with full traverse capability across the vessel diameters. All equipment is designed to minimize radiation exposure, contamination of surrounding equipment and reduce the number of workers required.

1.2.2.6.8 CRD Maintenance Facility

The CRD maintenance facility is located close to the primary containment and is designed and equipped to accommodate maintenance of the FMCRD, provide decontamination of the FMCRD component, perform the acceptance tests and provide storage. The facility uses manual and/or remote operation to minimize radiation exposure to the personnel and to minimize the contamination of surrounding equipment during operation. The layout of the facility is designed so as to maximize the efficiency of the personnel, thereby minimizing the number of workers required.

1.2.2.6.9 Internal Pump Maintenance Facility

The reactor internal pump (RIP) maintenance facility is located in the Reactor Building and is designed for performing maintenance work on the motor assembly and related parts. The facility is designed for one motor assembly, including decontamination in assembled and disassembled states. The facility is equipped with all tools needed for inspection of motor parts and heat exchanger tube bundles. RIP handling tools are stored outside this area.

1.2.2.6.10 Fuel Cask Cleaning Facility

The fuel cask cleaning facility provides for empty casks to be checked for contamination and cleaned of road dirt, moved into the Reactor Building airlock, inspected for damage, and raised to the refueling floor cask pit. The closure head is removed and stored in the adjacent cask washdown pit, while the canal gates between the cask pit and spent fuel pool are removed and the spent fuel is transferred to fill the cask. The canal gates and closure head are replaced and the cask is lifted to the washdown pit. The cask is decontaminated with high pressure water sprays, chemicals and hand scrubbing to the level required for offsite transport. Smear tests are performed to verify cleaning before the filled cask is lowered to the airlock, mounted on the transport vehicle and moved out of the Reactor Building.

1.2.2.6.11 Plant Startup Test Equipment

Plant startup test equipment is a combination of strain gauges, accelerometers, temperature detectors, photo cells, pressure transducers and other associated instrumentation for conducting special startup and reactor internal vibration tests.

1.2.2.6.12 Inservice Inspection Equipment

Inservice inspection equipment are coordinated ultrasonic, eddy current and visual systems needed for incore housing, stub tube, feedwater nozzle, RPV inside and outside diameters (GERIS 2000), RPV internals and head studs, shroud head bolts, and piping (SMART 2000) inspections and examinations.

1.2.2.7 Reactor Auxiliary Systems

1.2.2.7.1 Reactor Water Cleanup System

The Reactor Water Cleanup System (CUW) recirculates a portion of reactor coolant through a filter-demineralizer to remove particulate and dissolved impurities from the reactor coolant. It also removes excess coolant from the reactor system under controlled conditions and provides clean water for the reactor head spray nozzle.

1.2.2.7.2 Fuel Pool Cooling and Cleanup System

The Fuel Pool Cleanup (FPC) System maintains acceptable levels of temperature and clarity and minimizes radioactivity levels of the water in the spent fuel pool, reactor well and dryer/separator pit on top of the containment. The FPC System also maintains the temperature and water level in the service pool and equipment pool. The system includes two heat exchangers, each capable of removing the decay heat generated from an average discharge of spent fuel, and two filter/demineralizers, each unit having the capacity to process the system flow or greater to maintain the desired purity level.

1.2.2.7.3 Suppression Pool Cleanup System

The Suppression Pool Cleanup (SPCU) System provides a continuous purification of the suppression pool water. The system removes impurities by filtration, adsorption, and ion exchange processes. The system consists of a recirculation loop with a pump and isolation valves. Suppression pool water is passed through the Fuel Pool Cooling and Cleanup (FPC) System filter/demineralizers for treatment. Treated water may be diverted to refill the reactor well and the upper pool during refueling outage or provide makeup water to the fuel pool and reactor cooling water (RCW) surge tanks following a seismic event.

1.2.2.8 Control Panels

1.2.2.8.1 Main Control Room Panels

The main control room panel arrangement is summarized in Appendix 18C.

1.2.2.8.2 Control Room Backpanels

The control room backpanels are located in an area adjacent to the main control panels and convenient to the control room crew.

1.2.2.8.3 Radioactive Waste Control Panel

The Radioactive Waste Control Panel System provides the operator interface to the consolidated automatic and remote manual controlling of radioactive waste system mechanical, electrical, and chemical process components. It consists of one or more control panels, including panel-mounted meters and displays, CRT displays, status indicating lights, mode and display selector switches, actuating mechanical and electrical components, controllers, and control logic elements and signal conditioning devices and processors. It does not include equipment or process sensors, local panels or equipment-mounted actuators or power controllers.

It is expected that most of the panels of this system will be located in the radioactive waste control room; panels performing the above functions which are located in the main control room shall also belong to this system.

1.2.2.8.4 Local Control Panels

The local control panels provide facilities for the installation and operation of electrical equipment and interconnecting wiring which supports no primary man-machine interface during normal plant operations. Included within the scope of the local control panels shall be the physical panel structure and the wiring associated with the components installed within the panels. The local control panels do not include the major electrical components installed within the panels, which are instead defined and provided as part of the interfacing plant systems.

1.2.2.8.5 Instrument Racks

The instrument racks provide facilities for the installation and operation of locally mounted instrumentation. Included within the scope of the instrument racks shall be the physical structure upon which the instrumentation is mounted and the wiring associated with the instrument installations. The instrument racks do not include the locally mounted instrumentation, which is instead defined and provided as part of the interfacing plant systems.

1.2.2.8.6 Multiplexing System

The Multiplexing System provides redundant and distributed control and instrumentation data communications networks to support the monitoring and control of interfacing plant systems. The system includes electrical devices and circuitry (such as multiplexing units, bus controllers, formatters and data buses) that connect sensors, display devices, controllers, and actuators which are part of these plant systems. The Multiplexing System also includes the associated data acquisition and communication software required to support its function of plant-wide data and control distribution.

1.2.2.8.7 Local Control Boxes

Local control boxes are uniquely identified to provide operational control of an individual piece of electrical equipment.

1.2.2.9 Nuclear Fuel

1.2.2.9.1 Fuel Assembly

The nuclear fuel assembly contains fissionable material which produces thermal power while maintaining structural integrity. The configuration of the fuel bundle consists of fuel rods, spacers, water rods, upper and lower tie plates. The fuel bundle along with the channel and channel fastener, are assembled into a transportable, interchangeable assembly. The outer envelope of the fuel assembly is square with distinguishing features which provide support, identification, orientation and handling capabilities. The fuel design interface is described in Subsection 4.2.2.1.

1.2.2.9.2 Fuel Channel

The fuel channel encloses the fuel bundle and provides:

(1) A barrier between two parallel coolant flow paths, one for flow inside the fuel bundle and the other for flow in the bypass region between channels.

(2) A bearing surface for the control rod.

(3) Rigidity for the fuel bundle.

The channel fastener attaches the channel to the fuel bundle and, along with the channel spacer buttons, provides channel-to-channel spacing with resilient engagement.

1.2.2.10 Radioactive Waste System

1.2.2.10.1 Radwaste System

1.2.2.10.1.1 Liquid Waste Management System

The Liquid Waste Management System collects, monitors, and treats liquid radioactive wastes for return to the primary system whenever practicable. The radwaste processing equipment is located in the Radwaste Building. Processed waste volumes discharged to the environs are expected to be small. Any discharge is such that concentrations and quantities of radioactive material and other contaminants are in accord 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 facility.

Waste processing is done on a batch basis. Each batch is sampled as necessary in the collection tanks to determine concentrations of radioactivity and other contamination. Equipment drains and other low-conductivity wastes are treated by filtration and demineralization and are transferred to the condensate storage tank for reuse. Laundry drain wastes and other detergent wastes of low activity are treated by filtration, sampled and released via the liquid discharge pathway and demineralization and may be released from the plant on a batch basis. Protection against inadvertent release of liquid radioactive waste is provided by design redundancy, instrumentation for the detection and alarm of abnormal conditions, automatic isolation, and administrative controls.

Equipment is selected, arranged, and shielded to permit operation, inspection, and maintenance with minimum radiation exposure to personnel.

1.2.2.10.1.2 Gaseous Waste Management System

The objective of the Gaseous Waste Management System is to process and control the release of gaseous radioactive effluents to the site environs so as to maintain the exposure of persons in unrestricted areas to radioactive gaseous effluents as low as reasonably achievable (10CFR50, Appendix I). This shall be accomplished while maintaining occupational exposure as low as reasonably achievable and without limiting plant operation or availability.

The offgas system provides for holdup and decay of radioactive gases in the offgas from the air ejector system of a nuclear reactor and consists of process equipment along with monitoring instrumentation and control equipment (Section 11.3).

1.2.2.10.1.3 Solid Waste Management System

The Solid Waste Management System provides for the safe handling, packaging, and short-term storage of radioactive solid and concentrated liquid wastes that are produced. Wet waste processed by this system is transferred to the solidification system, where it is solidified in containers. Dry active waste is surveyed and disposed of whenever possible via the provisions of 10CFR20.302 (a). The remaining combustible waste is compacted. Incinerator ash is compacted waste and shipped in containers for offsite disposal.

1.2.2.11 Power Cycle Systems

1.2.2.11.1 Turbine Main Steam System

The Main Steam (MS) System delivers steam from the reactor to the turbine generator, the reheaters, and the steam jet air ejectors (SJAE) from warmup to full-load operation. The MS System also provides steam for the steam seal system and the auxiliary steam system when other steam sources are not available.

1.2.2.11.2 Condensate, Feedwater and Condensate Air Extraction System

The Condensate and Feedwater System provides a dependable supply of high-quality feedwater 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 SJAE condenser, gland steam condenser, offgas condenser, condensate filters and demineralizer, and through three parallel strings of four low pressure feedwater heaters to the reactor feed pumps' suction. The reactor feed pumps discharge through two stages of two parallel high pressure feedwater heaters to the reactor. The drains from the high pressure heaters are pumped backward to the suction of the reactors feed pumps.

1.2.2.11.2.1 Main Condenser Evacuation System

The Main Condenser Evacuation System removes the noncondensible gases from the main condenser and discharges them to the offgas system. This system consists of two 100% capacity, multiple-element, multi-stage steam jet air ejectors (SJAE) with intercondensers, for normal station operation, and mechanical vacuum pumps for use during startup.

1.2.2.11.3 Heater, Drain and Vent System

The Heater, Drain and Vent System permits efficient and dependable operation of the heat cycle balance-of-plant equipment and, particularly, the condensate and feedwater regenerative heaters. All process equipment drains and vents are collected and routed to the appropriate points in the cycle and flows are controlled for equipment protection.

1.2.2.11.4 Condensate Purification System

Each unit is served by a 100% capacity condensate cleanup system, consisting of high efficiency filters followed by deep-bed demineralizer vessels designed for parallel operation. One demineralizer vessel is a spare. The condensate cleanup system with instrumentation and automatic controls is designed to ensure a constant supply of high-quality water to the reactor.

1.2.2.11.5 Condensate Filter Facility

The condensate filter facility continuously removes suspended solids by processing the full-flow condensate through high efficiency filters. A fast acting full-flow bypass valve opens on high pressure differential across the filter to protect against sudden loss of condensate flow.

1.2.2.11.6 Condensate Demineralizer

The condensate demineralizers continuously process condensate to remove dissolved solids to reactor feedwater quality through demineralizers and an additional unit in manual standby. An emergency bypass line protects the equipment, and a demineralizer resin handling and cleaning system is included.

1.2.2.11.7 Main Turbine

The main turbine is a 188.5 rad/s, tandem compound six-flow, reheat steam turbine with 132.08 cm last-stage blades. The turbine generator is equipped with an electro-hydraulic control system and supervisory instruments to monitor performance. The gross electrical output of the turbine generator is approximately 1400 MW.

1.2.2.11.8 Turbine Control System

The Turbine Control System is summarized in Subsection 10.2.2.3.

1.2.2.11.9 Turbine Gland Steam System

The Turbine Gland Steam System provides steam to the turbine shaft glands and the turbine valve stems. The system prevents leakage of air into or radioactive steam out of the turbine shaft and turbine valves. The gland steam condenser collects air and steam mixture, condenses the steam, and discharges the air leakage to the atmosphere via the main vent by a motor-driven blower.

1.2.2.11.10 Turbine Lubricating Oil System

The Turbine Lubricating Oil System supplies oil to turbine-generator bearing lubrication lines and mainly consists of lube oil tank, oil pumps, oil coolers, and oil purifier equipment.

1.2.2.11.11 Moisture Separator Reheater

The moisture separator reheater is summarized in Subsection 10.2.2.2 (Subtopic Moisture Separator Reheater).

1.2.2.11.12 Extraction System

Extraction steam from the high pressure turbine supplies the last stage of feedwater heating and extraction steam from the low pressure turbines supplies the first four stages. An additional low pressure extraction drained directly to the condenser protects the last-stage bucket from erosion induced by water droplets.

1.2.2.11.13 Turbine Bypass System

The turbine bypass system is summarized in Subsection 10.4.4.2.1

1.2.2.11.14 Reactor Feedwater Pump Driver

Each reactor feedwater pump is driven by an electrical motor driven adjustable speed drive.

1.2.2.11.15 Turbine Auxiliary Steam System

The Turbine Auxiliary Steam System is used when required to supply steam to the steam jet air injectors for condenser deaeration and to the Turbine Gland Seal System, which prevents radioactive steam leakage out of the turbine casings and atmospheric air leakage into the casing at specific operating conditions.

The house boiler steam is a backup to the reactor generated steam during operation and would be used only when reactor steam is unavailable or too radioactive.

1.2.2.11.16 Generator

The generator is a direct-driven, three-phase, 60 Hz, 27 kV, 188.5 rad/s, conductor cooled, synchronous generator rated at approximately 1600 MVA, at 0.90 power factor, 537.4 kPaG hydrogen pressure, and 0.60 short circuit ratio.

1.2.2.11.17 Hydrogen Gas Cooling System

The Hydrogen Gas Cooling System is summarized in Subsection 10.2.2.2 (Subtopic Bulk Hydrogen System).

1.2.2.11.18 Generator Cooling System

The Generator Cooling System includes the hydrogen cooled rotor portion of the Hydrogen Gas Cooling System and the water cooled stator portion of the Turbine Building Cooling Water System.

1.2.2.11.19 Generator Sealing Oil System

The Generator Sealing Oil System prevents hydrogen gas from leaking from the generator. The sealing oil is vacuum-treated to maintain the hydrogen gas purity.

1.2.2.11.20 Exciter

The generator exciter is driven by the main turbine and will have a response ratio that meets the plant voltage regulation requirements and the site specific grid requirements.

Excitation power is provided by the output of a dedicated winding located in the main generator. This output is rectified by the stationary silicon-diode rectifiers. The DC output of the rectifier banks then is applied to the main generator field through the generator collectors.

1.2.2.11.21 Main Condenser

The main condenser is a multipressure three-shell deaerating type condenser or single pressure design as dictated by the site specific circulating water system and power generating heat sink. During plant operation, steam expanding through the low pressure turbines is directed downward into the main condenser and is condensed. The main condenser also serves as a heat sink for the turbine bypass system, emergency and high level feedwater heater and drain tank dumps, and various other startup drains and relief valve discharges.

1.2.2.11.22 Offgas System

The Offgas System is summarized in Subsection 11.3.

1.2.2.11.23 Circulating Water System

The Circulating Water System provides a continuous supply of cooling water to the condenser to remove the heat rejected by the steam cycle and transfers it to the power cycle heat sink.

1.2.2.11.24 Condenser Cleanup Facility

The condenser cleanup facility removes slime and sludge to prevent vacuum decline of the condenser and to suppress corrosion on the inner surface of the condenser tubes.

1.2.2.12 Station Auxiliary Systems

1.2.2.12.1 Makeup Water System (Preparation)

The Makeup Water System (preparation) is summarized in Subsection 9.2.8.3.

1.2.2.12.1.1 Makeup Water System (Purified)

The Makeup Water System (purified) is summarized in Subsection 9.2.10.2.

1.2.2.12.2 Makeup Water System (Condensate)

The Makeup Water System maintains the required capacity and flow of the condensate for the RCIC and HPCF Systems and maintains the required level in the condenser hotwell. The system also (1) stores and transfers water during refueling and cask storage pool water during fuel shipping cask loading, (2) receives and stores the process effluent from the liquid radwaste system, (3) provides makeup to other plant systems where required, and (4) provides condensate to the Control Rod Drive (CRD) Hydraulic System.

The system consists of a condensate storage tank, three condensate transfer pumps, and the necessary controls and instrumentation.

1.2.2.12.2.1 Condensate Storage Facilities and Distribution System

The condensate storage tank receives demineralized water from the purified water makeup system and may also receive low conductivity water from the condensate return of the primary loop, from the radwaste disposal system and the condensate system in the Turbine Building.

1.2.2.12.3 Reactor Building Cooling Water System

The Reactor Building Cooling Water (RCW) System provides cooling water to certain designated equipment located in the Reactor Building. Capacity and redundancy is provided in heat exchangers and pumps to ensure adequate performance of the cooling system under all postulated conditions. During loss of offsite power, emergency power for the system is available from the onsite emergency diesel generators. The closed loop design provides a barrier between radioactive systems and the reactor service water discharged to the environment. Heat is removed from the closed loop by the Reactor Service Water System. Radiation monitors are provided to detect contaminated leakage into the closed systems.

1.2.2.12.4 Turbine Building Cooling Water System

The Turbine Building Cooling Water System is summarized in Subsection 9.2.14.2.1.

1.2.2.12.5 HVAC Normal Cooling Water System

The HVAC Normal Cooling Water System provides chilled water to the air supply cooling coils of the reactor building, to the heating/cooling coils in the drywell, and the control building electrical equipment room.

1.2.2.12.6 HVAC Emergency Cooling Water System

The HVAC emergency cooling water system provides chilled water to the cooling coils in the control building essential electrical equipment room, the main control room and the diesel generator electrical equipment areas. The safety-related chilled-water system is designed to meet the requirements of Criterion 19 of 10CFR50.

1.2.2.12.7 Oxygen Injection System

The Oxygen Injection System is summarized in Subsection 9.3.10.2.

1.2.2.12.8 Ultimate Heat Sink

The Ultimate Heat Sink System is summarized in Subsection 9.2.5.3.

1.2.2.12.9 Reactor Service Water System

The Reactor Service Water System is summarized in Subsection 9.2.15.1.3 and 9.2.15.2.3.

1.2.2.12.10 Turbine Service Water System

The Turbine Service Water System is summarized ion Subsection 9.2.16.1.3 and 9.2.16.2.3.

1.2.2.12.11 Station Service Air System

The Station Service Air System provides a continuous supply of compressed air of suitable quality and pressure for general plant use. The service air compressor discharges into the air receivers and the air is then distributed throughout the plant.

1.2.2.12.12 Instrument Air System

The Instrument Air System is summarized in Subsection 9.3.6.2.

1.2.2.12.13 High Pressure Nitrogen Gas Supply System

Nitrogen gas is normally supplied by the Atmospheric Control System to meet the requirement of (1) the Main Steam System SRV automatic depressurization and relief function accumulators, (2) the main steam isolation valves, and (3) instruments and pneumatic valves using nitrogen in the Reactor Building. When this supply of pressurized nitrogen is not available, the High Pressure Nitrogen Gas Supply (HPIN) System automatically maintains nitrogen pressure to this equipment. The HPIN System consists of high pressure nitrogen storage bottles with piping, valves, instruments, controls and control panel.

1.2.2.12.14 Heating Steam and Condensate Water Return System

The Heating Steam and Condensate Water Return System supplies heating steam from the House Boiler for general plant use and recovers the condensate return to the boiler feedwater tanks. The system consists of piping, valves, condensate recovery set and associated controls and instrumentation.

1.2.2.12.15 House Boiler System

The House Boiler System consists of the house boilers, reboilers, feedwater components, boiler water treatment and control devices. The House Boiler System supplies turbine gland steam and heating steam, including the concentrating tanks and devices of the high conductivity waste equipment.

1.2.2.12.16 Hot Water Heating System

The Hot Water Heating System is a closed-loop hot water supply to the various heating coils of the HVAC systems. The system includes two heat exchangers, surge and chemical addition tanks and associated equipment, controls and instrumentation.

1.2.2.12.17 Hydrogen Water Chemistry System

The Hydrogen Water Chemistry System is summarized in Subsection 9.3.9.2.

1.2.2.12.18 Zinc Injection System

The Zinc Injection System is summarized in Subsection 9.3.11.1.

1.2.2.12.19 Breathing Air System

The Breathing Air System includes air compressors, dryers, purifiers and a distribution network. This network makes breathing air available in all plant areas where operations or maintenance must be performed and high radioactivity could occur in the ambient air. Special connections are provided to assure that this air is used only for breathing apparatus.

1.2.2.12.20 Sampling System (Includes PASS)

The Process Sampling System is furnished to provide process information that is required to monitor plant and equipment performance and changes to operating parameters. Representative liquid and gas samples are taken automatically and/or manually during plant operation for laboratory or online analyses.

1.2.2.12.21 Freeze Protection System

The Freeze Protection System provides insulation, steam and electrical heating for all external tanks and piping that may freeze during winter weather.

1.2.2.12.22 Iron Injection System

The Iron Injection System consists of an electrolytic iron ion solution generator and means to inject the iron solution into the feedwater system in controlled amounts.

1.2.2.13 Station Electrical Systems

1.2.2.13.1 Electrical Power Distribution System

The unit Class 1E AC power system supplies power to the unit Class 1E loads. The offsite power sources converge at the system. The system includes diesel generators that serve as standby power sources, independent of any onsite or offsite source. Therefore, the system has multiple sources. Furthermore, the system is divided into three divisions, each with its own independent distribution network, diesel generator, and redundant load group. A fourth division battery for the safety logic and control system bus receives charger power from the Division II source.

1.2.2.13.2 Unit Auxiliary Transformer

The unit auxiliary AC power system supplies power to unit loads that are non-safety-related and uses the main generator as the normal power source with the reserve auxiliary transformer as a backup source. The unit auxiliary transformer steps down the AC power to the 6900V station bus voltage.

1.2.2.13.3 Isolated Phase Bus

The isolated phase bus duct system provides electrical interconnection from the main generator output terminals to the generator breaker and from the generator breaker to the low voltage terminals of the main transformer, and the high voltage terminals of the unit auxiliary transformers. During the time the main generator is off line, the generator breaker is open and power is fed to the unit auxiliary transformers by backfeeding from the main transformer. During startup, the generator breaker is closed at about 7% power to provide power to the main and the unit auxiliary transformers for normal operation of the plant.

A package cooling unit is supplied with the isolated bus duct system.

1.2.2.13.4 Non-Segregated Phase Bus

The non-segregated phase bus provides the electrical interconnection between the unit auxiliary transformers and their associated 6.9 kV metal-clad switchgear.

1.2.2.13.5 Metal-clad Switchgear

The metal-clad switchgear distributes the 6.9 kV power. Circuit breakers are drawout type, stored energy vacuum breakers. The switchgear interrupting rating shall be determined in accordance with requirements of ANSI C37.10.

1.2.2.13.6 Power Center

The power center is summarized in Subsection 8.3.1.1.2.1.

1.2.2.13.7 Motor Control Center

The motor control center is summarized in Subsection 8.3.1.1.2.2.

1.2.2.13.8 Raceway System

The Raceway System is a plant-wide network comprised of metallic cable trays, metallic conduits and supports. Raceways are classified for carrying medium voltage power cables, low voltage power cables, control cables and low level signal/instrumentation cables. Divisional cables are routed in separate cable raceways for each division.

Fiber optic dataways are not restricted to raceway classifications, but would generally be run with control cables due to their common destinations.

1.2.2.13.9 Grounding Wire

Station grounding and surge protection are discussed in Section 8A.1.

1.2.2.13.10 Electrical Wiring Penetration

Electrical wiring penetrations are described in Subsection 8.3.3.6.1.2 (7).

1.2.2.13.11 Combustion Turbine Generator

The primary function of the Combustion Turbine Generator (CTG) is to act as a standby onsite non-safety power source to feed Plant Investment Protection (PIP) non-safety loads during Loss of Preferred Power (LOPP) events.

The unit also provides an alternate AC power source in case of a station blackout event, as defined by Appendix B of Regulatory Guide 1.155 (Appendix 1C).

1.2.2.13.12 Direct Current Power Supply

The plant has four independent Class 1E and three non-Class 1E 125 VDC power systems.

1.2.2.13.12.1 Unit Auxiliary DC Power System

The Unit Auxiliary DC Power System supplies power to unit DC loads that are non-safety-related. The system consists of three battery chargers, three batteries, and three distribution panels.

1.2.2.13.12.2 Unit Class 1E DC Power System

The Unit Class 1E DC Power System supplies 125 VDC power to the unit Class 1E loads. Battery chargers are the primary power sources. The system, which includes storage batteries that serve as standby power sources, is divided into four divisions, each with its own independent distribution network, battery, and charger.

1.2.2.13.13 Emergency Diesel Generator System

The Emergency Diesel Generator System is supplied by three diesel generators. Each Class 1E division is supplied by a separate diesel generator. There are no provisions for transferring Class 1E buses between standby AC power supplies or supplying more than one engineered safety feature (ESF) from one diesel generator. This one-to-one relationship ensures that a failure of one diesel generator can affect only one ESF division. The diesel generators are housed in the Reactor Building which is a Seismic Category I structure, to comply with applicable NRC and IEEE design guides and criteria.

1.2.2.13.14 Vital AC Power Supply

1.2.2.13.14.1 Safety System Logic and Control Power System

Four divisions of the Safety System Logic and Control (SSLC) Power System provide an uninterruptible Class 1E source of 120-VAC single-phase control power. The primary power source for the SSLC Power System is the Class 1E AC power system. On loss of AC power, the appropriate divisional battery immediately assumes load without interruption. When AC power is restored, it resumes the load without interruption.

1.2.2.13.14.2 Uninterruptible Power System

The Uninterruptible Power System (UPS) supplies regulated 120 VAC single-phase power to non-Class 1E instrument and control loads which require an uninterruptible source of power. The power sources for the UPS are similar to those for the SSLC, but are non-Class 1E.

1.2.2.13.14.3 Reactor Protection System Alternate Current Power Supply

The Reactor Protection System alternate current power supply is described in Subsection 8.3.1.1.4.2.1.

1.2.2.13.15 Instrument and Control Power Supply

The instrument and control (I&C) power supply provides 120 VAC single-phase power to I&C loads which do not require an uninterruptible power source.

1.2.2.13.16 Communication System

The communication system is summarized in Subsection 9.5.2.

1.2.2.13.17 Lighting and Servicing Power Supply

The design basis for the lighting facilities is the standard for the Illuminating Engineering Society. Special attention is given to areas where proper lighting is imperative during normal and emergency operations. The system design precludes the use of mercury vapor fixtures in the containment and the fuel handling areas. The normal lighting systems are fed from the unit auxiliary transformers. Emergency power is supplied by engineered safety buses backed-up by diesel generators. Normal operation and regular simulated offsite power loss tests verify system integrity.

1.2.2.14 Power Transmission Systems

1.2.2.14.1 Reserve Auxiliary Transformer

The reserve auxiliary transformer provides the alternate preferred feed for the Class 1E buses M/C, E, F, and G. It also provides an alternate feed to non-Class 1E 6.9 kV buses.

1.2.2.15 Containment and Environmental Control Systems

1.2.2.15.1 Primary Containment System

The ABWR primary containment system design incorporates the drywell/pressure suppression feature of previous BWR containment designs. In fulfilling its design basis as a fission product barrier, the primary containment is a low leakage structure even at the increased pressures that could follow a main steamline rupture or a fluid system line break.

1.2.2.15.1.1 Primary Containment Vessel

The main features of the primary containment design include:

(1) The drywell, a cylindrical steel-lined reinforced concrete structure surrounding the reactor pressure vessel (RPV).

(2) A suppression pool filled with water, which serves as a heat sink during normal operation and accident conditions.

(3) The air space above the suppression pool.

1.2.2.15.2 Containment Internal Structures

The containment internal structures are summarized in Subsections 3.8.3.1 and 6.2.1.1.2.3.

1.2.2.15.3 Reactor Pressure Vessel Pedestal

The reactor pressure vessel (RPV) pedestal is a prefabricated cylindrical steel structure filled with concrete which supports the RPV and is maintained below design temperature by cooling. The pedestal provides drywell connecting vents which lead to the horizontal vent pipes to the suppression pool.

1.2.2.15.4 Standby Gas Treatment System

The Standby Gas Treatment System (SGTS) minimizes exfiltration of contaminated air from the secondary containment to the environment following an accident or abnormal condition which could result in abnormally high airborne radiation in the Reactor Building. Because the fuel storage area is also in the secondary containment, it also can be exhausted to the SGTS.

All safety-related components of the SGTS are operable during loss of offsite power.

1.2.2.15.5 PCV Pressure and Leak Testing Facility

The PCV pressure and leak testing facility is a special area just outside the containment. It provides instrumentation for conducting the PCV pressure and integrated leak rate tests.

1.2.2.15.6 Atmospheric Control System

The Atmospheric Control System is designed to establish and maintain an inert atmosphere within the primary containment during all plant operating modes except during plant shutdown for refueling or maintenance.

The Atmospheric Control System is summarized in Subsection 6.2.5.2.1.

1.2.2.15.7 Drywell Cooling System

The Drywell Cooling System is summarized in Subsection 9.4.9.2.

1.2.2.15.8 Flammability Control System

A recombiner system is provided to control the concentration of hydrogen and oxygen produced by metal water reaction and radiolysis following a design basis accident in the primary containment.

1.2.2.15.9 Suppression Pool Temperature Monitoring System

The Suppression Pool Temperature Monitoring (SPTM) System is summarized in Subsection 7.6.1.7.1.

1.2.2.16 Structures and Servicing Systems

1.2.2.16.1 Foundation Work

The analytical design and evaluation methods for the containment and Reactor Building walls, slabs and foundation mat and foundation soil are summarized in Subsection 3.8.1.4.1.1.

1.2.2.16.2 Turbine Pedestal

The description for the turbine pedestal is the same as that for foundation work in Subsection 3.8.1.4.1.1.

1.2.2.16.3 Cranes and Hoists

The cranes and hoists are summarized in Subsection 9.1.

1.2.2.16.4 Elevator

The controlled elevators service the Reactor Building radiation controlled zones from the basemat to the refueling floor. Two additional clean elevators service all elevations of the clean zone.

1.2.2.16.5 Heating, Ventilating and Air Conditioning

The plant environmental control systems control temperature, pressure, humidity, and airborne contamination to ensure the integrity of plant equipment, provide acceptable working conditions for plant personnel, and limit offsite releases of airborne contaminants.

The following environmental systems are provided:

(1) The Control Room Habitability Area HVAC System, consisting of supply, recirculation/exhaust and makeup air cleanup units to ensure the habitability of the control room under normal and abnormal conditions of plant operation.

(2) The Reactor Building Secondary Containment HVAC System maintains a negative pressure in the secondary containment under normal and abnormal operating conditions, thereby isolating the environs from potential leak sources. This system removes heat generated during normal plant operation, shutdown, and refueling periods.

(3) The Drywell Cooling System to remove heat from the drywell generated during normal plant operations including startup, reactor scrams, hot standby, shutdown, and refueling periods.

(4) The R/B Safety-Related Equipment HVAC System to distribute air so that a negative pressure is maintained in the emergency core cooling equipment rooms, thereby isolating the potential airborne contamination in these rooms.

(5) The C/B Safety-Related Equipment Area HVAC System to pressurize the electrical rooms, thus allowing exfiltration of air to the battery rooms for exhaust to the outside atmosphere.

(6) The Spent Fuel Pool Area HVAC System to maintain the refueling floor at a negative pressure with respect to the outside atmosphere to prevent the potential release of airborne contamination.

(7) The R/B Satey-Related Diesel Generator HVAC System to provide cooling during operation of the diesel generators. A tempered air supply system controls the thermal environment when the diesel generators are not operating.

(8) Coolers in the steam tunnel and ECCS rooms to remove heat generated during operation of the equipment in these rooms.

1.2.2.16.5.1 Potable and Sanitary Water System

The potable and sanitary water includes conceptual site specific designs of a potable water system, a sanitary water system, a sewage treatment system, and a separate non-radioactive drain system. These systems are summarized in Subsections 9.2.4.1.3, 9.2.4.3.2, and 9.3.3.2.3 respectively.

1.2.2.16.6 Fire Protection System

The Fire Protection System is designed to provide an adequate supply of water or chemicals to points throughout the plant where fire protection is required. Diversified fire-alarm and fire-suppression types are selected to suit the particular areas or hazards being protected. Chemical fire-fighting systems are also provided as additions to or in lieu of the water fire-fighting systems. Appropriate instrumentation and controls are provided for the proper operation of the fire detection, annunciation and fire-fighting systems.

1.2.2.16.7 Floor Leakage Detection System

The drainage system is also used to detect abnormal leakage in safety-related equipment rooms and the fuel transfer area.

1.2.2.16.8 Vacuum Sweep System

A portable, submersible-type, underwater vacuum cleaner is provided to assist in removing crud and miscellaneous particulate matter from the pool floors or reactor vessel. The pump and the filter unit are completely submersible for extended periods. The filter “package” is capable of being remotely changed, and the filters will fit into a standard shipping container for offsite burial.

1.2.2.16.9 Decontamination System

The Decontamination System provides areas, equipment and services to support low radiation level decontamination activities. The services may include electrical power, service air, demineralized water, condensate water, radioactive and nonradioactive drains, HVAC and portable shielding.

1.2.2.16.10 Reactor Building

The Reactor Building includes the containment, drywell, and major portions of the nuclear steam supply system, steam tunnel, refueling area, diesel generators, essential power, non-essential power, emergency core cooling systems, HVAC and supporting systems. The secondary containment is a reinforced concrete building that forms the secondary containment boundary which surrounds the primary containment above the basemat. It permits monitoring and treating all potential radioactive leakage from the primary containment. Treatment consists of HEPA and activated charcoal filtration.

1.2.2.16.11 Turbine Building

The Turbine Building houses all equipment associated with the main turbine generator. Other auxiliary equipment is also located in this building.

1.2.2.16.12 Control Building

The Control Building includes the control room, the computer facility, the cable tunnels, some of the plant essential switchgear, some of the essential power, reactor building water system and the essential HVAC system.

1.2.2.16.13 Radwaste Building

The Radwaste Building houses all equipment associated with the collection and processing of solid and liquid radioactive waste generated by the plant.

1.2.2.16.14 Service Building

The Service Building houses the personnel facilities and portions of the non-essential HVAC System.

1.2.2.17 Yard Structures and Equipment

1.2.2.17.1 Stack

The plant stack is located on the Reactor Building and rises to an elevation of 76 meters above grade level. The stack is a steel shell construction supported by an external steel tubular frame work. The stack vents the Reactor Building, Turbine Building, Radwaste Building, and a small portion of the Control and Service buildings.

1.2.2.17.2 Oil Storage and Transfer System

The major components of this system are the fuel-oil storage tanks, pumps, and day tanks. Each diesel generator has its own individual supply components. Each storage tank is designed to supply the diesel needs during the post-LOCA period, and each day tank has capacity for 8 hours of diesel generator operation at maximum LOCA load demand. Each fuel oil pump is controlled automatically by day-tank level and feeds its day tank from the storage tank. Additional fuel oil pumps supply fuel to each diesel fuel manifold from the day tank.

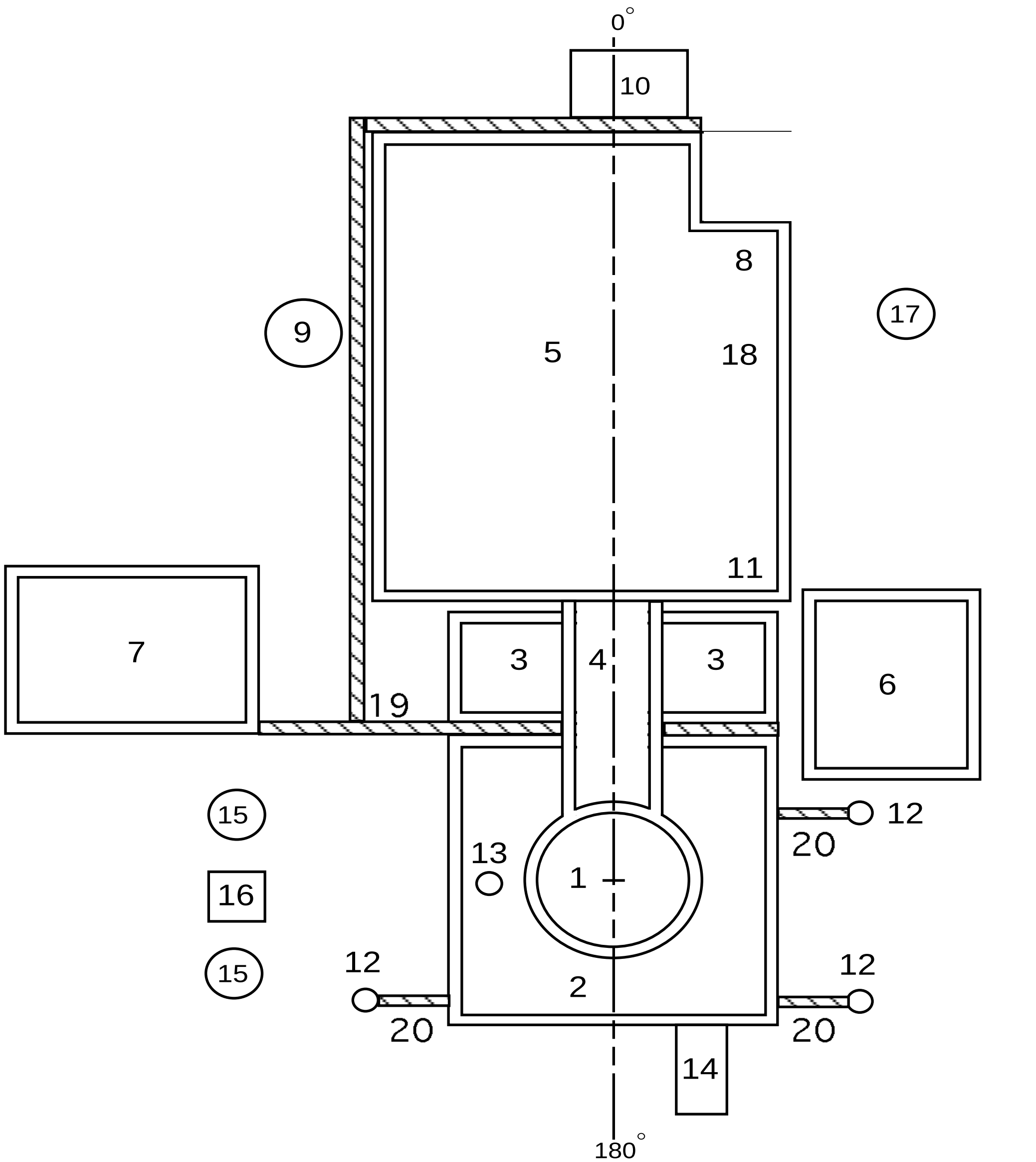
1.2.2.17.3 Site Security

Site Security is summarized in Subsection 13.6.3.1.

1.2.3 COL License Information

1.2.3.1 Plant Design and Aging Management

The COL applicant shall initiate the life cycle management program early in the design process and shall consider the design life requirements as outlined in Subsection 1.2.1.3. In addition, the aging management plan shall cover the structures and components, and the plan shall consider the potential causes of corrosion as outlined in Subsection 1.2.1.3.



No.	FACILITY
1	REACTOR CONTAINMENT
2	REACTOR BUILDING
3	CONTROL BUILDING
4	MAIN STEAM/FEEDWATER TUNNEL
5	TURBINE BUILDING
6	SERVICE BUILDING
7	RADWASTE BUILDING
8	HOUSE BOILER
9	CONDENSATE STORAGE TANK
10	UNIT AUXILIARY TRANSFORMERS
11	NORMAL SWITCHGEAR
12	DIESEL OIL STORAGE TANK (3)
13	STACK
14	EQUIPMENT ENTRY LOCK
15	FIRE PROTECTION WATER STORAGE TANK (2)
16	FIRE PROTECTION PUMPHOUSE
17	BUNKER FUEL TANK
18	COMBUSTION TURBINE GENERATOR
19	RADWASTE TUNNELS RB, CB, TB
20	DG OIL TRANSFER TUNNEL (3)

Figure 1.2-1 Site Plan

The following figures are located in Chapter 21 :

Figure 1.2-29 Turbine Building, General Arrangement, Section B-B

Figure 1.2-30 Turbine Building, General Arrangement, Section C-C

Figure 1.2-31 Turbine Building, General Arrangement, Section D-D

1.3 Comparison Tables

This section highlights the principal design features of the plant and compares its major features with those of other BWR facilities. The design of this facility is based on proven technology obtained during the development, design, construction, and operation of BWRs of similar types. The data, performance characteristics, and other information presented here represent a current, firm design.

1.3.1 Nuclear Steam Supply System Design Characteristics

Table 1.3-1 summarizes the design and operating characteristics for the nuclear steam supply systems. Parameters are related to power output for a single plant unless otherwise noted.

1.3.2 Engineered Safety Features Design Characteristics

Table 1.3-2 compares the engineered safety features design characteristics.

1.3.3 Containment Design Characteristics

Table 1.3-3 compares the containment design characteristics.

1.3.4 Structural Design Characteristics

Table 1.3-4 compares the structural design characteristics.

1.3.5 Instrumentation and Electrical Systems Design Characteristics

Table 7.1-1 compares the instrumentation and electrical systems design characteristics.

Table 1.3-1 Comparison of Nuclear Steam Supply System Design Characteristics

Design[1]	This Plant ABWR 278-872[2]	GESSAR BWR/6 238-748	NMP-2 BWR/5 251-764	Grand Gulf BWR/6 251-800
Thermal and Hydraulic (Section 4.4)				
Rated power (MWt)	3926	3579	3323	3833
Design power (MWt) (ECCS design basis)	4005	3729	3463	4025
Steam flow rate, Mlb/hr at 420°F (FW Temp)	16.843	15.40	14.263	16.491
Core coolant flow rate (Mlb/hr)	115.1	104.0	108.5	112.5
Feedwater flow rate (Mlb/hr)	16.807	15.372	14.564	16.455
System pressure, nominal in steam dome (psia)	1040	1040	1020	1040
Average power density (kW/l)	50.6	54.1	49.15	54.1
Maximum linear heat generation rate (kW/ft)	13.4	13.4	13.4	13.4
Average linear heat generation rate (kW/ft)	5.97	5.9	5.40	5.93
Maximum heat flux (Btu/hr/ft^2)	361,600	361,600	354,255	361,600
Average Heat flux (Btu/hr/ft^2)	161,100	159,500	144,032	160,300
Maximum UO_2 temperature (°F)	3365	3435	3325	3435
Average volumetric fuel temperature (°F)	2150	2185	2130	2185
Average cladding surface temperature (°F)	566	565	566	565
Minimum critical power ratio (MCPR)	1.17	1.20	1.24	1.20
Coolant enthalpy at core inlet (Btu/lb)	527.7	527.6	527.5	527.9
Core maximum voids within assemblies	75	79	76.2	76
Core average exit quality (% steam)	14.5	14.7	13.1	14.6
Feedwater temperature (°F)	420	420	420	420
Design power peaking factor				
Maximum relative assemble power	1.40	1.40	1.40	1.40
Local peaking factor	1.25	1.13	1.24	1.13
Axial peaking factor	1.40	1.40	1.40	1.40
Total peaking factor	2.43	2.26	2.43	2.26

Table 1.3-1 Comparison of Nuclear Steam Supply System Design Characteristics (Continued)

Design[1]	This Plant ABWR 278-872[2]	GESSAR BWR/6 238-748	NMP-2 BWR/5 251-764	Grand Gulf BWR/6 251-800
Nuclear (first core) (Section 4.3)				
Water/UO_2 volume ratio (cold)	2.95	2.70	2.55	2.70
Reactivity with strongest control rod out (k_{eff})	<0.99	<0.99	<0.99	<0.99
	–5.20c @102% rated output	–7.16	–8.57	–7.14
Dynamic void coefficient (c/%)at core average voids(%) (EOC-rated output)	39.2	40.95	40.54	41.31
Fuel temperature doppler coefficient (c/°C) (EOC-rated output)	–0.360	–0.412	–0.419	–0.396
Initial average U-235 enrichment (%)	2.22	1.90	1.90	1.70
Initial cycle exposure (MWd/short ton)	9950	9138	9200	7500
Fuel Assembly (Section 4.2)				
Number of fuel assemblies	872	748	764	800
Fuel rod array	8 x 8	8 x 8	8 x 8	8 x 8
Overall length (inches)	176	176	176	176
Weight of UO_2 per assembly (lb) (pellet type)	435	456	466	458
Weight of fuel assembly (lb) (includes channel)	675	697	698	697
Fuel Rods (Section 4.2)				
Number of fuel rods per assembly	62	62	63	62
Outside diameter (in.)	0.483	0.483	0.493	0.483
Cladding thickness (in.)	[3]	0.032	0.032	0.032
Diametral gap, pellet-to-cladding (in.)	[3]	0.009	0.009	0.009
Length of gas plenum (in.)	[3]	9.48	14	9.48
Cladding material[4]	Zircaloy-2	Zircaloy-2	Zircaloy-2	Zircaloy-2
Fuel Pellets (Section 4.2)				
Material	UO_2	UO_2	UO_2	UO_2
Density (% of theoretical)	[3]	95	95	95
Diameter (in.)	[3]	0.410	0.416	0.410
Length (in.)	[3]	0.410	0.420	0.410

Table 1.3-1 Comparison of Nuclear Steam Supply System Design Characteristics (Continued)

Design[1]	This Plant ABWR 278-872[2]	GESSAR BWR/6 238-748	NMP-2 BWR/5 251-764	Grand Gulf BWR/6 251-800
Fuel Channel (Section 4.2)				
Thickness (in.)	0.100	0.120	0.100	0.120
Cross section dimensions (in.)	5.48 x 5.48	5.45 x 5.45	5.48 x 5.48	5.45 x 5.45
Material	Zircaloy-4	Zircaloy-4	Zircaloy-4	Zircaloy-4
Core Assembly (Section 4.2)				
Fuel weight as UO_2 (lb)	379,221	341,640	265,551	365,693
Core diameter (equivalent) (in.)	203.3	185.2	160.2	191.5
Core height (active fuel) (in.)	146	150	146	150
Reactor Control System (Chapters 4 and 7)				
Method of variation of reactor power	Movable control rods and variable forced coolant flow	Movable control rods and variable forced coolant flow	Movable control rods and variable forced coolant flow	Movable control rods and variable forced coolant flow
Number of movable control rods	205	177	185	193
Shape of movable control rods	Cruciform	Cruciform	Cruciform	Cruciform
Pitch of movable control rods	12.2	12.0	12.0	12.0
Control material in movable rods	B_4C granules compacted in SS tubes	B_4C granules compacted in SS tubes	B_4C granules compacted in SS tubes	B_4C granules compacted in SS tubes
Type of control rod drives	Bottom entry electric hydraulic fine motion	Bottom entry locking piston	Bottom entry locking piston	Bottom entry locking piston
Type of temporary Reactivity control for initial core	Burnable poison; gadolinia-urania fuel rods	Burnable poison; gadolinia-urania fuel rods	Burnable poison; gadolinia-urania fuel rods	Burnable poison; gadolinia-urania fuel rods

Table 1.3-1 Comparison of Nuclear Steam Supply System Design Characteristics (Continued)

Design[1]	This Plant ABWR 278-872[2]	GESSAR BWR/6 238-748	NMP-2 BWR/5 251-764	Grand Gulf BWR/6 251-800
Incore Neutron Instrumentation (Chapters 4 and 7)				
Total number of LPRM detectors	208	164	172	176
Number of incore LPRM penetrations	52	41	43	44
Number of LPRM detectors per penetration	4	4	4	4
Number of SRM penetrations	5	4	4	6
Number of IRM penetrations	10[5]	8	8	8
Total nuclear instrument penetrations	62	53	43	58
Source range monitor range	N/A	[5]	[6]	[6]
Intermediate range monitor range	N/A	[6]	[6]	[6]
Startup range neutron monitor	8	N/A	N/A	N/A
Power range monitors range	Approximately 1% power to 125% power			
Local power range monitors	208	164	172	176
Average power range monitors	4	4	6	8
Number and type of incore neutron source	5 Sb-Be	7 Sb-Be	7 Sb-Be	7 Sb-Be
Reactor Vessel (Section 5.3)				
Material	Low-alloy steel/ stainless and Ni-Cr-Fe alloy clad	Low-alloy steel/ stainless clad	Low-alloy steel/ stainless clad	Low-alloy steel/ stainless clad
Design pressure (psig)	1250	1250	1250	1250
Design temperature (°F)	575	575	575	575
Inside diameter (ft-in.)	23-2	19-10	20-11	20-11
Inside height (ft-in.)	68-11	70-4	72-5	72-7
Minimum base metal thickness (cylindrical section) (in.)	7.50	6.0	6.19	6.19
Minimum cladding thickness (in.)	1/8	1/8	1/8	1/8

Table 1.3-1 Comparison of Nuclear Steam Supply System Design Characteristics (Continued)

Design[1]	This Plant ABWR 278-872[2]	GESSAR BWR/6 238-748	NMP-2 BWR/5 251-764	Grand Gulf BWR/6 251-800
Reactor Coolant Recirculation (Chapter 5)				
Number of recirculation loops	0	2	2	2
Design pressure				
inlet leg (psig)	N/A[7]	1250	1650	1250
outlet leg (psig)	N/A[7]	1650[9] 1550[8]	1650[9] 1550	1650[8] 1550[9]
Design temperature (°F)	N/A[7]	575	575	575
Pipe diameter (in.)	N/A[7]	22/24	24	24
Pipe material (ANSI)	N/A[7]	304/316	316k	304/316
Recirculation pump flow rate (gpm)	30,430/ pump	42,000	47,200	44,600
Number of jet pumps in reactor	N/A[7]	20	20	24
Main Steamlines (Subsection 5.4.9)				
Number of steamlines	4	4	4	4
Design Pressure(psig)	1250	1250	1250	1250
Design temperature (°F)	575	575	575	575
Pipe diameter (in.)	28	26	26/28	28
Pipe material	Carbon steel	Carbon steel	Carbon steel	Carbon steel

1 English units are utilized in this table since the data obtained from the comparative BWR operating facilities are in English units.
2 Parameters for the core loading in Figure 4.3-1 used in the sensitivity analysis.
3 Proprietary information not included in DCD, (Refer to SSAR Section 1.3, Amendment 32).
4 Free-standing loaded tubes.
5 Shutdown through criticality.
6 Prior criticality to low power.
7 ABWR design utilizes reactor internal pumps (RIPs).
8 Discharge piping from discharge block valve to vessel.
9 Pump and discharge piping to and including discharge block valve.

Table 1.3-2 Comparison of Engineered Safety Features Design Characteristics

System/Component[1]	This Plant ABWR 278-872	GESSAR BWR/6 238-748	NMP-2 BWR/5 251-764	Grand Gulf BWR/6 251-800
Emergency Core Cooling Systems (sized on design power-Section 6.3)				
Low Pressure Core Spray Systems[2]				
Number of loops	N/A	1	1	1
Flow rate(gpm)	N/A N/A	6000 at 122 psid	6350 at 128 psid	7000 at 122 psid
High Pressure Core Spray System[3]				
Number of loops	2	1	1	1
Flow rate (gpm)	800 at 1177 psid	1550 at 1147 psid	1550 at 1130 psid	1650 at 1147 psid
	3200 at 100 psid	6110 at 200 psid	6350 at 200 psid	7000 at 200 psid
Reactor Core Isolation Cooling System (Subsection 5.4.6)				
Flow rate (gpm)	800 at 165-1192 psia reactor pressure	700 at 165-1192 psia reactor pressure	600 at 1173 psia reactor pressure	800 at 165-1192 psia reactor pressure
Automatic Depressurization System				
Number of relief valves	8	8	7	8
Low Pressure Coolant Injection[4]				
Number of loops	3	3	3	3
Number of pumps	3	3	3	3
Flow rate (gpm/pump)	4200 at 40 psid	7100 at 20 psid	7450 at 26 psid	7450 at 20 psid

Table 1.3-2 Comparison of Engineered Safety Features Design Characteristics (Continued)

System/Component[1]	This Plant ABWR 278-872	GESSAR BWR/6 238-748	NMP-2 BWR/5 251-764	Grand Gulf BWR/6 251-800
Auxiliary Systems Residual Heat Removal System (Subsection 5.4.7)				
Reactor shutdown cooling mode				
Number of loops	3	2	2	2
Number of pumps [5]	3	2	2	2
Flow rate (gpm/pump)	4200	7100	7450	7450
Duty (MBtu/ hr heat exchanger)[6]	29.0	46.9	41.6	50.0
Number of heat exchangers	3	2	2	2
Primary containment cooling mode flow rate (gpm)	4200	7100	7450	7450
Flow rate (gpm/heat exchanger)	8000	7	7400	25,300 total
Number of pumps	3 loops RCW	7	6	2 at 12,000 gpm 1 at 1300 gpm
Fuel Pool Cooling and Cleanup System (Subsection 9.1.3)				
Capacity (MBtu/hr)	6.55	8.0	15.0	11.8

1 English units are utilized in this table since the data obtained from the comparative BWR operating facilities are in English units.
2 ABWR design utilizes the low pressure flooder mode of the RHR System.
3 ABWR design is a flooder system not a spray system.
4 ABWR design referred to as Low Pressure Flooder.
5 The design of the pumps is, in part, based on the required capacity during the reactor flooding mode.
6 Heat exchanger duty at 20 hours after reactor shutdown.

Table 1.3-3 Comparison of Containment Design Characteristics

Containment[1, 2]	This Plant ABWR 278-872	GESSAR BWR/6 238-748	NMP-2 BWR/5 251-764	Grand Gulf BWR/6 251-800
Primary				
Type	Over- and underpressure suppression	Mark III freestanding steel with reinforced concrete shield building	Over- and under-pressure Suppression Mark II	Mark III reinforced concrete containment with steel liner
Construction	Reinforced concrete with steel liner; steel structure	Cylindrical freestanding steel with ellipsoidal head	Reinforced concrete with steel liner	Reinforced concrete cylinder with hemispherical head; steel lined
Drywell	Concrete cylinder	Concrete cylinder[3]	Frustum of cone upper portion	Concrete cylinder[3]
Pressure suppression chamber	Concrete cylinder	Freestanding steel annulus with concrete backing	Cylindrical lower portion	Steel lined concrete annulus
Containment internal design pressure (psig)	45	15	45	15
Drywell internal design pressure (psig)	45	30	45	30
Drywell free volume (ft^3)	259,563	275,000	303,418	270,000
Pressure suppression chamber free volume (ft^3)(HWL)	210,475	1,140,000	192,028	1,400,000
Pressure suppression pool water volume (ft^3)(LWL)	126,426	129,600 (upper pool dump = 34,200)	154,794	136,000 (upper pool dump = 72,800)
Submergence of vent pipe below pressure pool surface (ft) (HWL)	11.8 to 20.8	7.5	11.0 max.	7.5 min.
Design temperature of drywell (°F)	340	330	340	330
Downcomer vent pressure loss factor	2.5–3.5	2.5–3.5	1.37	2.5–3.5
Break area/ total vent area	0.01	0.012	0.0108	0.008

Table 1.3-3 Comparison of Containment Design Characteristics (Continued)

Containment[1, 2]	This Plant ABWR 278-872	GESSAR BWR/6 238-748	NMP-2 BWR/5 251-764	Grand Gulf BWR/6 251-800
Primary (Continued)				
Calculated maximum drywell pressure after blowdown (psig).	39	23.0	39.7	22.0
Pressure suppression chamber (psig)	26	8.7	34.0	9.0
Initial pressure suppression pool temperature rise (°F) during LOCA	50	50	50	30
Leakage rate (% free volume/day)	0.5	1.0	1.1	0.35
Secondary				
Type	Controlled leakage	Controlled leakage	Controlled leakage elevated release	Controlled leakage
Construction				
Lower levels	Reinforced concrete	[4]	Reinforced concrete	Reinforced concrete
Upper levels	Reinforced concrete	[4]	Steel superstructure and siding	Steel superstructure and siding
Roof	Reinforced concrete	[4]	Steel decking	Steel decking
Internal design pressure (psig)	0.25	0.25	0.25	0.25
Design in leakage rate (% free volume/day at 0.25 in. H_2O)	50	100	100	100

1 English units are utilized in this table since the data obtained from the comparative BWR operating facilities are in English units.

2 Where applicable, containment parameters are based on design rated power.

3 Not part of containment boundary.

4 Not specified.

Table 1.3-4 Comparison of Structural Design Characteristics

	This Plant ABWR 278-872	GESSAR BWR/6 238-748	NMP-2 BWR/5 251-764	Grand Gulf BWR/6 251-800
Seismic Design (Section 3.7)[1]				
Operating Basis Earthquake				
horizontal g	None	0.15	0.075	0.075
vertical g	None	0.10	0.075	0.05
Safe Shutdown Earthquake				
horizontal g	0.3	0.30	0.15	0.15
vertical g	0.3	0.20	0.15	0.10
Wind Design (Subsection 3.3.2)				
Translation (mph)	60	70 max. 5 min.	70	60
Tangential (mph)	240	290	290	300

1 English units are utilized in this table since the data obtained from the comparative BWR operating facilities are in English units.

1.4 Identification of Agents and Contractors

GE has engaged in the development, design, construction, and operation of boiling water reactors since 1955. Table 1.4-1 lists the GE reactors completed, under construction, or on order. As can be seen, GE has substantial experience, knowledge, and capability to design, manufacture, and furnish technical assistance for the installation and startup of reactors.

Table 1.4-1 Commercial Nuclear Reactors Completed, Under Construction, or in Design by General Electric

Station	Utility	Rating (MWe)	Year of Order	Year of Low Power License
Dresden 1	Commonwealth Edison	207	1955	1959
Humboldt Bay	Pacific G&E	70	1958	1962
KHAL	Germany	15	1958	1961
Garigliano	Italy	150	1959	1964
Big Rock Point	Consumers Power	72	1959	1963
JPDR	Japan	11	1960	1963
KRB	Germany	237	1962	1967
Tarapur 1	India	190	1962	1967
Tarapur 2	India	190	1962	1969
GKN	Holland	52	1963	1968
Oyster Creek	JCP&L	640	1963	1969
Nine Mile Point	Niagara Mohawk	610	1963	1969
Dresden 2	Commonwealth Edison	794	1965	1969
Pilgrim	Boston Edison	670	1965	1972
Millstone 1	NUSCO	652	1965	1970
Tsuruga	Japan	340	1965	1970
Nuclenor	Spain	440	1965	1971
Fukushima 1	Japan	439	1966	1971
BKW KKM	Switzerland	306	1966	1972
Dresden 3	Commonwealth Edison	794	1966	1971
Monticello	Northern States	548	1966	1970
Quad Cities 1	Commonwealth Edison	789	1966	1972
Browns Ferry 1	TVA	1067	1966	1973
Browns Ferry 2	TVA	1067	1966	1974
Quad Cities 2	Commonwealth Edison	789	1966	1972
Vermont Yankee	Vermont Yankee	515	1966	1972
Peach Bottom 2	Philadelphia Electric	1065	1966	1973
Peach Bottom 3	Philadelphia Electric	1065	1966	1974
FitzPatrick	PASNY	821	1968	1974
Shoreham	LILCO	820	1967	1984

Table 1.4-1 Commercial Nuclear Reactors Completed, Under Construction, or in Design by General Electric (Continued)

Station	Utility	Rating (MWe)	Year of Order	Year of Low Power License
Cooper	Nebraska PPD	778	1967	1974
Browns Ferry 3	TVA	1067	1967	1977
Limerick 1	Philadelphia Electric	1100	1967	1984
Hatch 1	Georgia Power	786	1967	1974
Fukushima 2	Japan	762	1967	1975
Brunswick 1	Carolina P&L	821	1968	1977
Brunswick 2	Carolina P&L	821	1968	1974
Duane Arnold	Iowa ELP	545	1968	1974
Fermi 2	Detroit Edison	1093	1968	1987
Hope Creek 1	PSE&G	1067	1969	1984
Hope Creek 2	PSE&G	1067	1969	1986
Chinshan 1	Taiwan	610	1969	1978
Caorso	Italy	822	1969	1977
Hatch 2	Georgia Power	786	1970	1978
La Salle 1	Commonwealth Edison	1078	1970	1982
La Salle 2	Commonwealth Edison	1078	1970	1983
Susquehanna 1	Pennsylvania P&L	1050	1967	1982
Susquehanna 2	Pennsylvania P&L	1050	1968	1984
Chinshan 2	Taiwan	610	1970	1979
Hanford 2	WPPSS	1100	1971	1983
Nine Mile Point 2	Niagara Mohawk	1100	1971	1987
Grand Gulf 1	Mississippi P&L	1250	1971	1982
Fukushima 6	Japan	1135	1971	1979
Tokai	Japan	1135	1971	1977
River Bend 1	Gulf States	940	1972	1985
Perry 1	Cleveland Electric	1205	1972	1981
Laguna Verde1	Mexico	660	1972	1988
Leibstadt	Switzerland	940	1972	1984
Kuosheng 1	Taiwan	992	1972	1981
Kuosheng 2	Taiwan	992	1972	1982

Table 1.4-1 Commercial Nuclear Reactors Completed, Under Construction, or in Design by General Electric (Continued)

Station	Utility	Rating (MWe)	Year of Order	Year of Low Power License
Clinton 1	Illinois Power	950	1973	1986
Cofrentes	Spain	975	1973	1985
Laguna Verde 2	Mexico	660	1973	1994
Kashiwazaki - Kariwa 6	Japan	1300	1987	1996
Kashiwazaki - Kariwa 7	Japan	1300	1987	1997

1.5 Requirements for Further Technical Information

In the December 1986 technical description of the Advanced Boiling Water Reactor (ABWR) GE, in Section 3, provided a description of the test and development program associated with the ABWR. Of the efforts described in that report, all have been satisfactorily completed.

1.6 GE Topical Reports and Other Documents

Table 1.6-1 is a list of all GE topical reports and any other reports or documents which are referenced in Tier 2 and which contain information utilized for the ABWR.

Table 1.6-1 Referenced Reports

Report No.	Title	Tier 2 Section No.
22A7007	"GESSAR II, 238 Nuclear Island, BWR/6 Standard Plant, General Electric Company", March 1980, & Amendments 1-21.	3.7 19.2 19.3 19D.3 19D.7 19E.2 19E.3
APED-5750	"Design and Performance of General Electric Boiling Water Reactor Main Steam Line Isolation Valves", General Electric Company, Atomic Power Equipment Department, March 1969.	5.4
NEDO-10029	An Analytic Study on Brittle Fracture of GE-BWR Vessel Subject to the Design Basis Accident, June 1969.	5.3
NEDO-10299A	H.T. Kim, "Core Flow Distribution in a Modern BWR as Measured in Monticello", October 1976.	4.4
NEDO-10527	C.J. Paone and J.A. Woolley, "Rod Drop Accident Analysis for Large Boiling Water Reactors", Licensing Topical Report, March 1972.	15.4
NEDO-10585	F.G. Brutchscy, et al., "Behavior of Iodine in Reactor Water During Plant Shutdown and Startup", August 1972.	15.2
NEDO-10722	H.T. Kim, "Core Flow Distribution in a Large Boiling Water Reactor as Measured in Quad Cities Unit 1", December 1972.	4.4
NEDO-10722A	H.T. Kim, "Core Flow Distribution in a Large Boiling Water Reactor as Measured in Quad Cities Unit 1", August 1976.	4.4
NEDO-10802 - A	R.B. Linford, "Analytical Methods of Plant Transients Evaluations for the GE BWR", December 1986.	4.4
NEDO-10802-01A	R.B. Linford, "Analytical Methods of Plant Transients Evaluations for the GE BWR", Amendment 1, December 1986.	4.4
NEDO-10802-02A	R.B. Linford, "Analytical Methods of Plant Transients Evaluations for the GE BWR", Amendment 2, December 1986.	4.4
NEDO-10871	J.M. Skarpelos and R.S. Gilbert, "Technical Derivation of BWR 1971 Design Basis Radioactive Material Source Terms", March 1973.	11.1
NEDO-10958-A	H.T. Kim, "General Electric Thermal Analysis Basis (GETAB): Data, Correlation and Design Applications (LTR)", January 1977.	4.4 4B
NEDO-11209-04-A	"GE Nuclear Energy Quality Assurance Program Description", the latest NRC-accepted version.	17.1

Table 1.6-1 Referenced Reports (Continued)

Report No.	Title	Tier 2 Section No.
NEDE13426P	"T. R. McIntyre, et. al., Mark III Conformatory Test Program -1/3 Scale Impact Tests - Test Series 5805", August 1975.	3B
NEDO-20206	D.R. Rogers, "BWR Turbine Equipment N-16 Radiation Shielding Studies", December 1973.	12.2
NEDO-20340	J. Carew, "Process Computer Performance Evaluation Accuracy" June 1974.	4.3
NEDO-20533	W.J.Bilanin, "The GE Mark III Pressure Suppression Containment Analytical Model", June 1974.	6.2
NEDO-20533-1	W.J.Bilanin, "The GE Mark III Pressure Suppression Containment Analytical Model", Supplement 1, September 1975	6.2
NEDE-20566-A	"General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K", September 1986.	6.3
NEDM-20609-01	P.P. Stancavage and D.G. Abbott, "Liquid Discharge Doses LIDSR Code", August 1976.	12.2
NEDO-20953A	J.A. Woolley, "Three-Dimensional BWR Core Simulator", January 1977.	4A.4
NEDO-21052	F.J. Moody, "Maximum Discharge Rate of Liquid-Vapor Mixtures from Vessels", General Electric Company, September 1975.	6.2
NEDO-21143-1	H. Careway, V. Nguyen, and P. Stancavege, "Radiological Accident-The CONACO3 CODE", December 1981.	15.2 15.6
NEDO-21159	"Airborne Releases from BWRs for Environmental Impact Evaluations", March 1976.	11.1
NEDO-21159	"Airborne Releases from BWRs for Environmental Impact Evaluations" - Amendment 2 - Iodine	12.2
NEDE-21175-P	"BWR/6 Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings", November 1976.	3.9
NEDC-21215	"Brunswick Steam Electric Plant Unit 1 Safety Analysis Report for Plant Modifications to Eliminate Significant In-Core Vibrations", March 1976.	4.4
NEDC-21251	J. Charnley, "KKM Safety Analysis Report", April 1976.	4.4
NEDE-21354-P	"BWR Fuel Channel Mechanical Design and Deflection", September 1976.	3.9
NEDE-21471-1	L. Lasher,et. al, "Analytical Model for Estimating Drag Forces on Rigid Submerged Structures Caused Supplement for X-Quencher Air Discharge", October 1979.	3B

Table 1.6-1 Referenced Reports (Continued)

Report No.	Title	Tier 2 Section No.
NEDO-21471	F. Moody, "Analytical Model for Estimating Drag Forces on Rigid Submerged Structures Caused by a LOCA", September 1977.	3B
NEDO-21506	"Stability and Dynamic Forces of the GE Boiling Water Reactor (LTR)", October 1976.	4.1
NEDE-21514 - 1&2	"BWR Scram System Reliability Analysis", December 1976, General Electric Company.	19D.6
NEDE-21526	J. Dougherty, "SCAM - Subcompartment Analysis Method", January 1977.	6.2
NEDE-21544-P	R.J Ernst, et. al., "Mark II Pressure Suppression Containment Systems: An Analytical Model of the Pool Swell Phenomenon", December 1977.	3B
NEDO-21778-A	"Transient Pressure Rises Affecting Fracture Toughness Requirements for Boiling Water Reactors", December 1978.	5.3
NEDO-21985	"Functional Capability Criteria for Essential Mark II Piping", September 1978, prepared by Battelle Columbus Laboratories for General Electric Company.	3.9
NEDE-22056	"Failure Rate Data Manual for GE BWR Components", Rev. 2 January 17, 1986, Class III, General Electric Company.	19.3 19D.3 19E.2
NEDO-22155	"GE Report, Generation and Mitigation of Combustible Gas Mixtures in Inerted BWR Mark I Containments", June 1982.	6.2
NEDE-22277-P-I	G. A. Watford, "Compliance of the GE BWR Fuel Design to Stability Licensing Criteria", October 1984.	20.3.7
NEDE-23819	P.D. Knecht, "BWR/6 Drywell and Containment Maintenance and Testing Access Time Estimates", May 1978.	12.4
NEDE-23996-1	P.D. Knecht, "Maintenance Access Time Estimates, BWR/6 Auxiliary and Fuel Buildings", May 1979.	12.4
NEDE-23996-2	A. Chappori, "Maintenances Access Time Estimates, BWR/6 Radwaste Building", May 1979.	12.4
NEDO-24057	"Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants", November 1977.	3.9
NEDO-24057-P	"Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants", November 1977.	3.9
NEDE-24131	"Basis for 8x8 Retrofit Fuel Thermal Analysis Application", September 1978.	4D.2
NEDO-24154	"Qualification of the One-Dimensional Core Transient Model for BWRs", Vol. 1 & 2, October 1978.	4.4

Table 1.6-1 Referenced Reports (Continued)

Report No.	Title	Tier 2 Section No.
NEDO-24154-P	"Qualification of the One-Dimensional Core Transient Model for BWRs", Vol. 3 October 1978. (Proprietary)	4.4
NEDE-24222	J. Weiss, "Assessment of BWR Mitigation of ATWS", December 1979.	15E 19.3
NEDE-24302-P	"Mark II Containment Program, Generic Chugging Load Definition Report", April 1981.	3B
NEDE-24326-1-P	"General Electric Environmental Qualification Program", Proprietary Document, January 1983.	3.9 3.11
NEDE-24351	D. Hale, "Fatigue Crack Growth in Piping and RPV Steels in Simulated BWR Water Environment Update", July 1981.	3E
NEDE-24679	"Study of Advanced BWR Features, Plant Definition/Feasibility Results", Vol.III, Part G, October 1979.	12.4
NEDO-24708	P. W. Marriot, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors", August 1979.	7.3
NEDE-25100-P	"Mark II Containment Supporting Program, Caorso SRV Discharge Tests Phase I Test Report", May 1979.	3B
NEDE-25118	"Mark II Containment Supporting Program, Caorso SRV Discharge Tests Phase II ATR", August 1979.	3B
NEDO-25132A	E. W. Bradley, "Gamma & Beta Dose to Man from Noble Gas Release to the Atmosphere GEMAN Code", April 1980.	12.2
NEDO-25153	L. E. Lasher, "Analytical Model for Estimating Drag Forces on Rigid Structures Caused by Steam Condensation and Chugging", July 1979.	3B
NEDE-25250	A. Javid, "Generic Criteria for High Frequency Cutoff of BWR Equipment", January 1980. (Proprietary)	3.9
NEDO-25257	E. W. Bradley and V. D. Nguyen, "Radiation Exposure from Airborne Effluents-the REFAE Code", July 1980.	12.2
NEDE-25273	F. T. Dodge, "Scaling Study of the General Electric Pressure Suppression Test Facility - Mark III Long Range Program, Task 2.2.1", SwRI, March 1980. (Proprietary)	3B
NEDE-30090	"Alto Lazio Station Reliability Analysis", December 1984	19D.6
NEDO-30130-A	Bill Zarbis, "Steady-State Nuclear Methods", May 1985. (Proprietary)	4.3 4.4
NEDC-30259	H.A. Careway, D.B. Townsend, B.W. Shaffer, "A Technique for Evaluation of BWR MSIV Leakage Contribution to Radiological Dose Rate Calculations", September 1983.	15.6
NEDE-30637	B.M. Gordon, "Corrosion and Corrosion Control in BWRs", December 1984.	5.2

Table 1.6-1 Referenced Reports (Continued)

Report No.	Title	Tier 2 Section No.
NEDE-30640	"Evaluation of Proposed Modification to the GESSAR II Design", Class III, June 1984.	19P
NEDO-30832	J.E. Torbeck, "Elimination of Limit on BWR Suppression Pool Temperature for SRV Discharge With Quenchers", December 1984.	3B
NEDC-30851P-A	W. P. Sullivan, "Technical Specification Improvement Analyses for BWR Reactor Protection System", March 1988.	19D.6
NEDE-31096-A	"GE Licensing Topical Report ATWS Response to NRC ATWS Rule 10CFR 50.62", February 1987.	19B.2
NEDE-31152-P	"GE Bundle Designs", December 1988.	4.2
NEDO-31331	Gerry Burnette, "BWR Owner's Group Emergency Procedure Guidelines", March 1987.	18A
NEDC-31336	Julie Leong, "General Electric Instrument Setpoint Methodology", October 1986.	7.3
NEDC-31393	"ABWR Containment Horizontal Vent Confirmatory Test, Part I", March 1987.	3B
NEDO-31439	C. VonDamm, "The Nuclear Measurement Analysis & Control Wide Range Neutron Monitoring System (NUMAC-WRNMS)", May 1987	20.3
NEDC-31858P	Louis Lee, "BWROG Report for Increasing MSIV Leakage Rate Limits and Elimination of Leakage Control System", 1991	15.6
NEDE-31906-P	A. Chung, "Laguna Verde Unit I Reactor Internals Vibration Measurement", January 1991.	7.4
NEDO-31960	Glen Watford, "BWR Owners' Group Long-Term Stability Solutions Licensing Methodology", June 1991.	4.4
NEDC-32267P	"ABWR Project Application Engineering Organization and Procedures Manual", December 1993.	17.1

1.7 Drawings

1.7.1 Piping and Instrumentation and Process Flow Drawings

Table 1.7-1 contains a list of system piping and instrumentation diagrams (P&ID) and process flow diagrams (PFD) provided in Tier 2. Figure 1.7-1, sheets 1 and 2 define the symbols used on these drawings.

1.7.2 Instrument, Control and Electrical Drawings

Interlock block diagrams (IBD), instrument engineering diagrams (IED) and single-line diagrams (SLD) are listed in Table 1.7-2. Figure 1.7-2 defines the graphic symbols used in the IBDs.

1.7.3 ASME Standard Units to Preferred Metric Conversion Factors

The ASME standard units are applied with the numerical values converted to the preferred metric units system as listed in Table 1.7-3.

1.7.4 Preferred Metric Conversion to ASME Standard Units

Selected flow, pressure, temperature, and length preferred metric units are converted to ASME standard units as listed in Table 1.7-4.

1.7.5 Drawing Standards

Guidelines for identifying systems, facilities, equipment types, and numbers and for drawing P&IDs and PFDs are treated in Table 1.7-5.

1.7.6 COL License Information

1.7.6.1 P&ID Pipe Schedules

COL applicants shall complete P&ID pipe schedules indicated as: COL applicant.

Table 1.7-1 Piping and Instrumentation and Process Flow Diagrams

Tier 2 Fig. No.	Title	Type
4.6-8	CRD System	P&ID
4.6-9	CRD System	PFD
5.1-3	Nuclear Boiler System	P&ID
5.4-4	Reactor Recirculation System	P&ID
5.4-5	Reactor Recirculation System	PFD
5.4-8	Reactor Core Isolation Cooling System	P&ID
5.4-9	Reactor Core Isolation Cooling System	PFD
5.4-10	Residual Heat Removal System	P&ID
5.4-11	Residual Heat Removal System	PFD
5.4-12	Reactor Water Cleanup System	P&ID
5.4-13	Reactor Water Cleanup System	PFD
6.2-39	Atmospheric Control System	P&ID
6.2-40	Flammability Control System	P&ID
6.3-1	High Pressure Core Flooder System	PFD
6.3-7	High Pressure Core Flooder System	P&ID
6.5-1	Standby Gas Treatment System	P&ID
6.7-1	High Pressure Nitrogen Gas Supply System	P&ID
9.1-1	Fuel Pool Cooling and Cleanup System	P&ID
9.1-2	Fuel Pool Cooling and Cleanup System	PFD
9.2-1	Reactor Building Cooling Water System	P&ID
9.2-2	HVAC Normal Cooling Water System	P&ID
9.2-3	HVAC Emergency Cooling Water System	P&ID
9.2-4	Makeup Water System (Condensate)	P&ID
9.2-5	Makeup Water System (Purified)	P&ID
9.2-7	Reactor Service Water System	P&ID
9.3-1	Standby Liquid Control System	P&ID
9.3-1A	Standby Liquid Control System	PFD
9.3-6	Instrument Air System	P&ID
9.3-7	Service Air System	P&ID
9.4-1	Control Building HVAC	PFD
9.4-8	Drywell Cooling System	P&ID
9.5-1	Suppression Pool Cleanup System	P&ID

Table 1.7-1 Piping and Instrumentation and Process Flow Diagrams (Continued)

Tier 2 Fig. No.	Title	Type
11.2-1	Radwaste System	PFD Simplified
11.3-1	Offgas System	PFD
11.3-2	Offgas System	P&ID

Table 1.7-2 Instrument Engineering, Interlock Block and Single-Line Diagrams

Tier 2 Fig. No.	Title	Type
5.2-8	Leak Detection and Isolation System	IED
5.4-14	Reactor Water Cleanup System	IBD
7.2-9	Reactor Protection System	IED
7.2-10	Reactor Protection System	IBD
7.3-1	High Pressure Core Flooder System	IBD
7.3-2	Nuclear Boiler System	IBD
7.3-3	Reactor Core Isolation Cooling System	IBD
7.3-4	Residual Heat Removal System	IBD
7.3-5	Leak Detection and Isolation System	IBD
7.3-6	Standby Gas Treatment System	IBD
7.3-7	Reactor Building Cooling Water/Reactor Service Water System	IBD
7.3-9	HVAC Emergency Cooling Water System	IBD
7.3-10	High Pressure Nitrogen Gas System	IBD
7.4-1	Standby Liquid Control System	IBD
7.4-2	Remote Shutdown System	IED
7.4-3	Remote Shutdown System	IBD
7.6-1	Neutron Monitoring System	IED
7.6-2	Neutron Monitoring System	IBD
7.6-5	Process Radiation Monitoring System	IED
7.6-7	Containment Atmosphere Monitoring System	IED
7.6-8	Containment Atmosphere Monitoring System	IBD
7.6-11	Suppression Pool Temperature Monitoring System	IED
7.6-12	Suppression Pool Temperature Monitoring System	IBD
7.7-2	Rod Control and Information System	IED
7.7-3	Rod Control and Information System	IBD
7.7-4	Control Rod Drive System	IBD
7.7-5	Recirculation Flow Control System	IED
7.7-7	Recirculation Flow Control System	IBD
7.7-8	Feedwater Control System	IED

Table 1.7-2 Instrument Engineering, Interlock Block and Single-Line Diagrams (Continued)

Tier 2 Fig. No.	Title	Type
7.7-9	Feedwater Control System	IBD
7.7-12	Steam Bypass and Pressure Control System	IED
7.7-13	Steam Bypass and Pressure Control System	IBD
7.7-14	Fuel Pool Cooling and Cleanup System	IBD
8.3-1	Electrical Power Distribution System	SLD
8.3-2	Instrument and Control Power Supply System	SLD
8.3-3	Plant Vital AC Power Supply System	SLD
8.3-4	Plant Vital DC Power Supply System	SLD

Table 1.7-3 Conversion to ASME Standard Units

	From	To convert to	Divide by
(1)	**Pressure/Stress**		
	kilopascal	1 Pound/Square Inch	6.894757
	kilopascal	1 Atmosphere (STD)	101.325
	kilopascal	1 Foot of Water (39.2°F)	2.98898
	kilopascal	1 Inch of Water (60°F)	0.24884
	kilopascal	1 Inch of HG (32°F)	3.38638
(2)	**Force/Weight**		
	newton	1 Pound - force	4.448222
	kilogram	1 Ton (Short)	907.1847
	kilogram	1 Tons (Long)	1016.047
(3)	**Heat/Energy**		
	joule	1 Btu	1055.056
	joule	1 Calorie	4.1868
	kilowatt-hour	1 Btu	0.0002930711
	kilowatts	1 Horsepower (U.K)	0.7456999
	kilowatt-hour	1 Horsepower-Hour	0.7456999
	kilowatt	1 Btu/Min	0.0175725
	joule/gram	1 Btu/Pound	2.326
(4)	**Length**		
	millimeter	1 Inch	25.4
	centimeter	1 Inch	2.54
	meter	1 Inch	0.0254
	meter	1 Foot	0.3048
	centimeter	1 Foot	30.48
	meter	1 Mile	1609.344
	kilometer	1 Mile	1.609344
(5)	**Volume**		
	liter	1 Cubic Inch	0.01638706
	cubic centimeter	1 Cubic Inch	16.38706
	cubic meter	1 Cubic Foot	0.02831685

Table 1.7-3 Conversion to ASME Standard Units (Continued)

	From	To Convert to	Divide by
	cubic centimeter	1 Cubic Foot	28316.85
	liter	1 Cubic Foot	28.31685
	cubic meter	1 Cubic Yard	0.7645549
	liter	1 Gallon (US)	3.785412
	cubic centimeter	1 Gallon (US)	3785.412
	E-03 cubic centimeter	1 Gallon (US)	3.785412
(6)	**Volume Per Unit Time**		
	cubic centimeter/s	1 Cubic Foot/Min	471.9474
	cubic meter/h	1 Cubic Foot/Min	1.69901
	liter/s	1 Cubic Foot/Min	0.4719474
	cubic meter/s	1 Cubic Foot/Sec	0.02831685
	E-05 cubic meter/s	1 Gallon/Min (US)	6.30902
	cubic meter/h	1 Gallon/Min (US)	0.22712
	liter/s (101.325 kPaA, 15.56°C)	1 STD CFM (14.696 psia, 60°F)	0.4474
	cubic meter/h (101.325 kPaA, 15.56°C)	1 STD CFM (14.696 psia, 60°F)	1.608
(7)	**Velocity**		
	centimeter/s	1 Foot/Sec	30.48
	centimeter/s	1 Foot/Min	0.508
	meter/s	1 Foot/Min	0.00508
	meter/min	1 Foot/Min	0.3048
	centimeter/s	1 Inches/Sec	2.45
(8)	**Area**		
	square centimeter	1 Square Inch	6.4516
	E-04 square meter	1 Square Inch	6.4516
	square centimeter	1 Square Foot	929.0304
	E-02 square meter	1 Square Foot	9.290304
(9)	**Torque**		
	newton-meter	1 Foot Pound	1.355818
(10)	**Mass Per Unit Time**		
	kilogram/s	1 Pound/Sec	0.4535924

Table 1.7-3 Conversion to ASME Standard Units (Continued)

	From	To Convert to	Divide by
	kilogram/min	1 Pound/Min	0.4535924
	kilogram/h	1 Pound/Min	27.215544
(11)	**Mass Per Unit Volume**		
	kilogram/cubic meter	1 Pound/Cubic Inch	27679.90
	kilogram/cubic meter	1 Pound/Cubic Foot	16.01846
	kilogram/cubic centimeter	1 Pound/Cubic Inch	0.0276799
	liter/s	1 Gallon/Min	0.0630902
(12)	**Dynamic Viscosity**		
	Pa•s	1 Pound-Sec/Sq Ft	47.88026
(13)	**Specific Heat/Heat Transfer**		
	joule/kilogram kelvin	1 Btu/Pound-Deg F	4168.8
	watt/square meter kelvin	1 Btu/Hr-Sq Ft-Deg F	5.678263
	watt/square meter kelvin	1 Btu/Sec-Sq Ft-Deg F	2.044175E+4
	watt/square meter	1 Btu/Hr-Sq Ft	3.154591
(14)	**Temperature**		
	degrees celsius	Degrees Fahrenheit	$T_{\circ F}=T_{\circ C}x1.8+32$
	Degree C Increment	1 Degree F Increment	0.555556
(15)	**Electricity**		
	coulomb	1 ampere hour	3600
	seimens/meter	1 mho/centimeter	100
(16)	**Light**		
	candels/square meter	1 candela/square inch	1550.003
	lux	1 footcandle	10.76391
(17)	**Radiation**		
	megabequerel	1 curie	37,000
	gray	1 rad	0.01
	sievert	1 rem	0.01

Note:

Rounding of Calculated values per Appendix C of ANSI/IEEE Std 268.

Table 1.7-4 Conversion Tables—Metric to ASME Standard Units

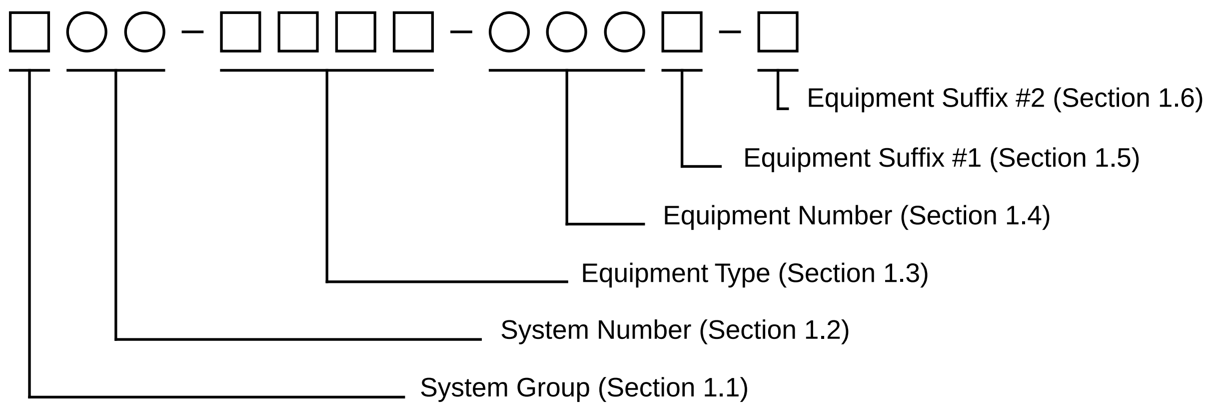
Flow-Volume Per Unit Time							
m^3/h	gal/min	m^3/h	gal/min	m^3/h	gal/min	m^3/h	gal/min
1	4.4	10	44	100	440	1000	4402
2	8.8	20	88	200	881	2000	8805
3	13.2	30	132	300	1321	3000	13207
4	17.6	40	176	400	1761	4000	17610
5	22.0	50	220	500	2201	5000	22012
6	26.4	60	264	600	2641	6000	26414
7	30.8	70	308	700	3082	7000	30817
8	35.2	80	352	800	3522	8000	35219
9	39.6	90	396	900	3962	9000	39621
Temperature							
°C	°F	°C	°F	°C	°F	°C	°F
0.1	32.18	1	33.8	10	50	100	212
0.2	32.36	2	35.6	20	68	200	392
0.3	32.54	3	37.4	30	86	300	572
0.4	32.72	4	39.2	40	104	400	752
0.5	32.90	5	41.0	50	122	500	932
0.6	33.08	6	42.8	60	140	600	1112
0.7	33.26	7	44.6	70	158	700	1292
0.8	33.44	8	46.4	80	176	800	1472
Pressure							
kPa	psi	kPa	psi	kPa	psi	kPa	psi
1	0.145	10	1.45	100	14.51	1000	145.1
2	0.290	20	2.90	200	29.01	2000	290.1
3	0.435	30	4.35	300	43.52	3000	435.2
4	0.580	40	5.80	400	58.02	4000	580.2
5	0.725	50	7.25	500	72.53	5000	725.3
6	0.870	60	8.70	600	87.03	6000	870.3
7	1.015	70	10.15	700	101.54	7000	1015.4
8	1.160	80	11.60	800	116.04	8000	1160.4
9	1.306	90	13.06	900	130.55	9000	1305.5

Table 1.7-4 Conversion Tables—Metric to ASME Standard Units (Continued)

Length							
cm	inch	cm	inch	m	ft	m	ft
0.01	0.004	0.1	0.039	1	3.28	10	32.81
0.02	0.008	0.2	0.079	2	6.56	20	65.62
0.03	0.012	0.3	0.118	3	9.84	30	98.43
0.04	0.016	0.4	0.157	4	13.12	40	131.2
0.05	0.020	0.5	0.197	5	16.40	50	164.0
0.06	0.024	0.6	0.236	6	19.69	60	196.9
0.07	0.028	0.7	0.276	7	22.97	70	229.7
0.08	0.032	0.8	0.315	8	26.25	80	262.5
0.09	0.035	0.9	0.354	9	29.53	90	295.3

Table 1.7-5 Drawing Standards

1.0 Equipment Identification



1.1 System Groups

The plant systems and facilities are divided into several major groups. Each group is represented by a single alphabetical letter as follows:

A	Plant in general
B	Reactor steam-generating systems
C	Control systems
D	Radiation monitoring systems
E	Core cooling systems
F	Reactor handling equipment
G	Reactor auxiliary systems
H	Control panels
J	Fuels
K	Waste-processing systems
N	Plant main systems
P	Plant auxiliary systems
R	Onsite electrical systems
S	Power transmission and receiving systems
T	Reactor containment vessel and ancillary facilities
U	Various buildings and ancillary facilities
W	Water intake facilities and ancillary facilities
Y	Other facilities on the grounds

Note: The following letters are not used: I, L, M, O, Q, V, X, Z

Table 1.7-5 Drawing Standards (Continued)

1.2 System Numbers

The system number for each system or facility consists of a two-digit number. Table 3.2-1 shows the system group and system numbers (MPL numbers) for each system and facility.

1.3 Equipment Type

The equipment type is represented by from one to four alphabetical letters as follows:

Mechanical Equipment

Identifying Letter	Description	
A	Tanks	Such as collection tanks, sample tanks, surge tanks, precoat tanks, backwashing tanks, sludge and resin tanks, other tanks, lining vats
B	Heat transfer equipment	Various types of heat exchangers, coolers, condensers, heaters
C	Rotating equipment	Such as various types of pumps and prime movers, fans and blowers, generators, exciters
D	Other equipment	Such as reactor pressure vessel, reactor internals, steam separators, dryers, control rod drive mechanisms, hydraulic control units, control rods, flow-limiting orifices, strainers, filters, demineralizers, agitators, extractors, ejectors, dispersers, and other types of equipment
E	Tools and servicing equipment	
F	Valves and their operators (where supplied)	
G	Pipes, hangers and supports	
H	Insulation	

Structural Equipment

Identifying Letter	Description
U	Foundation and supporting structure
V	Steel structures
W	Structural concrete and reinforcement bars
X	Equipment structures such as flues, chimneys, ducts, louvers, and cable trays

Table 1.7-5 Drawing Standards (Continued)

Electrical Equipment			
Identifying Letter	Description	**Identifying Letter**	Description
J	Electrical equipment-buses, transformers, power supply facilities	MV/I	Millivolt/current converters
K	Auxiliary relays	O/E	Optic/electric converters
L	Limiters	P/E	Pneumatic-electric converters (including air-pressure-to-current and air-pressure-to-voltage converters)
P	Panels and racks	R/I	Resistance/current converters
S	Operation switch	RMC	Remote controllers
T	Timing relays	RMS	Remote operating switches
Z	Complicated controllers such as ratio setters, function generators, division/multiplication calculators, time-lag calculators, addition/subtraction calculators. All microprocessor based algorithms.	R/P	Resistance/pneumatic converters
AM	Analog memory	RY	Relay modules
D/D	Converters	SQRT	Square-root calculators
E/O	Electric/optic converters	SRU	Resistance units
E/P	Electropneumatic converters (including current-to-air-pressure and voltage-to-air-pressure convertors)	S/S	Selector switch
E/S	Power supply for instrumentation	SSA	Selector-selector switch automatic
E/T	Relay terminal boards	TDS	Time delay switches
I/O	I/O module	TMC	Cycle timers
I/V	Current/voltage converters	TPR	Program timers
M/A	Manual and manual/automatic controllers	V/V	Voltage/voltage converters
MRY	Deviation monitor		

Table 1.7-5 Drawing Standards (Continued)

1.3.1 Instrumentation

The identification for “equipment type” provides information about the measured variable as well as the instrument function (Figure 1.7-1, sheet 2).

1.4 Equipment Number

Mechanical, structural and electrical equipment is numbered from 001 to 999 within the system for each equipment type.

1.4.1 Flow Direction Numbering Method

Equipment numbers are assigned in the direction of flow starting from the reactor vessel (or the upstream flow boundary) and moving in sequence from upstream to downstream. In systems which have two flow paths, the main flow path takes priority. The “Flow-Direction” method takes priority over the “Alternate” method below.

1.4.2 Alternate Numbering Method

For items having the same “Equipment Type” with different specifications and arranged in parallel, the equipment numbers are assigned according to equipment layout following a priority according to the direction, either from north to south or from the sea to the mountains. The north-to-south direction takes priority over the sea-to-mountain direction. However, within a system, the degree of importance of the individual pieces of equipment takes priority over the aforesaid rule, and the numbers are assigned in order of priority from the more important pieces of equipment. In the case in which the items are in parallel and are arranged above and below each other (e.g., upper and lower floors), the priority in numbering from more important to less important supersedes numbering from upper to lower floor.

1.4.3 Rules for Adding and Eliminating Equipment

When equipment is added to a system as the design progresses, the sequential numbers in the system upstream and downstream of the added equipment are not changed, and the added equipment is given the number following the last of the sequential numbers of the equipment at that time. When equipment is eliminated, its equipment number shall not be used again, and the numbers of the equipment on the downstream side remain unchanged.

Table 1.7-5 Drawing Standards (Continued)

1.4.4 Valve Numbering

Valves are divided into three categories—(1) Process Valves, (2) Drain Valves and Vent Valves, and (3) Instrument Valves having the following sets of numbers:

Process Valves	001 to 499
Drain and Vent Valves	500 to 699
Instrument Valves	700 to 999

1.4.5 Instrument Numbering

Equipment numbers for instruments are assigned in a series, for instruments only, from the upstream side of the system. They are assigned without relation to the symbols for the type of equipment; that is, without regard to the variables measured and measuring functions. The following sets of numbers are used for instruments according to their location and equipment classification:

001 to 299	Instruments installed in local panels.
301 to 399	Instruments installed locally, attached to equipment only.
601 to 999	Instruments installed in main control room, including instrument functions performed by multiplexer. The instrument number assigned to the latter is prefixed by the letter Z.

For a system having more than one fluid stream, instruments are numbered in sequence with those used for water first, then for steam and then for air. Within any of the above categories, for local instruments mounted on equipment, the priority is for level instruments first, then pressure and then temperature. Instruments measuring the same quantities, in this case, are numbered in sequence from those which have higher setting values, or from those which have higher upper limit values.

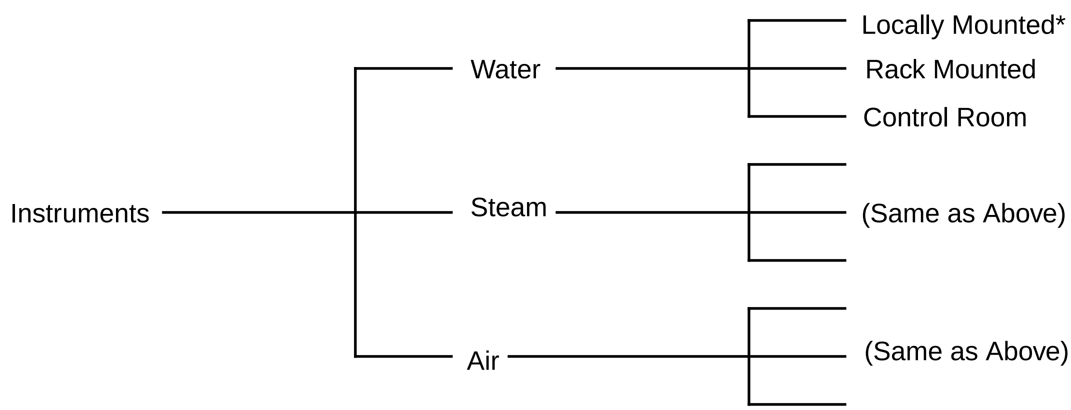
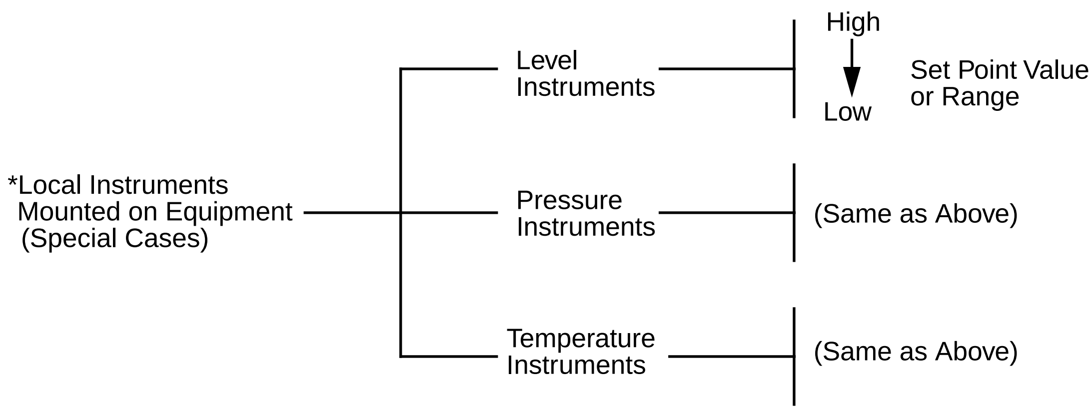


Table 1.7-5 Drawing Standards (Continued)



1.4.6 Numbering Within a Loop

Instruments or instrument functions performed by the multiplexer in the same loop have the same last two numbers. Instruments located in the main control room that receive signals from locally installed instruments in the same loop are numbered by adding 600 to the local or local panel-mounted instrument number in the loop.

1.5 Equipment Suffix #1

Equipment suffix #1 shall consist of a single letter (A, B, C, etc.). This is assigned when equipment or instruments in a system have the same equipment numbers and are required to be differentiated because of safety/separation considerations or because there are redundant instrument or mechanical loops. The following set of guidelines is followed in assigning equipment suffix #1:

(a) Equipment suffix #1 is the same as the suffix assigned to the reactor vessel nozzle to which the associated system or subsystem containing that equipment or instrument is connected.

(b) For equipment or instruments arranged in parallel and their systems having the same flow direction, the equipment suffix #1 is assigned by the equipment arrangement. The numbering is done from north-to-south or from sea-to-mountain. The north-to-south direction takes priority over the sea-to-mountain direction.

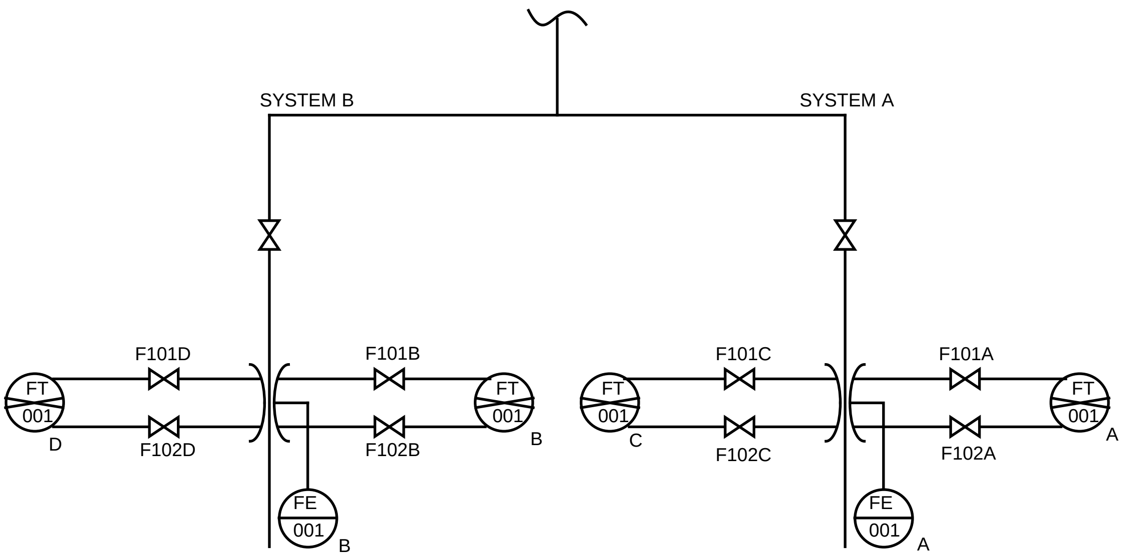
(c) For equipment or instruments having the same flow direction but which are installed at upper and lower levels, the suffix #1 is assigned as “A”, “B”, “C”, etc., from the lower level up.

(d) For the secondary system, the equipment suffix #1 is the same as that for the equipment in the primary system.

(e) If equipment or instruments belonging to an interfacing system are connected to equipment with suffix A and B in the primary system, their equipment suffix #1 may be omitted if they can be differentiated without it.

Table 1.7-5 Drawing Standards (Continued)

(f) When components connected to a dual system are further divided, the equipment suffix #1 is assigned in a staggered fashion. That is, component elements of the secondary system which are connected to system A have suffix A, C, E, G, J, L,...., while those which are connected to system B have suffix as B, D, F, H, K, M.....



(g) The Hydraulic Control Units (HCUs) in the Control Rod Drive (CRD) System shall be assigned a different type of equipment suffix #1. The core-coordinates of the two fuel bundles to which a particular HCU belongs shall be used as suffix. For example, C12D0010722/2718 represents an HCU for control rods belonging to fuel bundles at core coordinates 07,22 and 27,18.

1.6 Equipment Suffix #2

Equipment suffix #2 is only used for instruments if necessary. This number will differentiate instruments of the same type in an instrument loop. A single digit number is used in specifying the equipment suffix #2.

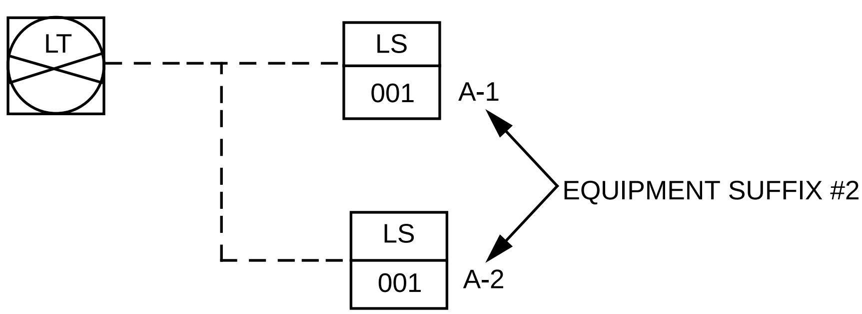


Table 1.7-5 Drawing Standards (Continued)

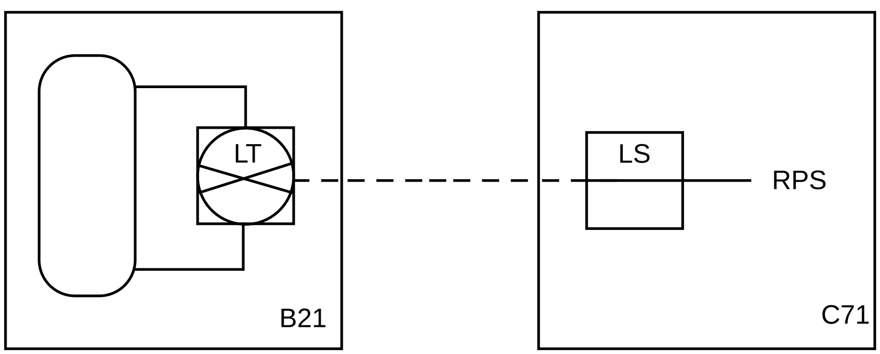
1.7 Numbering of Shared Equipment

The following rules are followed in assigning numbers to equipment shared between systems or between loops within a system:

(a) “System group and number” of shared instruments—Assign the system number of whichever system has the largest number of instruments using the shared component. If the number of instruments is the same, use the system number which has the system group and system number closest to A00.

(b) “Equipment Number” of shared instruments—Except for instruments with a recording function, the same rule as outlined for unshared instruments is followed. All recorders, regardless of the “measured variable”, are numbered in a single series from 001 to 999.

(c) In instrumentation systems which monitor the process quantities of one system and perform interlock controls with another system, the primary instruments (elements and transmitters, or local switches) are assigned the system and equipment number of the system being monitored; and the other instruments are assigned the system and equipment number of the latter system. However, switch functions sending signals to multiple systems are excepted from the above rule and are considered as a part of the primary system.



2.0 Piping and Instrument Diagram Standards

2.1 P&ID

The P&ID provides a schematic illustration of a specific system. It may contain the following information:

(a) Equipment, valves, piping and instrumentation required for system function.

(b) Interface between components and other systems to show control and function of each valve.

(c) Electrical and instrumental interlocks, protective features and logic connections.

(d) Valves and associated components shown in plant normal operating mode (e.g., valve open-valve closed) or as defined on the drawing or specified in the notes. An exception to this is a three-way solenoid valve supplied with associated air or nitrogen-operated valve, which is shown in the de-energized mode.

Table 1.7-5 Drawing Standards (Continued)

(e) The P&ID shows the location of the valves, pipe junctions, pumps, instruments, tanks and other equipment in actual sequence along the pipeline. Piping takeoff connections from equipment are shown at their proper locations relative to the equipment whenever practicable.

(f) System design (maximum) conditions such as design pressure, design temperature, material and seismic class are given in the P&ID. The changing points for these items are defined.

(g) The identification of building(s) (including yard) is defined.

(h) Equipment, valves and instrumentation belonging to another system or used in common are shown by broken line with two dots between each line break, and the system group and number(s) clearly stated for other system or systems.

(i) Instrument root valves in the instrument piping branching from the process piping are shown. Valves on the instrument side are not shown.

(j) Drain, vent and test connections are shown on P&IDs. The discharge of drains and vents is assigned to the appropriate drain system whose system acronym is written at the end of the line.

(k) System (group and number) and system acronyms are given in the upper right-hand corner of the first sheet of the P&ID.

(l) Use of a black box on a P&ID is allowed when other sheets of the same drawing or a different drawing contains complete information about the contents of the black box. A note is added that specifies the drawing number of the contents of the black box.

(m) Piping is divided into three categories—Process piping, drain and vent piping and instrument piping. The following sets of numbers are used for these categories:

Process Piping	001 to 499
Drain and Vent Piping	500 to 699
Instrument Piping	700 to 999

When numbers in a series run out, four-digit pipe numbers may be used. For example, for process piping, after 499, the numbers from 1001 to 1499 are used.

(n) The pipe numbering is done using the flow direction method, same as the equipment numbering method described in Subsection 1.4.1.

(p) Piping is basically identified by a single number (Example 1 below). If the P&ID is changed during the detailed design after the initial numbering and if an additional pipe number is required due to the change, a suffix number may be applied (Example 2):

Example 1	400A-MUWC-001
Example 2	400A-MUWC-001-1
	400A-MUWC-001-2

(q) Nominal pipe diameter is identified by the symbol “A” preceeded by a millimeter dimension consistent with inches or the ANSI standard outside diameter as shown by the following examples:

Table 1.7-5 Drawing Standards (Continued)

Nominal Diameter Inches	ABWR P&ID Symbol mm	ANSI OD mm
2	50A	60.3
4	100A	114.3
8	200A	219.1
16	400A	406.4
24	600A	609.6

3.0 Process Flow Diagram Standards

3.1 PFD

The process flow diagram shows the engineering requirements or conditions (e.g., modes of operation, flow, pressure and temperature) at specified locations throughout the system using the following guidelines:

(a) Main flow lines of the system are shown. Drain lines, vent lines and instrument lines are not shown.

(b) Identification numbers of the main valves are included. All symbols used are the same as the P&ID.

(c) Operating conditions for each mode of operation are shown in a tabular form.

(d) The position nodes for the key locations at which the operating conditions are given are shown by the symbol ◯ (circle) or ⬡ (hexagon).

3.6 Operating Conditions

The operating conditions include the following items:

(a) Flow (m^3/h)

(b) Pressure (kPaG or kPaA)

(c) Temperature (°C)

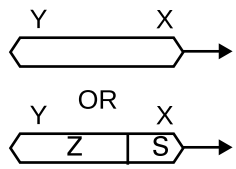
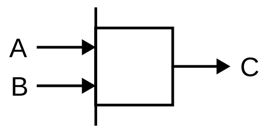
(d) Valve opening/closing conditions.

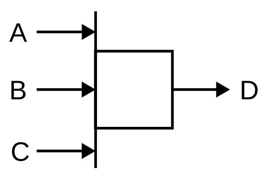
(e) Maximum pressure drop (m) if necessary.

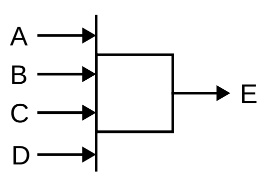
The following figure is located in Chapter 21 :

Figure 1.7-1 Piping and Instrumentation Diagram Symbols (Sheets 1–2)

Figure 1.7-2 Graphical Symbols for Use in IBDs (Sheet 1 of 8)

<table>
<tr><th>No.</th><th>Function</th><th>Graphic Symbol</th><th>Explanation of Function</th></tr>
<tr><td>1</td><td>Condition Symbol or Signal</td><td>
</td><td>Symbol indicates signal condition or action (e.g. valve close signal). Action shifts to right when condition is met.
Y – Is instrument number
Z – Represents the name of condition signal
S – Shows above or below setpoint for transfer of signal condition</td></tr>
<tr><td>2A</td><td>AND</td><td>

2-Input AND</td><td>Output exists if and only if all specified inputs exist
<table>
<tr><th>A</th><th>B</th><th>C</th></tr>
<tr><td>0</td><td>0</td><td>0</td></tr>
<tr><td>0</td><td>1</td><td>0</td></tr>
<tr><td>1</td><td>0</td><td>0</td></tr>
<tr><td>1</td><td>1</td><td>1</td></tr>
</table>
2-Input AND Truth Table</td></tr>
<tr><td>2B</td><td></td><td>

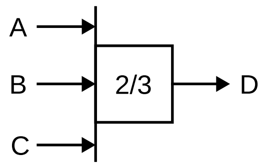
3-Input AND</td><td><table>
<tr><th>A</th><th>B</th><th>C</th><th>D</th></tr>
<tr><td>0</td><td>0</td><td>0</td><td>0</td></tr>
<tr><td>0</td><td>0</td><td>1</td><td>0</td></tr>
<tr><td>0</td><td>1</td><td>0</td><td>0</td></tr>
<tr><td>0</td><td>1</td><td>1</td><td>0</td></tr>
<tr><td>1</td><td>0</td><td>0</td><td>0</td></tr>
<tr><td>1</td><td>0</td><td>1</td><td>0</td></tr>
<tr><td>1</td><td>1</td><td>0</td><td>0</td></tr>
<tr><td>1</td><td>1</td><td>1</td><td>1</td></tr>
</table>
3-Input AND Truth Table</td></tr>
<tr><td>2C</td><td></td><td>

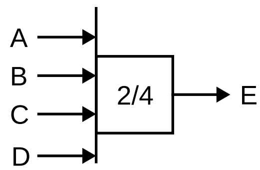
4-Input AND</td><td>4-Input AND Truth Table (Not Shown)</td></tr>
</table>

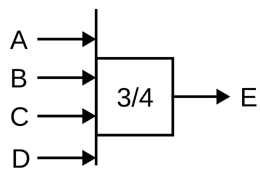
Figure 1.7-2 Graphical Symbols for Use in IBDs (Sheet 2 of 8)

<table>
<tr><th>No.</th><th>Function</th><th>Graphic Symbol</th><th>Explanation of Function</th></tr>
<tr><td>2D

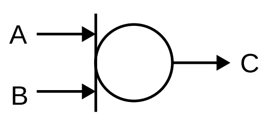
2E

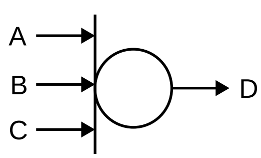
2F</td><td>Coincident Variable Gate (CVG)</td><td>

2/3 AND


2/4 AND


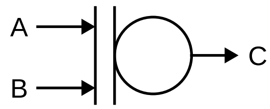
3/4 AND</td><td>Output exists if specified number of inputs exist (2 of 3, 2 of 4, or 3 of 4)
<table>
<tr><th>A</th><th>B</th><th>C</th><th>D</th></tr>
<tr><td>0</td><td>0</td><td>0</td><td>0</td></tr>
<tr><td>0</td><td>0</td><td>1</td><td>0</td></tr>
<tr><td>0</td><td>1</td><td>0</td><td>0</td></tr>
<tr><td>1</td><td>0</td><td>0</td><td>0</td></tr>
<tr><td>0</td><td>1</td><td>1</td><td>1</td></tr>
<tr><td>1</td><td>0</td><td>1</td><td>1</td></tr>
<tr><td>1</td><td>1</td><td>0</td><td>1</td></tr>
<tr><td>1</td><td>1</td><td>1</td><td>1</td></tr>
</table>
Truth Table 2/3 AND

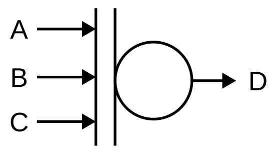
2/4 and 3/4 AND’s Truth Tables (Not Shown)</td></tr>
<tr><td>3A</td><td>OR</td><td>

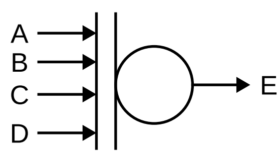
2-Input OR</td><td>Output exists only when at least one input exist
<table>
<tr><th>A</th><th>B</th><th>C</th></tr>
<tr><td>0</td><td>0</td><td>0</td></tr>
<tr><td>0</td><td>1</td><td>1</td></tr>
<tr><td>1</td><td>0</td><td>1</td></tr>
<tr><td>1</td><td>1</td><td>1</td></tr>
</table>
Truth Table 2-Input OR</td></tr>
<tr><td>3B</td><td></td><td>

3-Input OR</td><td>
<table>
<tr><th>A</th><th>B</th><th>C</th><th>D</th></tr>
<tr><td>0</td><td>0</td><td>0</td><td>0</td></tr>
<tr><td>0</td><td>0</td><td>1</td><td>1</td></tr>
<tr><td>0</td><td>1</td><td>0</td><td>1</td></tr>
<tr><td>1</td><td>0</td><td>0</td><td>1</td></tr>
<tr><td>0</td><td>1</td><td>1</td><td>1</td></tr>
<tr><td>1</td><td>0</td><td>1</td><td>1</td></tr>
<tr><td>1</td><td>1</td><td>0</td><td>1</td></tr>
<tr><td>1</td><td>1</td><td>1</td><td>1</td></tr>
</table>
Truth Table 3-Input OR</td></tr>
</table>

Figure 1.7-2 Graphical Symbols for Use in IBDs (Sheet 3 of 8)

<table>
<tr><th>No.</th><th>Function</th><th>Graphic Symbol</th><th>Explanation of Function</th></tr>
<tr><td>3C</td><td>X OR (Exclusive "OR")</td><td>

2-Input X OR</td><td>These logic symbols represent an Exclusive "OR" Gate whose output assumes 1 state if one and only one of the logic input assumes the 1 state
<table>
<tr><th>A</th><th>B</th><th>C</th></tr>
<tr><td>0</td><td>0</td><td>0</td></tr>
<tr><td>0</td><td>1</td><td>1</td></tr>
<tr><td>1</td><td>0</td><td>1</td></tr>
<tr><td>1</td><td>1</td><td>0</td></tr>
</table>
Truth Table 2-Input Exclusive "OR"</td></tr>
<tr><td>3D</td><td></td><td>

3-Input X OR</td><td><table>
<tr><th>A</th><th>B</th><th>C</th><th>D</th></tr>
<tr><td>0</td><td>0</td><td>0</td><td>0</td></tr>
<tr><td>0</td><td>0</td><td>1</td><td>1</td></tr>
<tr><td>0</td><td>1</td><td>0</td><td>1</td></tr>
<tr><td>1</td><td>0</td><td>0</td><td>1</td></tr>
<tr><td>1</td><td>0</td><td>1</td><td>0</td></tr>
<tr><td>0</td><td>1</td><td>1</td><td>0</td></tr>
<tr><td>1</td><td>1</td><td>0</td><td>0</td></tr>
<tr><td>1</td><td>1</td><td>1</td><td>0</td></tr>
</table>
Truth Table 3-Input Exclusive "OR"</td></tr>
<tr><td>3E</td><td>Exclusive "OR"</td><td>

4-Input X OR</td><td><table>
<tr><th>A</th><th>B</th><th>C</th><th>D</th><th>E</th></tr>
<tr><td>0</td><td>0</td><td>0</td><td>0</td><td>0</td></tr>
<tr><td>0</td><td>0</td><td>0</td><td>1</td><td>1</td></tr>
<tr><td>0</td><td>0</td><td>1</td><td>0</td><td>1</td></tr>
<tr><td>0</td><td>1</td><td>0</td><td>0</td><td>1</td></tr>
<tr><td>1</td><td>0</td><td>0</td><td>0</td><td>1</td></tr>
<tr><td>0</td><td>0</td><td>1</td><td>1</td><td>0</td></tr>
<tr><td>0</td><td>1</td><td>0</td><td>1</td><td>0</td></tr>
<tr><td>1</td><td>0</td><td>0</td><td>1</td><td>0</td></tr>
<tr><td>0</td><td>1</td><td>1</td><td>0</td><td>0</td></tr>
<tr><td>1</td><td>0</td><td>1</td><td>0</td><td>0</td></tr>
<tr><td>1</td><td>1</td><td>0</td><td>0</td><td>0</td></tr>
<tr><td>0</td><td>1</td><td>1</td><td>1</td><td>0</td></tr>
<tr><td>1</td><td>0</td><td>1</td><td>1</td><td>0</td></tr>
<tr><td>1</td><td>1</td><td>0</td><td>1</td><td>0</td></tr>
<tr><td>1</td><td>1</td><td>1</td><td>0</td><td>0</td></tr>
<tr><td>1</td><td>1</td><td>1</td><td>1</td><td>0</td></tr>
</table>
Truth Table 4-Input Exclusive "OR"</td></tr>
</table>

Figure 1.7-2 Graphical Symbols for Use in IBDs (Sheet 4 of 8)

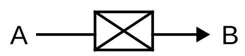
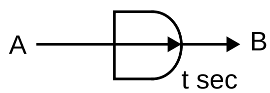
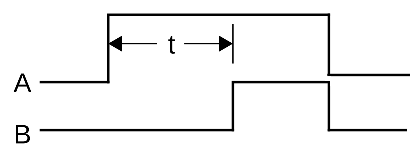
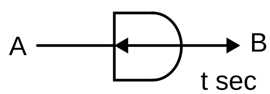
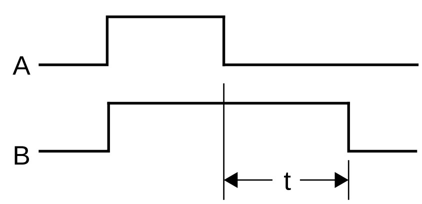
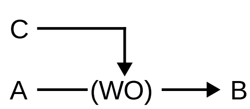
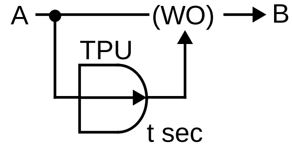
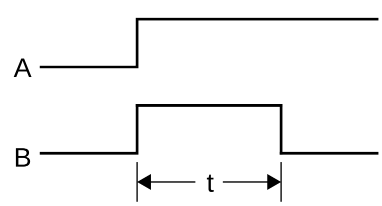
<table>
<tr><th>No.</th><th>Function</th><th>Graphic Symbol</th><th>Explanation of Function</th></tr>
<tr><td>4</td><td>Not</td><td></td><td>This symbol shows the “NOT” condition. Output B is opposite to input A<table><tr><th>A</th><th>B</th></tr><tr><td>0</td><td>1</td></tr><tr><td>1</td><td>0</td></tr></table></td></tr>
<tr><td>5A</td><td>Timer Elements</td><td>TPU
Delayed Initiation</td><td>TPU– Signal B is energized within specified time limit (t) after signal A is energized. B terminates when A terminates.
</td></tr>
<tr><td>5B</td><td></td><td>TDO
Delayed Termination (Reset)</td><td>TDO– Initially B is energized when A is energized. signal B is de-energized within specified time limit (t) after signal A is de-energized
</td></tr>
<tr><td>6A</td><td>Wipe-Out (Signal Block)</td><td></td><td>When signal C is not present, signal A is transmitted to B. When signal C is present, signal A is stopped and does not flow to B. (WO: Wipe-out)<table><tr><th>A</th><th>C</th><th>B</th></tr><tr><td>0</td><td>0</td><td>0</td></tr><tr><td>0</td><td>1</td><td>0</td></tr><tr><td>1</td><td>0</td><td>1</td></tr><tr><td>1</td><td>1</td><td>0</td></tr></table></td></tr>
<tr><td>6B</td><td>Delayed Wipe-Out (One-Shot)</td><td></td><td>The output signal to B is stopped after time interval “t”.
</td></tr>
</table>

Figure 1.7-2 Graphical Symbols for Use in IBDs (Sheet 5 of 8)

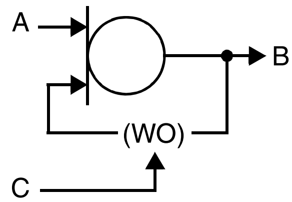
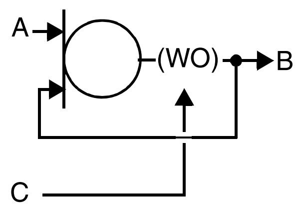
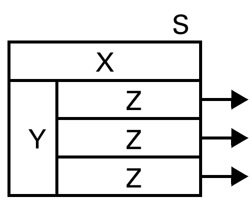
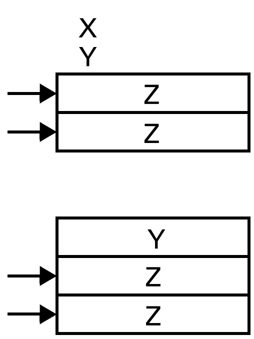
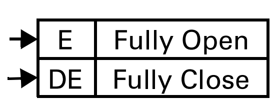
No.	Function	Graphic Symbol	Explanation of Function
7A	Self-Hold or Reset		When condition C does not exist, Condition A holds itself and there is output to B. The self holding is released when condition C is established and there is an output to B only when there is an A condition (A takes priority).
7B			When condition C does not exist, Condition A holds itself and there is output to B. The self holding is released when condition C is established and there is no output to B (C takes priority).
8	Operating Switch		S – Place of installation X – Switch operation name Y – Switch type, e.g. CS—Control Switch Spring Return COS—Control Operating Switch Position Hold PBS—Pushbutton Switch PBL—Pushbutton Illuminated Type KS—Key Switch (Spring Return) KOS—Key Operating Switch (Position Hold) CRT—CRT Touch-Screen Z – Switch Position: On, Off, Pull Hold, … etc.
9	Control Component or Device		Shows a component or device to be controlled X – Part # of controlled device Y – Controlled device name e.g. pump, valve, etc. Z – Controlled condition, e.g., Start, Stop, On, Off, Open, Close, … etc.
10	Electromagnetic Valve		This symbol represents an electromagnetic valve E – Energize DE – D energized

Figure 1.7-2 Graphical Symbols for Use in IBDs (Sheet 6 of 8)

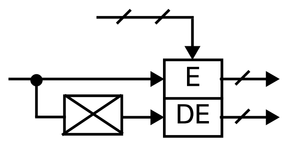
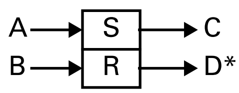
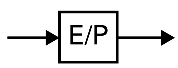
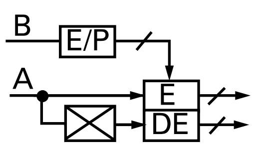
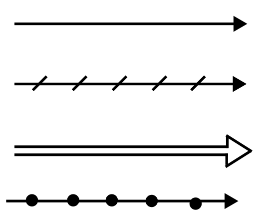
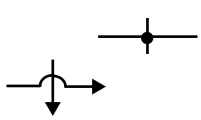
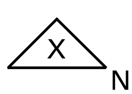
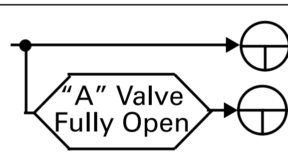
No.	Function	Graphic Symbol	Explanation of Function
11	Electromagnetic Pilot Valve		This symbol represents an electromagnetic pilot valve E – Energized DE – De-energized
12	Memory (Flip-Flop)	 *Output D shall not be shown if not used.	S Represents "Set Memory" R Represents "Reset Memory" Logic output C exists when logic input A exists. C continues to exist regardless of subsequent state of A and until reset by input at B C remains terminated regardless of subsequent state of B, until A causes memory to reset. Logic output D, if used, exist when C does not exist, and D does not exist when C exists.
13	Static Transducer (Converter)		This device converts "E" (Electrical Signal) to "P" (Pneumatic Signal)
14	Electromagnetic Pilot Valve for Control		Shows a pilot electromagnetic valve for a control valve. When the pilot electromagnetic valve is energized by a signal from A, opening of the control valve is adjusted by a signal from B.
15	Transmission Signals or Lines		Indicates electrical signal and flow direction Indicates pneumatic line and flow direction Indicated oil hydraulic pressure line and flow direction Indicates mechanical linkage
16	Electrical Signal Connection		Signal is connected electrically Signal is not connected electrically
17	Signal Input		This symbol represents an input signal to a computer, display, test panel, etc. as designated by the letter X inside the triangle. The letter N indicates the assigned signal number.
18	Operational Condition		This graphical presentation in used in sequential control

Figure 1.7-2 Graphical Symbols for Use in IBDs (Sheet 7 of 8)

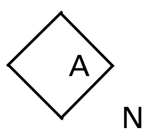
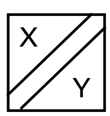
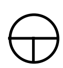
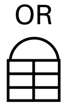
No.	Function	Graphic Symbol	Explanation of Function
19	Virtual Condition Signal		Used only for signals which do not actually exist but are convenient to show on IBD.
20	Panel Indicator Lights	R	Red indication light: Shows actuation, input and valve opening
		G	Green indication light: Shows stop, interruption and valve closure
		W	White (milk-white) indication light: Shows condition indication, automatic mode operation… etc.
		O	Orange indication light: Shows caution and failure
		T	Colorless or transparent indicating light
	CRT Indicator Lights	X CRT	Light indicator to be shown on CRT. X represents the color of the light to be indicated.
	Alarm		Indicates an annunciated alarm or warning. The letter N indicates the alarm number.
21	Isolator		This symbol represents that the input signal shall be divisionally isolated from the output signal X – Input division number Y – Isolation output division number
22 to 32		Intentionally left blank for future additions	
33	Signal Transfer	 OR 	These symbols indicate signal transfer to other location(s). The upper half of the symbol is used to enter the transfer code. The lower left portion of symbol is used to reference the sheet number to go to, and the lower right hand portion will indicate the location where the signal can be found. The transfer code shall utilize either an English letter or a number if signal transfer is within the same sheet or to other sheets of the IBD. For signal transfer from/or to other MPL systems, the transfer code shall be expressed with 2 English letters starting with “AA”. Also indicate the system MPL reference where the signal goes to or originates next to the symbol.

Figure 1.7-2 Graphical Symbols for Use in IBDs (Sheet 8 of 8)

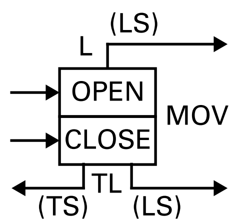
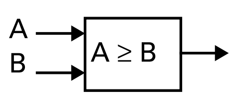
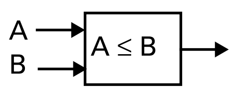
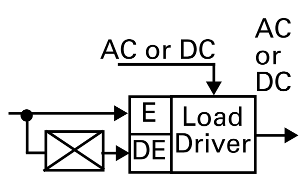
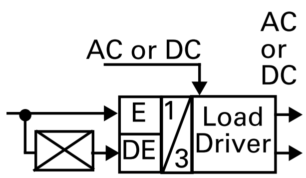
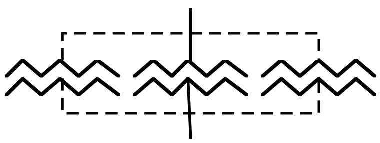
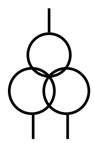
No.	Function	Graphic Symbol	Explanation of Function
34	MOV's Position Indication Designators (See Appendix "D" for Application Examples)	 LS: Limit SW TS: Torque SW	L — Limit Off L(T) — Limit Off With Torque Backup TL — Both Limit and Torque Off T — Torque Off TL(C) — With Chattering Prevention at TL T(C) — With Chattering Prevention at T The above letter designators are used to show the control methods of motor-driven valves. The control method should be indicated above the left side of the component block for valve "Opening" and below the left side for valve "Closure".
35A 35B	Comparator	 	These symbols represent a comparator that provide an output when the condition A ≥ B or A≤ B is met.
36A	Load Driver	 Standard Load Driver	This symbol represents a standard load driver.
36B		 Isolated Load Driver	This Symbol represents a load driver whose output power signal DIV 3 is isolated internally from the input logic signal division 1.

Figure 1.7-3 Graphical Symbols for Use in Electrical SLDs (Sheet 1 of 4)

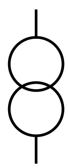
Transformer



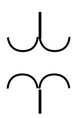
Three 1ø



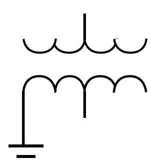
Three winding - 3ø



Two winding - 3ø or 1ø

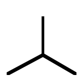


Two winding - 3ø

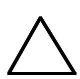


Two winding - 3ø center tap ground

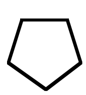
Note: Symbol in transformer = winding connection type i.e.



Star



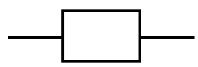
Delta



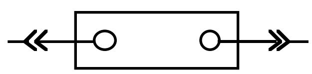
Special

Figure 1.7-3 Graphical Symbols for Use in Electrical SLDs (Sheet 2 of 4)

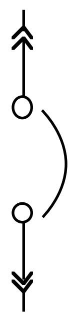
Circuit Breakers/Disconnect Devices



Main Generator/Switchyard Circuit Breaker

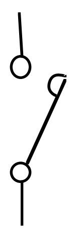


Metal Clad Switchgear Circuit Breaker (Drawout Type)



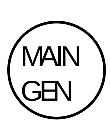
Air Circuit Breaker (Drawout Type)

Isolated Phase Bus Disconnect Link



Disconnect Switch

Generators

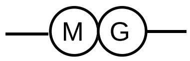


Main Plant Generator

Diesel Generator

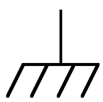
Figure 1.7-3 Graphical Symbols for Use in Electrical SLDs (Sheet 3 of 4)

Combustion Turbine Generator

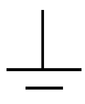


Motor Generator

Ground



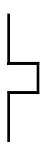
M/C, P/C Switchgear Ground



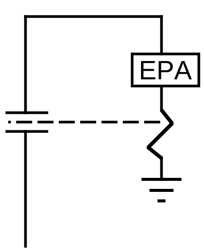
Transformer Ground

Overload/Protection Devices

Magnetic Overload Devices

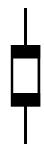


Thermal Overload Devices



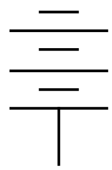
Electrical Protection Assembly

Figure 1.7-3 Graphical Symbols for Use in Electrical SLDs (Sheet 4 of 4)

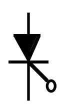


Fuse

DC Supply



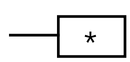
Station Battery



Full Wave Rectifier

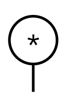
Devices

Current Transformer (CT)



*Function Identifier

STD	-	Static Trip Device
27D	-	DC under voltage relay
64	-	Ground detector relay
76	-	DC overcurrent relay
84	-	Selector switch



* Monitoring Function

V	-	Voltmeter
A	-	Ampmeter

Abbreviations

M/C	Metal Clad
P/C	Power Center
MCC	Motor Control Center
R/B	Reactor Building
C/B	Control Building
T/B	Turbine Building
RW/B	Radwaste Building
UHS	Ultimate Heat Sink

1.8 Conformance with Standard Review Plan and Applicability of Codes and Standards

1.8.1 Conformance with Standard Review Plan

This subsection provides the information required by 10CFR50.34(g) showing conformance with the Standard Review Plan (SRP). The summary of differences from the SRP section is presented by SRP section in Tables 1.8-1 through 1.8-18. (See Subsection 1.8.4.1 for COL license information.)

1.8.2 Applicability of Codes and Standards

Standard Review Plans, Branch Technical Positions, Regulatory Guides and Industrial Codes and Standards which are applicable to the ABWR design are provided in Tables 1.8-19, 1.8-20 and 1.8-21. Applicable revisions are also shown.

1.8.3 Applicability of Experience Information

Experience information is routinely made available and distributed to design personnel in the design process. Nuclear field experience is maintained in hard copy form in functional component and library files and in the GE world-wide computer retrieval system.

Generic Letters and IE Bulletins, Information Notices and Circulars covering the decade including 1980 through the current issues (late 1991) were reviewed for applicability to the ABWR design. The review was enhanced by associating related experiences and tracing referenced occurrences. This was accomplished starting with the current issues of the Generic Letters and proceeding back into the decade. The Circulars, Bulletins and Notices were reviewed in that order. Interfacing experience was included in the review. The selection of ABWR information was based on the significance to future design and operation guidance. Included is a list of NUREGs related to the closing of current safety issues. Experience that resulted in applicable rules, codes and standards was not repeated. Table 1.8-22 lists the experience information that has been included in the ABWR design or impacts the COL applicant. (See Subsection 1.8.4.2 for COL license information.)

A systematic procedure encompassing available resources was used to identify the applicability of experience information resulting in Table 1.8-22. Engineering management surveyed the indices of annual experience information to identify those very likely to be applicable to the ABWR. The remaining potentially applicable experiences were reviewed individually. Experience information not deemed applicable to the ABWR design (issues pertaining to other reactor types, scram discharge volume, etc.) were not included in Table 1.8-22. The experience information categories applicable to the ABWR design in Table 1.8-22 include experience information accommodated by a design change, covered by review of USIs/GSIs or an issue that impacts the ABWR design but must be addressed by the COL applicant. This latter category is included as COL license information.

Experiences related to identified regulatory or industry developed resolutions were eliminated to avoid repetition except for selected experiences that have a nuisance potential for reoccurring. Lead system engineers classified the more complex experiences.

Reference to the new or novel design features used in the ABWR are provided below:

Feature	Tier 2 Section
Fine Motion Control Rod Drive	4.6
Internal Reactor Pumps	5.4.1
Multiplexing	7A.2
Digital/Solid-State Control	7A.7
Overpressure Protection System	6.2.5.2.6, 6.2.5.3, 6.2.5.4
AC-Independent Water Addition System	5.4.7.1.1.10
Lower Drywell Flooder	9.5.1.2

1.8.4 COL License Information

1.8.4.1 SRP Deviations

The SRP sections to be addressed by the COL applicant are indicated in the comments column of Table 1.8-19 as “COL Applicant”. Where applicable the COL applicant will provide the information required by 10CFR50.34(g) similar to Tables 1.8-1 through 1.8-18 (see Subsection 1.8.1).

1.8.4.2 Experience Information

The experience information to be addressed by the COL applicant are indicated in the comment column of Table 1.8-22 as “COL Applicant” (see Subsection 1.8.3).

Table 1.8-1 Summary of Differences from SRP Section 1

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
None	None	None	None

Table 1.8-2 Summary of Differences from SRP Section 2

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
2.2.1-2.2.2	See Table 2.1-1.	Limits imposed on selected SRP Section II acceptance criteria by (1) the envelope of the ABWR Standard Plant site parameters and (2) evaluations assumptions.	2.1
2.2.3			
2.3.1			
2.3.4			
2.4.1			
2.4.4			
2.4.5			
2.4.6			
2.4.8			
2.4.11.6			
2.4.12			
2.5.2.7		OBE is not a design requirement.	

Table 1.8-3 Summary of Differences from SRP Section 3

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
3.6.1 and 3.6.2	II—Postulated pipe rupture.	Large bore piping can utilize leak before break option as provided in GDC-4 October 27, 1987, “Modification of General Design Criterion 4”.	3.6 and 3.6.3
3.6.1 and 3.6.2	ASB 3-1 and MEB 3-1 Consider 1/2 SSE for postulating pipe ruptures.	Earthquake stresses considered only in cumulative usage factor calculations when postulating pipe ruptures	3.6.1.1, 3.6.2.1.4.2, 3.6.2.1.4.3, 3.6.2.1.4.4., 3.6.2.1.5.2, 3.6.2.1.5.3

Table 1.8-3 Summary of Differences from SRP Section 3 (Continued)

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
3.6.2	MEB 3-1, B.1.c.(1).(b) - Pipe ruptures must be postulated if Eq.(10) of NB-3653 exceeds 2.4 Sm.	Pipe ruptures postulated only if, in NB-3653, Eq. (10) and either (12) or Eq. (13) exceed 2.4 Sm.	3.6.1.1 and 3.6.2.1.4.3
3.7.1 and 3.7.3	II.2 - Two earthquakes, the SSE and the OBE shall be considered in the design.	The ABWR will be based on a single earthquake (SSE) design.	3.6, 3.7, 3.9
3.9.2	II-E.2.g - For multiply supported equipment use envelope RS and;	Independent Support Motion Response Spectrum methods acceptable for use.	3.7.3.8.1.10
	Combine responses from inertia effects with anchor displacements by Absolute Sum.	Combine responses from inertia effects with anchor displacements by SRSS.	3.7.3.8.1.8
3.7.3	II.2.b—Determination of number of OBE cycles.	The ABWR is based on a single earthquake (SSE) design, two SSE events with 10 peak stress cycles per event are used.	3.7.3.2

Table 1.8-4 Summary of Differences from SRP Section 4

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
None	None	None	None

Table 1.8-5 Summary of Differences from SRP Section 5

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
5.2.3	II.3.b.(3)—Reg Guide 1.71, Welding Qualification for Areas of Limited Accessibility.	Alternate position employed.	5.2.3.4.2.3
5.2.4	II.1—Inspection of Class 1 pressure-containing components.	Some welds inaccessible for volumetric examination.	5.2.4.2.2
5.4.6	Deleted		

Table 1.8-5 Summary of Differences from SRP Section 5 (Continued)

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
5.4.7	Branch Technical Position RSB 5-1, B.1.(b) and (c)—Diverse interlocks for RHR suction isolation valves.	No diversity of interlocks.	5.4.7.1.1.7

Table 1.8-6 Summary of Differences from SRP Section 6

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
6.2.1.1	Design provision for automatic actuation of wetwell spray 10 minutes following a LOCA signal	Manual actuation of wetwell spray 30 minutes following a LOCA signal	6.2.1.1.5.6.1
6.2.4	One isolation valve inside and one isolation valve outside containment	Both isolation valves located outside the containment	6.2.4.3.2.2.2.3
	Purge and vent valves to close in ð5 seconds	Purge and vent valves will close in ð20 seconds	6.2.4.3.2.2.2.3
6.2.1.1	Monthly vacuum valve operability test	Operability tests only performed during refueling outages	6.2.1.1.5.6.3

Table 1.8-7 Summary of Differences from SRP Section 7

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
7.1	Table 7-1: 1a IEEE-279, 4.19	RHR Annunciation at loop level.	7.3.2.3.2 (1) 7.3.2.4.2 (1) 7.4.2.3.2 (1)
7.1	Table 7-1: 2i GDC 20	Some modes of RHR are not automatic.	7.3.2.3.2 (2)(b) 7.3.2.4.2 (2)(b) 7.4.2.3.2 (1)
7.1	Table 7-1: 3a Reg Guide 1.22	Clarification of requirements.	7.3.2.1.2. (3)(a)
7.1	Table 7-1: 3a Reg Guide 1.22	HP/LP interlocks cannot be tested during power operation.	7.6.2.3.2 (3)
7.1	Table 7-1: 3c Reg Guide 1.53	Continuity testing of certain solenoids.	7.3.2.1.2. (3)(c)
7.1	Table 7-1: 3c Reg Guide 1.53	Some components are not redundant.	7.3.2.5.2 (3) 7.4.2.2.2 (3)

Table 1.8-7 Summary of Differences from SRP Section 7 (Continued)

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
7.1	Table 7-1: 3c Reg Guide 1.53	Limited redundancy of remote shutdown.	7.4.2.4.2 (1) 7.4.2.4.2 (3)
7.1	Table 7-1: 3e Reg Guide 1.75	Alternate positions employed.	7.1.2.10.5
7.1	Table 7-1: 3h Reg Guide 1.118	Some sensors cannot be tested at power operation.	7.2.2.2.1 (7) 7.2.2.2.3.1 (10) 7.2.2.2.3.1 (21)
7.1	Table 7-1: 4i BTP ICSB 22	Some actuators cannot be exercised during power operation.	7.3.2.1.2 (4)(d) 7.4.2.3.2 (4)(c)

Table 1.8-8 Summary of Differences from SRP Section 8

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
8.1	Table 8-1: 2f Reg Guide 1.75	Exception to LOCA trip for certain non-1E loads.	8.1.3.1.2.2 (6) Appendix 9A
8.1	Table 8-1: 2f Reg Guide 1.75	4.572 m cable marking intervals.	8.3.3.5.1.3
8.1	Table 8-1: 2f IEEE-384	LDS divisional separation in steam tunnel.	8.3.3.6.1.2 (2)

Table 1.8-9 Summary of Differences from SRP Section 9

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
9.3.1	II.1—Particles shall not exceed 3 micrometer.	Instrument air is filtered to 5 micrometer.	9.3.6.2
9.3.2	II.k.5—Capable of sampling liquid of 370,000 MBq/cm^3.	Capable of sampling liquids of 37,000 MBq/cm^3.	9.3.2.3.1
9.4.1	GDC 19	Site specific.	6.4.7.1
9.5.1	Section 7.b	Control Room Complex 1. Peripheral rooms 2. Underfloor (subfloor) 3. Consoles & cabinets	9.5.1
	Section 7.j	Diesel fuel oil tank capacity	9.5.1
	Section 7.i	Diesel Generator operation	9.5.1

Table 1.8-10 Summary of Differences from SRP Section 10

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
None	None	None	None

Table 1.8-11 Summary of Differences from SRP Section 11

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
11.1	II.9—BWR GALE Code	Alternate computer code.	20.3.7 (Response to Question 460.1)

Table 1.8-12 Summary of Differences from SRP Section 12

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
None	None	None	None

Table 1.8-13 Summary of Differences from SRP Section 13

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
None	None	None	None

Table 1.8-14 Summary of Differences from SRP Section 14

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
None	None	None	None

Table 1.8-15 Summary of Differences from SRP Section 15

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
15.1.1-15.1.4	Acceptable analytical model.	3D simulator instead of REDY Code is used. 3D simulator was approved for use in GESTAR review by NRC.	15.1.1.3.2

Table 1.8-15 **Summary of Differences from SRP Section 15** (Continued)

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
15.2.6	All recirculation pumps are tripped simultaneously by the initiating event.	Only four of ten RIPs are tripped. This is based on ABWR design.	15.2.6.1.1
15.3.1-15.3.2	Complete recirculation pumps trip is considered as a moderate-frequency transient.	Trip of all RIPs is classified as an infrequent low probability event with special acceptance for fuel failure.	15.3.1.1.2
15.3.3-15.3.4	II.10—coincident turbine trip, loss of offsite power and coastdown of undamaged pumps.	Not analyzed with the assumption. If the assumption is made, the consequence would be similar to the event shown in 15.2.6.	15.3.3.2.2.
15.4.2	Analysis of uncontrolled control rod withdrawal at power.	No quantitative analysis is provided because ABWR's ATLM design prevents this transient from occurring.	15.4.2.2
15.4.4-15.4.5	II.2.(b)—Fuel cladding integrity.	MCPR not calculated, since transients are very mild.	15.4.4.3 15.4.5.3.2.1 and 15.4.5.3.2.2
15.4.9	Not applicable SRP for BWR.	Discussion is provided to show this event cannot occur with ABWR FMCRD design.	15.4.9
15.4.10	Analysis of rod drop accidents.	No quantitative analysis is provided because ABWR's FMCRD design prevents this accident from occurring.	15.4.10.1 & 15.4.10.2
15.6.5	II.2—Use of assumptions outlined in Reg Guide 1.3.	ABWR LOCA analysis incorporates suppression pool scrubbing IAW SRP 6.5.5 and in variance from R.G. 1.3. Fission product plateout and removal is incorporated in the analysis of leakage sources through the main steamlines and into the turbine condenser based upon BWROG analysis of acceptability of the steamlines and condenser to mitigate releases without requiring Seismic Category I structures.	15.6.5.5

Table 1.8-16 Summary of Differences from SRP Section 16

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
None	None	None	None

Table 1.8-17 Summary of Differences from SRP Section 17

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
17.1	II.1 - Applicant is responsible for overall QA program.	GE & major technical associates are responsible for their own QA programs.	17.0 17.1.1 17.1.2
17.1	II.2,3,4,7,13,17,18-Meet identified quality related Reg Guides.	Reg Guide 1.28, Rev. 3 and alternative positions employed.	Table 17.0-1 17.1.2, 17.1.3 17.1.4, 17.1.7 17.1.13, 17.1.17 17.1.18
17.1	II.2 - Meet identified regulations and codes.	Differences between domestic and international designs are identified in a controlled list.	17.1.3

Table 1.8-18 Summary of Differences from SRP Section 18

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Subsection Where Discussed
None	None	None	None

Table 1.8-19 Standard Review Plans and Branch Technical Positions Applicable to ABWR

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
Chapter 1 Introduction and General Description of Plant					
1.8	Interfaces for Standard Design	1	7/81	Yes	
Chapter 2 Site Characteristics					
2.1.1	Site Location and Description	2	7/81	—	COL Applicant
2.1.2	Exclusion Area Authority and Control	2	7/81	—	COL Applicant
2.1.3	Population Distribution	2	7/81	—	COL Applicant
2.2.1–2.2.2	Identification of Potential Hazards in Site Vicinity	2	7/81	—	COL Applicant
2.2.3	Evaluation of Potential Accidents	2	7/81	—	COL Applicant
2.3.1	Regional Climatology	2	7/81	—	COL Applicant
2.3.2	Local Meteorology	2	7/81	—	COL Applicant
2.3.3	Onsite Meteorological Measurements Programs	2	7/81	—	COL Applicant
	Appendix A	2	7/81	—	COL Applicant
2.3.4	Short-Term Diffusion Estimates for Accidental Atmospheric Releases	1	7/81	—	COL Applicant
2.3.5	Long-Term Diffusion Estimates	2	7/81	—	COL Applicant
2.4.1	Hydrologic Description	2	7/81	—	COL Applicant
	Appendix A	2	7/81	—	COL Applicant
2.4.2	Floods	2	7/81	—	COL Applicant
2.4.3	Probable Maximum Flood (PMF) on Streams and Rivers	2	7/81	—	COL Applicant

Table 1.8-19 Standard Review Plans and Branch Technical Positions Applicable to ABWR (Continued)

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
2.4.4	Potential Dam Failures	2	7/81	—	COL Applicant
2.4.5	Probable Maximum Surge and Seiche Flooding	2	7/81	—	COL Applicant
2.4.6	Probable Maximum Tsunami Flooding	2	7/81	—	COL Applicant
2.4.7	Ice Effects	2	7/81	—	COL Applicant
2.4.8	Cooling Water Canals and Reservoirs	2	7/81	—	COL Applicant
2.4.9	Channel Diversions	2	7/81	—	COL Applicant
2.4.10	Flood Protection Requirements	2	7/81	—	COL Applicant
2.4.11	Cooling Water Supply	2	7/81	—	COL Applicant
2.4.12	Groundwater	2	7/81	—	COL Applicant
	BTP HGEB 1	2	7/81	—	COL Applicant
2.4.13	Accidental Releases of Liquid Effluents in Ground and Surface Waters	2	7/81	—	COL Applicant
2.4.14	Technical Specifications and Emergency Operation Requirements	2	7/81	—	COL Applicant
2.5.1	Basic Geologic and Seismic Information	2	7/81	—	COL Applicant
2.5.2	Vibratory Ground Motion	2	8/89	—	COL Applicant
2.5.3	Surface Faulting	2	7/81	—	COL Applicant
2.5.4	Stability of Subsurface Materials and Foundations	2	7/81	—	COL Applicant
2.5.5	Stability of Slopes	2	7/81	—	COL Applicant
Chapter 3 Design of Structures, Components, Equipment, and Systems					
3.2.1	Seismic Classification	1	7/81	Yes	

Table 1.8-19 Standard Review Plans and Branch Technical Positions Applicable to ABWR (Continued)

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
3.2.2	System Quality Group Classification	1	7/81	Yes	
	Appendix A (Formerly BTP RSB 3-1)	1	7/81	Yes	
	Appendix B (Formerly BTP RSB 3-2)	1	7/81	Yes	
3.3.1	Wind Loadings	2	7/81	Yes	
3.3.2	Tornado Loadings	2	7/81	Yes	
3.4.1	Flood Protection	2	7/81	Yes	
3.4.2	Analysis Procedures	2	7/81	Yes	
3.5.1.1	Internally Generated Missiles (Outside Containment)	2	7/81	Yes	
3.5.1.2	Internally Generated Missiles (Inside Containment)	2	7/81	Yes	
3.5.1.3	Turbine Missiles	2	7/81	Yes	
3.5.1.4	Missiles Generated by Natural Phenomena	2	7/81	Yes	
3.5.1.5	Site Proximity Missiles (Except Aircraft)	1	7/81	Yes	
3.5.1.6	Aircraft Hazards	2	7/81	Yes	
3.5.2	Structures, Systems, and Components to be Protected from Externally Generated Missiles	2	7/81	Yes	
3.5.3	Barrier Design Procedures	1	7/81	Yes	
	[*Appendix A*	*0*	*7/81*][(1)]	Yes	
3.6.1	Plant Design for Protection Against Postulated Piping Failures in Fluid Systems Outside Containment	1	7/81	Yes	
	BTP ASB-3-1	1	7/81	Yes	
3.6.2	Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping	1	7/81	Yes	
	BTP MEB-3-1	2	6/87	Yes	
3.7.1	Seismic Design Parameters	2	8/89	Yes	
3.7.2	Seismic System Analysis	2	8/89	Yes	
3.7.3	Seismic Subsystem Analysis	2	8/89	Yes	
3.7.4	Seismic Instrumentation	1	7/81	Yes	
3.8.1	Concrete Containment	1	7/81	Yes	
	[*Appendix*	*0*	*7/81*][(1)]	Yes	

Table 1.8-19 Standard Review Plans and Branch Technical Positions Applicable to ABWR (Continued)

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
3.8.2	Steel Containment	1	7/81	Yes	
3.8.3	Concrete and Steel Internal Structures of Steel or Concrete Containments	1	7/81	Yes	
3.8.4	Other Seismic category I Structures	1	7/81	Yes	
	Appendix A	0	7/81	Yes	
	Appendix B	0	7/81	Yes	
	Appendix C	0	7/81	Yes	
	Appendix D	0	7/81	Yes	
3.8.5	Foundations	1	7/81	Yes	
3.9.1	Special Topics for Mechanical Components	2	7/81	Yes	
3.9.2	Dynamic Testing and Analysis of Systems, Components, and Equipment	2	7/81	Yes	
3.9.3	ASME Code Class 1, 2, and 3 Components, Component Supports, and Core Support Structures	1	7/81	Yes	
	Appendix A	1	4/84	Yes	
3.9.4	Control Rod Drive Systems	2	4/84	Yes	
3.9.5	Reactor Pressure Vessel Internals	2	7/81	Yes	
3.9.6	Inservice Testing of Pumps and Valves	2	7/81	Yes	
3.10	Seismic Qualification of Category I Instrumentation and Electrical Equipment	2	7/81	Yes	
3.11	Environmental Design of Mech. and Elec. Equip.	2	7/81	Yes	
Chapter 4 Reactor					
4.2	Fuel System Design	2	7/81	Yes	
	[Appendix A	*0*	*7/81]*(2)	Yes	
4.3	Nuclear Design	2	7/81	Yes	
	BTP CPB 4.3-1	2	7/81	Yes	
4.4	Thermal and Hydraulic Design	1	7/81	Yes	
4.5.1	Control Rod Drive Structural Materials	2	7/81	Yes	
4.5.2	Reactor Internal and Core Support Materials	2	7/81	Yes	
4.6	Functional Design of Control Rod Drive System	1	7/81	Yes	

Table 1.8-19 Standard Review Plans and Branch Technical Positions Applicable to ABWR (Continued)

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
Chapter 5 Reactor Coolant System and Connected Systems					
5.2.1.1	Compliance with the codes and Standard Rule, 10CFR50.55a	2	7/81	Yes	
5.2.1.2	Applicable Code Cases	2	7/81	Yes	
5.2.2	Overpressure Protection	2	7/81	Yes	
	BTP RSB 5-2	0	7/81	No	PWR only
5.2.3	Reactor Coolant Pressure Boundary Materials	2	7/81	Yes	
	BTP MTEB 5-7 (Superseded by NUREG-0313)				
5.2.4	Reactor Coolant Pressure Boundary Inservice Inspection and Testing	1	7/81	Yes	
5.2.5	Reactor Coolant Pressure Boundary Leakage Detection	1	7/81	Yes	
5.3.1	Reactor Vessel Materials	1	7/81	Yes	
5.3.2	Pressure-Temperature Limits	1	7/81	Yes	
	BTP MTEB 5-2	1	7/81	Yes	
5.3.3	Reactor Vessel Integrity	1	7/81	Yes	
5.4	Deleted				
5.4.1.1	Pump Flywheel Integrity (PWR)	1	7/81	No	PWR only
5.4.2.1	Steam Generator Materials	2	7/81	No	PWR only
	BTP MTEB 5-3	2	7/81	No	
5.4.2.2	Steam Generator Tube Inservice Inspection	1	7/81	No	PWR only
5.4.6	Reactor Core Isolation Cooling System (BWR)	3	7/81	Yes	
5.4.7	Residual Heat Removal (RHR) System	3	7/81	Yes	
	BTP RSB 5-1				
5.4.8	Reactor Water Cleanup System (BWR)	2	7/81	Yes	
5.4.11	Pressurizer Relief Tank	2	7/81	No	PWR only
5.4.12	Reactor Coolant System High Point Vents	0	7/81	Yes	
Chapter 6 Engineered Safety Features					
6.1.1	Engineered Safety Features Materials	2	7/81	Yes	
	BTP MTEB 6-1	2	7/81	Yes	

Table 1.8-19 Standard Review Plans and Branch Technical Positions Applicable to ABWR (Continued)

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
6.1.2	Protective Coating Systems (Paints)—Organic Materials	2	7/81	Yes	
6.2.1	Containment Functional Design	2	7/81	Yes	
6.2.1.1.A	PWR Dry Containments, Including Subatmospheric Containments	2	7/81	No	PWR only
6.2.1.1.B	Ice Condenser Containments	2	7/81	No	PWR only
6.2.1.1.C	Pressure Suppression Type BWR Containments	6	8/84	Yes	
	Appendix A	2	1/83	Yes	
	Appendix B	0	1/83	Yes	
6.2.1.2	Subcompartment Analysis	2	7/81	Yes	
6.2.1.3	Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents	1	7/81	Yes	
6.2.1.4	Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures	1	7/81	Yes	
6.2.1.5	Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies	2	7/81	No	PWR Only
	BTP CSB 6-1	2	7/81	No	PWR only
6.2.2	Containment Heat Removal Systems	4	10/85	Yes	
6.2.3	Secondary Containment Functional Design	2	7/81	Yes	
	BTP CSB 6-3	2	7/81	Yes	
6.2.4	Containment Isolation System	2	7/81	Yes	
	BTP CSB 6-4	2	7/81	Yes	
6.2.5	Combustible Gas Control in Containment	2	7/81	Yes	
	Appendix A	2	7/81	Yes	
	BTP CSB 6-2 (Superseded by Reg. Guide 1.7)				
6.2.6	Containment Leakage Testing	2	7/81	Yes	
6.2.7	Fracture Prevention of Containment Pressure Boundary	0	7/81	Yes	
6.3	Emergency Core Cooling System	2	4/84	Yes	
	BTP RSB 6-1	1	7/81	Yes	
6.4	Control Room Habitability Systems	2	7/81	Yes	

Table 1.8-19 Standard Review Plans and Branch Technical Positions Applicable to ABWR (Continued)

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
	Appendix A	2	7/81	Yes	
6.5.1	ESF Atmosphere Cleanup Systems	2	7/81	Yes	
6.5.2	Containment spray as a Fission Product Cleanup System	1	7/81	Yes	
6.5.3	Fission Product Control Systems and Structures	2	7/81	Yes	
6.5.4	Ice Condenser as a Fission Product Cleanup System	2	7/81	No	PWR only
6.5.5	Pressure Suppression Pool as a Fission Product Cleanup System	0	12/88	Yes	
6.6	Inservice Inspection of Class 2 and 3 Components	1	7/81	Yes	
6.7	Main Steam Isolation Valve Leakage Control System (BWR)	2	7/81	No	
Chapter 7 Instrumentation and Controls					
7.1	Instrumentation and Controls Introduction	3	2/84	Yes	
	Table 7-1 Acceptance Criteria and Guidelines for Instrumentation and Controls Systems Important to Safety	3	2/84	Yes	
	Table 7-2 TMI Action Plan Requirements for Instrumentation and Controls Systems Important to Safety	0	7/81	Yes	
	Appendix A	1	2/84	Yes	
	Appendix B	0	7/81	Yes	
7.2	Reactor Trip System	2	7/81	Yes	
	Appendix A (Superseded by SRP 7.1 App. B)				
7.3	Engineered Safety Features Systems	2	7/81	Yes	
	Appendix A (Superseded by SRP 7.1 App. B)				
7.4	Safe Shutdown Systems	2	7/81	Yes	
7.5	Information Systems Important to Safety	3	2/84	Yes	
7.6	Interlock Systems Important to Safety	2	7/81	Yes	
7.7	Control Systems	3	2/84	Yes	
	Appendix 7-A Branch Technical Positions (ICSB)	2	7/81	Yes	
	BTP ICSB 1 (DOR) (Deleted)				

Table 1.8-19 Standard Review Plans and Branch Technical Positions Applicable to ABWR (Continued)

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
	BTP ICSB 3	2	7/81	Yes	
	BTP ICSB 4 (PSB)	2	7/81	Yes	
	BTP ICSB 5 (Superseded by Std. Tech Specs)				
	BTP ICSB 9 (Superseded by Std. Tech Specs)				
	BTP ICSB 12	2	7/81	Yes	
	BTP ICSB 13	2	7/81	Yes	
	BTP ICSB 14	2	7/81	Yes	
	BTP ICSB 16 (Deleted)				
	BTP ICSB 19 (Deleted)				
	BTP ICSB 20	2	7/81	Yes	
	BTP ICSB 21	2	7/81	Yes	
	BTP ICSB 22	2	7/81	Yes	
	BTP ICSB 25 (Superseded by Std. Tech Specs)				
	BTP ICSB 26	2	7/81	Yes	
	Appendix 7-B General Agenda, Station Site Visits	1	7/81	Yes	
Chapter 8 Electric Power					
8.1	Electric Power-Interaction	2	7/81	Yes	
	Table 8-1 Acceptance Criteria and Guidelines for Electric Power Systems	2	7/81	Yes	
8.2	Offsite Power System	3	7/83	Yes	ABWR and COL Applicant
	Appendix A	0	7/83	Yes	ABWR and COL Applicant
8.3.1	AC Power Systems (Onsite)	2	7/81	Yes	
	Appendix (Superseded by BTP PSB-2)				
8.3.2	DC Power Systems (Onsite)	2	7/81	Yes	
	Appendix 8 — A Branch Technical Positions (PSB)	2	7/81	Yes	
	BTP ICSB 2 (PSB) (Superseded by IEEE-387)				
	BTP ICSB 4 (PSB)	2	7/81	Yes	

Table 1.8-19 Standard Review Plans and Branch Technical Positions Applicable to ABWR (Continued)

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
	BTP ICSB 8 (PSB)	2	7/81	Yes	
	BTP ICSB 11 (PSB)	2	7/81	Yes	
	BTP ICSB 15 (PSB) (Deleted)				
	BTP ICSB 17 (PSB) (Superseded by Reg. Guide 1.9)				
	BTP ICSB 18 (PSB)	2	7/81	Yes	
	BTP ICSB 21 (PSB)	2	7/81	Yes	
	BTP PSB 1	0	7/81	Yes	
	BTP PSB 2	0	7/81	Yes	
	Appendix 8 — B General Agenda, Station Site Visits	0	7/81	Yes	
Chapter 9 Auxiliary Systems					
9.1.1	New Fuel Storage	2	7/81	Yes	
9.1.2	Spent Fuel Storage	3	7/81	Yes	
9.1.3	Spent Fuel Pool Cooling and Cleanup System	1	7/81	Yes	
9.1.4	Light Load Handling System (Related to Refueling)	2	7/81	Yes	
	BTP ASB 9-1 (Superseded by NUREG-0554)				
9.1.5	Overhead Heavy Load Handling Systems	0	7/81	Yes	
9.2.1	Station Service Water System	4	6/85	Yes	ABWR and COL Applicant
9.2.2	Reactor Auxiliary Cooling Water Systems	3	6/86	Yes	ABWR and COL Applicant
9.2.3	Demineralized Water Makeup System	2	7/81	Yes	ABWR and COL Applicant
9.2.4	Potable and Sanitary Water Systems	2	7/81	Yes	ABWR and COL Applicant
9.2.5	Ultimate Heat Sink	2	7/81	—-	COL Applicant
	BTP ASB 9-2	2	7/81	Yes	

Table 1.8-19 Standard Review Plans and Branch Technical Positions Applicable to ABWR (Continued)

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
9.2.6	Condensate Storage Facilities	2	7/81	Yes	
9.3.1	Compressed Air System	1	7/81	Yes	
9.3.2	Process and Post-Accident Sampling Systems	2	7/81	Yes	
9.3.3	Equipment and Floor Drainage System	2	7/81	Yes	ABWR and COL Applicant
9.3.4	Chemical and Volume Control System (PWR) (Including Boron Recovery System)	2	7/81	No	PWR only
9.3.5	Standby Liquid Control System (BWR)	2	7/81	Yes	
9.4.1	Control Room Area Ventilation System	2	7/81	Yes	ABWR and COL Applicant
9.4.2	Spent Fuel Pool Area Ventilation System	2	7/81	Yes	
9.4.3	Auxiliary and Radwaste Area Ventilation System	2	7/81	Yes	ABWR and COL Applicant
9.4.4	Turbine Area Ventilation System	2	7/81	Yes	
9.4.5	Engineered Safety Feature Ventilation System	2	7/81	Yes	
9.5.1	Fire Protection Program	3	7/81	Yes	
	BTP CMEB 9.5-1	2	7/81	Yes	
	Appendix A (Deleted)				
9.5.2	Communication Systems	2	7/81	Yes	ABWR and COL Applicant
9.5.3	Lighting Systems	2	7/81	Yes	
9.5.4	Emergency Diesel Engine Fuel Oil Storage and Transfer System	2	7/81	Yes	
9.5.5	Emergency Diesel Engine Cooling Water System	2	7/81	Yes	
9.5.6	Emergency Diesel Engine Starting System	2	7/81	Yes	
9.5.7	Emergency Diesel Engine Lubrication System	2	7/81	Yes	
9.5.8	Emergency Diesel Engine Combustion Air Intake and Exhaust System	2	7/81	Yes	

Table 1.8-19 Standard Review Plans and Branch Technical Positions Applicable to ABWR (Continued)

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
Chapter10 Steam and Power Conversion System					
10.2	Turbine Generator	2	7/81	Yes	
10.2.3	Turbine Disk Integrity	1	7/81	Yes	
10.3	Main Steam Supply System	3	4/84	Yes	
10.3.6	Steam and Feedwater System Materials	2	7/81	Yes	
10.4.1	Main Condensers	2	7/81	Yes	
10.4.2	Main Condenser Evacuation System	2	7/81	Yes	
10.4.3	Turbine Gland Sealing System	2	7/81	Yes	
10.4.4	Turbine Bypass System	2	7/81	Yes	
10.4.5	Circulating Water System	2	7/81	Yes	ABWR and COL Applicant
10.4.6	Condensate Cleanup System	2	7/81	Yes	
10.4.7	Condensate and Feedwater System	3	4/84	Yes	
	BTP ASB 10-2	3	4/84	Yes	
10.4.8	Steam Generator Blowdown System (PWR)	2	7/81	No	PWR only
10.4.9	Auxiliary Feedwater System (PWR)	2	7/81	No	PWR only
	BTP ASB 10-1	2	7/81	No	PWR only
Chapter 11 Radioactive Waste Management					
11.1	Source Terms	2	7/81	Yes	
11.2	Liquid Waste Management Systems	2	7/81	Yes	
11.3	Gaseous Waste Management Systems	2	7/81	Yes	
	BTP ETSB 11-5	0	7/81	No	
11.4	Solid Waste Management Systems	2	7/81	Yes	
	BTP ETSB 11-3	2	7/81	Yes	
	Appendix 11.4-A	0	7/81	Yes	
11.5	Process and Effluent Radiological Monitoring Instrumentation and Sampling Systems	3	7/81	Yes	
	Appendix 11.5-A	1	7/81	Yes	

Table 1.8-19 Standard Review Plans and Branch Technical Positions Applicable to ABWR (Continued)

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
Chapter 12 Radiation Protection					
12.1	Assuring That Occupational Radiation Exposures Are as Low as Reasonably Achievable	2	7/81	Yes	
12.2	Radiation Sources	2	7/81	Yes	
12.3–12.4	Radiation Protection Design Features	2	7/81	Yes	
12.5	Operational Radiation Protection Program	2	7/81	—	COL Applicant
Chapter 13 Conduct of Operations					
13.1.1	Management and Technical Support Organization	2	7/81	—	COL Applicant
13.1.2–13.1.3	Operating Organization	2	7/81	—	COL Applicant
13.2	Training (Replaced by SRP Sections 13.2.1 and 13.2.2)				
13.2.1	Reactor Operator Training	0	7/81	—	COL Applicant
13.2.2	Training For Non-Licensed Plant Staff	0	7/81	—	COL Applicant
13.3	Emergency Planning	2	7/81	—	COL Applicant
13.4	Operational Review	2	7/81	—	COL Applicant
13.5	Plant Procedures (Replaced by SRP Sections 13.5.1 and 13.5.2)				
13.5.1	Administration Procedures	0	7/81	—	COL Applicant
13.5.2	Operating and Maintenance Procedures	1	7/85	—	COL Applicant
	Appendix A	0	7/85	—	COL Applicant
13.6	Physical Security	2	7/81	Yes	ABWR and COL Applicant

Table 1.8-19 Standard Review Plans and Branch Technical Positions Applicable to ABWR (Continued)

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
Chapter 14 Initial Test Program					
14.1	Initial Plant Test Programs—PSAR (Deleted)				
14.2	Initial Plant Test Programs—FSAR	2	7/81	Yes	
14.3	Standard Plant Design, Initial Test Program—Final Design Approval (FDA) (Deleted)				
Chapter 15 Accident Analysis					
15.0	Introduction	2	7/81	Yes	
15.1.1–15.1.4	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow, and Inadvertent Opening of a Steam Generator Relief or Safety Valve	1	7/81	Yes	
15.1.5	Steam System Piping Failures Inside and Outside of Containment (PWR)	2	7/81	No	PWR only
	Appendix A	2	7/81	No	PWR only
15.2.1–15.2.5	Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR), and Steam Pressure Regulator Failure (Closed)	1	7/81	Yes	
15.2.6	Loss of Nonemergency AC Power to the Station Auxiliaries	1	7/81	Yes	
15.2.7	Loss of Normal Feedwater Flow	1	7/81	Yes	
15.2.8	Feedwater system Pipe Breaks Inside and Outside Containment (PWR)	1	7/81	No	PWR only
15.3.1–15.3.2	Loss of Forced Reactor Coolant Flow Including Trip of Pump and Flow Controller Malfunctions	1	7/81	Yes	
15.3.3–15.3.4	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	2	7/81	Yes	
15.4.1	Uncontrolled control Rod Assembly Withdrawal from a Subcritical of Low Power Startup Condition	2	7/81	Yes	
15.4.2	Uncontrolled Control Rod Assembly Withdrawal at Power	2	7/81	Yes	
15.4.3	Control Rod Misoperation (System Malfunction or Operator Error)	2	7/81	Yes	

Table 1.8-19 Standard Review Plans and Branch Technical Positions Applicable to ABWR (Continued)

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
15.4.4–15.4.5	Startup of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	1	7/81	Yes	
15.4.6	Chemical and Volume Control System Malfunction That Results in a Decrease in the Boron Concentration in the Reactor Coolant (PWR)	1	7/81	No	PWR only
15.4.7	Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position	1	7/81	Yes	
15.4.8	Spectrum of Rod Ejection Accidents (PWR)	2	7/81	No	PWR only
	Appendix A	1	7/81	No	PWR only
15.4.9	Spectrum of Rod Drop Accidents (BWR)	2	7/81	Yes	
	Appendix A	2	7/81	Yes	
15.5.1–15.5.2	Inadvertent Operation of ECCS and Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory	1	7/81	Yes	
15.6.1	Inadvertent Opening of a PWR Pressurizer Relief Valve or a BWR Relief Valve	1	7/81	Yes	
15.6.2	Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment	2	7/81	Yes	
15.6.3	Radiological Consequences of Steam Generator Tube Failure (PWR)	2	7/81	No	PWR only
15.6.4	Radiological Consequences of Main Steam Line Failure Outside Containment (BWR)	2	7/81	Yes	
15.6.5	Loss-of-Coolant Accidents Resulting from Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	2	7/81	Yes	
	Appendix A	1	7/81	Yes	
	Appendix B	1	7/81	Yes	
	Appendix C (Deleted)				
	Appendix D	1	7/81	Yes	
15.7.1	Waste Gas System Failure (Deleted)				
15.7.2	Radioactive Liquid Waste System Leak or Failure (Released to Atmosphere) (Deleted)				

Table 1.8-19 Standard Review Plans and Branch Technical Positions Applicable to ABWR (Continued)

SRP No.	SRP Title	Appl. Rev.	Issued Date	ABWR Appli-cable?	Comments
15.7.3	Postulated Radioactive Release Due to Liquid-Containing Tank Failures	2	7/81	Yes	
15.7.4	Radiological Consequences of Fuel Handling Accidents	1	7/81	Yes	
15.7.5	Spent Fuel Cask Drop Accidents	2	7/81	Yes	
15.8	Anticipated Transients Without Scram	1	7/81	Yes	
	Appendix (Deleted)				
Chapter 16 Technical Specifications					
16.0	Technical Specifications	1	7/81	Yes	
Chapter 17 Quality Assurance					
17.1	Quality Assurance During the Design and Construction Phases	2	7/81	Yes	
17.2	Quality Assurance During the Operations Phase	2	7/81	—	COL Applicant
Chapter 18 Human Factors Engineering					
18.0	Human Factors Engineering/Standard Review Plan Development	1	9/84	Yes	
18.1	Control Room	0	9/84	Yes	
	Appendix A	0	9/84	Yes	
18.2	Safety Parameter Display System	0	11/84	Yes	
	Appendix A	0	11/84	Yes	

Notes:

(1) See Subsection 3.8.3.2

(2) See Subsection 4.2

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps	0	11/70	Yes	
1.3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors	2	6/74	Yes	
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors	2	6/74	No	PWR only
1.5	Assumptions Used for Evaluating the Potential Radiological Consequences of a Steamline Break Accident for Boiling Water Reactors	0	3/71	Yes	
1.6	Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems	0	3/71	Yes	
1.7	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident	2	11/78	Yes	
1.8	Personnel Selection and Training	--	--	--	See Table 17.0-1
1.9	Selection, Design, Qualification, and Testing of Emergency Diesel-Generator Units Used As Class 1E Onsite Electric Power Systems at Nuclear Plants	3	7/93	Yes	
1.11	Instrument Lines Penetrating Primary Reactor Containment	0	3/71	Yes	
1.12	Instrumentation for Earthquakes	1	4/74	No	NA
1.13	Spent Fuel Storage Facility Design Basis	1	12/75	Yes	
1.14	Reactor Coolant Pump Flywheel Integrity	1	8/75	No	PWR only
1.16	Reporting of Operating Information — Appendix A Technical Specifications	4	8/75	---	COL Applicant
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing	2	5/76	Yes	

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR (Continued)

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
1.21	Measuring, Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light Water Nuclear Power Plants	1	6/74	Yes	
1.22	Periodic Testing of Protection System Actuation-Functions	0	2/72	Yes	
1.23	Onsite Meteorological Programs	0	2/72	Yes	
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure	0	3/72	No	PWR only
1.25	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors	0	3/72	Yes	
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants	---	---	---	See Table 17.0-1
1.27	Ultimate Heat Sink for Nuclear Power Plants	2	1/76	Yes	
1.28	Quality Assurance Program Requirements (Design and Construction)	---	---	---	See Table 17.0-1
1.29	Seismic Design Classification	---	---	---	See Table 17.0-1
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment	---	---	---	See Table 17.0-1
1.31	Control of Ferrite Content in Stainless Steel Weld Metal	3	4/78	Yes	
1.32	Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants	2	2/77	Yes	
1.33	Quality Assurance Program Requirements (Operations)	2	2/78	---	COL Applicant
1.34	Control of Electroslag Weld Properties	0	12/72	Yes	
1.35	Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures	2	1/76	Yes	

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR (Continued)

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
1.36	Non-Metallic Insulation for Austenitic Stainless Steel	0	2/73	Yes	
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants	---	---	---	See Table 17.0-1
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants	---	---	---	See Table 17.0-1
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants	---	---	---	See Table 17.0-1
1.40	Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants	0	3/73	Yes	
1.41	Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments	0	3/73	Yes	
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components	0	5/73	Yes	
1.44	Control of Use of Sensitized Stainless Steel	0	5/73	Yes	
1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems	0	5/73	Yes	
[*1.47*	*Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems*	*0*	*5/73*	*Yes*](4)	
1.49	Power Levels of Nuclear Power Plants	1	12/73	Yes	
1.50	Control of Preheat Temperature Welding of Low-Alloy Steel	0	5/73	Yes	
1.52	Design, Testing, Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants	2	3/78	Yes	
1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems	0	6/73	Yes	
1.54	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants	0	6/73	Yes	

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR (Continued)

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
1.56	Maintenance of Water Purity in Boiling Water Reactors	1	7/78	Yes	
1.57	Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components	0	6/73	Yes	
1.58	Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel		Superceded		See Table 17.0-1
1.59	Design Basis Floods for Nuclear Power Plants	2	8/77	Yes	
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants	1	12/73	Yes	
1.61	Damping Values for Seismic Design of Nuclear Power Plants	0	10/73	Yes	
1.62	Manual Initiation of Protective Actions	0	10/73	Yes	
1.63	Electric Penetration Assemblies in Containment Structures of Nuclear Power Plants	3	2/87	Yes	
1.64	Quality Assurance Requirements for the Design of Nuclear Power Plants		Superceded		See Table 17.0-1
1.65	Materials and Inspections for Reactor Vessel Closure Studs	0	10/73	Yes	
1.68	Initial Test Programs for Water-Cooled Reactor Power Plants	2	8/78	Yes	
1.68.1	Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants	1	1/77	Yes	
1.68.2	Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants	1	7/78	Yes	
1.68.3	Preoperational Testing of Instrument and Control Air Systems	0	4/82	Yes	
1.69	Concrete Radiation Shields for Nuclear Power Plants	0	12/73	Yes	
1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants	3	11/78	Yes	
1.71	Welder Qualifications for Areas of Limited Accessibility	0	12/73	---	COL Applicant

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR (Continued)

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
1.72	Spray Pond Piping Made From Fiberglass-Reinforced Thermosetting Resin	2	11/78	Yes	
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants	0	1/74	Yes	
1.74	Quality Assurance Terms and Definitions		Super-ceded		See Table 17.0-1
[*1.75*	*Physical Independence of Electric Systems*	*2*	*9/78*	*Yes*](4)	
1.76	Design Basis Tornado for Nuclear Power Plants	0	4/74	Yes	
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors	0	5/74	No	PWR only
1.78	Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release	0	6/74	Yes	
1.79	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors	1	9/75	No	PWR only
1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Power Plants	1	1/75	Yes	
1.82	Water Sources for Long-Term Recirculation Cooling Following Loss-of-Coolant Accident	1	11/85	Yes	
1.83	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes	1	7/75	No	PWR only
[*1.84*	*Design and Fabrication Code Case Acceptability, ASME Section III, Division 1*	*27*	*11/90*	*Yes*](1)	
1.85	Materials Code Case Acceptability, ASME Section III, Division 1	27	11/90	Yes	
1.86	Termination of Operating Licenses for Nuclear Reactors	0	6/74	----	COL Applicant
1.87	Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595, and 1596)	1	6/75	No	

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR (Continued)

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
1.88	Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records		Superceded		See Table 17.0-1
[*1.89*	*Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants*	*1*	*6/84*	*Yes*][(2)]	
1.90	Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons	1	8/77	---	COL Applicant
1.91	Evaluations of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants	2	2/78	Yes	
[*1.92*	*Combining Modal Responses and Spatial Components in Seismic Response Analysis*	*1*	*2/76*	*Yes*][(1)]	
1.93	Availability of Electric Power Sources	0	12/74	Yes	
1.94	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Steel During the Construction Phase of Nuclear Power Plants	---	---	---	See Table 17.0-1
1.95	Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release	1	1/77	Yes	
1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants	1	6/76	Yes	
1.97	Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident	3	5/83	Yes	
1.98	Assumptions for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor	0	3/76	Yes	
1.99	Radiation Embrittlement of Reactor Vessel Materials	2	5/88	Yes	
[*1.100*	*Seismic Qualification of Electric Equipment for Nuclear Power Plants*	*2*	*6/88*	*Yes*][(2)]	
1.101	Emergency Planning and Preparedness for Nuclear Power Reactors	3	8/92	Yes	
1.102	Flood Protection for Nuclear Power Plants	1	9/76	Yes	

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR (Continued)

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
[*1.105*	*Instrument Setpoints for Safety-Related Systems*	*2*	*2/86*	*Yes*](3)	
1.106	Thermal Overload Protection for Electric Motors on Motor-Operated Valves	1	3/77	Yes	
1.107	Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures	1	2/77	Yes	
1.108	Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants	1	8/77	Yes	
1.109	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I	1	10/77	Yes	
1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Plants	0	3/76	Yes	
1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors	1	7/77	Yes	
1.112	Calculation for Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors	0	5/77	Yes	
1.113	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I	1	4/77	Yes	
1.114	Guidance On Being Operator At the Controls of a Nuclear Power Plant	1	11/76	---	COL Applicant
1.115	Protection Against Low-Trajectory Turbine Missiles	1	7/77	Yes	
1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems	---	---	---	See Table 17.0-1
1.117	Tornado Design Classification	1	4/78	Yes	
1.118	Periodic Testing of Electric Power and Protection Systems	2	6/78	Yes	
1.120	Fire Protection Guidelines for Nuclear Power Plants	1	11/87	Yes	

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR (Continued)

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes	0	8/76	No	PWR only
1.122	Development of floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components	1	2/78	Yes	
1.123	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants		Superceded		See Table 17.0-1
1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports	1	1/78	Yes	
1.125	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants	1	11/78	Yes	
1.126	An Acceptable Model and Related Statistical Methods for the Analysis for Fuel Densification	1	3/78	Yes	
1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants	1	3/78	---	COL Applicant
1.128	Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants	1	10/78	Yes	
1.129	Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants	1	2/78	Yes	
1.130	Service Limits and Loading Combination for Class 1 Plate-and-Shell-Type Component Supports	1	10/78	Yes	
1.131	Qualification Tests of Electric Cable, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants	0	8/77	Yes	
1.132	Site Investigations for Foundations of Nuclear Power Plants	1	3/79	Yes	
1.133	Loose-Part Detection Program for the Primary Systems of Light-Water-Cooled Reactors	1	6/81	Yes	
1.134	Medical Evaluation of Licensed Personnel for Nuclear Power Plants	2	5/87	---	COL Applicant
1.135	Normal Water Level and Discharge at Nuclear Power Plants	0	9/77	Yes	

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR (Continued)

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
1.136	Materials, Construction, and Testing of Concrete Containments (Articles CC-1000, -2000, and -4000 through -6000 of the "Code for Concrete Reactor Vessels and Containment")	2	7/81	Yes	
1.137	Fuel-Oil Systems for Standby Diesel Generators	1	10/79	Yes	
1.138	Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants	0	4/78	Yes	
1.139	Guidance for Residual Heat Removal	0	5/78	Yes	
1.140	Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants	1	10/79	No	No charcoal filtration required for normal operation
1.141	Containment Isolation Provisions for Fluid Systems	0	4/78	Yes	
1.142	Safety-Related Concrete Structures for Nuclear Power Plants (Other Than Reactor Vessels and Containments)	1	11/81	Yes	
1.143	Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants	1	10/79	Yes	
1.144	Auditing of Quality Assurance Programs Nuclear Power Plants		Super-ceded		See Table 17.0-1
1.145	Atmospheric Dispersion Models for Potential Accident Consequences Assessments at Nuclear Power Plants	1	12/82	Yes	
1.146	Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants		Super-ceded		See Table 17.0-1
1.147	Inservice Inspection Code Case Acceptability-ASME Section XI, Division 1	8	11/90	Yes	
1.148	Functional Specifications for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants	0	4/81	Yes	
1.149	Nuclear Power Plant Simulation Facilities for Use in Operator License Examinations	1	5/87	---	COL Applicant

Table 1.8-20 NRC Regulatory Guides Applicable to ABWR (Continued)

RG No.	Regulatory Guide Title	Appl. Rev.	Issued Date	ABWR Applicable?	Comments
1.150	Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations	1	2/83	Yes	
1.151	Instrument Sensing Lines	0	7/83	Yes	
[*1.152*	*Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants*	*0*	*11/85*	*Yes*](4)	
[*1.153*	*Criteria for Power, Instrumentation, and Control Portions of Safety Systems*	*0*	*12/85*	*Yes*](4)	
1.154	Format and Contents of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors	0	3/87	No	PWR only
1.155	Station Blackout	0	8/88	Yes	
1.160	Monitoring the Effectiveness of Maintenance at Nuclear Power Plants	0	6/93	Yes	
5.1	Serial Numbering of Fuel Assemblies for Light-Water-Cooled Nuclear Power Plants	0	12/72	Yes	
5.7	Control of Personnel Access to Protected Areas, Vital Areas, and Material Access Areas	1	5/80	Yes	
5.12	General use of Locks in the Protection and Control of Facilities and Special Nuclear Materials	0	11/73	Yes	
5.44	Perimeter Intrusion Alarm Systems	2	6/80	Yes	
5.65	Vital Area Access Controls, Protection of Physical Security Equipment, and Key and Lock Controls	0	9/86	Yes	
8.5	Criticality and Other Interior Evacuation Signals	0	2/73	Yes	
8.8	Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Reasonably Achievable	3	6/78	Yes	
8.12	Criticality Accident Alarm Systems	1	2/81	Yes	
8.19	Occupational Radiation Dose Assessment in Light-Water Reactor Power Plants Design Stage Man-Rem Estimates	1	6/79	Yes	

Table 1.8-20 Notes:

(1) See Subsection 3.9.1.7 for restriction of change to this revision. The change restriction to R.G 1.84 applies only in regard to Code Case N-420 (See DCD/Introduction, Table 7).

(2) See Section 3.10 for restriction of change to this revision.

(3) See Subsection 7.1.2.10.9 for restriction to change this revision.

(4) See Section 7A.1(1).

Table 1.8-21 Industrial Codes and Standards* Applicable to ABWR

Code or Standard Number	Year	Title
American Concrete Institute (ACI)		
211.1†	1981	Practice for Selecting Proportions for Normal, Heavy Weight, and Mass Concrete.
212	1981	Guide for Admixtures in Concrete
214	1977	Recommended Practice for Evaluation of Strength Test Results of Concrete
301†	1984	Specifications for Structural Concrete for Buildings
304	1973	Practice for Measuring, Mixing, Transporting, and Placing of Concrete
305	1977	Recommended Practice for Hot Weather Concreting
306	1978	Recommended Practice for Cold Weather Concreting
307	1979	Specification for the Design and Construction of Reinforced Concrete Chimneys
308†	1981	Practice for Curing Concrete
309	1972	Practice for Consolidation of Concrete
311.1R	1981	ACI Manual of Concrete Inspection
311.4R	1981	Guide for Concrete Inspection
315†	1980	Details and Detailing of Concrete Reinforcement
318†	1989	Building Code Requirements for Reinforced Concrete
[*349*†	*1980*	*Code Requirements for Nuclear Safety-Related Concrete Structures*](1)
359		(See ASME BPVC Section III)
American Institute of Steel Construction (AISC)		
[*N690*†	*1984*	*Specifications for the Design, Fabrication, and Erection of Steel Safety-Related Structures for Nuclear Facilities*](1)
--	--	Manual of Steel Construction
American Iron and Steel Insitute		
SG-673	1986	Cold-Formed Steel Design Manual

Table 1.8-21 Industrial Codes and Standards* Applicable to ABWR (Continued)

Code or Standard Number	Year	Title
American Nuclear Society (ANS)		
2.3†	1983	Standard for Estimating Tornado and Other Extreme Wind Characteristics at Nuclear Power Sites
2.8†	1981	Determining Design Basis Flooding at Power Reactor Sites
4.5†	1988	Criteria for Accident Monitoring Functions in Light-Water-Cooled Reactors
5.1†	1979	Decay Heat Power in LWRs
[7-4.3.2†	*1982*	*Application Criteria for Programmable Digital Computer Systems in Safety Systems of NPGS]*(3)(4)
18.1 (ANSI N237)	1984	Radioactive Source Term for Normal Operation of LWRs
52.1†	1983	Nuclear Safety Design Criteria for the Design of Stationary Boiling Water Reactor Plants
55.4	1979	Gaseous Radioactive Waste Processing Systems for Light Water Reactors
56.5	1979	PWR and BWR Containment Spray System Design Criteria
56.11†	1988	Standard Design Criteria for Protection Against the Effects of Compartment Flooding in Light Water Reactor Plants
57.1†(ANSI N208)	1980	Design Requirements for LWR Fuel Handling Systems
57.2(ANSI N210)	1976	Design Requirements for LWR Spent Fuel Storage Facilities at NPP
57.3	1983	Design Requirements for New Fuel Storage Facilities at LWR Plants
[57.5†	*1981*	*Light Water Reactor Fuel Assembly Mechanical Design and Evaluation]*(2)
[58.2†	*1988*	*Design Basis for Protection of Light Water NPP Against Effects of Postulated Pipe Rupture]*(8)
58.8†	1984	Time Response Design Criteria for Nuclear Safety Related Operator Actions
59.51 (ANSI N195)	1976	Fuel Oil Systems for Standby Diesel-Generators
American National Standards Institute (ANSI)‡		
A40	1993	National Plumbing Code
A58.1	1982	Minimum Design Loads for Buildings and other Structures, revised and redesigned as ASCE 7-1988
AG-1		(See ASME AG-1)
B3.5	1960	American Standard Tolerance for Ball and Roller Bearings
B30.2		(See ASME B30.2)

Table 1.8-21 Industrial Codes and Standards* Applicable to ABWR (Continued)

Code or Standard Number	Year	Title
B30.9		(See ASME B30.9)
B30.10		(See ASME B30.10)
B30.11		(See ASME B30.11)
B30.16		(See ASME B30.16)
B31.1		(See ASME B31.1)
B96.1		(See ASME B96.1)
C1 /ASQC	1985	Specifications of General Requirements for a Quality Program
C37.01		(See IEEE C37.01)
C37.04		(See IEEE C37.04)
C37.06	1987	Preferred Ratings of Power Circuit Breakers
C37.09		(See IEEE C37.09)
C37.11	1979	Power Circuit Breaker Control Requirements
C37.13		(See IEEE C37.13)
C37.16	1988	Preferred Ratings and Related Requirements for Low Voltage AC Power Circuit Breakers
C37.17	1979	Trip Devices for AC and General-Purpose DC Low-Voltage Power Circuit Breakers
C37.20		(See IEEE C37.20)
C37.50	1989	Test Procedures for Low Voltage AC Power Circuit Breakers Used in Enclosures
C37.100		(See IEEE C37.100)
C57.12		(See IEEE C57.12)
C57.12.11		(See IEEE C57.12.11)
C57.12.80		(See IEEE C57.12.80)
C57.12.90		(See IEEE C57.12.90)
C62.41		(See IEEE C62.41)
C62.45		(See IEEE C62.45)
C63.12		(See IEEE C63.12)
D975 /ASTM	1981	Diesel Fuel Oils, Specifications for
HE I	1970	Standards for Steam Surface Condenser, 6th Ed., Heat Exchangers Institute
[HFS-100	*1988*	*Human Factors Engineering of Visual Display Terminal Workstations*][(5)]
MC11.1	1976	Quality Standard for Instrument Air

Table 1.8-21 Industrial Codes and Standards* Applicable to ABWR (Continued)

Code or Standard Number	Year	Title
N5.12	1972	Protective Coatings (Paint) for Nuclear Industry
N13.1	1969	Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities
N14.6	1986	Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Materials
N18.7	1976	Administrative Controls and Quality Assurance for the Operation Phase of Nuclear Power Plants
N45.2.1[f] (RG 1.37)	1973	Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants
N45.2.2[f] (RG 1.38)	1972	Packaging, Shipping, Receiving, Storage, and Handling of Items for Nuclear Power Plants During the Construction Stage
N45.2.3	1973	Housekeeping During the Construction Phase of Nuclear Power Plants
N45.2.4	1972	Quality Assurance Program Requirements for Nuclear Power Plants
N45.2.5	1974	Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants
N45.2.8[f] (RG 1.116)	1976	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems
N45.4		(See ASME N45.4)
N101.2	1972	Protective Coatings (Paints) for Light Water Nuclear Containment Facilities
N101.4	1972	QA for Protective Coatings Applied to Nuclear Facilities
N195		(See ANS 59.51)
N237		(See ANS 18.1)
N270		(See ANS 52.2)
N509		(See ASME N509)
N510		(See ASME N510)
N690		(See AISC N690)
NQA-1		(See ASME NQA-1)
NQA-1a		(See ASME NQA-1a)
NQA-2a		(See ASME NQA-2a)
OM3	1990	Requirements for preoperational and Initial Startup Vibration Test Program for Water-Cooled Power Plants

Table 1.8-21 Industrial Codes and Standards* Applicable to ABWR (Continued)

Code or Standard Number	Year	Title
OM7	1986	Requirements for Thermal Expansion Testing of Nuclear Plant Piping Systems [September 1986 (Draft-Revision 7)]
[X3.139	*1987*	*Fiber Distributed Data Interface (FDDI) - Token Ring Media Access Control (MAC)]*(3)(4)
[X3.148	*1988*	*Fiber Distributed Data Interface (FDDI) - Token Ring Physical Layer Protocol (PHY)]*(3)(4)
[X3.166	*1990*	*Fiber Distributed Data Interface (FDDI) - Physical Layer Medium Dependent (PMD)]* (3)(4)
[X3T9.5/84-49	*Rev. 7.1 May 7, 1992*	*FDDI Station Mangement (SMT), Preliminary Draft]*(3)(4)
American Petroleum Institute (API)		
620†	1986	Rules for Design and Construction of Large, Welded, Low-Pressure Storage Tanks
650†	1980	Welded Steel Tanks for Oil Storage
American Society of Heating, Refrigerating and Air-Conditioning Engineers, Inc. (ASHRAE)		
30	1978	Methods of Testing Liquid Chilling Packages
33	1978	Methods of Testing Forced Circulation Air Cooling and Air Heating Coils
American Society of Mechanical Engineers (ASME)		
AG-1†	1991	Code on Nuclear Air and Gas Treatment
B30.2†	1983	Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Grider, Top Running Trolley Hoist)
B30.9†	1984	Slings
B30.10†	1982	Hooks
B30.11†	1980	Monorails and Underhung Cranes
B30.16†	1981	Overhead Hoists
B31.1†	1986	Power Piping
B96.1†	1986	Specification for Welded Aluminum-Alloy Storage Tanks
N45.4	1972	Leakage-Rate Testing of Containment Structures for Nuclear Reactors
N509†	1989	Nuclear Power Plant Air-Cleaning Units and Components
N510†	1989	Testing of Nuclear Air-Cleaning Systems
NQA-1†	1983	Quality Assurance Program Requirements for Nuclear Facilities
NQA-1a†	1983	Addenda to ANSI/ASME NQA-1-1983

Table 1.8-21 Industrial Codes and Standards* Applicable to ABWR (Continued)

Code or Standard Number	Year	Title
[NQA-2a†	*1990*	*Quality Assurance Requirements of Computer Software for Nuclear Facility Application]*(3)(4)
OMa	1988	Operation and Maintenance of Nuclear Power Plants (Addenda to OM-1987)
Sec II	1989	BPVC Section II, Material Specifications
[*Sec III*	*1989*	*BPVC Section III, Rules for Construction of Nuclear Power Plant Components*](6)(8)
Sec VIII	1989	BPVC Section VIII, Rules for Construction of Pressure Vessel
Sec IX	1989	BPVC Section IX, Qualification Standard for Welding and Brazing Procedures Welder, Brazers and Welding and Brazing Operators
Sec XI	1989	BPVC Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components
American Society for Testing and Materials (ASTM)		
[*C776*	*1979*	*Sintered Uranium Dioxide Pellets*](2)
[*C934*	*1980*	*Design and Quality Assurance Practices for Nuclear Fuel Rods*](2)
E84 REV. A	1991	Methods of Test of Surface Burning Characteristics of Building Materials
E119	1988	Standard Test Methods for Fire Tests of Building Construction and Materials
E152	1981	Standard Methods of Fire Tests of Door Assemblies
		(See ASME BPVC Section III for ASTM Material Specifications)
American Welding Society (AWS)		
A4.2†	1986	Procedures for Calibrating Magnetic Instruments to Measure the Delta Ferrite content of Anstenitic Stainless Steel Weld Metal
D1.1†	1986	Steel Structural Welding Code
D14.1†	1985	Welding of Industrial and Mill Cranes and other Material Handling Equipment
American Water Works Association (AWWA)		
D100†	1984	Welded Steel Tanks for Water Storage
CMAA70	1983	Specification for Electric Overhead Traveling Cranes
Insulated Cable Engineer Association (ICEA)		
P-46-426/IEEE S-135	1982	Ampacities Including Effect of Shield Losses for Single Conductor Solid-Dielectric Power Cable 15 kV through 69 kV

Table 1.8-21 Industrial Codes and Standards* Applicable to ABWR (Continued)

Code or Standard Number	Year	Title
P-54-440/NEMA WC-51	1987	Ampacities of Cables in Open-Top Cable Trays
S-61-402/NEMA WC-5	1973	Thermoplastic Insulated Wire & Cable for the Transmission and Distribution of Electrical Energy
S-66-524/NEMA WC-7	1982	Cross Linked Thermosetting Polyethylene Insulated Wire and Cable for Transmission and Distributor of Electrical Energy
Institute of Electrical and Electronics Engineers (IEEE)		
C37.01†	1979	Application Guide for Power Circuit Breakers
C37.04†	1979	AC Power Circuit Breaker Rating Structure
C37.09†	1979	Test Procedure For Power Circuit Breakers
C37.13†	1989	Low Voltage Power Circuit Breakers
C37.20†	1987	Switchgear Assemblies and Metal-Enclosed Bus
[C37.90.2	*1987*	*Trial-Use Standard, Withstand Capability of Relay Systems to Radiated Electromagnetic Interference form Transceivers*](3)(4)
C37.100†	1992	Definitions for Power Switchgear Transformers
C57.12†	1987	General Requirements for Distribution, Power, and Regulating Transformers
C57.12.11†	1980	Guide for Installation of Oil-immersed Transformers(10MVA & Larger, 69-287 kV Rating)
C57.12.80†	1978	Terminology for Power and Distribution Transformers
C57.12.90†	1987	Test Code for Distribution, Power, and Regulating Transformers
[C62.41†	*1991*	*Guide for Surge Voltage in Low-Voltage AC Power Circuits*](3)(4)
[C62.45†	*1987*	*Guide on Surge Testing for Equipment Connected to Low-Voltage AC Power Curcuits*](3)(4)
[C63.12†	*1987*	*American National Standard for Electromagnetic Compatibility Limits-Recommended Practice*](3)(4)
7-4.3.2	1982	Application Criteria for Digital Computers in Safety Systems for Nuclear Facilities (to be replaced by the issued version of P 7-4.3.2, “Standard Criteria for Digital Computers Used in Safety Systems of Nuclear Power Generation Stations”)
80†	1986	Guide for Safety in AC Substation Grounding
81†	1983	Guide for Measuring Earth Resistivity, Ground Impedance, and Earth Surface Potentials of a Ground System
S-135		(See ICEA P-46-426)
141†	1986	Recommended Practice for Electric Power Distribution for Industrial Plants (IEEE Red Book)

Table 1.8-21 Industrial Codes and Standards* Applicable to ABWR (Continued)

Code or Standard Number	Year	Title
242†	1986	Recommended Practice for Protection and Coordination of Industrial and Commercial Power Systems
[279	*1971*	*Criteria for Protection Systems for NPGS]*(3)(4)
308†	1980	Criteria for Class 1E Power Systems for NPGS
317†	1983	Electrical Penetration Assemblies in Containment Structures for NPGS
[323†	*1974*	*Qualifying Class 1E Equipment for NPGS]*(3)(4)(7)
334†	1974	Motors for NPGS, Type Tests of Continuous Duty Class 1E
[338†	*1977*	*Criteria for the Periodic Testing of NPGS Safety Systems]*(3)(9)
[344†	*1987*	*Recommended Practices for Seismic Qualifications of Class 1E Equipment for NPGS]*(7)
352†	1987	General Principles for Reliability Analysis of Nuclear Power Generating Station Protection Systems
379†	1977	Standard Application of the Single-Failure Criterion to NPGS Safety Systems
382†	1985	Qualification of Actuators for Power-Operated Valve Assemblies with Safety-Related Functions for NPP
383†	1974	Type Test of Class 1E Cables; Field Splices and Connections for NPGS
[384†	*1981*	*Criteria for Independence of Class 1E Equipment and Circuits]*(3)
387†	1984	Criteria for Diesel-Generator Units Applied as Standby Power Supplies for NPGS
399†	1990	Recommended Practice for Industrial and Commercial Power Systems Analysis (IEEE Brown Book)
450†	1987	Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations
484†	1987	Recommended Practice for the Design and Installation of Large Lead Storage Batteries for NPGS
485†	1983	Recommended Practice for Sizing Large Lead Storage Batteries for NPGS
500	1984	Guide to the Collection and Presentation of Electronic, Sensing Component, and Mechanical Equipment Reliability Data for Nuclear Power Generating Stations.
[518	*1982*	*Guide for the Installation of Electrical Equipment to Minimize Electrical Noise Inputs to Controllers from External Sources]*(3)(4)
519†	1981	IEEE Standard Recommended Practices and Requirements for Harmonic Control in Electrical Power Systems

Table 1.8-21 Industrial Codes and Standards* Applicable to ABWR (Continued)

Code or Standard Number	Year	Title
[603†	*1980*	*IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations]*(3)
622†	1987	Recommended Practice for the Design and Installation of Electric Heat Tracing Systems in Nuclear Power Generating Stations
622A†	1984	Recommended Practice for the Design and Installation of Electric Pipe Heating Control and Alarm Systems in Nuclear Power Generating Stations
[730	*1984*	*Standard for Software Quality Assurance Plans]*(3)(4)
741†	1986	Standard Criteria for the Protection of Class 1E Power Systems and Equipment in Nuclear Power Generating Stations
765†	1983	Standard for Preferred Power Supply for Nuclear Power Generating Stations
[802.2†	*1985*	*Standards for Local Area Networks: Logic Link Control]*(3)
[802.5†	*1985*	*Token Ring Access Method and Physical Layer Specifications]*(3)
[828†	*1983*	*Standard for Software Configuration Management Plans]*(3)(4)
[829†	*1983*	*Standard for Software Test Documentation]*(3)(4)
[830†	*1984*	*Standard for Software Requirements Specifications]*(3)(4)
[845†	*1988*	*Guide to Evaluation of Man-Machine Performance in Nuclear Power Generating Station Control Rooms and Other Peripheries]*(5)
944†	1986	Recommended Practice for the Application and Testing of Uninterruptable Power Supplies for Power Generating Station
946†	1985	Recommended Practice for the Design of Safety-Related DC Auxiliary Power Systems for Nuclear Power Generating Stations
[1012†	*1986*	*Standard for Software Verification and Validation]*(3)(4)
[1023†	*1988*	*IEEE Guide to the Application of Human Factors Engineering to Systems, Equipment and Facilities of Nuclear Power Generating Stations]*(5)
[1042	*1987*	*Guide to Software Configuration Management]*(3)(4)
[1050	*1989*	*Guide for Instrumentation and Control Equipment in Generating Stations]*(3)(4)
[1228 (Draft)	*1992*	*Standard for Software Safety Plans]*(3)(4)
Instrument Society of America (ISA)		
S7.3†	1981	Quality Standard for Instrument Air
S67.02-80	1980	Nuclear-Safety-Related Instrument Sensing Line Piping and Tubing Standards for Use in Nuclear Power Plants
National Electrical Manufacturers Association (NEMA)		
AB 1	1986	Molded Case Circuit Breakers

Table 1.8-21 Industrial Codes and Standards* Applicable to ABWR (Continued)

Code or Standard Number	Year	Title
FB1	1977	Fittings and Support for Conduit and Cable Assemblies
ICS 1†	1983	General Standards for Industrial Control
ICS 2†	1988	Standards for Industrial Control Devices, Controllers and Assemblies
MG 1	1987	Motors and Generators
WC-5		(See ICEA S-61-402)
WC 7		(See ICEA S-66-524)
WC 51		(See ICEA P-54-440)
National Fire Protection Association (NFPA)		
10†	1981	Portable Fire Extinguishers - Installation
10A	1973	Portable Fire Extinguishers - Maintenance and Use
11†	1988	Low Expansion Foam and Combined Agent Systems-Foam Extinguishing System
12†	1985	Carbon Dioxide Extinguishing Systems
13†	1985	Installation of Sprinklers Systems
14†	1986	Installation of Standpipe and Hose Systems
15†	1985	Standard for Water Spray Fixed Systems
16†	1991	Deluge Foam-Water Sprinkler and Foam-Water Spray Systems
16A†	1988	Recommended Practice for the Installation of Closed Head Foam-Water Sprinkler Systems
20†	1990	Standard for the Installation of Centrifugal Fire Pumps
24†	1984	Private Service Mains and their Appurtenances
26†	1988	Recommended Practice for the Supervision of Valves Controlling Water Supplies for Fire Protection
37†	1984	Stationary Combustion Engines and Gas Turbines
70†	1987	National Electrical Code-Handbook 1987
72†	1990	Protective Signaling Systems
72D	1986	Proprietary Protective Signaling Systems
78†	1986	Lightning Protection Code
80†	1986	Fire Doors and Windows
80A†	1993	Protection of Buildings from Exterior Fire Exposures
90A†	1985	Installation of Air Conditioning and Ventilating Systems
91†	1983	Blower and Exhaust Systems

Table 1.8-21 Industrial Codes and Standards* Applicable to ABWR (Continued)

Code or Standard Number	Year	Title
92A†	1988	Smoke Control Systems
101†	1985	Life Safety Code
251†	1985	Fire Test, Building Construction and Materials
252†	1984	Fire Tests, Door Assemblies
255†	1984	Building Materials, Test of Surface Burning Characteristics
321†	1987	Classification of Flammable Liquids
801†	1986	Facilities Handling Radioactive Materials
802†	1988	Nuclear Research Reactors
803†	1993	Fire Protection for Light Water Nuclear Power Plants
1961†	1979	Fire Hose
1963†	1985	Screw Threads and Gaskets for Fire Hose Connections
Steel Structures Painting Council (SSPC)		
PA-1	1972	Shop, Field and Maintenance Painting
PA-2	1973	Measurements of Paint Film Thickness with Magnetic Gages
SP-1	1982	Solvent Cleaning
SP-5	1985	White Metal Blast Cleaning
SP-6	1986	Commercial Blast Cleaning
SP-10	1985	Near-White Blast Cleaning
U.S. Department of Defense (DOD)		
[5000.2	*1991*	*Defense Acquisition Management Policies and Procedures*](5)
[AD/A223168	*1990*	*System Engineering Management Guide*](5)
[AR602-1	*1983*	*Human Factors Engineering Program*](5)
[DI-HFAC-80740	*1989*	*Human Factors Engineering Program Plan*](5)
[ESD-TR-86-278	*1986*	*Guidelines for Designing User Interface Software*](5)
[HDBK-761A	*1990*	*Human Engineering Guidelines for Management Information Systems*](5)
[HDBK-763	*1991*	*Human Engineering Procedures Guide, Ch. 5-7 & Appendix. A&B*](5)
[STD-2167A	*1988*	*Defense System Software Development*](3)(4)
[TOP 1-2-610	*1990*	*Test Operating Procedure Part 1*](5)
U.S. Military (MIL)		
F-51068	Latest Edition	Filter, Particulate High-Efficiency, Fire-Resistant

Table 1.8-21 Industrial Codes and Standards* Applicable to ABWR (Continued)

Code or Standard Number	Year	Title
[H-46855B	1979	Human Engineering Requirements for Military Systems, Equipment and Facilities][(5)]
[HDBK-217	Latest Edition	Reliability Prediction of Electronic Equipment][(3)]
[HDBK-251	Latest Edition	Reliability/Design: Thermal Applications][(3)]
[HDBK-759A	1981	Human Factors Engineering Design for Army Material][(5)]
STD-282	1956	Filter Units, Protective Clothing Gas-Mask Components and Related Products: Performance-Test Methods
[STD-461C	1987	Electromagnetic Emission and Susceptibility Requirements for the Control of Electromagnetic Interference][(3)(4)]
[STD-462	1967	Measurement of Electromagnetic Interference Characteristics][(3)(4)]
[STD-1472D	1989	Human Engineering Design Criteria for Military Systems, Equipment and Facilities][(5)]
[STD-1478	1991	Task Performance Analysis][(5)]
Others		
ASCE 7	1988	Minimum Design Loads for Buildings and Other Structures
ERDA 76-21	1976	Testing of Ventilation Systems, Section 9 of Industrial Ventilation Systems
[IEC 801-2	1991	Electronic Capability for Industrial-Process Measurement and Control Equipment][(3)]
[IEC 880	1986	Software for Computers in the Safety Systems of Nuclear Power Stations][(3)(4)]
[IEC 964	1989	Design for Control Rooms of Nuclear Power Plants, Bureau Central de la Commission Electrotechnique Internationale][(5)]
[ISO 7498	1984	Open Systems Interconnection-Basic Refence Model, as the Data Link Layer and Physical Layer][(3)]
OSHA 1910.179	1990	Overhead and Gantry Cranes
TEMA C	1978	Standards of Tubular Exchanger Manufacturers Association
UL-44	1983	Rubber-Insulated Wires and Cables
UL-489	1991	Molded-Case Circuit Breakers and Circuit Breaker Enclosures
UL-845	1988	Standard for Safety Motor Control Centers - Low Voltage Circuit Breakers
--	--	Crane Manufacturers Association of America, Specification No. 70
--	--	Aluminum Construction Manual by Aluminum Association

Table 1.8-21 Industrial Codes and Standards* Applicable to ABWR (Continued)

Code or Standard Number	Year	Title
NCIG-01	Rev. 2	Visual Weld Acceptance Criteria for Structural Welding at Nuclear Power Plants
UBC	1991	Uniform Building Code

* The listing of a code or standard does not necessarily mean that it is applicable in its entirety.

† Also an ANSI code (i.e. ANSI/ASME, ANSI/ANS, ANSI/IEEE etc.).

‡ ANSI, ANSI/ANS, ANSI/ASME, and ANSI/IEEE codes are included here. Other codes that approved by ANSI and another organization are listed under the latter.

ƒ As modified by NRC accepted alternate positions to the related Regulatory Guide and identified in Table 2-1 of Reference 1 to Chapter 17.

Notes:

(1) See Subsection 3.8.3.2 for restriction to use of these.

(2) See Subsection 4.2.

(3) See section 7A.1(1).

(4) See Section 7A.1(2).

(5) See Section 18E.1 for required use of this document.

(6) See Subsection 3.8.1.1.1 for specific restriction of change to this edition.

(7) See Section 3.10 for restriction of change to this revision.

(8) See Subsection 3.9.1.7 for specific restriction of change to this edition in application to piping design. See Table 3.2-3 for the restricted Subsections of this Code as applied to piping design only.

(9) See Subsection 7.1.1.2.

Table 1.8-22 Experience Information Applicable to ABWR

No.	Issue Date	Title	Comment
Type: Generic Letters			
80-06	4/25/80	Clarification of NRC Requirement for Emergency Response Facilities at Each Site	
80-30	12/15/80	Periodic Updating of Final Safety Analysis Reports (FSARs)	COL Applicant
80-31	12/22/80	Control of Heavy Loads	
81-03	2/26/81	Implementation of NUREG-0313m, Rev. 1	
81-04	2/25/81	Emergency Procedures and Training for Station Blackout Events	COL Applicant
81-07	2/3/81	Control of Heavy Loads	
81-10	2/18/81	Post-TMI Requirements for the Emergency Operations Facility	
81-11	2/22/81	Error in NUREG-0619	
81-20	4/1/81	Safety Concerns Associated with Pipe Breaks in the BWR Scram System	
81-37	12/29/81	ODYN Code Reanalysis Requirements	
81-38	11/10/81	Storage of Low-Level Radioactive Wastes at Power Reactor Sites	COL Applicant
82-09	4/20/82	Environmental Qualification of Safety-Related Electrical Equipment	
82-21	10/6/82	Technical Specifications for Fire Protection Audits	COL Applicant
82-22	10/30/82	Inconsistency Between Requirements of 10CFR73.40(d) and Standard Technical Specifications for Performing Audits of Safeguard Contingency Plans	
82-27	11/15/82	Transmittal of NUREG-0763, “Guidelines for Confirmatory In-Plant Tests of Safety-Relief Valve Discharges for BWR Plants,” and NUREG-0783, “Suppression Pool Temperature Limits for BWR Containments.”	
82-33	12/17/82	Supplement 1 to NUREG-0737	
82-39	12/22/82	Problems with the Submittals of 10CFR73.21 Safeguards Information Licensing Review	COL Applicant
83-05	2/83	Safety Evaluation of “Emergency Procedure Guidelines,” Revision 2, NEDO-24934, June 1982	COL Applicant
83-07	2/16/83	The Nuclear Waste Policy Act of 1982	COL Applicant
83-13	3/2/83	Clarification of Surveillance Requirements for HEPA Filters and Charcoal Absorber Units in Standard Technical Specifications on ESF Cleanup Systems	

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
83-28	7/8/83	Required Actions Based on Generic Implications of Salem ATWS Events	
83-33	10/19/83	NRC Positions on Certain Requirements of Appendix R to 10 CFR 50	COL Applicant
84-15	7/2/84	Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability	
84-23	10/26/84	Reactor Vessel/Water Level Instrumentation in BWRs	
85-01	1/9/85	Fire Protection Policy Steering Committee Report	
86-10	4/24/86	Implementation of Fire Protection Requirements	
87-06	3/13/87	Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves	COL Applicant
87-09	6/4-87	Sections 3.0 and 4.0 of the Standard Technical Specifications (STS) on the Applicability of Limiting Conditions for Operations and Surveillance Requirements	
88-01	1/25/88	NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping	
88-02	1/20/88	Integrated Safety Assessment Program II (ISAP II)	
88-14	8/8/88	Instrument Air Supply System Problems Affecting Safety-Related Equipment Past Related Correspondence: IE Notice 87-28, Supp. 1 NUREG-1275, Volume 2	
88-15	9/12/88	Electric Power Systems — Inadequate Control Over Design Process Past Related Correspondence: IE Notice 88-45	
88-16	10/4/88	Removal of Cycle-Specific Parameter Limits from Technical Specifications	
88-18	10/20/88	Plant Record Storage on Optical Disks Past Related Correspondence: NUREG-0800 Reg. Guide 1.28, Rev. 3	COL Applicant
88-20	11/23/88	Individual Plant Examination for Severe Accident Vulnerabilities-10CFR Para. 50.54(f)	
88-20	8/29/89	Generic 88-20 Supplement No. 1	
89-01	1/31/89	Implementation of programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Offsite Dose Calculation Manual or to the Process Control Program	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
89-02	3/21/89	Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products Past Related Correspondence: EPRI-NP-5652, “Guideline for the Utilization of Commercial-Grade Items in Nuclear Safety-Related Applications”. Bulletins 87-02 and Supplements 1 and 2, 88-05 and Supplements 1 and 2, 88-10 IE Notices 87-66, 88-19, 88-35, 88-46 and Supplements 1 and 2, 88-48 and Supplement 1, 88-97	COL Applicant
89-04	4/3/89	Guidance on Developing Acceptable Inservice Testing Program	COL Applicant
89-06	4/12/89	Task Action Plan Item I.D.2 – Safety Parameter Display System CFR 50.54(f)	1A.2.3
89-07	4/28/89	Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs	
89-07 Supp I	4/21/89	Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs	
89-08	5/2/89	Erosion/Corrosion-Induced Pipe Wall Thinning	
89-10	6/28/89	Safety-Related Motor-Operated Valve Testing and Surveillance	COL Applicant
89-13	7/18/89	Service Water System Problems Affecting Safety-Related Equipment	COL Applicant
89-14	8/21/89	Line Item Improvements in Technical Specifications Removal of the 3.25 Limit on Extending Surveillance Intervals	
89-15	8/21/89	Emergency Response Data System	COL Applicant
89-16	9/1/89	Installation of a Hardened Wetwell Vent	
89-18	9/6/89	Resolution of USI A-17, Systems Interactions	Subsection 19B.2.59
89-19	9/20/89	Request for Action Related to Resolution of Unresolved Safety Issue A-47, “Safety Implication of Control Systems in LWR Nuclear Power Plants”, Pursuant to 10CFR50.54(f)	Subsection 19B.2.17
89-22	10/19/89	Potential for Increased Roof Loads and Plant Area Flood Runoff Depth at Licensed Nuclear Power Plants Due to Recent Change in Probable Maximum Precipitation Criteria Developed By The National Weather Service	
90-09	12/11/90	Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions	
91-03	03/06/91	Reporting of Safeguards Events	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
91-04	04/02/91	Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle	
91-05	04/04/91	Licensee Commercial Grade Procurement and Dedication Programs	
91-06	04/29/91	Resolution of Generic Issue A-30, "Adequacy of Safety-Related DC Power Supplies", Pursuant to 10CFR50.54(f)	Subsection 19B.2.52
91-10	07/08/91	Explosive Searches at Protected Area Portals	COL Applicant
91-11	07/19/91	Resolution of Generic Issue 48, "LCOs for Class 1E Tie Breakers", Pursuant to 10CFR50.54(f)	Subsection 19B.2.52
91-14	09/23/91	Emergency Telecommunications	
91-16	10/03/91	Licensed Operators' and Other Nuclear Facility Personnel Fitness for Duty	COL Applicant
91-17	10/17/91	Generic Safety Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants"	Subsection 19B.2.62
92-04	8/19/92	Resolution of the Issues Related to Reactor Vessel Level Instrumentation in BWRs Pursuant to 10CFR50.54(f)	
Type: IE Bulletins			
79-02	3/8/79	Pipe Support Base Plate Designs Using Concrete Expansion Anchor Bolts	
79-08	4/14/79	Events Relevant to BWR Identified During TMI Incident	
80-01	1/11/80	ADS Valve Pneumatic Supply	
80-03	2/6/80	Loss of Charcoal from Absorber Cells	
80-05	3/10/80	Vacuum Condition Resulting in Damage to Chemical and Volume Control System (CVCS) Holdup Tanks	COL Applicant
80-06	3/13/80	ESF Reset Controls	
80-08	4/7/80	Containment Lines Penetration Welds	COL Applicant
80-10	5/6/80	Non-Radioactive System – Potential for Unmonitored Release	COL Applicant
80-12	5/9/80	Decay Heat Removal System Operability	COL Applicant
80-13	5/12/80	Cracking in Core Spray Spargers	
80-15	6/18/80	Possible Loss of Emergency Notification System with Loss of Offsite Power	
80-20	7/31/80	Westinghouse Type W-2 Switch Failures	

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
80-21	11/6/80	Valve Yokes Supplied by Mole	COL Applicant
80-22	9/11/80	Automatic Industries, Model 200-500-008 Sealed Source Containers	COL Applicant
80-24	11/21/80	Prevention of Damage due to H2O Leakage Inside Containment	NUREG/CR-4524
80-25	12/19/80	Operating Problems with Target Rock SRVs at BWRs	
81-01	1/27/81	Surveillance of Mechanical Snubbers	
81-02	4/9/81	Failure of Gate Type Valves to Close	COL Applicant
81-02, Supp 1	8/19/81	Failure of Gate Type Valves to Close Against Differential Pressure	COL Applicant
81-03	4/10/81	Flow Blockage of Cooling Water to Safety System	COL Applicant
82-04	12/3/82	Deficiencies in Primary Containment Electrical Penetration Assemblies	COL Applicant
83-06	7/22/83	Non-Conforming Materials Supplied by Tube-Line Corp.	COL Applicant
84-01	2/3/84	Cracks in Boiling Water Reactor Mark I Containment Vent Header	
84-03	8/24/84	Refueling Cavity Water Seal	
85-03	11/15/85	Motor-Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings	COL Applicant
85-03, Supp 1	4/27/88	Motor-Operated Valve Common Mode Failure During Plant Transients Due to Improper Switch Settings Past Related Correspondence: IE Bulletin 85-03, IE Notice 86-29, and IE Notice 87-01	COL Applicant
86-01	5/23/86	Minimum Flow Logic Problems That Could Disable RHR Pumps	
86-03	10/8/86	Potential Failure of Multiple ECCS Pumps Due to Single Failure of Air-Operated Valve in Minimum Flow Recirculation Line	
87-01	7/9/87	Thinning of Pipe Walls in Nuclear Power Plants	
87-02	11/6/87	Fastener Testing to Determine Conformance with Applicable Material Specifications	COL Applicant
87-02, Supp 1	4/22/88	Fastener Testing to Determine Conformance with Applicable Material Specifications Past Related Correspondence: IE Notice 88-17	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
87-02, Supp 2	6/10/88	Fastener Testing to Determine Conformance with Applicable Material Specifications	COL Applicant
88-04	5/5/88	Potential Safety-Related Pump Loss Past Related Correspondence: IE Notice 87-59	
88-07	6/15/88	Power Oscillations in Boiling Water Reactors (BWRs) Past Related Correspondence: IE Notice 88-39	
88-07, Supp 1	12/30/88	Power Oscillations in Boiling Water Reactors (BWRs)	Subsections 7.1.2.6.1.4 and 7.1.2.1.1.2.2
90-01	03/09/90	Loss of Fill-Oil in Transmitters Manufactured by Rosemount	
90-02	03/20/90	Loss of Thermal Margin Caused by Channel Box Bow	
91-01	10/18/91	Reporting Loss of Criticality Safety Controls	
Type: IE Information Notices			
79-22	9/14/79	Qualifications of Control Systems	COL Applicant
80-12	3/31/80	Instrumentation Failure Causes PORV Opening	
80-21	5/16/80	Anchorage and Support of Safety-Related Electrical Equipment	
80-22	5/28/80	Breakdowns in Contamination Control Programs	COL Applicant
80-40	11/7/80	Excessive N_2 Supply Pressure	
80-42	11/24/80	Effect of Radiation on Hydraulic Snubber Fluid	
81-05	3/13/81	Degraded DC Systems at Palisades	COL Applicant
81-07	3/16/81	Potential Problem with Water Soluble Purge Dam Materials Used During Inert Gas Welding	COL Applicant
81-10	3/25/81	Inadvertent Containment Spray	COL Applicant
81-20	7/13/81	Test Failures of Electrical Penetrations	
81-21	7/21/81	Potential Loss of Direct Access to Ultimate Heat Sink	COL Applicant
81-31	10/8/81	Failure of Safety Injection Valves	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
81-38	12/17/81	Potential Significant Equipment Failures Resulting from Contamination of Air-Operated Systems	COL Applicant
82-03	3/22/82	Environmental Tests of Electrical Terminal Block	
82-10	3/3/82	Following Up Symptomatic Repairs	COL Applicant
82-12	4/21/82	Surveillance of Hydraulic Snubbers	
82-22	7/9/82	Failures in Turbine Exhaust Lines	
82-23	7/16/82	Main Steam Isolation Valve Leakage	
82-25	7/20/82	Failures of Hiller Actuators Upon Gradual Loss of Air Pressure	
82-32	8/19/82	Contamination of Reactor Coolant System by Organics	COL Applicant
82-40	9/22/82	Deficiencies in Primary Containment Electrical Penetration Assemblies	
82-43	11/16/82	Deficiencies in LWR Air Filtration/Vent System	
82-49	12/16/82	Correction for Sample Conditions for Air & Gas Monitor	COL Applicant
83-03	1/28/83	Calibration of Liquid Level Instruments	COL Applicant
83-07	3/7/83	Nonconformities with Materials Supplied by Tube Line Corp.	COL Applicant
83-08	3/9/83	Component Failures Caused by Elevated DC Control Voltage	
83-17	3/31/83	Electrical Control Logic Problem Resulting in Inoperable Auto Start of Emergency Diesel Generator	
83-30	5/11/83	Misapplication of Generic EOP Guidelines	COL Applicant
83-35	5/31/83	Fuel Movement with Control Rods Withdrawn at BWRs	COL Applicant
83-44	7/1/83	Damage to Redundant Safety Equipment from Backflow Through the Equipment	
83-46	7/11/83	Common Mode Valve Failures Degrade Surry's Recirculation Spray Subsystem	COL Applicant
83-50	8/1/83	Failure of Class 1E Circuit Breakers to Close	
83-51	8/5/83	Diesel Generator Events	
83-62	9/26/83	Failure of Toxic Gas Detectors	Subsection 19B.2.40
83-64	9/29/83	Lead Shielding Attached to Safety-Related Systems	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
83-70	10/25/83	Vibration-Induced Valve Failures	
83-70, Supp 1	3/4/85	Vibration-Induced Valve Failures	
83-72	10/28/83	Environmental Qualification Testing Experience	
83-75	11/3/83	Improper Control Rod Manipulation	COL Applicant
83-80	11/23/83	Use of Specialized “Stiff” Pipe Clamps	
84-09	2/13/84	Lessons Learned from NRC Inspections of Fire Protection Safe Shutdown Systems (10CFR50, App. R)	
84-09, Rev. 1	3/7/84	Lessons Learned from NRC Inspections of Fire Protection Safe Shutdown Systems (10CFR50, App. R)	
84-10	2/24/84	Motor-Operated Valve Torque Switches Set Below the Manufacturer’s Recommended Value	COL Applicant
84-17	3/5/84	Problems with Liquid Nitrogen Cooling Components Below the Nil Ductility Temperature	
84-22	3/29/84	Deficiency in Comsip, Inc. Standard Bed Catalyst	
84-23	4/5/84	Results of the NRC-Sponsored Qualification Methodology on ASCO Solenoid Valves	
84-32	4/18/84	Auxiliary Feedwater Sparger and Pipe Hanger Damage	
84-35	4/23/84	BWR Post-Scram Drywell Pressurization	
84-38	5/17/84	Problems With Design, Maintenance, and Operation of Offsite Power Systems	
84-47	6/15/84	Environmental Qualification Tests of Electrical Terminal Blocks	
84-67	8/17/84	Recent Snubber Inservice Testing With High Failure Rates	COL Applicant
84-69	8/29/84	Operation of Emergency Diesel Generators	COL Applicant
84-69, Supp. 1	2/24/86	Operation of Emergency Diesel Generators	COL Applicant
84-70	9/4/84	Reliance on Water Level Instrumentation with a Common Reference Leg	COL Applicant
84-70, Supp. 1	8/26/85	Reliance on Water Level Instrumentation with a Common Leg	COL Applicant
84-76	10/19/84	Loss of All AC Power	
84-87	12/3/84	Piping Thermal Deflection Induced by Stratified Flow	
84-88	12/3/84	Standby Gas Treatment System Problems	

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
84-93	12/17/84	Potential for Loss of Water from the Refueling Cavity	
85-08	1/30/85	Industry Experience on Certain Materials Used in Safety-Related Equipment	
85-13	2/21/85	Consequences of Using Soluble Dams	COL Applicant
85-17	4/1/85	Possible Sticking of ASCO Solenoid Valves	
85-17, Supp. 1	10/1/85	Possible Sticking of ASCO Solenoid Valves	
85-24	3/26/85	Failures of Protective Coatings in Pipes and Heat Exchangers	COL Applicant
85-25	4/2/85	Consideration of Thermal Conditions in the Design and Installation of Supports for Diesel Generator Exhaust Silencers	
85-28	4/9/85	Partial Loss of AC Power and Diesel Generator Degradation	
85-30	4/19/85	Microbiologically Induced Corrosion of Containment Service Water System	
85-32	4/22/85	Recent Engine Failures of Emergency Diesel Generators	
85-33	4/22/85	Undersized Nozzle-to-Shell Welded Joints in Tanks and Heat Exchangers Constructed Under the Rules of the ASME Boiler and Pressure Vessel Code	
85-34	4/30/85	Heat Tracing Contributes to Corrosion Failure of Stainless Steel Piping	COL Applicant
85-35	4/30/85	Failure of Air Check Valves to Seat	
85-35, Supp. 1	5/17/88	Failure of Air Check Valves to Seat	
85-47	6/18/85	Potential Effect of Line-Induced Vibration on Certain Target Rock Solenoid-Operated Valves	
85-51	7/10/85	Inadvertent Loss of Improper Actuation of Safety-Related Equipment	COL Applicant
85-59	7/17/85	Valve Stem Corrosion Failures	
85-66	8/7/85	Discrepancies Between As-Built Construction Drawings and Equipment Installations	COL Applicant
85-76	9/19/85	Recent Water Hammer Events	
85-77	9/20/85	Possible Loss of Emergency Notification System Due to Loss of AC Power	COL Applicant
85-81	10/17/85	Problems Resulting in Erroneously High Reading With Thermoluminscent Dosimeters	COL Applicant
85-84	10/30/85	Inadequate Inservice Testing of Main Steam Isolation Valves	

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
85-85	10/31/85	Systems Interaction Event Resulting in Reactor System Safety/Relief Valve Opening Following a Fire-Protection Deluge System Malfunction	
85-86	11/5/85	Lightning Strikes at Nuclear Power Generating Stations	
85-87	11/18/85	Hazards of Inerting Atmospheres	COL Applicant
85-89	11/19/85	Potential Loss of Solid-State Instrumentation Following Failure or Control Room Cooling	Subsection 19B.2.40
85-90	11/19/85	Use of Sealing Compounds in an Operating Plant	COL Applicant
85-91	11/27/85	Load Sequencers for Emergency Diesel Generators	COL Applicant
85-92	12/2/85	Surveys of Wastes Before Disposal From Nuclear Reactor Facilities	COL Applicant
85-94	12/13/85	Potential for Loss of Minimum Flow Paths Leading to ECCS Pump Damage During a LOCA	
85-96	12/23/85	Temporary Strainers Left Installed in Pump Suction Piping	COL Applicant
86-01	1/6/86	Failure of Main Feedwater Check Valves Causes Loss of Feedwater System Integrity and Water-Hammer Damage	
86-03	1/14/86	Potential Deficiencies in Environmental Qualification of Limitorque Motor Valve Operator Wiring	
86-09	2/3/86	Failure of Check and Stop Valves Subjected to Low Flow Conditions	
86-10	2/13/86	Safety Parameter Display System Malfunctions	
86-29	4/25/86	Effects of Changing Valve Motor-Operator Switch Settings Past Related Correspondence: IE Bulletin 85-03	COL Applicant
86-30	4/29/86	Design Limitations of Gaseous Effluent Monitoring System	
86-39	5/20/86	Failures of RHR Pump Motors and Pump Internals	
86-43	6/10/86	Problems with Silver Zeolite Sampling of Airborne Radioiodine	COL Applicant
86-48	6/13/86	Inadequate Testing of Boron Solution Concentration in the Standby Liquid Control System	
86-50	6/18/86	Inadequate Testing to Detect Failures of Safety-Related Pneumatic Components or Systems Past Related Correspondence: IE Notices 82-25, 85-35, 85-84, 85-94	

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
86-51	6/18/86	Excessive Pneumatic Leakage in the Automatic Depressurization System Past Related Correspondence: IE Bulletins 80-01, 80-25; IE Notice 85-35; IE Inspection Report 50-458/84-18 (8/16/84)	
86-53	6/26/86	Improper Installation of Heat Shrinkable Tubing	COL Applicant
86-57	7/11/86	Operating Problems With Solenoid-Operated Valves at Nuclear Power Plants	
86-60	7/28/86	Unanalyzed Post-LOCA Release Paths Past Related Correspondence: NUREG-0737	
86-68	8/15/86	Stuck Control Rod	
86-70	8/18/86	Potential Failure of All Emergency Diesel Generators	
86-71	8/19/86	Recent Identified Problems With Limitorque Motor Operators Past Related Correspondence: IE Notice 86-03	
86-76	8/20/86	Problems Noted in Control Room Emergency Ventilation Systems Past Related Correspondence: Item III D.3.4 of NUREG-0737 Generic Issue 83, IE Notice 85-89	Subsection 19B.2.40
86-83	9/16/86	Underground Pathways into Protected Areas, Vital Areas, Material Access Areas, and Controlled Access Areas Past Related Correspondence: NUREG-0908, ANSI 3.3	COL Applicant
86-87	10/10/86	Loss of Offsite Power Upon An Automatic Bus Transfer	
86-89	10/16/86	Uncontrolled Rod Withdrawal Because of A Single Failure	
86-96	11/20/86	Heat Exchanger Fouling Can Cause Inadequate Operability of Service Water Systems Past Related Correspondence: IE Bulletin 81-03, IE Notice 81-21	COL Applicant
86-100	12/12/86	Loss of Offsite Power to Vital Buses at Salem 2	
86-104	12/16/86	Unqualified Butt Splice Connectors Identified in Qualified Penetrations	
86-106	12/16/86	Feedwater Line Break	
86-106, Supp. 1	2/13/87	Feedwater Line Break Past Related Correspondence: E Notice 82-22 EPRI Report NP-3944, 4/85	
86-106, Supp. 2	3/18/87	Feedwater Line Break	

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
86-106, Supp. 3	10/10/88	Feedwater Line Break	
86-109	12/29/86	Diaphragm Failure in Scram Outlet Valve Causing Rod Insertion Past Related Correspondence: IE Notice 86-08	COL Applicant
87-06	1/30/87	Loss of Suction to Low-Pressure Service Water System Pumps Resulting From Loss of Siphon	COL Applicant
87-08	2/4/87	Degraded Motor Leads in Limitorque DC Motor Operators Past Related Correspondence: (Unrelated problems involving wiring installed in Limitorque motor actuators) IE Notices 83-72, 86-03 and 86-71	
87-09	2/5/87	Emergency Diesel Generator Room Cooling Deficiency Past Related Correspondence: IE Notice 86-50, 86-51 and 86-89	
87-10	2/11/87	Potential for Water Hammer During Restart of Residual Heat Removal Pumps Past Related Correspondence: AEOD/E309, 4/83	
87-13	2/24/87	Potential For High Radiation Fields Following Loss of Water From Fuel Pool Past Related Correspondence: IE Notice 84-93, IE Bulletin 84-03	
87-14	3/23/87	Actuation of Fire Suppression System Causing Inoperability of Safety-Related Ventilation Equipment Past Related Correspondence: IE Notice 83-41, 85-85, 86-106 Supp. 2	
87-28	6/22/87	Air Systems Problems at U.S. Light Water Reactors Past Related Correspondence: AEOD-C701	
87-28, Sup. 1	12/28/88	Air Systems Problems at U.S. Light Water Reactors Past Related Correspondence: AEOD-C701 NUREG-1275 Vol. 2	
87-36	8/4/87	Significant Unexpected Erosion of Feedwater Lines Past Related Correspondence: IE Notice 82-22, 86-106 plus Supp. 1&2 IE Bulletin 87-01	
87-43	9/8/87	Gaps in Neutron-Absorbing Material in High-Density Spent Fuel Storage Racks Past Related Correspondence: EPRI NP-4724	
87-49	10/9/87	Deficiencies in Outside Containment Flooding Protection	

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
87-50	10/9/87	Potential LOCA at High- and Low-Pressure COL Applicants from Fire Damage	
87-59	11/17/87	Potential RHR Pump Loss	
88-01	1/27/88	Safety Injection Pipe Failure	
88-04	2/5/88	Inadequate Qualification and Documentation of Fire Barrier Penetration Seals Past Related Correspondence: 10CFR50 Appendix R, Appendix A to BTP APCSB 9.5-1, NUREG-0800, ASTM E-119, BTP CMEB 9.5-1, Generic Letter 86-10	
88-04, Supp. 1	8/9/88	Inadequate Qualification and Documentation of Fire Barrier Penetration Seals	
88-05	2/12/88	Fire in Annunciator Control Cabinets	
88-12	4/12/88	Overgreasing of Electrical Motor Bearings Past Related Correspondence: LER 387/84-036	COL Applicant
88-13	4/18/88	Water Hammer and Possible Piping Damage Caused by Misapplication of Kerotest Packless Metal Diaphragm Globe Valves	
88-17	4/22/88	Summary of Responses to NRC Bulletin 87-01, “Thinning of Pipe Walls in Nuclear Power Plants” Past Related Correspondence: IE Bulletin 87-01; IE Notice 82-22, 86-106, 87-36	
88-21	5/9/88	Inadvertent Criticality Events at Oskarshamn and at U.S. Nuclear Power Plants	COL Applicant
88-24	5/13/88	Failures of Air-Operated Valves Affecting Safety-Related Systems Past Related Correspondence: IE Notice 87-28 & Supp. 1, NUREG-1275	
88-27	5/18/88	Deficient Electrical Terminations Identified in Safety-Related Components	COL Applicant
88-35	6/3/88	Inadequate Licensee Performed Vendor Audits Past Related Correspondence: IE Bulletin 88-05	COL Applicant
88-37	6/14/88	Flow Blockage of Cooling Water to Safety System Components Past Related Correspondence: IE Notice 81-21, 86-96; IE Bulletin 81-03	COL Applicant
88-39	6/15/88	LaSalle Unit 2 Loss of Recirculation Pumps With Power Oscillation Event Past Related Correspondence: Generic Issue B-19, Generic Letter 86-02	

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
88-43	6/23/88	Solenoid Valve Problems Past Related Correspondence: IE Notices 85-17 & Supp. 1, 86-57; IE Circular 81-14	
88-51	7/21/88	Failures of Main Steam Isolation Valves	
88-61	8/11/88	Control Room Habitability-Recent Reviews of Operating Experience	Subsection 19B.2.40
88-63	8/15/88	High Radiation Hazards from Irradiated Incore Detectors and Cables	COL Applicant
88-65	8/18/88	Inadvertent Drainings of Spent Fuel Pools	
88-70	8/29/88	Check Valve Inservice Testing Program Deficiencies Past Related Correspondence: IE Notice 86-01, Generic Letter 87-06	
88-72	9/2/88	Inadequacies in the Design of DC Motor-Operated Valves	
88-76	9/19/88	Recent Discovery of a Phenomenon Not Previously Considered in the Design of Secondary Containment Pressure Control Past Related Correspondence: NUREG-0800	
88-77	9/22/88	Inadvertent Reactor Vessel Overfill	
88-81	10/7/88	Failure of AMP Window Indent Kynar Splices and Thomas and Betts Nylon Wire Caps During Environmental Qualification Testing	
88-85	10/14/88	Broken Retaining Block Studs on Anchor Darling Check Valves	
88-86	10/21/88	Operating with Multiple Grounds in Direct Current Distribution Systems and Supplement 1	
88-89	11/21/88	Degradation of Kapton Electrical Insulation Past Related Correspondence: IE Notices 87-08, 87-16	
88-92	11/22/88	Potential for Spent Fuel Pool Draindown	
88-95	12/8/88	Inadequate Procurement Requirements Imposed by Licensees on Vendors	COL Applicant
89-01	1/4/89	Valve Body Erosion Past Related Correspondence: IE Notice 88-17	
89-04	1/17/89	Potential Problems from the Use of Space Heaters	COL Applicant
89-07	1/25/89	Failures of Small-Diameter Tubing in Control Air, Fuel Oil, and Lube Oil Systems Which Render Emergency Diesel Generators Inoperable	
89-08	1/26/89	Pump Damage Caused by Low-Flow Operation	

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
89-10	1/27/89	Undetected Installation Errors in Main Steam Line Pipe Tunnel Differential Temperature Sensing Elements at Boiling Water Reactors	
89-11	2/2/89	Failure of DC Motor-Operated Valves to Develop Rated Torque Because of Improper Cabling Sizing	
89-14	2/16/89	Inadequate Dedication Process for Commercial Grade Components Which Could Lead to Common Mode Failure of a Safety System	
89-16	2/16/89	Excessive Voltage Drop in DC Systems Past Related Correspondence: Generic Letter 88-15	
89-17	2/22/89	Contamination and Degradation of Safety-Related Battery Cells	
89-20	2/24/89	Weld Failures in a Pump of Byron-Jackson Design	
89-21	2/27/89	Changes in Performance Characteristics of Molded Case Circuit Breakers	
89-26	3/7/89	Instrument Air Supply to Safety-Related Equipment Past Related Correspondence: Generic Letter 88-14	
89-30	3/15/89	High Temperature Environments at Nuclear Power Plants	
89-36	4/4/89	Excessive Temperatures in Emergency Core Cooling System Piping Located Outside Containment	
89-37	4/4/89	Proposed Amendments to 40CFR Part 61, Air Emission Standards for Radionuclides	
89-39	4/5/89	List of Parties Excluded from Federal Procurement of Non-procurement Programs	COL Applicant
89-52	6/8/89	Potential Fire Damper Operational Problems	
89-61	8/30/89	Failure of Borg-Warner Gate Valves to Close Against Differential Pressure	
89-63	9/5/89	Possible Submergence of Electrical Circuits Located Above the Flood Level Because of Water Intrusion and Lack of Drainage	COL Applicant
89-64	9/7/89	Electrical Bus Bar Failures	COL Applicant
89-66	9/11/89	Qualification Life of Solenoid Valves	
89-68	9/25/89	Evaluation of Instrument Setpoints During Modifications	COL Applicant
89-69	9/29/89	Loss of Thermal Margin Caused by Channel Box Bow	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
89-70	10/11/89	Possible Indications of Misrepresented Vendor Products	COL Applicant
89-71	10/19/89	Diversion of the Residual Heat Removal Pump Seal Cooling Water Flow During Recirculation Operation Following a Loss-of-Coolant Accident	
89-72	10/24/89	Failure of Licensed Senior Operators to Classify Emergency Events Properly	COL Applicant
89-73	11/1/89	Potential Overpressurization of Low Pressure Systems	COL Applicant
89-76	11/21/89	Biofouling Agent: Zebra Mussel	COL Applicant
89-77	11/21/89	Debris in Containment Emergency Sumps and Incorrect Screen Configurations	
89-79	12/1/89	Degraded Coatings and Corrosion of Steel Containment Vessels	
89-80	12/1/89	Potential for Water Hammer, Thermal Stratification, and Steam Binding in High-Pressure Coolant Injection Piping	
89-81	12/6/89	Inadequate Control of Temporary Modifications to Safety-Related Systems	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
Type: IE Information Notices			
89-83	12/11/89	Sustained Degraded Voltage on the Offsite Electrical Grid and Loss of Other Generating Stations as a Result of a Plant Trip	COL Applicant
89-87	12/19/89	Disabling of Emergency Diesel Generators by Their Neutral Ground-Fault Protection Circuitry	
89-88	12/16/89	Recent NRC-Sponsored Testing of Motor-Operated Valves	
90-02	01/22/90	Potential Degradation of Secondary Containment	
90-05	01/29/90	Inter-System Discharge of Reactor Coolant	
90-07	01/30/90	New Information Regarding Insulation Material Performance and Debris Blockage of PWR Containment Sumps	
90-8	02/01/90	KR-85 Hazards From Decayed Fuel	
90-13	03/05/90	Importance of Review and Analysis of Safeguards Event Logs	COL Applicant
90-20	03/22/90	Personnel Injuries Resulting From Improper Operation of Radwaste Incinerators	COL Applicant
90-21	03/22/90	Potential Failure of Motor-Operated Butterfly Valves to Operate Because Valve Seat Friction was Underestimated	COL Applicant
90-22	03/23/90	Unanticipated Equipment Actuation Following Restoration of Power to Rosemount Transmitter Trip Units	COL Applicant
90-25	04/16/90	Loss of Vital AC Power With Subsequent Reactor Coolant System Heatup	COL Applicant
90-25 Supp.1	03/11/90	Loss of Vital AC Power With Subsequent Reactor Coolant System Heatup	COL Applicant
90-26	04/24/90	Inadequate Flow of Essential Service Water to Room Coolers and Heat Exchangers for Engineered Safety-Feature Systems	COL Applicant
90-30	05/01/90	Ultrasonic Inspection Techniques for Dissimilar Metal Welds	
90-33	05/09/90	Sources of Unexpected Occupational Radiation Exposure at Spent Fuel Storage Pools	COL Applicant
90-39	06/01/90	Recent Problems with Service Water Systems	COL Applicant
90-40	06/05/90	Results of NRC-Sponsored Testing of Motor-Operated Valves	COL Applicant
90-42	06/19/90	Failure of Electrical Power Equipment Due to Solar Magnetic Disturbances	
90-47	07/27/90	Unplanned Radiation Exposures to Personnel Extremities Due to Improper Handling of Potentially Highly Radioactive Sources	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
90-50	08/08/90	Minimization of Methane Gas in Plant Systems and Radwaste Shipping Containers	COL Applicant
90-53	08/16/90	Potential Failures of Auxiliary Steam Piping and the Possible Effects on the Operability of Vital Equipment	
90-54	08/28/90	Summary of Requalification Program Deficiencies	COL Applicant
90-61	09/20/90	Potential for Residual Heat Removal Pump Damage Caused by Parallel Pump Interaction	
90-63	10/03/90	Management Attention to the Establishment and Maintenance of a Nuclear Criticality Safety Program	COL Applicant
90-67	10/29/90	Potential Security Equipment Weaknesses	
90-68	10/30/90	Stress Corrosion Cracking of Reactor Coolant Pump Bolts	
90-69	10/31/90	Adequacy of Emergency and Essential Lighting	
90-70	11/06/90	Pump Explosions Involving Ammonium Nitrate	
90-72	11/28/90	Testing of Parallel Disc Gate Valves in Europe	
90-74	12/04/90	Information on Precursors to Severe Accidents	
90-78	12/18/90	Previously Unidentified Release Path From Boiling Water Reactor Control Rod Hydraulic Units	
90-81	12/24/90	Fitness For Duty	COL Applicant
90-82	12/31/90	Requirements For Use of Nuclear Regulatory Commission-(NRC)-Approved Transport Packages For Shipment of Type A Quantities of Radioactive Material	COL Applicant
91-04	01/28/91	Reactor Scram Following Control Rod Withdrawal Associated With Low Power Turbine Testing	
91-06	01/31/91	Lockup of Emergency Diesel Generator and Load Sequencer Control Circuits Preventing Restart of Tripped Emergency Diesel Generator	
91-12	02/15/91	Potential Loss of Net Positive Suction Head (NPSH) of Standby Liquid Control System Pumps	
91-13	03/04/91	Inadequate Testing of Emergency Diesel Generators (EDGs)	
91-14	03/05/91	Recent Safety-Related Incidents at Large Irradiators	
91-17	03/11/91	Fire Safety of Temporary Installation of Services	COL Applicant
91-19	03/12/91	High-Energy Piping Failures Caused by Wall Thinning	
91-22	03/19/91	Four Plant Outage Events Involving Loss of AC Power or Coolant Spills	

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
91-23	03/26/91	Accident Radiation Overexposures to Personnel Due to Industrial Radiography Accessory Equipment Malfunctions	COL Applicant
91-29	04/15/91	Deficiencies Identified During Electrical Distribution System Functional Inspections	
91-33	05/31/91	Reactor Safety Information for States During Exercises and Emergencies	COL Applicant
91-34	06/03/91	Potential Problems in Identifying Causes of Emergency Diesel Generator Malfunctions	
91-37	06/10/91	Compressed Gas Cylinder Missile Hazards	COL Applicant
91-38	06/13/91	Thermal Stratification in Feedwater System Piping	
91-40	06/19/91	Contamination of Nonradioactive System and Resulting Possibility for Unmonitored, Uncontrolled Release to the Environment	COL Applicant
91-41	06/27/91	Potential Problems with the Use of Freeze Seals	COL Applicant
91-42	07/27/91	Plant Outage Events Involving Poor Coordination Between Operations and Maintenance Personnel During Valve Testing and Manipulations	COL Applicant
91-46	07/18/91	Degradation of Emergency Diesel Generator Fuel Oil Delivery Systems	COL Applicant
91-47	08/06/91	Failure of Thermo-Lag Fire Barrier Material to Pass Fire Endurance Test	
91-49	08/15/91	Enforcement of Safety Requirements for Radiographers	COL Applicant
91-50	08/20/91	A Review of Water Hammer Events After 1985	
91-51	08/20/91	Inadequate Fuse Control Programs	COL Applicant
91-57	09/19/91	Operational Experience on Bus Transfers	
91-58	09/20/91	Dependency of Offset Disc Butterfly Valve's Operation of Orientation With Respect to Flow	
91-59	09/23/91	Problems With Access Authorization Programs	COL Applicant
91-60	11/01/91	Reissuance of Information Notice 91-60: False Alarms of Alarm Ratemeters Because of Radio Frequency Interference	COL Applicant
91-61	09/30/91	Preliminary Results of Validation Testing of Motor-Operated Valve Diagnostic Equipment	

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
91-63	10/03/91	Natural Gas Hazards at Fort St. Vrain Nuclear Generating Station	COL Applicant
91-64	10/09/91	Site Area Emergency Resulting From a Loss of Non-Class 1E Uninterruptable Power Supplies	
91-65	10/17/91	Emergency Access to Low-Level Radioactive Waste Disposal Facilities	COL Applicant
91-66	10/18/91	(1) Erroneous Date in "Nuclear Safety Guide, TID-7016, Revision 2," (NUREG/CR-0095, ORNL/NUREG/CSD-6 (1978) And (2) Thermal Scattering Data Limitation in the Cross-Section Sets Provided With the Keno and Scale Codes	
91-68	10/28/91	Careful Planning Significantly Reduces the Potential Adverse Impacts of Loss of Offsite Power Events During Shutdown	COL Applicant
91-72	11/19/91	Issuance of a Revision to the EPA Manual of Protective Action Guides and Protective Actions for Nuclear Incidents	
Type: IE Circulars			
80-03	3/6/80	Protection from Toxic Gas Hazards	COL Applicant
80-05	4/1/80	Emergency D/G Lube Oil	COL Applicant
80-08	4/18/80	RPS Response Time	
80-09	4/28/80	Problems with Plant Internal Communications Systems	COL Applicant
80-10	4/29/80	Failure to Maintain Environmental Qualification of Equipment	COL Applicant
80-11	5/13/80	Emergency Diesel Generator Lube Oil Cooler Failures	COL Applicant
80-14	6/24/80	Radioactive Contamination of Demin Water System	COL Applicant
80-18	8/22/80	10 CFR 50.59 Safety Evaluation for Changes to Radioactive Waste Treatment Systems	COL Applicant
81-03	3/2/81	Inoperable Seismic Monitoring Instrument	COL Applicant
81-05	3/31/81	Self-Aligning Rod End Bushing for Pipe Supports	COL Applicant
81-07	5/14/81	Control of Radioactivity Contaminated Material	COL Applicant

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
81-08	5/29/81	Foundation Materials	COL Applicant
81-09	7/10/81	Containment Effluent Water	
81-11	7/24/81	Inadequate Decay Heat Removal	COL Applicant
81-13	9/25/81	Torque Switch Electrical Bypass Circuit	COL Applicant
81-14	11/5/81	Main Steam Isolation Valve Failures to Close	COL Applicant
NUREG			
0313 Rev. 2	6/88	Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping	
0371	10/78	Task Action Plans for Generic Activities Category A	
0471	6/78	Generic Task Problem Description: Category B, C & D Tasks	
0578	9/80	Performance Testing of BWR and PWR Relief and Safety Valves.	
0588	12/79	Interim Staff Position On Environmental Qualification of Safety-Related Electrical Equipment	
0619	4/80	BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking	
0626	1/80	Generic Evaluation of Feedwater Transients and Small Break LOCA in GE-Designed Operating Plants and Near-Term Operating License Applications	
0660	5/80	NRC Action Plan Developed as a Result of the TMI-2 Accident	
0661 Supp. 1	8/82	Safety Evaluation Report – Mark I Containment Long-Term Program – Resolution of Generic Technical Activity A-7	Subsection 19B.2.3
0654	10/80	Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants	COL Applicant
0696	12/80	Functional Criteria for Emergency Response Facilities	COL Applicant
0710 Rev. 1	6/81	Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License	
0737 Supp.1	12/82	Clarification of TMI Action Plan Requirements	
0744 Rev. 1	10/82	Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue	
0800	7/81	Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition	

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
0808	8/81	Mark II Containment Program Load Evaluation and Acceptance Criteria	
0813	9/81	Draft Environmental Statement Related to the Operation of Calloway Plant, Unit No. 1	
0933	4/93	A prioritization of Generic Safety Issues	Appendix 19B
0977	3/83	NRC Fact-Finding Task Force Report on the ATWS Events at the Salem Nuclear Generating Station, Unit 1, on February 22 and 25, 1983	
1150	6/89	Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, Vol. 1 & 2	
1161	5/80	Recommended Revisions to USNRC-Seismic Design Criteria	Subsection 19B.2.14
1174	5/89	Evaluation of Systems Interactions in Nuclear Power Plants	Subsection 19B.2.59
1212	6/86	Status of Maintenance in the US Nuclear Power Industry, 1985, Vol. 1, 2	
1216	8/86	Safety Evaluation PP2 Related to Operability and Reliability of Emergency Diesel Generators	
1217	4/88	Evaluation of Safety Implications of Control Systems in LWR Nuclear Power Plants-Technical Findings Related to USI A-47	Subsection 19B.2.17
1218	4/88	Regulatory Analysis for Proposed Resolution of USI A-47	Subsection 19B.2.17
1229	8/89	Regulatory Analysis for Resolution of USI A-17	Subsection 19B.2.59 & 19B.2.14
1233	9/89	Regulatory Analysis for USI A-40	Subsection 19B.2.14
1273	4/88	Containment Integrity Check-Technical Finds Regulatory Analysis	
1296	2/88	Peer Review of High Level Nuclear Waste	
1341	5/89	Regulatory Analysis for Resolution of Generic Issue 115, Enhancement	
1353	4/89	Regulatory Analysis for the Resolution of Generic Issue 82, “Beyond Design Basis Accidents in Spent Fuel Pools”	Subsection 19B.2.63
1370	9/89	Resolution of USI A-48	Subsection 19B.2.18
1275	2/91	Volume 6, Operating Experience Feedback Report Solenoid Operated Valve Problems	

Table 1.8-22 Experience Information Applicable to ABWR (Continued)

No.	Issue Date	Title	Comment
1339	6/90	Resolution of Generic Safety Issue 29: Bolting Degradation of Failure in Nuclear Power Plants	Subsection 19B.2.62
CR-3922	1/85	Survey and Evaluation of System Interaction Events and Sources, Vol. 1, 2	Subsection 19B.2.59
CR-4261	3/86	Assessment of Systems Interactions in Nuclear Power Plants	Subsection 19B.2.59
CR-4262	5/85	Effects of Control System Failures on Transients, Accidents at a GE BWR, Vol. 1 and 2	
CR-4387	12/85	Effects of Control System Failures on Transient and Accidents and Core-Melt Frequencies at a GE BWR	
CR-4470	5/86	Survey and Evaluation of Vital Instrumentation and Control Power Supply Events	
CR-5055	5/88	Atmospheric Diffusion for Control Room Habitability Assessment	Subsection 19B.2.40
CR-5088	1/89	Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues.	
CR-5230	4/89	Shutdown Decay Heat Removal Analysis: Plant Case Studies and Special Issues	
CR-5347	6/89	Recommendations for Resolution of Public Comments on USI A-40	Subsection 19B.2.14
CR-5458	12/89	Value-Impact Assess for Candidate Operating Procedure Upgrade Program	
CR-4674	84/89	Precusors to Potential Severe Core Damage Accidents: Series	

1.9 COL License Information

Tier 2 presents the ABWR Standard Plant design incorporating the Nuclear Island, Turbine Island and radwaste facility. Although this scope is essentially a total plant, there is a modest amount of information that must be addressed by the COL applicant. The purpose of this section is to identify the Tier 2 sections where descriptions of the COL license information are presented.

The COL license information is summarized in Table 1.9-1 in the order it is presented in Tier 2. An item number has been assigned to each entry to facilitate future identification.

Table 1.9-1 Summary of ABWR Standard Plant COL License Information

Item No.	Subject	Subsection
1.1	Design Process to Establish Detailed Design Documentation	1.1.11.1
1.1a	Plant Design and Aging Management	1.2.3.1
1.2	P&ID Pipe Schedule	1.7.6.1
1.3	SRP Deviations	1.8.4.1
1.4	Experience Information	1.8.4.2
1.5	Emergency Procedures and Emergency Procedures Training Program	1A.3.1
1.6	Review and Modify Procedures for Removing Safety-Related Systems from Service	1A.3.2
1.7	In-plant Radiation Monitoring	1A.3.3
1.8	Reporting Failures of Reactor System Relief Valves	1A.3.4
1.9	Report on ECCS Outages	1A.3.5
1.10	Procedure for Reactor Venting	1A.3.6
1.11	Testing of SRV and Discharge Piping	1A.3.7
1.12	RCIC Bypass Start System Test	1A.3.8
1.13	Station Blackout Procedures	1C.4.1
2.1	Non-Seismic Design Parameters	2.3.1.1
2.2	Seismic Design Parameters	2.3.1.2
2.3	Site Location and Description	2.3.2.1
2.4	Exclusion Area Authority and Control	2.3.2.2
2.5	Population Distribution	2.3.2.3
2.6	Identification of Potential Hazards in Site Vicinity	2.3.2.4
2.7	Evaluation of Potential Accidents	2.3.2.5
2.8	External Impact Hazards	2.3.2.6
2.9	Local Meteorology	2.3.2.7
2.10	Onsite Meteorological Measurements Program	2.3.2.8
2.11	Short-Term Dispersion Estimates for Accident Atmosphere Releases	2.3.2.9
2.12	Long-Term Diffusion Estimates	2.3.2.10
2.13	Hydrologic Description	2.3.2.11
2.14	Floods	2.3.2.12

Table 1.9-1 Summary of ABWR Standard Plant COL License Information (Continued)

Item No.	Subject	Subsection
2.15	Probable Maximum Flood on Streams and Rivers	2.3.2.13
2.16	Ice Effects	2.3.2.14
2.17	Cooling Water Channels and Reservoirs	2.3.2.15
2.18	Channel Division	2.3.2.16
2.19	Flooding Protection Requirements	2.3.2.17
2.20	Cooling Water Supply	2.3.2.18
2.21	Accidental Release of Liquid Effluents in Ground and Surface Waters	2.3.2.19
2.22	Technical Specifications and Emergency Operation Requirement	2.3.2.20
2.23	Basic Geological and Seismic Information	2.3.2.21
2.24	Vibratory Ground Motion	2.3.2.22
2.25	Surface Faulting	2.3.2.23
2.26	Stability of Subsurface Material and Foundation	2.3.2.24
2.27	Site and Facilities	2.3.2.25
2.28	Field Investigations	2.3.2.26
2.29	Laboratory Investigations	2.3.2.27
2.30	Subsurface Conditions	2.3.2.28
2.31	Evacuation and Backfilling for Foundation Construction	2.3.2.29
2.32	Effect of Groundwater	2.3.2.30
2.33	Liquefaction Potential	2.3.2.31
2.34	Response of Soil and Rock to Dynamic Loading	2.3.2.32
2.35	Minimum Static Bearing Capacity	2.3.2.33
2.36	Earth Pressures	2.3.2.34
2.37	Soil Properties for Seismic Analysis of Buried Pipes	2.3.2.35
2.38	Static and Dynamic Stability of Facilities	2.3.2.36
2.39	Subsurface Instrumentation	2.3.2.37
2.40	Stability of Slopes	2.3.2.38
2.41	Embankments and Dams	2.3.2.39
2.42	CRAC 2 Computer Code Calculations	2.3.3
3.1	Site-Specific Design Basis Wind	3.3.3.1
3.2	Site-Specific Design Basis Tornado	3.3.3.2

Table 1.9-1 Summary of ABWR Standard Plant COL License Information (Continued)

Item No.	Subject	Subsection
3.3	Effect of Remainder of Plant Structures, Systems and Components Not Designed for Wind Loads	3.3.3.3
3.4	Effects of Remainder of Plant Structures, Systems and Components Not Designed for Tornado Loads	3.3.3.4
3.5	Flood Elevation	3.4.3.1
3.6	Ground Water Elevation	3.4.3.2
3.7	Flood Protection Requirements for Other Structures	3.4.3.3
3.8	Not Used	
3.9	Protection of Ultimate Heat Sink	3.5.4.1
3.10	Missiles Generated by Other Natural Phenomena	3.5.4.2
3.11	Site Proximity Missiles and Aircraft Hazards	3.5.4.3
3.12	Impact of Failure of Out of ABWR Standard Plant Scope Non-Safety-Related Structures, Systems, and Components Due to Design Basis Tornado	3.5.4.4
3.13	Turbine System Maintenance Program	3.5.4.5
3.14	Maintenance Equipment Missile Prevention Inside Containment	3.5.4.6
3.15	Failure of Structures, Systems, and Components Outside ABWR Standard Plant Scope	3.5.4.7
3.16	Details of Pipe Break Analysis Results and Protection Methods	3.6.5.1
3.17	Leak-Before-Break Analysis Report	3.6.5.2
3.18	Inservice Inspection of Piping in Containment Penetration Areas	3.6.5.3
3.19	Seismic Design Parameters	3.7.5.1
3.20	Pre-Earthquake Planning and Post-Earthquake Actions	3.7.5.2
3.21	Piping Analysis, Modeling of Piping Supports	3.7.5.3
3.22	Assessment of Interaction Due to Seismic Effects	3.7.5.4
3.23	Foundation Waterproofing	3.8.6.1
3.24	Site Specific Physical Properties and Foundation Settlement	3.8.6.2
3.25	Structural Integrity Pressure Results	3.8.6.3
3.26	Identification of Seismic Category I Structures	3.8.6.4
3.27	Reactor Internals Vibration Analysis, Measurement and Inspection Programs	3.9.7.1

Table 1.9-1 Summary of ABWR Standard Plant COL License Information (Continued)

Item No.	Subject	Subsection
3.28	ASME Class 2 or 3 Quality Group Components with 60-Year Design Life	3.9.7.2
3.29	Pump and Valve Testing Program	3.9.7.3
3.30	Audits of Design Specifications and Design Reports	3.9.7.4
3.31	Not Used	3.9.7.5
3.32	Not Used	3.9.7.6
3.33	Not Used	3.9.7.7
3.34	Not Used	3.9.7.8
3.35	Not Used	3.9.7.9
3.36	Not Used	3.9.7.10
3.37	Equipment Qualification	3.10.5.1
3.38	Dynamic Qualification Report	3.10.5.2
3.39	Qualification by Experience	3.10.5.3
3.40	Environmental Qualification Document (EQD)	3.11.6.1
3.41	Environmental Qualification Records	3.11.6.2
3.42	Surveillance, Maintenance, and Experience Information	3.11.6.3
3.43	Radiation Environment Conditions	3I.3.3.1
4.1	Thermal Hydraulic Stability	4.3.5.1
4.2	Power/Flow Operating Map	4.4.7.1
4.3	Thermal Limits	4.4.7.2
4.4	CRD Inspection Program	4.5.3.1
4.5	CRD and FMCRD Installation and Verification During Maintenance	4.6.6.1
5.1	Conversion of Indicators	5.2.6.1
5.2	Plant Specific ISI/PSI	5.2.6.2
5.3	Reactor Vessel Water Level Instrumentation	5.2.6.3
5.4	Fracture Toughness Data	5.3.4.1
5.5	Materials and Surveillance Capsule	5.3.4.2
5.6	Plant Specific Pressure-Temperature Information	5.3.4.3
5.7	Testing of Mainsteam Isolation Valves	5.4.15.1
5.8	Analyses of 8-hour RCIC Capability	5.4.15.2
5.9	ACIWA Flow Reduction	5.4.15.3

Table 1.9-1 Summary of ABWR Standard Plant COL License Information (Continued)

Item No.	Subject	Subsection
5.10	RIP Installation and Verification During Maintenance	5.4.15.4
6.1	Protection Coatings and Organic Materials	6.1.3.1
6.2	Alternate Hydrogen Control	6.2.7.1
6.3	Administrative Control Maintaining Containment Isolation	6.2.7.2
6.4	Suppression Pool Cleanliness	6.2.7.3
6.5	Wetwell-to-Drywell Vacuum Breaker Protection	6.2.7.4
6.5a	Containment Penetration Leakage Test (Type B)	6.2.7.5
6.6	ECCS Performance Results	6.3.6.1
6.7	ECCS Testing Requirements	6.3.6.2
6.7a	Limiting Break Results	6.3.6.3
6.8	Toxic Gases	6.4.7.1
6.9	SGTS Performance	6.5.5.1
6.9a	SGTS Exceeding 90 Hours of Operation per Year	6.5.5.2
6.10	PSI and ISI Program Plans	6.6.9.1
6.11	Access Requirement	6.6.9.2
7.1	Cooling Temperature Profiles for Class 1E Digital Equipment	7.3.3.1
7.2	APRM Oscillation Monitoring Logic	7.6.3.1
7.3	Effects of Station Blackout on HVAC	7.8.1
7.4	Electrostatic Discharge on Exposed Equipment Components	7.8.2
7.5	Localized High Heat Spots in Semiconductor Material for Computing Devices	7.8.3
8.1	Diesel Generator Reliability	8.1.4.1
8.2	Periodic Testing of Offsite Equipment	8.2.4.1
8.3	Procedures When a Reserve or Unit Auxiliary Transformer is Out of Service	8.2.4.2
8.4	Offsite Power Systems Design Bases	8.2.4.3
8.5	Offsite Power Systems Scope Split	8.2.4.4
8.6	Capacity of Auxiliary Transformers	8.2.4.5
8.7	Not Used	8.3.4.1
8.8	Diesel Generator Design Details	8.3.4.2

Table 1.9-1 Summary of ABWR Standard Plant COL License Information (Continued)

Item No.	Subject	Subsection
8.9	Not Used	8.3.4.3
8.10	Protective Devices for Electrical Penetration Assemblies	8.3.4.4
8.11	Not Used	8.3.4.5
8.12	Not Used	8.3.4.6
8.13	Not Used	8.3.4.7
8.14	Not Used	8.3.4.8
8.15	Offsite Power Supply Arrangements	8.3.4.9
8.16	Not Used	8.3.4.10
8.17	Not Used	8.3.4.11
8.18	Not Used	8.3.4.12
8.19	Load Testing of Class 1E Switchgear and Motor Control Centers	8.3.4.13
8.20	Administrative Controls for Bus Grounding Circuit Breakers	8.3.4.14
8.21	Administrative Controls for Manual Interconnections	8.3.4.15
8.22	Not Used	8.3.4.16
8.23	Common Industrial Standards Referenced in Purchase Specifications	8.3.4.17
8.24	Administrative Control for Switching 125Vdc Standby Charger	8.3.4.18
8.25	Control of Access to Class 1E Power Equipment	8.3.4.19
8.26	Periodic Testing of Voltage Protection Equipment	8.3.4.20
8.27	Diesel Generator Parallel Test Mode	8.3.4.21
8.28	Periodic Testing of Diesel Generator Protective Relaying	8.3.4.22
8.29	Periodic Testing of Diesel Generator Synchronizing Interlocks	8.3.4.23
8.30	Periodic Testing of Thermal Overloads and Bypass Circuitry	8.3.4.24
8.31	Periodic Inspection/Testing of Lighting System	8.3.4.25
8.32	Controls for Limiting Potential Hazards Into Cable Chases	8.3.4.26
8.33	Periodic Testing of Class 1E Equipment Protective Relaying	8.3.4.27
8.34	Periodic Testing of CVCF Power Supplies and EPAs	8.3.4.28
8.35	Periodic Testing of Class 1E Circuit Breakers	8.3.4.29

Table 1.9-1 Summary of ABWR Standard Plant COL License Information (Continued)

Item No.	Subject	Subsection
8.36	Periodic Testing of Electrical Systems & Equipment	8.3.4.30
8.37	Not Used	8.3.4.31
8.38	Class 1E Battery Installation and Maintenance Requirements	8.3.4.32
8.39	Periodic Testing of Class 1E Batteries	8.3.4.33
8.40	Periodic Testing of Class 1E CVCF Power Supplies	8.3.4.34
8.41	Periodic Testing of Class 1E Battery Chargers	8.3.4.35
8.42	Periodic Testing of Class 1E Diesel Generators	8.3.4.36
9.1	New Fuel Storage Racks Criticality Analysis	9.1.6.1
9.2	Dynamic and Impact Analysis of New Fuel Storage Racks	9.1.6.2
9.3	Spent Fuel Storage Racks Criticality Analysis	9.1.6.3
9.4	Spent Fuel Rack Load Drop Analysis	9.1.6.4
9.5	New Fuel Inspection Stand Seismic Capability	9.1.6.5
9.6	Overhead Load Handling System Information	9.1.6.6
9.7	Spent Fuel Racks Structural Evaluation	9.1.6.7
9.8	Spent Fuel Racks Thermal-Hydraulic Analysis	9.1.6.8
9.9	Spent Fuel Firewater Makeup Procedures and Training	9.1.6.9
9.10	Protection of RHR System Connections to FPC System	9.1.6.10
9.11	HECW System Refrigerator Requirements	9.2.17.1
9.12	Reactor Service Water System Requirements	9.2.17.2
9.12a	Not Used	9.3.12.1
9.13	Not Used	9.3.12.2
9.14	Not Used	9.3.12.3
9.15	Radioactive Drain Transfer System	9.3.12.4
9.16	Service Building HVAC System	9.4.10.1
9.17	Radwaste Building HVAC System	9.4.10.2
9.18	Contamination of DG Combustion Air Intake	9.5.13.1
9.19	Use of Communication System in Emergencies	9.5.13.2
9.20	Maintenance and Testing Procedures for Communication Equipment	9.5.13.3
9.21	Use of Portable Hand Light in Emergency	9.5.13.4
9.22	Vendor Specific Design of Diesel Generator Auxiliaries	9.5.13.5

Table 1.9-1 Summary of ABWR Standard Plant COL License Information (Continued)

Item No.	Subject	Subsection
9.23	Diesel Generator Cooling Water System Design Flow and Heat Removal Requirements	9.5.13.6
9.24	Fire Rating for Penetration Seals	9.5.13.7
9.25	Diesel Generator Requirements	9.5.13.8
9.26	Applicant Fire Protection Program	9.5.13.9
9.27	HVAC Pressure Calculations	9.5.13.10
9.28	Plant Security System Criteria	9.5.13.11
9.29	Not Used	9.5.13.12
9.30	Diesel Fuel Refueling Procedures	9.5.13.13
9.31	Portable and Fixed Emergency Communication Systems	9.5.13.14
9.32	Identification of Chemicals	9.5.13.15
9.33	NUREG/CR-0660 Diesel Generator Reliability Recommendations	9.5.13.16
9.34	Sound-Powered Telephone Units	9.5.13.17
9.35	Fire-Related Administrative Controls	9.5.13.18
9.36	Periodic Testing of Combustion Turbine Generator (CTG)	9.5.13.19
9.37	Operating Procedures for Station Blackout	9.5.13.20
9.38	Quality Assurance Requirements for CTG	9.5.13.21
10.1	Low Pressure Turbine Disk Fracture Toughness	10.2.5.1
10.2	Turbine Design Overspeed	10.2.5.2
10.3	Turbine Inservice Test and Inspection	10.2.5.3
10.4	Procedures to Avoid Steam Hammer and Discharge Loads	10.3.7.1
10.5	MSIV Leakage	10.3.7.2
10.6	Radiological Analysis of the TGSS Effluents	10.4.10.1
11.1	Plant-Specific Liquid Radwaste Information	11.2.5.1
11.2	Compliance With Appendix I to 10CFR50	11.3.11.1
11.3	Plant-Specific Solid Radwaste Information	11.4.3.1
11.4	Calculation of Radiation Release Rates	11.5.6.1
11.5	Compliance with the Regulatory Shielding Design Basis	11.5.6.2
11.6	Provisions for Isokinetic Sampling	11.5.6.3
11.7	Sampling of Radioactive Iodine and Particulates	11.5.6.4

Table 1.9-1 Summary of ABWR Standard Plant COL License Information (Continued)

Item No.	Subject	Subsection
11.8	Calibration Frequencies and Techniques	11.5.6.5
12.1	Regulatory Guide 8.10	12.1.4.1
12.2	Regulatory Guide 1.8	12.1.4.2
12.3	Occupational Radiation Exposure	12.1.4.3
12.4	Regulatory Guide 8.8	12.1.4.4
12.5	Compliance with 10CFR20 and 10CFR50 Appendix I	12.2.3.1
12.6	Airborne Radionuclide Concentration Calculation	12.3.7.1
12.7	Operational Considerations	12.7.3.2
12.8	Requirements of 10CFR70.24	12.3.7.3
12.9	Radiation Protection Program	12.5.3.1
12.10	Compliance With Para. 50.34(f)(xxvii) of 10CFR50 and NUREG-0737 Item III.D.3.3	12.5.3.2
13.1	Incorporation of Operating Experience	13.2.3.1
13.2	Emergency Plans	13.3.1.1
13.2a	Review and Audit	13.4.1
13.3	Plant Operating Procedures Development Plan	13.5.3.1
13.4	Emergency Procedures Development	13.5.3.2
13.5	Implementation of the Plan	13.5.3.3
13.6	Procedures Included in Scope of Plan	13.5.3.4
13.7	Physical Security Interfaces	13.6.3
14.1	Other Testing	14.2.13.1
14.2	Test Procedures/Startup Administrative Manual	14.2.13.2
14.3	Not Used	14.2.13.3
15.1	Anticipated Operational Occurrences (AOO)	15.0.5.1
15.2	Operating Limits	15.0.5.2
15.3	Design Basis Accidents	15.0.5.3
15.4	Radiological Effects of MSIV Closure	15.2.10.1
15.5	Mislocated Fuel Bundle Accident	15.4.11.1
15.6	Misoriented Fuel Bundle Accident	15.4.11.2
15.7	Iodine Removal Credit	15.6.7.1
15.8	Not Used	
15.9	Radiological Consequences of Non-Line Break Accidents	15.7.6.1

Table 1.9-1 Summary of ABWR Standard Plant COL License Information (Continued)

Item No.	Subject	Subsection
16.1	COL Information Required for Plant Specific Technical Specifications	16.1.1
17.1	QA Programs For Construction And Operation	17.0.1.1
17.2	Policy and Implementation Procedures for D-RAP	17.3.13.1
17.3	D-RAP Organization	17.3.13.2
17.4	Provision for O-RAP	17.3.13.3
18.1	HSI Design Implementation Process	18.8.1
18.2	Number of Operators Needing Controls Access	18.8.2
18.3	Automation Strategies and Their Effects on Operator Reliability	18.8.3
18.4	SPDS Integration With Related Emergency Response Capabilities	18.8.4
18.5	Standard Design Features Design Validation	18.8.5
18.6	Remote Shutdown System Design Evaluation	18.8.6
18.7	Local Valve Position Indication	18.8.7
18.8	Operator Training	18.8.8
18.9	Safety System Status Monitoring	18.8.9
18.10	PGCS Malfunction	18.8.10
18.11	Local Control Stations	18.8.11
18.12	As-Built Evaluation of MCR and RSS	18.8.12
18.13	Accident Monitoring Instrumentation	18.8.13
18.14	In-Core Cooling Instrumentation	18.8.14
18.15	Performance of Critical Tasks	18.8.15
18.16	Plant Status and Post-Accident Monitoring	18.8.16
19.1	Post Accident Recovery Procedure for Unisolated CUW Line Break	19.9.1
19.2	Confirmation of CUW Operation Beyond Design Bases	19.9.2
19.3	Event Specific Procedures for Severe External Flooding	19.9.3
19.4	Confirmation of Seismic Capacities Beyond the Plant Design Bases	19.9.4
19.5	Plant Walkdowns	19.9.5
19.6	Confirmation of Loss of AC Power Event	19.9.6
19.7	Procedures and Training for Use of AC-Independent Water Addition System	19.9.7

Table 1.9-1 Summary of ABWR Standard Plant COL License Information (Continued)

Item No.	Subject	Subsection
19.8	Action to Avoid Common-Cause Failures in the Essential Multiplexing System (EMUX) and Other Common-Cause Failures	19.9.8
19.9	Action to Mitigate Station Blackout Events	19.9.9
19.10	Actions to Reduce Risk of Internal Flooding	19.9.10
19.11	Actions to Avoid Loss of Decay Heat Removal and Minimize Shutdown Risk	19.9.11
19.12	Procedures for Operation of RCIC from Outside the Control Room	19.9.12
19.13	ECCS Test and Surveillance Intervals	19.9.13
19.14	Accident Management	19.9.14
19.15	Manual Operation of MOVs	19.9.15
19.16	High Pressure Core Flooder Discharge Valve	19.9.16
19.17	Capability of Containment Isolation Valves	19.9.17
19.18	Procedures to Ensure Sample Lines and Drywell Purge Lines Remain Closed During Operation	19.9.18
19.19	Procedures for Combustion Turbine Generator to Supply Power to Condensate Pumps	19.9.19
19.19a	Actions to Assure Reliability of the Supporting RCW and Service Water Systems	19.9.20
19.19b	Housing of AICWA Equipment	19.9.21
19.19c	Procedures to Assure SRV Operability During Station Blackout	19.9.22
19.19d	Procedures for Ensuring Integrity of Freeze Seals	19.923
19.19e	Procedures for Controlling Combustibles During Shutdown	19.9.24
19.19f	Outage Planning and Control	19.9.25
19.19g	Reactor Service Water Systems Definition	19.9.26
19.19h	Capability of Vacuum Breakers	19.9.27
19.19i	Capability of the Containment Atmosphere Monitoring System	19.9.28
19.19j	Plant Specific Safety-Related Issues and Vendors Operating Guidance	19.9.29
19.20	Long-Term Training Upgrade	19A.3.1

Table 1.9-1 Summary of ABWR Standard Plant COL License Information (Continued)

Item No.	Subject	Subsection
19.21	Long-Term Program of Upgrading of Procedures	19A.3.2
19.22	Purge System Reliability	19A.3.3
19.23	Licensing Emergency Support Facility	19A.3.4
19.24	In-Plant Radiation Monitoring	19A.3.5
19.25	Feedback of Operating, Design and Construction Experience	19A.3.6
19.26	Organization and Staffing to Oversee Design and Construction	19A.3.7
19.27	Develop More Detailed QA Criteria	19A.3.8
19.28	COL Applicant Safety Issues	19B.3.1
19.28a	Testing of Isolators	19B.3.2
19.29	Seismic Capacity	19H.5.1
19.30	PRA Update	19.9.30

1A Response to TMI Related Matters

1A.1 Introduction

The investigations and studies associated with the TMI accident produced several documents specifying results and recommendations, which prompted the issuances by the NRC of various bulletins, letters, and NUREGs providing guidance and requiring specific actions by the nuclear power industry. In May 1980, the issuance of NUREG-0660 (Reference 1A-1) provided a comprehensive and integrated plan and listing requirements to correct or improve the regulation and operation of nuclear facilities based on the experience from the accident at TMI and the studies and investigations of the accident. NUREG-0737(Reference 1A-2), issued in November 1980, listed items from NUREG-0660 approved by the NRC for implementation, and included additional information concerning schedules, applicability, method of implementation review, submittal dates, and clarification of technical positions. Finally, NUREG-0718 (Reference 1A-3) was issued in June 1981 to provide guidance that the NRC staff believes should be followed to account for the lessons learned from the TMI accident.

This Appendix 1A provides GE's responses for the ABWR Standard Plant required by Section II of the NRC Standard Review Plan, those satisfying 10CFR50.34(f) are addressed in Appendix 19A. The remaining TMI issues satisfying severe accident requirements are addressed in Appendix 19B.

1A.2 NRC Positions/Responses

1A.2.1 Short-Term Accident Analysis Procedure Revision [I.C.1(3)]

NRC Position

In letters of September 13 and 27, October 10 and 30, and November 9, 1979 (References 1A-7 through 1A-11), the Office of Nuclear Reactor Regulation required licensees of operating plants, applicants for operating licenses and licensees of plants under construction to perform analyses of transients and accidents, prepare emergency procedure guidelines, upgrade emergency procedures, including procedures for operating with natural circulation conditions, and to conduct operator retraining (see also Item I.A.2.1). Emergency procedures are required to be consistent with the actions necessary to cope with the transients and accidents analyzed. Analyses of transients and accidents were to be completed in early 1980 and implementation of procedures and retraining were to be completed 3 months after emergency procedure guidelines were established; however, some difficulty in completing these requirements has been experienced. Clarification of the scope of the task and appropriate schedule revisions are being developed. In the course of review of these matters on Babcock and Wilcox (B&W)-designed plants, the staff will follow up on the bulletin and order matters relating to analysis methods and results, as listed in NUREG-0660, Appendix C

(Table C.1, Items 3, 4, 16, 18, 24, 25, 26, 27; Table C.2, Items 4, 12, 17, 18, 19, 20; and Table C.3, Items 6, 35, 37, 38, 39, 41, 47, 55, 57).

Response

In the clarification of the NUREG-0737 requirement for reanalysis of transients and accidents and inadequate core cooling and preparation of guidelines for development of emergency procedures, NUREG-0737 states:

> Owners' group or vendor submittals may be referenced as appropriate to support this reanalysis. If owners' group or vendor submittals have already been forwarded to the staff for review, a brief description of the submittals and justification of their adequacy to support guideline development is all that is required.

GE has participated, and continues to participate, in the BWR Owners' Group (BWROG) program to develop emergency procedure guidelines for GE BWRs. The resulting emergency procedure guidelines are generally applicable to the ABWR, as are the transient and accident analyses. Following is a brief description of the submittals to date, and a justification of their adequacy to support guideline development.

(1) Description of Submittals

(a) NEDO-24708, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors", August 1979.

(b) NEDO-24708A, Revision 1,"Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors", December 1980. This report was issued via the letter from D. B. Waters (BWR Owners' Group) to D. G. Eisenhut (NRC) dated March 20, 1981.

(c) "BWR Emergency Procedure Guidelines (Rev. 0)"—submitted in prepublication form June 30, 1980.

(d) "BWR Emergency Procedure Guidelines (Rev. 1)"—Issued via the letter from D. B. Waters (BWR Owners' Group) to D. G. Eisenhut (NRC) dated January 31, 1981.

(e) "BWR Emergency Procedure Guidelines (Rev. 2)"—submitted in prepublication form June 1, 1982, Letter BWROG-8219 from T. J. Dente (BWR Owners' Group) to D. G. Eisenhut (NRC).

(f) "BWR Emergency Procedure Guidelines (Rev. 3)", submitted in prepublication form December 22, 1982, Letter BWROG-8262 from T. J. Dente (BWR Owners' Group) to D. G. Eisenhut (NRC).

(g) NEDO-31331, “BWR Emergency Procedure Guidelines (Rev. 4)”, submitted April 23, 1987, Letter BWROG-8717, from T. A. Pickens (BWR Owners’ Group) to T. Murley (NRC).

(2) Adequacy of Submittals

The submittals described in (1) above have been discussed and reviewed extensively among the BWR Owners’ Group, the General Electric Company, and the NRC Staff.

The NRC has extensively reviewed the latest revision (Revision 4) of the Emergency Procedures Guidelines and issued a SER, “Safety Evaluation of BWR Owners’ Group Emergency Procedure Guidelines, Revision 4, NEDO-31331, March 1987”, letter from A. C. Thadani, NRC Office of Nuclear Reactor Regulation, to D. Grace, Chairman of BWR Owners’ Group, dated September 12, 1988. The SER concludes that this document is acceptable for implementation. It further states that the SER closes all the open items carried from the previous revisions of the EPG.

GE believes that, in view of these findings, no further detailed justification of the analyses or guidelines is necessary at this time. COL license information requirements pertaining to emergency procedures are discussed in Subsection 1A.3.1.

1A.2.2 Control Room Design Reviews—Guidelines and Requirements [I.D.1(1)]

NRC Position

In accordance with task Action Plan I.D.1(1), all licensees and applicants for operating licenses will be required to conduct a detailed control room design review to identify and correct design deficiencies. This detailed control room design review is expected to take about a year. Therefore, the Office of Nuclear Reactor Regulation (NRR) requires that those applicants for operating licenses who are unable to complete this review prior to issuance of a license make preliminary assessments of their control rooms to identify significant human factors and instrumentation problems and establish a schedule approved by NRC for correcting deficiencies. These applicants will be required to complete the more detailed control room reviews on the same schedule as licensees with operating plants.

Response

The design of the main control room will utilize accepted human factors engineering principles, incorporating the results of a full systems analysis similar to that described in Appendix B of NUREG-0700. A DCRDR specified in NUREG-0737 is not required by SRP Section 18.1. Details are described in Chapter 18.

1A.2.3 Control Room Design—Plant Safety Parameter Display Console [I.D.2]

NRC Position

In accordance with Task Action Plan I.D.2, each applicant and licensee shall install a Safety Parameter Display System (SPDS) that will display to operating personnel a minimum set of parameters which define the safety status of the plant. This can be attained through continuous indication of direct and derived variables as necessary to assess plant safety status.

Response

The functions of the SPDS will be integrated into the overall control room design, as permitted by SRP Section 18.2. Details are found in Chapter 18.

1A.2.4 Scope of Test Program—Preoperational and Low Power Testing [I.G.1]

NRC Position

Supplement operator training by completing the special low-power test program. Tests may be observed by other shifts or repeated on other shifts to provide training to the operators.

Response

The initial test program presents an excellent opportunity for licensed operators and other plant staff members to gain valuable experience and training and, in fact, these benefits are objectives of the program (Subsection 14.2.1). The degree to which the potential benefit is realized will depend on such plant-specific factors as the organizational makeup of the startup group and overall plant staff (Subsections 14.2.2 and 13.1), as well as how the test program is conducted (Subsection 14.2.4).

The BWR Owners' Group response to Item I.G.1 of NUREG-0737 is documented in a letter of February 4, 1981 from D.B. Waters to D.G. Eisenhut. For the most part, this issue concerns training requirements, although in the context of the initial test program. Thus, the BWROG response primarily deals with operator training issues. The exception is Appendix E of the BWROG response which describes additional tests to be conducted during the preoperational and/or startup phase.

The specific training requirements for reactor operators are discussed in Section 13.2 of the SRP, which is outside the scope of the ABWR Standard Plant (See Table 1.9-1 for COL license information requirements.). Details are found in Chapter 13.2

The additional tests specified in Appendix E of the BWROG response are contained within the initial test program described in Chapter 14. See specifically Subsections 14.2.12.1.1(3)(a), 14.2.12.1.9(3)(j), and 14.2.12.1.44(3)(a) for the relevant testing.

1A.2.5 Reactor Coolant System Vents [II.B.1]

NRC Position

Each applicant and licensee shall install Reactor Coolant System (RCS) and reactor vessel head high point vents remotely operated from the control room. Although the purpose of the system is to vent noncondensible gases from the RCS, which may inhibit core cooling during natural circulation, the vents must not lead to an unacceptable increase in the probability of a loss-of-coolant accident (LOCA) or a challenge to containment integrity. Since these vents form a part of the reactor coolant pressure boundary, the design of the vents shall conform to the requirements of Appendix A to 10CFR50, "General Design Criteria". The vent system shall be designed with sufficient redundancy to assure a low probability of inadvertent or irreversible actuation.

Each licensee shall provide the following information concerning the design and operation of the high point vent system:

(1) Submit a description of the design, location, size, and power supply for the vent system along with results of analyses for LOCAs initiated by a break in the vent pipe. The results of the analyses should demonstrate compliance with the acceptance criteria of 10CFR50.46.

(2) Submit procedures and supporting analyses for operator use of the vents that also include the information available to the operator for initiating or terminating vent usage.

Response

The capability to vent the ABWR reactor coolant system is provided by the safety/relief valves (SRVs) and reactor coolant vent line, as well as other systems. The COL applicant will develop plant-specific procedures to govern the operator's use of the relief mode for venting the reactor (Subsection 1A.3.6). The capability of these systems and their satisfaction of Item II.B.1 are discussed below.

The ABWR design includes various means of high-point venting. Among these are:

(1) Normally closed reactor vessel head vent valves, operable from the control room, which discharge to the drywell. The reactor coolant vent line is located at the very top of the reactor vessel as shown in the nuclear boiler system P&ID (Figure 5.1-3). This 50A line contains two safety-related Class 1E motor-operated valves that are operated from the control room. The location of this line permits it to vent the entire reactor core system normally connected to the reactor pressure vessel. In addition, since this vent line is part of the original design, it has already been considered in all the design basis accident (DBA) analyses contained elsewhere in this document.

(2) Normally open reactor head vent line, which discharges to a main steamline.

The conclusions from this vent evaluation are as follows:

(1) Reactor vessel head vent valves exist to relieve head pressure (at shutdown) to the drywell via remote operator action.

(2) The reactor vessel head is continuously swept to the main condenser and can be vented during operating conditions.

(3) The size of the vents is not a critical issue because BWR SRVs have substantial capacity, exceeding the full power steaming rate of the nuclear boiler.

(4) No new 10CFR50.46 conformance calculations are required, because the vent provisions are part of the plant’s original design and are covered by the original design bases.

(5) Plant-specific procedures govern the operator’s use of the relief mode for venting reactor pressure.

1A.2.6 Plant Shielding to Provide Access to Vital Areas and Protect Safety Equipment for Post-Accident Operation [II.B.2]

NRC Position

With the assumption of a post-accident release of radioactivity equivalent to that described in Regulatory Guides 1.3 and 1.4 (i.e., the equivalent of 50% of the core radioiodine, 100% of the core noble gas inventory, and 1% of the core solids are contained in the primary coolant), each licensee shall perform a radiation and shielding-design review of the spaces around systems that may, as a result of an accident, contain highly radioactive materials. The design review should identify the location of vital areas and equipment, such as the control room, radwaste control stations, emergency power supplies, motor control centers, and instrument areas, in which personnel occupancy may be unduly limited or safety equipment may be unduly degraded by the radiation fields during post-accident operation of these systems.

Each licensee shall provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent or temporary shielding, or post-accident procedural controls. The design review shall determine which types of corrective actions are needed for vital areas throughout the facility.

Response

A review of the radiation and shielding of the ABWR Standard Plant post- accident operations has been made. It has been found that there is adequate access to vital areas and that safety equipment is adequately protected. No need for corrective action was identified. Details of the review may be found in Attachment A to Appendix 1A.

1A.2.7 Post-Accident Sampling [II.B.3]

NRC Position

A design and operational review of the reactor coolant and containment atmosphere sampling line systems shall be performed to determine the capability of personnel to obtain (less than 1 hour) a sample under accident conditions without incurring a radiation exposure to any individual in excess of 0.05 and 0.50 Sv to the whole body or extremities, respectively. Accident conditions should assume a Regulatory Guide 1.3 or 1.4 release of fission products. If the review indicates that personnel could not promptly and safely obtain the samples, additional design features or shielding should be provided to meet the criteria.

A design and operational review of the radiological spectrum analysis facilities shall be performed to determine the capability to quantify (in less than 2 hours) certain radionuclides that are indicators of the degree of core damage. Such radionuclides are noble gases (which indicate cladding failure), iodines and cesiums (which indicate high fuel temperatures), and nonvolatile isotopes (which indicate fuel melting). The initial reactor coolant spectrum should correspond to a Regulatory Guide 1.3 or 1.4 release. The review should also consider the effects of direct radiation from piping and components in the auxiliary building and possible contamination and direct radiation from airborne effluents. If the review indicates that the analyses required cannot be performed in a prompt manner with existing equipment, then design modifications or equipment procurement shall be undertaken to meet the criteria.

In addition to the radiological analyses, certain chemical analyses are necessary for monitoring reactor conditions. Procedures shall be provided to perform boron and chloride chemical analyses assuming a highly radioactive initial sample (Regulatory Guide 1.3 or 1.4 source term). Both analyses shall be capable of being completed promptly (i.e., the boron sample analysis within an hour and the chloride sample analysis within a shift).

Response

Discharges From Plant and Containment— During the development of an accident, samples of liquid and gaseous discharges from both the plant and containment will be obtained. Chemical and radiochemical analyses will be performed for protection of the health and safety of the public and the plant operators. These samples will be obtained from the Process Sampling System. The Post Accident Sampling Systems will not be required to obtain these samples.

Core Damage Assessment—During this initial period, instrumentation will provide sufficient information for the operators to perform their duties. For example, the containment high range radiation meters will give instant information concerning the radiation level in containment (To obtain data from the PASS several hours may be required for sampling and analyses.). Calculations can be performed to relate containment radiation level with the probable extent of core damage. Core damage assessment instrumentation is described in Section 18.4.6. This section describes the safety parameter display system (SPDS). The principle purpose of the SPDS is to aid the control room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing whether abnormal conditions warrant corrective action by operators to avoid a degraded core. The following critical safety functions are provided at the wide screen display panel in the main control room:

(1) Reactivity control

(2) Reactor core cooling and heat removal from the primary system

(3) Reactor coolant system integrity

(4) Radioactivity control

(5) Contamination conditions

This instrumentation and the PASS work together to obtain complementary information. After this initial period during the development of an accident, the ABWR PASS will be used to obtain samples of reactor water and containment atmosphere to assess the extent of core damage. The ABWR PASS has been designed to safely obtain samples with radioactivity levels up to 37,000 M Bq/g. Approximately one day after a serious core damage accident, it is expected that sample radioactivity levels will be no more than this value. Early in such an accident, the plant instrumentation in the main control room would be indicating that abnormal conditions exist. If a reactor coolant sample were obtained which had excessive radioactivity, as measured by the area radiation monitor in the PASS area, the plant operators would use this high radiation information as confirmatory evidence that severe core damage has occurred and continue following the emergency operating procedures. It would not be necessary to perform any radiochemical analyses to reach this conclusion. During less severe accidents, in which only some cladding damage has occurred, samples may be obtained from either the Process Sampling System or PASS.

NUREG-0737 Requirements— The ABWR PASS has been designed to meet the eleven requirements listed in NUREG-0737 except as noted below:

(1) The combined time allotted for sampling and analysis should be 3 hours or less from the time a decision is made to take a sample. Meets the requirements of NUREG-0737.

(2) There shall be onsite capability to perform the following within the 3 hour time period:

(a) Determine the presence and amount of certain radionuclides in the reactor coolant and containment atmosphere that may be indicators of the degree of core damage. Meets the requirements of NUREG-0737.

(b) Hydrogen in containment atmosphere. Hydrogen in containment atmosphere is measured by the Containment Atmospheric Monitoring System. Meets the requirements of NUREG-0737.

(c) Dissolved gases, chloride and boron in liquids. Dissolved gases are discussed in item 4 below. Meets the requirements concerning chloride and boron of NUREG-0737.

(d) Inline monitoring capability is acceptable. No inline monitors are provided in PASS.

(3) Sampling need not depend upon an isolated auxiliary system being put into operation. Meets the requirements of NUREG-0737.

(4) Reactor coolant samples and analyses for total dissolved gases and hydrogen are required. During a severe core damage accident for the ABWR, the reactor water will become mixed with the suppression pool water. The pressure in the reactor vessel will decrease to approximately the pressure within the wetwell and the drywell. As a result of this decrease in pressure, dissolved gases will partially pass out of the water phase into the gas phase. Data on gases dissolved in the reactor water under these conditions will have little meaning in responding to the accident. During accidents in which only a small amount of cladding damage has occurred or in accidents in which the reactor vessel has not been depressurized, pressurized reactor water samples may be obtained from the Process Sampling System. Therefore, the ability to obtain pressurized or depressurized reactor water samples for dissolved gas analyses has not been provided.

(5) If both of the following are present:

(a) There is only a single barrier between primary containment and the cooling water.

(b) If the cooling water is sea water or brackish water, chloride analysis within 24 hours after sampling shall be provided. If both are not present, the time to complete the analyses is increased to 4 days. Analysis does not have to be done onsite. Meets the requirements of NUREG-0737. (Note that there are several barriers to prevent chloride intrusion from the power cycle cooling water into the reactor vessel. These barriers are: the main condenser tubing, the condensate polishing demineralizers and the pumps and valves in the condensate/feedwater systems. These pumps are stopped and these valves closed during upset conditions. Thus, because both factors are not present, the time to complete the analysis is increased to 4 days.)

(6) It must be possible to obtain and analyze a sample without radiation exposures to any individual exceeding 0.05 Sv for whole body and 0.50 Sv for extremities. Meets the requirements of 50.34(f)(2)(viii).

(7) Ability to sample and analyze for reactor coolant boron must be provided. Meets the requirements of NUREG-0737.

(8) If inline monitoring is used, backup sampling and analysis capability must be provided. Inline monitoring is not used. Meets the requirements of NUREG-0737.

(9)

(a) Capability to identify and quantify a specified number of isotopes over a range of nuclide concentrations from approximately 37,000 Bq/g to 370,000 M Bq/g. Capability is provided to identify and quantify the desired isotopes in samples over a range from approximately 37,000 Bq/g to 37,000 M Bq/g. Samples obtained during the accident recovery phase would be within this range for most core damage accidents. If the gross radioactivity levels are higher than 37,000 M Bq/g, this would confirm that severe core damage has occurred.

(b) Restrict background levels of radiation in the laboratory and provide proper ventilation. Meets the requirements of NUREG-0737.

(10) Provide adequate accuracy, range and sensitivity to provide pertinent information. Meets the requirements of NUREG-0737.

(11)

(a) Provide sample lines with proper features for sampling during accident conditions. Meets the requirements of NUREG-0737.

(b) PASS ventilation exhaust should be filtered with charcoal adsorbers and HEPA filters. Meets the requirements of NUREG-0737.

Summary

The post-accident sampling system meets the requirements of the NRC position with the following exceptions:

(1) The upper limit of activity in the samples at the time they are taken is as follows:

(a) Liquid sample 37,000 M Bq / ml

(b) Gas sample 3700 M Bq / ml.

(2) Radiological measurements could be performed 24 hours following the accident.

(3) Boron concentration measurements could be performed 8 hours following the accident.

(4) There is no need to perform chloride measurements.

(5) There is no need to analyze dissolved gases.

1A.2.8 Rule Making Proceeding or Degraded Core Accidents [II.B.8]

Response to this TMI action plan item is addressed in Appendix 19A.

1A.2.9 Coolant System Valves—Testing Requirements [II.D.1]

NRC Position

Pressurized water reactor and boiling water reactor licensees and applicants shall conduct testing to qualify the reactor coolant system relief and safety valves under expected operating conditions for design basis transients and accidents.

Response

The ABWR safety / relief valve (SRV) is postulated to discharge steam only, not liquid or two-phase flow under expected operating conditions for design basis transients and accidents.

A generic test program was conducted through the BWR Owners' Group (Reference 1A-10) to satisfy the discharge of steam. These steam discharge test results will be used as the qualification basis for plant-specific SRV models and discharge piping that are sufficiently similar to those reported in Reference 1A-11. [*Plant-specific SRV models and discharge piping that are not similar will be tested in accordance with NUREG-0737 requirements.*]* See Subsection 1A.3.7 for COL license information.

The ABWR system logic for response to high water level conditions is described in Subsection 7.3.1.1.1.1(3) and is considered to be sufficiently redundant that the

* See Subsection 3.9.1.7.

probability of steamline flooding by ECCS is extremely low. There is no high drywell pressure signal that would inhibit this logic system.

In the ABWR design, each of three RHR shutdown cooling lines has its own separate containment penetration and its own separate source of suction from the reactor vessel. Alternate shutdown using the SRV is therefore not required for the ABWR in order to meet single failure rules. Hence, the ABWR does not require SRV testing with liquid under low pressure conditions associated with this event as required in past BWRs.

1A.2.10 Relief and Safety Valve Position Indication [II.D.3]

NRC Position

Reactor Coolant System relief and safety valves shall be provided with a positive indication in the control room derived from a reliable valve-position detection device or a reliable indication of flow in the discharge pipe.

Response

The ABWR Standard Plant SRVs are equipped with position sensors which are qualified as Class 1E components. These are used to monitor valve position.

In addition, the downstream pipe from each valve line is equipped with temperature elements which signal the annunciator and the plant process computer when the temperature in the tailpipe exceeds the predetermined setpoint.

These sensors are shown on Figure 5.1-3 (Nuclear Boiler System P&ID).

1A.2.11 Systems Reliability [II.E.3.2]

This TMI action plan item superseded by USI A-45. USI A-45 is addressed in Appendix 19B.

1A.2.12 Coordinated Study of Shutdown Heat Removal Requirements [II.E.3.3]

This TMI action plan item superseded by USI A-45. USI A-45 is addressed in Appendix 19B.

1A.2.13 Containment Design—Dedicated Penetration [II.E.4.1]

NRC Position

For plant designs with external hydrogen recombiners, provide redundant dedicated containment penetrations so that, assuming a single failure, the recombiner systems can be connected to the containment atmosphere.

Response

A Flammability Control System is provided to control the concentration of oxygen in the primary containment. The FCS utilizes two permanently installed recombiners located in the secondary containment. The FCS is operable in the event of a single active failure. The FCS is described in Subsection 6.2.5.

1A.2.14 Containment Design—Isolation Dependability [II.E.4.2]

NRC Position

(1) Containment isolation system designs shall comply with the recommendations of the Standard Review Plan, Subsection 6.2.4 (i.e., that there be diversity in the parameters sensed for the initiation of containment isolation).

(2) All plant personnel shall give careful consideration to the definition of essential and non-essential systems, identify each system determined to be non-essential, describe the basis for selection of each essential system, modify their containment isolation designs accordingly, and report the results of the reevaluation to the NRC.

(3) All nonessential systems shall be automatically isolated by the containment isolation signal.

(4) The design of control systems for automatic containment isolation valves shall be such that resetting the isolation signal will not result in the automatic reopening of containment isolation valves. Reopening of containment isolation valves shall require deliberate operator action.

(5) The containment setpoint pressure that initiates containment isolation for non-essential penetrations must be reduced to the minimum compatible with normal operating conditions.

(6) Containment purge valves that do not satisfy the operability criteria set forth in Branch Technical Position CSB 6-4 or the Staff Interim Position of October 23, 1979 must be sealed closed as defined in SRP 6.2.4, Item II.6.f during operational conditions 1, 2, 3, and 4. Furthermore, these valves must be verified to be closed at least every 31 days.

(7) Containment purge and vent isolation valves must close on a high radiation signal.

Response

(1) The isolation provisions described in the Standard Review Plan, Subsection 6.2.4 (i.e., that there be diversity in the parameters sensed for the initiation of containment isolation) were reviewed in conjunction with the ABWR Standard Plant design. It was determined that the ABWR Standard Plan is designed in accordance with these recommendations of the SRP.

(2) This request appears to be directed primarily toward operating plants. However, the classification of structures, systems and components for the ABWR Standard Plant design is addressed in Section 3.2. The basis for classification is also presented in Section 3.2. The ESF system, with remote manual valves with leakage detection outside the containment, is delineated in Tables 6.2-7. The ABWR Standard Plant fully conforms with the NRC position so far as it relates to the new equipment supplier.

(3) All non-essential systems comply with the NRC position to automatically isolate by the containment isolation signals, and by redundant safety grade isolation valves.

(4) Control systems for automatic containment isolation valves are designed in accordance with this position for the ABWR Standard Plant Design.

(5) The ABWR Standard Plant design is consistent with this position.

(6) All ABWR containment purge valves meet the criteria provided in BTP CSB 6-4. The main 550A purge valves are fail-closed and are maintained closed through power operation as defined in the plant technical specifications. All purge and vent valves are remote pneumatically-operated, fail closed and receive containment isolation signals. Certain vent valves can be opened manually in the presence of an isolation signal, to permit venting through the SGTS.

(7) In the ABWR design, the containment purge and vent isolation valves will be automatically isolated on high radiation levels detected in the Reactor Building HVAC air exhaust or in the fuel handling area air exhaust.

1A.2.15 Additional Accident—Monitoring Instrumentation [II.F.1]

NRC Position

Noble gas effluent monitors shall be installed with an extended range designed to function during accident conditions, as well as during normal operating conditions. Multiple monitors are considered necessary to cover the ranges of interest.

(1) Noble gas effluent monitors with an upper range capacity of 3.7E+09 M BQ/cc (Xe133) are considered to be practical and should be installed in all operating plants.

(2) Noble gas effluent monitoring shall be provided for the total range of concentration extending from normal condition (as low as reasonably achievable (ALARA)) concentrations to a maximum of 3.7E+09 M Bq/cc (Xe-

133). Multiple monitors are considered to be necessary to cover the ranges of interest. The range capacity of individual monitors should overlap by a factor of ten.

Because iodine gaseous effluent monitors for the accident condition are not considered to be practical at this time, capability for effluent monitoring of radioiodines for the accident condition shall be provided with sampling conducted by absorption on charcoal or other media, followed by onsite laboratory analysis.

In-containment radiation-level monitors with a maximum range of 1E+06 Gy/h shall be installed. A minimum of two such monitors that are physically separated shall be provided. Monitors shall be developed and qualified to function in an accident environment.

A continuous indication of containment pressure shall be provided in the control room of each operating reactor. Measurement and indication capability shall include three times the design pressure of the containment for concrete, four times the design pressure for steel, and –34.32 kPa G for all containments.

A continuous indication of containment water level shall be provided in the control room for all plants. A narrow range instrument shall be provided for BWRs and cover the range from the bottom to the top of the containment sump. A wide range instrument shall also be provided for BWRs and shall cover the range from the bottom of the containment to the elevation equivalent to a 2.27 x 10^6 liter capacity. For BWRs, a wide range instrument shall be provided and cover the range from the bottom to 1.52 meters above the normal water level of the suppression pool.

A continuous indication of hydrogen concentration in the containment atmosphere shall be provided in the control room. Measurement capability shall be provided over the range of 0 to 10% hydrogen concentration under both positive and negative ambient pressure.

Response

GE believes the requirements of Regulatory Guide 1.97, incorporate the above requirements. Section 7.5 compares the ABWR design against this Regulatory Guide and Subsection 18.8.13 addresses the operator interface of the instrumentation.

1A.2.16 Identification of and Recovery from Conditions Leading to Inadequate Core Cooling [II.F.2]

NRC Position

Licensees shall provide a description of any additional instrumentation controls (primary or backup) proposed for the plant to supplement existing instrumentation (including primary coolant saturation monitors) in order to provide an unambiguous, easy-to-interpret indication of inadequate core cooling (ICC). A description of the functional design requirements for the system shall also be included. A description of the procedures to be used with the proposed equipment, the analysis used in developing these procedures, and a schedule for installing the equipment shall be provided.

Response

The direct water level instrumentation provided in the ABWR design is capable of detecting conditions indicative of inadequate core cooling.

The ABWR has two sets of four wide range reactor water level sensing units (eight total) which are used in two separate two-out-of-four logics which initiate ECCS and other safety functions. Each set of four sensors are used in two separate two-out-of-four logics which initiate ECCS operation. Four separate sets of sensing lines, one from each quadrant of the reactor pressure vessel, supply the pressure to the eight sensors for reliability. The ABWR arrangement of reactor water level sensing complies with the NRC Generic Letter 84-23. The vertical drop inside the drywell of the reactor pressure vessel reference leg water level instrument lines is limited to 0.9 meters. Analog level transmitters are employed to monitor the reactor vessel water level. For the safety related functions initiated automatically upon receipt of a reactor pressure vessel water level trip signal, two-out-of-four trip initiation logic is employed, utilizing a signal from a level transmitter in each of the four instrument divisions. This provides high reliability for initiation upon demand, and high tolerance against inadvertent initiation.

To address the US NRC staff’s concerns about the potential for reactor pressure vessel water level measurement errors resulting from dissolved non-condensable gasses in the water column in the reactor pressure vessel reference leg water level instrument lines (NRC Generic Letter 92-04 and NRC Information Notice 93-27), the ABWR has implemented continuous purging of the reactor pressure vessel reference leg water level instrument lines. Water is continuously injected into the reactor water level reference leg water level instrument lines by means of the Control Rod Drive (CRD) System. This is shown in Figure 5.1-3 and discussed in Subsection 7.7.1.1.

Based on the above information, the existing highly redundant direct water level instrumentation already provides an unambiguous, easy to interpret indication of inadequate core cooling, and there are no plans to include additional instrumentation in the ABWR design. Subsection 18.8.14 addresses the COL license information requirements for these instruments and controls.

1A.2.17 Instruments for Monitoring Accident Conditions [II.F-3]

NRC Position

Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage.

Response

The ABWR Standard Plant is designed in accordance with Regulatory Guide 1.97 . A detailed assessment of the Regulatory Guide, including the list of instruments, is found in Section 7.5.

1A.2.18 Safety-Related Valve Position Indication [II.K.1(5)]

NRC Position

(1) Review all valve positions and positioning requirements and positive controls and all related test and maintenance procedures to assure proper ESF functioning, if required.

(2) Verify that AFW valves are in the open position.

Response

(1) The ABWR Standard Plant is equipped with status monitoring that satisfies the requirements of Regulatory Guide 1.47. See Subsection 7.1.2 for detailed information on the status monitoring equipment and capability provided in the ABWR Standard Plant design.

In addition to the status monitoring, the COL applicant plant-specific procedures (Subsection 1A.3.2) will assure that independent verification of system lineups is applied to valve and electrical lineups for all safety-related equipment, to surveillance procedures, to restoration following maintenance and to comply with IE Bulletin 79-08. Through these procedures, approval will be required for the performance of surveillance tests and maintenance, including equipment removal from service and return to service.

(2) This requirement is not applicable to the ABWR. It applies only to Babcock & Wilcox designed reactors.

1A.2.19 Review and Modify Procedures for Removing Safety-Related Systems from Service [II.K.1(10)]

NRC Position

Review and modify (as required) procedures for removing safety-related systems from service (and restoring to service) to assure operability status is known. The COL applicant must verify the operability of safety-related systems after performing maintenance or tests as part of the test to restore a system to service.

Response

See Subsection 1A.3.2 for COL license information requirements.

1A.2.20 Describe Automatic and Manual Actions for Proper Functioning of Auxiliary Heat Removal Systems When Feedwater System Not Operable [II.K.1(22)]

NRC Position

For boiling water reactors, describe the automatic and manual actions necessary for proper functioning of the auxiliary heat removal systems that are used when the main feedwater system is not operable (Bulletin 79-08, Item 3).

Response

If the main feedwater system is not operable, a reactor scram will be automatically initiated when reactor water level falls to Level 3. The operator can then manually initiate the RCIC System from the main control room, or the system will be automatically initiated as hereinafter described. Reactor water level will continue to decrease due to boiloff until the low-low level setpoint (Level 2) is reached. At this point, the RCIC System will be automatically initiated to supply makeup water to the RPV. This system will continue automatic injection until the reactor water level reaches Level 8, at which time the RCIC steam supply valve is closed.

In the nonaccident case, the RCIC System is normally the only makeup system utilized to furnish subsequent makeup water to the RPV. When the water level reaches Level 2 again due to loss of inventory through the main steam relief valves or to the main condenser, the RCIC System automatically restarts as described in Subsection 1A.2.22. This system then maintains the coolant makeup supply. RPV pressure is regulated by the automatic or manual operation of the main turbine bypass valves which discharge to the condenser.

To remove decay heat during a planned isolation event, assuming that the main condenser is not available, the SRVs are utilized to dump the residual steam to the suppression pool. Suppression pool temperature is monitored by the Suppression Pool Temperature Monitoring (SPTM) System. When the pool temperature increases to a selected setpoint, the SPTM System signals the RHR System to cause automatic initiation of the suppression pool cooling (SPC) mode of RHR. The SPC mode cools the suppression pool by routing pool water through the RHR heat exchanger to cool it, and returning it to the suppression pool. SPC may also be affected by manual alignment of the RHR System. Makeup water to the RPV is still supplied by the RCIC System.

For the accident case with the RPV at high pressure, the HPCF Systems can also be utilized to automatically provide the required makeup flow when the water level drops below Level 1.5 setpoint. No manual operations are required. If the HPCF Systems are postulated to fail at these same conditions and the RCIC capacity is insufficient, the Automatic Depressurization System (ADS) will automatically initiate depressurization of the RPV to permit the low pressure ECCS/LPFL mode of the RHR System to provide makeup coolant.

Therefore, it can be seen that, although manual actions can be taken to mitigate the consequences of a loss of feedwater, there are no short-term manual actions which must be taken. Sufficient systems exist to automatically mitigate these consequences.

1A.2.21 Describe Uses and Types of RV Level Indication for Automatic and Manual Initiation of Safety Systems [II.K.1(23)]

NRC Position

For boiling water reactors, describe all uses and types of reactor vessel level indication for both automatic and manual initiation of safety systems. Describe other instrumentation that might give the operator the same information on plant status (Bulletin 79-08, Item 4).

Response

The water-level measurement for BWRs, in general, is fully described in NEDO-24708A, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors". An outline of this description as applicable to the ABWR Standard Plant is provided in the following paragraphs.

Figure 7.7-1 illustrates the reactor vessel elevations covered by each water-level range. Additional details may be found in Figure 5.1-3 (Nuclear Boiler System P&ID). The instruments that sense the water level are differential pressure devices calibrated to be accurate at a specific vessel pressure and liquid temperature condition. The following is a description of each water-level range.

(1) **Shutdown Water-Level Range—**This range is used to monitor the reactor water level during the shutdown condition when the reactor system is flooded for maintenance and head removal. The water-level measurement design is the condensate reference chamber leg type that is not compensated for changes in density. The vessel temperature and pressure conditions that are used for the calibration are 48.9°C and 0 kPaG water in the vessel. The two vessel instrument penetration elevations used for this water-level measurement are located at the top of the RPV head and the instrument tap just below the bottom of the dryer skirt.

(2) **Narrow Water-Level Range—**This range uses for its RPV taps the elevation above the main steam outlet nozzle and the tap at an elevation near the bottom of the dryer skirt. The instruments are calibrated to be accurate at the normal operating points. The water-level measurement design is the condensate reference chamber type, is not density compensated, and uses differential pressure devices as its primary elements. The Feedwater Control System uses this range for its water-level control and indication inputs.

(3) Wide Water-Level Range—This range uses for its RPV taps the elevation above the main steam outlet nozzle and the taps at an elevation near the top of the active fuel. The instruments are calibrated to be accurate at the normal power operating point. The water-level measurement design is the condensate reference type, is not density compensated, and uses differential pressure devices as its primary elements. These instruments provide inputs to various safety systems and engineered safeguards systems.

(4) Fuel-Zone, Water-Level Range—This range used for its RPV taps the elevation above the main steam outlet nozzle and the taps just above the reactor internal pump (RIP) deck. The zero of the instrument is the bottom of the active fuel and the instruments are calibrated to be accurate at 0 MPaG and saturated condition. The water- level measurement design is the condensate reference type, is not density compensated, and uses differential pressure devices as its primary elements. These instruments provide input to water-level indication only.

There are common condensate reference chambers for the narrow-range, wide-range and fuel-zone range water-level transmitters.

The elevation drop from RPV penetration to the drywell penetration is uniform for the narrow-range and wide-range water-level instrument lines in order to minimize the change in water-level with changes in drywell temperature.

Reactor water-level instrumentation that initiates safety systems and engineered safeguards is shown in Figure 5.1-3.

1A.2.21.1 Failure of PORV or Safety to Close [II.K.3.(3)]

NRC Position

Assure that any failure of a PORV or safety valve to close will be reported to the NRC promptly. All challenges to the PORVs safety valves should be documented in the annual report. This requirement is to be met before fuel load.

Response

See Subsection 1A.3.4 for COL license information requirements.

1A.2.22 Separation of HPCI and RCIC System Initiation Levels [II.K.3(13)]

NRC Position

Currently, the Reactor Core Isolation Cooling (RCIC) System and the High-Pressure Coolant Injection (HPCI) System both initiate on the same low-water-level signal and both isolate on the same high-water-level signal. The HPCI System will restart on low water level but the RCIC System will not. The RCIC System is a low-flow system when compared to the HPCI System. The initiation levels of the HPCI and RCIC Systems should be separated so that the RCIC System initiates at a higher water level than the HPCI System. Further, the initiation logic of the RCIC System should be modified so that the RCIC System will restart on low-water-level. These changes have the potential to reduce the number of challenges to the HPCI System and could result in less stress on the vessel from cold water injection. Analyses should be performed to evaluate these changes. The analyses should be submitted to the NRC staff and changes should be implemented if justified by the analyses.

Response

The ABWR Standard Plant design is consistent with this position. The High-Pressure Core Flooder (HPCF) System is initiated at Level one and one half , and the RCIC System is initiated at Level 2. At Level 8, the injection valves for the HPCF and the RCIC steam supply and injection valves will automatically close in order to prevent water from entering the main steamlines.

In the unlikely event that a subsequent low level recurs, the RCIC steam supply and injection valves will automatically reopen to allow continued flooding of the vessel. The HPCF injection valves will also automatically reopen unless the operator previously closed them manually. Refer to Subsections 7.3.1.1.1.1 (HPCF) and 7.3.1.1.1.3 (RCIC) for additional details regarding system initiation and isolation logic.

1A.2.23 Modify Break-Detection Logic to Prevent Spurious Isolation of HPCI and RCIC Systems [II.K.3(15)]

NRC Position

The High-Pressure Coolant Injection (HPCI)and Reactor Core Isolation Cooling (RCIC) Systems use differential pressure sensors on elbow taps in the steamlines to their turbine drives to detect and isolate pipe breaks in the systems. The pipe-break-detection circuitry has resulted in spurious isolation of the HPCI and RCIC Systems due to the pressure spike which accompanies startup of the systems. The pipe-break-detection circuitry should be modified to that pressure spikes resulting from HPCI and RCIC System initiation will not cause inadvertent system isolation.

Response

The ABWR design utilizes the motor-driven HPCF System rather than the turbine-driven HPCI System for high pressure inventory maintenance. Therefore, this position is only applicable to the turbine-driven RCIC System.

In the ABWR Standard Plant design, the high differential pressure signals which isolate the RCIC turbine are processed through the Leak Detection and Isolation System (LDS). Spurious trips are avoided because the RCIC has a bypass start system controlled by valves F037 and F045 (Figure 5.4-8, RCIC P&ID). On receipt of RCIC start signals, bypass valve F045 opens to pressurize the line downstream and accelerate the turbine. The bypass line via F045 is small (25A) and naturally limits the initial flow surge such that a differential pressure spike in the upstream pipe will not occur. After a predetermined delay (approximately 5–10 seconds), steam supply valve F037 opens to admit full steam flow to the turbine. At this stage, the line downstream is already pressurized. Thus, it is highly unlikely that a differential pressure spike could occur during any phase of the normal startup process. See Subsection 1A.3.8 for COL license information requirements.

1A.2.24 Reduction of Challenges and Failures of Relief Valves—Feasibility Study and System Modification [II.K.3(16)]

NRC Position

The record of relief-valve failures to close for all boiling water reactors (BWRs) in the past 3 years of plant operation is appproximately 30 in 73 reactor-years (0.41 failures per reactor-year). This has demonstrated that the failure of a relief valve to close would be the most likely cause of a small-break loss-of-coolant accident (LOCA). The high failure rate is the result of a high relief-valve challenge rate and a relatively high failure rate per challenge (0.16 failures per challenge). Typically, five valves are challenged in each event. This results in an equivalent failure rate per challenge of 0.03. The challenge and failure rates can be reduced in the following ways:

(1) Additional anticipatory scram on loss of feedwater

(2) Revised relief-valve actuation setpoints

(3) Increased emergency core cooling (ECC) flow

(4) Lower operating pressures

(5) Earlier initiation of ECC systems

(6) Heat removal through emergency condensers

(7) Offset valve setpoints to open fewer valves per challenge

(8) Installation of additional relief valves with a block or isolation-valve feature to eliminate opening of the safety / relief valves (SRVs), consistent with the ASME Code

(9) Increasing the high steamline flow setpoint for main steamline isolation valve (MSIV) closure

(10) Lowering the pressure setpoint for MSIV Closure

(11) Reducing the testing frequency of the MSIVs

(12) More stringent valve leakage criteria

(13) Early removal of leaking valves

An investigation of the feasibility and constraints of reducing challenges to the relief valves by use of the aforementioned methods should be conducted. Other methods should also be included in the feasibility study. Those changes which are shown to reduce relief-valve challenges without compromising the performance of the relief valves or other systems should be implemented. Challenges to the relief valves should be reduced substantially (by an order of magnitude).

Response

GE and the BWR Owners' Group responded to this requirement in Reference 1A-6. This response, which was based on a review of existing operating information on the challenge rate of relief valves, concluded that the BWR/6 product line had already achieved the "order of magnitude" level of reduction in SRV challenge rate. The ABWR relief valve system also has similar design features which also reduce the SRV challenge rate. With regard to inadvertently opened relief valves (IORV), the BWR/6 plant design evaluated for the Owners' Group report reflected a reduced level if IORV compared with the previous design because of elimination of the pilot-operated relief valve type of design. The ABWR design has also eliminated the pilot-operated relief valve type of design.

For the ABWR, which has a solid-state logic design with redundancy, the likelihood of an IORV is the same or less than the BWR/6 design evaluated in connection with the Owners' Group report. The redundant-solid state design has been selected in order that the frequency of IORV with solid state-logic becomes low enough so as to achieve the order of magnitude reduction in total SRV challenge rate required by NUREG-0737.

The redundant solid-state design for SRV operation in the pressure relief mode consists of two duplicated microprocessor channels. Each microprocessor channel activates a separate load driver and both load drivers must be activated to cause operation of the SRVs in the relief mode. Operation of the SRVs in the ADS mode also requires activation of two microprocessor channels with separate load drivers to prevent unwanted SRV operation; however, two separate dual channel systems are used to assure reliable operation in the ADS mode. Reliable operation in the pressure relief mode is assured by direct opening of the SRV against spring force.

1A.2.25 Report on Outages of Emergency Core Cooling Systems Licensee Report and Proposed Technical Specification Changes [II.K.3(17)]

NRC Position

Several components of the Emergency Core Cooling (ECC) Systems are permitted by technical specifications to have substantial outage times (e.g., 72 hours for one diesel-generator; 14 days for the HPCI System). In addition, there are no cumulative outage time limitations for ECC Systems. Licensees should submit a report detailing outage dates and lengths of outages for all ECC Systems for the last 5 years of operation. The report should also include the causes of the outages (i.e., controller failure, spurious isolation).

Clarification

The present technical specifications contain limits on allowable outage times for ECC Systems and components. However, there are no cumulative outage time limitations on these same systems. It is possible that ECC equipment could meet present technical specification requirements but have a high unavailability because of frequent outages within the allowable technical specifications.

The licensees should submit a report detailing outage dates and length of outages for all ECC Systems for the last 5 years of operation, including causes of the outages. This report will provide the staff with a quantification of historical unreliability due to test and maintenance outages, which will be requirements in the technical specifications.

Based on the above guidance and clarification, a detailed report should be submitted. The report should contain:

(1) Outage dates and duration of outages

(2) Causes of the outage

(3) ECC Systems or components involved in the outage

(4) Corrective action taken

Tests and maintenance outages should be included in the above listings which are to cover the last 5 years of operation. The licensee should propose changes to improve the availability of ECC equipment, if needed.

Applicants for an operating license shall establish a plan to meet these requirements.

Response

See Subsection 1A.3.5 for COL license information requirements.

1A.2.26 Modification of Automatic Depressurization System Logic—Feasibility for Increased Diversity for Some Event Sequences [II.K.3(18)]

NRC Position

The Automatic Depressurization System (ADS) actuation logic should be modified to eliminate the need for manual actuation to assure adequate core cooling. A feasibility and risk assessment study is required to determine the optimum approach. One possible scheme that should be considered is ADS actuation on low reactor-vessel water level, provided no HPCI or HPCS flow exists and a low-pressure ECC System is running. This logic would complement, not replace, the existing ADS actuation logic.

Response

An 8 minute high drywell pressure bypass timer has been added to the ADS initiation logic to address TMI action Item II.K.3.18. This timer will initiate on a Low Water Level 1 signal. When it times out, it bypasses the need for a high drywell signal to initiate the standard ADS initiation logic.

For all LOCAs inside the containment, a high drywell signal will be present and ADS will actuate 29 seconds after a Low Water Level 1 signal is reached. All LOCAs outside the containment become rapidly isolated and any one of the three high pressure ECCS can control the water level. The high drywell pressure bypass timer in the ADS initiation logic will only affect the LOCA response if all high pressure ECCS fail following a break outside the containment. For this case, the ADS will automatically initiate within 509 seconds (8 minute timer plus 29 second standard ADS logic delay) following a Low Water Level 1 signal.

1A.2.27 Restart of Core Spray and LPCI Systems on Low Level Design and Modification [II.K.3(21)]

NRC Position

The Core Spray and Low Pressure Coolant Injection (LPCI) Systems flow may be stopped by the operator. These systems will not restart automatically on loss of water level if an initiation signal is still present. The Core Spray and LPCI system logic should be modified so that these systems will restart, if required, to assure adequate core cooling. Because this design modification affects several core-cooling modes under accident conditions, a preliminary design should be submitted for staff review and approval prior to making the actual modification.

Response

The ABWR Standard Plant Emergency Core Cooling System (ECCS) is made up of the High Pressure Core Flooder (HPCF) System, the Reactor Core Isolation Cooling (RCIC) System, the Automatic Depressurization System (ADS) and the low pressure flooder (LPFL) mode of the Residual Heat Removal (RHR) System.

A high water level (Level 8) signal will automatically close the HPCF injection valves and the RCIC steam supply and injection valves. These systems may also be shut down manually. Manual action is required to shut down the RHR once it is initiated.

In the unlikely event that a subsequent low level reoccurs, the RCIC steam supply and injection valves will automatically reopen to allow continued flooding of the vessel. The HPCF injection valves will also automatically reopen unless the operator previously closed them manually.

The NRC has suggested certain modifications to the LPCI (LPFL for the ABWR) and core flooder systems provided as part of the ECCS network. These NRC suggestions center on control system logic modifications that would provide automatic restart capability following manual termination of system operation.

General Electric and the BWR Owners' Group reviewed this issue on a generic basis in 1980, and concluded that the NRC suggestions were not required for plant safety considerations. Justification is provided in the December 29, 1980, BWR Owners' Group submittal to the NRC (Reference 1A-8). Plant variations between the BWR and the ABWR designs are not significant to the overall technical conclusions of the study.

This conclusion is based on the adequacy of the current ECCS logic design coupled with the potentially negative impact on overall safety of the proposed changes. For the low pressure ECCS, these negative impacts include a significant escalation of control system complexity and restricted operator flexibility when dealing with anticipated events.

A full understanding of the significance of these logic changes must be based on a recognition that these systems must consider the possible interactive effects among the other systems making up the overall ECCS network. This must also include the potential impact on supporting systems such as the standby power supplies and the shutdown service water systems. Furthermore, the LPFL is one mode of the RHR System which has other safety-related functions such as suppression pool cooling and wetwell / drywell spray cooling. Clearly, these other safety functions must not be compromised by any changes in the LPFL mode of operation.

The referenced systems analysis took into consideration these potential interactive effects, impacts on supporting systems, plant instrumentation and emergency procedure guidelines. The study concluded that auto-restart of these systems is not necessary or appropriate. Therefore, GE concludes that no modifications should be made with respect to automatic restart of the low pressure ECCS.

1A.2.28 Automatic Switchover of Reactor Core Isolation Cooling System Suction—Verify Procedures and Modify Design [II.K.3(22)]

NRC Position

The Reactor Core Isolation Cooling (RCIC) System takes suction from the condensate storage tank (CST) with manual switchover to the suppression pool when the CST level is low. This switchover should be made automatically. Until the automatic switchover is implemented, licensees should verify that clear and cogent procedures exist for the manual switchover of the RCIC System suction from the condensate storage tank to the suppression pool.

Response

The RCIC System provided in the ABWR Standard Plant includes an automatic switchover feature which will change the pump suction source from the condensate storage pool to the suppression pool. The safety-grade switchover will automatically occur upon receipt of a low-level signal from the condensate storage tank or a high-level signal from the suppression pool.

See Subsection 7.3.1.1.1.3 for additional information.

1A.2.29 Confirm Adequacy of Space Cooling for High Pressure Coolant Injection and Reactor Core Isolation Cooling Systems [II.K.3(24)]

NRC Position

Long-term operation of the Reactor Core Isolation Cooling (RCIC) and High-Pressure Coolant Injection (HPCI) Systems may require space cooling to maintain the pump-room temperatures within allowable limits. Licensees should verify the acceptability of the consequences of a complete loss of alternating-current power. The RCIC and HPCI Systems should be designed to withstand a complete loss of offsite alternating-current power to their support systems, including coolers, for at least 2 hours.

Response

The ABWR HPCF and the RCIC systems are provided space cooling via individual room safety grade air-conditioning systems (Subsection 9.4.5). If all offsite power is lost, space cooling for the HPCF and RCIC System equipment would not be lost because the motor power supply for each system is from separate essential power supplies.

1A.2.30 Effect of Loss of Alternating Current Power on Pump Seals [II.K.3(25)]

NRC Position

The licensees should determine, on a plant-specific basis, by analysis or experiment, the consequences of a loss of cooling water to the reactor recirculation pump seal coolers. The pump seals should be designed to withstand a complete loss of alternating-current (AC) power for at least 2 hours. Adequacy of the seal design should be demonstrated.

Response

The ABWR design features internal recirculation pumps called Reactor Internal Pumps (RIP) which do not require shaft seals. During a Loss of Preferred Power (LOPP), the RIPs shutdown automatically but there are no shaft seals which require cooling water restoration.

A plant AC power failure would temporarily disrupt the operation of the reactor recirculation subsystems but their failure would not generate a LOCA, as the following describes.

(1) **RMC Failure—**Subsection 5.4.1.3.1 describes the RMC operation during normal running or stopped condition. This operation assumes that the Reactor Building Cooling Water (RCW) System is in operation continuously during these operating or stopped conditions. Normal LOCA and LOPP operation of the RCW are described in Subsection 9.2.11.2.

A loss of AC power or Loss of preferred power (LOPP) will stop all RIPs. The LOPP will also temporarily stop the RCW and RSW pumps. The onsite emergency power sources will automatically restart the RCW/RSW pumps, which will restore cooling for the stopped RIP motors. The RCW primary containment isolation valves will not close on LOPP (only on LOCA).

The RMC Subsystem for each RIP includes a motor cooling inlet and outlet temperature detectors TE 301 and 302, which will automatically run back individual RIPs to minimum speed and subsequently trip on high coolant temperature and prevent motor damage.

The RIP motors are designed and will be plant tested to not be damaged in the stopped hot standby condition indefinitely with RCW cooling available.

(2) RMP Failure—Subsection 5.4.1.3.2 describes the RMP operation. Since the RIP and motor have no seals, the water in the RIP motor communicates directly with the reactor water at the same pressure but at much lower temperature. There is no possibility of this water escaping from the coolant pressure boundary such as in conventional pumps, which include seals.

The RMP water is supplied from the CRD System. The CRD pumps will stop temporarily during a LOPP, which will cause the normal RMP flow to each RIP to temporarily stop. The CRD pumps are subsequently restarted automatically, after several minutes time delay, powered by onsite power sources and the RMP water will restart.

This temporary interruption of RMP flow will not initiate a LOCA. The only effect of loosing the RMP flow temporarily to the RIPs, from a LOPP, is that it will allow reactor contamination, by diffusion, to enter the RIP motor during the RMP flow interruption.

(3) RMISS Failure—Subsection 5.4.1.3.3 describes the operation of the RMISS, which is used only during plant shutdown and RIP maintenance. The power source for the inflatable seal is a portable air-operated water pump which is moved from RIP to RIP. A LOPP would therefore not cause a direct loss of RMISS pressure, since (1) the plant air system has a finite passive storage capacity in the air receivers, and (2) the RMISS air-operated pump only operates when the RMISS pressure drops below a preset value.

The RMISS is a secondary seal. Even with a long time RMISS failure RIP maintenance, the passive backseat seal of the RIP shaft on the stretch tube will preclude draining the reactor.

1A.2.31 Study and Verify Qualification of Accumulators on Automatic Depressurization System Valves [II.K.3(28)]

NRC Position

Safety analysis reports claim that air or nitrogen accumulators for the Automatic Depressurization System (ADS) valves are provided with sufficient capacity to cycle the valves open five times at normal drywell pressures. GE has also stated that the Emergency Core Cooling (ECC) Systems are designed to withstand a hostile environment and still perform their function for 100 days following an accident. The licensee should verify that the accumulators on the ADS valves meet these requirements, even considering normal leakage. If this cannot be demonstrated, the licensee must show that the accumulator design is still acceptable.

Response

The accumulators for the ADS valves are sized to provide one actuation at drywell design pressure and five actuations at normal drywell pressure. This cyclic capability is validated during preoperational testing at the station. The accumulators are safety-grade components.

The 100-day, post-accident functional operability requirement is met through conservative design and redundancy; eight ADS valves are provided with code-qualified accumulators and Seismic Category I piping within primary containment. Two redundant seven-day supplies of bottled air are available to compensate for leakage during long-term usage, with replacement capability being provided for the remainder of the postulated accident to assure system functional operability. A maximum of three of eight ADS valves need to function to meet short-term demands (Subsection 19.3.1.3.1) and the functional operability of only one ADS valve will fulfill longer term needs. Loss of pneumatic supply pressure to the ADS SRV accumulator is alarmed to provide the reactor operator with indication of the failure of any of the redundant systems under hostle environmental conditions.

The BWR Owners' Group sponsored an evaluation of the adequacy of the ADS configurations. Evaluation results are summarized in the following paragraph.

The accumulators are designed to provide one actuation at drywell design pressure and five actuations at normal drywell pressure. See Subsection 6.7.1 for a description of the ADS N_2 pneumatic supply.

1A.2.32 Revised Small-Break Loss-of-Coolant-Accident Methods to Show Compliance with 10CFR50, Appendix K [II.K.3(30)]

NRC Position

The analysis methods used by Nuclear Steam Supply System (NSSS) vendors and/or fuel suppliers for small break loss-of-coolant accident (LOCA) analysis for compliance with Appendix K to 10CFR50 should be revised, documented, and submitted for NRC approval. The revisions should account for comparisons with experimental data, including data from the LOFT Test and Semiscale Test facilities.

Response

GE has evaluated the NRC request requiring that the BWR small-break LOCA analysis methods are to be demonstrated to be in compliance with Appendix K to 10CFR50 or that they be brought into compliance by analysis methods changes. The specific NRC concerns are contained in NUREG-0626, Appendix F. The specific NRC concerns identified in Subsection 4.2.10 of NUREG-0626 (Appendix F) relate to the following: counter current flow limiting (CCFL) effects, core bypass modeling, pressure variation in the reactor pressure vessel, integral experimental verification, quantification of uncertainties in predictions, the recirculation line inventory modeling, and the homogeneous/equilibrium model.

The response to the NRC small-break model concerns was provided at a meeting between the NRC and GE on June 18, 1981. Information provided at this meeting showed that, based on the TLTA small-break test results and sensitivity studies, the existing GE small-break LOCA model already satisfies the concerns of NUREG-0626 and is in compliance with 10CFR50, Appendix K. Therefore, the GE model is acceptable relative to the concerns of Item II.K.3(30), and no model changes need be made to satisfy this item.

Documentation of the information provided at the June 18, 1981 meeting was provided via the letter from R. H. Buchholz (GE) to D. G. Eisenhut (NRC), dated June 26, 1981.

1A.2.33 Plant-Specific Calculations to Show Compliance with 10CFR50.46 [II.K.3(31)]

NRC Position

Plant-specific calculations using NRC-approved models for small break loss-of-coolant accidents (LOCAs) as described in Item II.K.3.30 to show compliance with 10CFR50.46 should be submitted for NRC approval by all licensees.

Response

The ABWR standard safety small-break LOCA calculations are discussed in Subsection 6.3.3.7.

The references listed in Subsection 6.3.6 describe the currently approved Appendix K methodology used to perform these calculations. Compliance with 10CFR50.46 has previously been established for that methodology.

Since, as noted in the previous Item (1A.2.32), no model changes are needed to satisfy NUREG-0737, Item II.K.3(30), there is no need to revise the calculations presented in Subsection 6.3.3.7.

1A.2.33.1 Evaluation of Anticipated Transients with Single Failure to Verify No Fuel Failure [II.K.3 (44)]

NRC Position

For anticipated transients combined with the worst single failure and assuming proper operator actions, licensees should demonstrate that the core remains covered or provide analysis to show that no significant fuel damage results from core uncovery. Transients which occur in a stuck-open relief valve should be included in this category. The results of the evaluation are due January 1, 1981.

Response

GE and the BWROG have concluded, based on a representative BWR/6 plant study, that all anticipated transients in Regulatory Guide 1.70, Revision 3, combined with the worst single failure, the reactor core remains covered with water until stable conditions are achieved. Furthermore, even with more degraded conditions involving a stuck-open relief valve in addition to the worst transient (loss of feedwater) and worst single failure (of high pressure core spray), studies show that the core remains covered and adequate core cooling is available during the whole course of the transient (NEDO-24708, March 31, 1980). The conclusion is applicable to the ABWR. Since the ABWR has more high pressure makeup systems (2HPCFs and 1 RCIC), the core covering is further assured.

Other discussions of transients with single failure is presented in the response to NRC Question 440.111.

1A.2.33.2 Evaluate Depressurization Other Than Full ADS [II.K.3 (45)]

NRC Position

Provide an evaluation of depressurization methods, other than by full actuation of the Automatic Depressurization System, that would reduce the possibility of exceeding vessel integrity limits during rapid cooldown (Applicable to BWRs only).

Response

This response is provided in Subsection 19A.2.11.

1A.2.33.3 Responding to Michelson Concerns [II.K.3 (46)]

NRC Position

General Electric should provide a response to the Michelson concerns as they relate to boiling water reactors.

Clarification: General Electric provided a response to the Michelson concerns as they relate to boiling water reactors by letter dated February 21, 1980. Licensees and applicants should assess applicability and adequacy of this response to their plants.

Response

All of the generic February 21, 1980 GE responses were reviewed and updated for the ABWR Standard Plant. The specific responses are provided in Table 1A-1.

1A.2.34 Primary Coolant Sources Outside Containment Structure [III.D.1.1(1)]

NRC Position

Applicants shall implement a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as-practical levels. This program shall include the following:

(1) Immediate Leak Reduction

   (a) Implement all practical leak reduction measures for all systems that could carry radioactive fluid outside of containment

   (b) Measure actual leakage rates with systems in operation and report them to the NRC

(2) Continuing Leak Reduction—establish and implement a program of preventive maintenance to reduce leakage to as-low-as-practical levels. This program shall include periodic integrated leak tests at intervals not to exceed each refueling cycle.

Response

Leak reduction measures of the ABWR Standard Plant include a number of barriers to containment leakage in the closed systems outside the containment. These closed systems include:

(1) Residual Heat Removal

(2) High Pressure Core Flooder

(3) Low Pressure Core Flooder

(4) Reactor Core Isolation Cooling

(5) Suppression Pool Cleanup

(6) Reactor Water Cleanup

(7) Fuel Pool Cooling and Cleanup

(8) Post-Accident Sampling

(9) Process Sampling

(10) Containment Atmospheric Monitoring

(11) Fission Product Monitor (Part of LDS)

(12) Hydrogen Recombiner

(13) Standby Gas Treatment

Plant procedures will prescribe the method of leak testing these systems. The testing will be performed on a schedule appropriate to 10CFR50 Appendix J type B and C penetrations (i.e., at each refueling outage). When leakage paths are discovered, including during these tests, they will be investigated and necessary maintenance will be performed to reduce leakage to its lowest practical level.

In addition, lines which penetrate the primary containment are equipped with inboard and outboard isolation valves that are designed in accordance with GDC 55, 56 and 57 to provide reliable isolation in the event of a line break or leakage. The containment isolation provisions are discussed in detail in Subsection 6.2.4, which also identifies all the system lines that penetrate the containment, together with their respective isolation valves.

Leakage within and outside the primary containment are continuously monitored by the Leak Detection and Isolation System (LDS) for breach in the integrity of the containment. Upon detection of a leakage parameter, the LDS will automatically initiate the necessary control functions to isolate the source of the break and alerts the operator for corrective action. The MSL tunnel area is monitored for high radiation levels and for high ambient temperatures that are indicative of steam leakage. The Turbine Building is also monitored for high area ambient temperatures for MSL leakage. The resulting action causes isolation of the MSIVs and subsequent shutdown of the reactor.

The radiation levels in the HVAC air exhaust ducts of the Reactor Building and the fuel handling area are monitored for use in isolating the secondary containment. The results in closure of the HVAC air ducts in the Reactor Building, closure of the containment purge and vent lines, and activation of the Standby Gas Treatment System (SGTS).

The leak detection methods and associated instrumentation that monitor leakage from the reactor coolant pressure boundary are described in Subsection 5.2.5.

For small line breaks in the secondary containment that could cause significant release of radioactive material will be detected by process radiation monitors in the Reactor Building HVAC air ducts. As indicated above, a high radiation level will activate the SGTS (Subsection 6.5.1) prior to the release of radiation to the environment. Also, any fluid leakage will drain into the sumps in the Reactor Building and will be monitored by sumps instrumentation for level and flow rate. The operator will be alerted to any abnormal condition for action. All lines which pass outside of the secondary containment contain leakage detection systems or loop seals. These systems allow the SGTS to maintain a negative pressure relative to the environment and thus limit the amount of leakage through the secondary containment. These systems are discussed in Subsection 6.2.3. Finally, expected liquid leakoff from equipment outside the containment is directed to equipment drain sumps and processed by the Radwaste

System. These multiple design features of the ABWR Standard Plant provide substantial capability to limit any potential release to the environment from systems likely to contain radioactive material.

Additionally, pressure boundary components of radioactive waste systems are purchased as augmented Class D systems to assure their capability to provide integrity.

1A.2.35 In-Plant Radiation Monitoring [II.D.3.3(3)]

NRC Position

(1) Each licensee shall provide equipment and associated training and procedures for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during an accident.

(2) Each applicant for a fuel-loading license to be issued prior to January 1, 1981 shall provide the equipment, training, and procedures necessary to accurately determine the presence of airborne radioiodine in areas within the plant where plant personnel may be present during an accident.

Response

(1) See Subsection 1A.3.3 for COL license information requirements.

(2) Not applicable.

1A.2.36 Control Room Habitability [III.D.3.4(1)]

NRC Position

In accordance with Task Action Plan Item III.D.3.4 and control room habitability, licensees shall assure that control room operators will be adequately protected against the effects of accidental release of toxic and radioactive gases and that the nuclear power plant can be safely operated or shut down under design basis accident conditions (Criterion 19, “Control Room” of Appendix A, General Design Criteria for Nuclear Power Plants, to 10CFR50).

Response

This requirement is satisfied for the ABWR by the provisions of the Control Building Atmospheric Control System(ACS).

Section 7.1 describes the Control Building ACS instrumentation and controls for assuring control room habitability. Subsection 6.4 provides HVAC design details. As demonstrated by the analyses provided in these subsections, these systems satisfy Criterion 19, Appendix A of 10CFR50.

The ABWR design satisfies this item.

1A.3 COL License Information

1A.3.1 Emergency Procedures and Emergency Procedures Training Program

Emergency procedures, developed from the emergency procedures guidelines, shall be provided and implemented prior to fuel loading (Subsection 1A.2.1).

1A.3.2 Review and Modify Procedures for Removing Safety-Related Systems from Service

Procedures shall be reviewed and modified (as required) for removing safety-related systems from service (and restoring to service) to assure operability status is known (Subsections 1A.2.18 and 1A.2.19).

1A.3.3 In-Plant Radiation Monitoring

Equipment and training procedures shall be provided for accurately determining the airborne iodine concentration in areas within the facility where plant personnel may be present during the accident (Subsection 1A.2.35).

1A.3.4 Reporting Failures of Reactor System Relief Valves

Failures of reactor system relief valves shall be reported in the annual report to the NRC (Subsection 1A.2.21.1).

1A.3.5 Report on ECCS Outages

Starting from the date of commercial operations, an annual report should be submitted which includes instance of ECCS unavailability because of component failure, maintenance outage (both forced or planned), or testing, the following information shall be collected:

(1) Outage date

(2) Duration of outage

(3) Cause of outage

(4) Emergency core cooling system or component involved

(5) Corrective action taken

The above information shall be assembled into a report, which will also include a discussion of any changes, proposed or implemented, deemed appropriate, to improve the availability of the emergency core cooling equipment (Subsection 1A.2.25).

1A.3.6 Procedure for Reactor Venting

Procedures shall be developed for the operators use of the reactor vents. (See Subsection 1A.2.5)

1A.3.7 Testing of SRV and Discharge Piping

The COL applicant will confirm that any SRVs or discharge piping installed that is not similar to those that have been tested will be tested in accordance with Subsection 1A2.9.

1A.3.8 RCIC Bypass Start System Test

The COL applicant shall perform the RCIC bypass start system test described in Subsection 1A.2.23 during plant startup.

1A.4 References

1A-1 Memo, C. Michelson to D. Okrent, “Possible Incorrect Operator Action Such as Pipe Break Isolation”, June 4,1979.

1A-2 Letter, D. G. Eisenhut (NRC)to R. L. Gridley(GE), “Potential for Break Isolation and Resulting GE-Recommended BWR/3 ECCS Modifications”, June 14, 1978.

1A-3 “Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors”, NEDO-24708, August 1979.

1A-4 U. S. Nuclear Regulatory Commission, “NRC Action Plan Developed as a Result of the TMI-2 Accident”, USNRC report NUREG-0660, Vols. 1 and 2, May 1980.

1A-5 U. S. Nuclear Regulatory Commission, “Clarification of TMI Action Plan Requirements”, USNRC Report NUREG-0737, November 1980.

1A-6 U. S. Nuclear Regulatory Commission, “Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License”, NUREG-0718, Revision 1, June 1981.

1A-7 Letter, R. H. Buchholz (GE) to D. F. Ross (NRC), Subject: Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors, November 30, 1979, MFN-290-79.

1A-8 Letter, D. B. Waters (Chairman, BWR Owners’ Group) to NRC staff, dated December 29, 1980, “BWR Owners’ Group Evaluation of NUREG-0737 Requirements.”

1A-9 Letter, D. B. Waters (Chairman, BWR Owners' Group) to D. G. Eisenhut (NRC), dated March 31, 1981, "BWR Owners' Group Evaluation of NUREG-0737 Requirements II.K.3.16 and II.K.3.18."

1A-10 Letter, D.B.Waters (Chairman, BWR Owners' Group) to R.H. Vollmer (NRC), dated September 17, 1980, NUREG-0578 "Requirements 2.12-Performance Testing of BWR and PWR Relief and Safety Valves."

1A-11 NEDE-24988-P, "Analysis of Generic BWR Safety / Relief Valve Operability Test Results", Proprietary Document, October 1981.

Table 1A-1 Responses to Questions Posed by Mr. C. Michelson [II.K.3(46)]

Question 1

Pressurizer level is an incorrect measure of primary coolant inventory.

Response 1

BWRs do not have pressurizers. BWRs measure primary coolant inventory directly using differential pressure sensors attached to the reactor vessel. Thus, this concern does not apply to the ABWR.

Question 2

The isolation of small breaks (e.g., letdown line; PORV) not addressed or analyzed.

Response 2

Automatic isolation only occurs for breaks outside the containment. Such breaks are addressed in Section 3.1.1.1.2 of NEDO-24708. It was shown that if high pressure systems are available, no operator actions are required. If it is assumed that all high pressure systems fail, the operator must manually depressurize to allow the low pressure systems to inject and maintain vessel water level. Analyses in Section 3.5.2.1 of NEDO-24708 show that the operator has sufficient information and time to perform these manual actions. The necessary manual actions have been included in the operator guidelines for small-break accidents.

Question 3

Pressure boundary damage due to loadings from (1) bubble collapse in subcooled liquid and 2) injection of ECC water in steam-filled pipes.

Response

The BWR has no geometry equivalent to that identified in Michelson's report on B&W reactors relative to bubble collapse (steam bubbling upward through the pressurizer surge line and pressurizer). Thus, the first concern is not applicable to the ABWR.

ECC injection in the ABWR at high pressure is either into the reactor vessel throuh water-filled lines (RHR-B+C;HPCF-B+C) or into the feedwater lines (RHR-A:RCIC). The feedwater lines are normally filled with relatively cold liquid (251.5°C or less). ECCS injection in the ABWR at low pressure is either directly into the reactor vessel or into the feedwater lines. Thus, the second concern is not applicable to the ABWR.

Question 4

In determining need for steam generators to remove decay heat, consider that break flow enthalpy is not core exit enthalpy.

Response 4

BWRs do not use steam generators to remove decay heat, so this concern does not apply to the ABWR.

Question 5

Are sources of auxiliary feedwater adequate in the event of a delay in cooldown subsequent to a small LOCA?

Table 1A-1 Responses to Questions Posed by Mr. C. Michelson [II.K.3(46)] (Continued)

Response 5

BWRs do not need feedwater to remove heat from the reactor following a LOCA, whether the subsequent cooldown is delayed or not. Therefore, this concern is not applicable to the ABWR. BWRs have a closed cooling system in which vessel water flows out the postulated break to the suppression pool. The suppression pool is cooled and water is pumped back to the vessel with ECCS pumps.

Question 6

Is the recirculation mode of operation of the HPCI pumps at high pressure an established design requirement?

Response 6

The high-pressure injection systems utilized in the ABWR are the Reactor Core Isolation Cooling (RCIC) and High Pressure Core Flooder (HPCF) Systems.

The RCIC and HPCF Systems normally take suction from the condensate storage tank and have an alternate suction source from the suppression pool. A recirculation mode of operation of these systems is established when the system suction is from the suppression pool. Following a LOCA when system suction is from the suppression pool, water injected into the reactor is discharged through the break and flows back to the suppression pool, forming a closed recirculation loop.

Other recirculation modes include test modes (e.g., suction from and discharge to the suppression pool) and system operation on low flow bypass with discharge to the suppression pool.

All of these modes are established design requirements.

Question 7

Are the HPCI pumps and RHR pumps run simultaneously? Do they share common piping?/suction? If so, is the system properly designed to accommodate this mode of operation (i.e., are any NPSH requirements violated, etc...)?

Response 7

For the ABWR, the high-pressure injection systems (RCIC/HPCF) do not share any common suction piping with the low pressure RHR and they can operate simultaneously with this low pressure system.

The RCIC and HPCF Systems share common suction line from the condensate storage tank. The RHR shutdown cooling operating mode does not share any common suction piping with the RCIC or HPCF Systems. It is an established design requirement to size the suction piping, including shared piping, such that adequate NPSH is available to RCIC and HPCF pumps for all operating modes of these systems.

Pre-operational and/or startup tests are conducted that demonstrate that the NPSH requirements are met.

Question 8

Mechanical effects of slug flow on steam generator tubes need to be addressed (transitioning from solid natural circulation to reflux boiling and back to solid natural circulation may cause slug flow in the hot leg pipes).

Table 1A-1 Responses to Questions Posed by Mr. C. Michelson [II.K.3(46)] (Continued)

Response 8

BWRs do not have steam generators, so this concern does not apply to ABWR.

Question 9

Is there minimum flow protection for the HPCI pumps during the recirculation mode of operation?

Response 9

For the ABWR, the RCIC, RHR, and HPCF pumps all contain valves, piping, and automatic logic that bypasses flow to the suppression pool as required to provide minimum flow protection for all design basis operating modes of the systems.

Question 10

The effect of the accumulators dumping during small-break LOCAs is not taken into account.

Response 10

BWRs do not use accumulators to mitigate LOCAs. Therefore, this concern does not apply to the ABWR.

Question 11

What is the impact of continued running of the RC pumps during a small LOCA?

Response 11

The impact of continued running of the recirculation pumps has been addressed in Subsections 3.3.2.2, 3.3.2.3, and Subsection 3.5.2.1.5.1 of NEDO-24708. The conclusions were that continued running of the recirculation pumps results in little change in the time available for operator actions and does not significantly change the overall system response.

Question 12

During a small break LOCA in which offsite power is lost, the possibility and impact of pump seal damage and leakage has not been evaluated.

Response 12

The RCIC, HPCF, and RHR pumps are provided with mechanical seals. No external cooling from auxiliary support systems, such as site service water or room air coolers, is required for RCIC pump seals. The HPCF and RHR seals are cooled by connections to the three separate divisions of the Reactor Building Cooling Water (RCW) System to protect against single failures. RHR Divisions A, B and C, and HPCF-B and C are connected to their corresponding RCW divisions. If offsite power is lost, onsite diesel generators maintain the RCW three-divisional function. These types of seals have demonstrated (in nuclear and other applications) their capability to operate for an extended period of time at temperatures in excess of those expected following a LOCA.

Should seal failure occur, it can be detected by room sump high level alarms. The RCIC, HPCF, and RHR individual pumps are arranged, and motor-operated valves provided, so that a pump with a failed seal can be shut down and isolated without affecting the proper operation of the other redundant pumps/systems.

Table 1A-1 Responses to Questions Posed by Mr. C. Michelson [II.K.3(46)] (Continued)

Considering the low probability of seal failure during a LOCA, the fact that a pump with a failed seal can be isolated without affecting other redundant equipment, and the substantial redundancy provided in the BWR emergency cooling systems, pump seal failure is not considered a significant concern.

Question 13

During transitioning from solid natural circulation to reflux boiling and back again, the vessel level will be unknown to the operators, and emergency procedures and operator training may be inadequate. This needs to be addressed and evaluated.

Response 13

There is no similar transition in the BWR case. In addition, the BWR has water level measurement within the vessel and the indication of the water level is incorporated into the operator guidelines. Consequently, this concern does not apply to ABWR.

Question 14

The effect of non-condensable gas accumulation in the steam generators and its possible disruption of decay heat removal by natural circulation needs to be addressed.

Response 14

The effect of non-condensable gas accumulation is addressed in Subsection 3.3.1.8.2 of NEDO-24708. For a BWR, vapor is present in the core during both normal operation and natural circulation conditions. Non-condensables may change the composition of the vapor but would have an insignificant effect on the natural or forced circulation itself, since the non-condensables would rise with the steam to the top of the vessel after leaving the steam separators.

Concern 15

Delayed cooldown following a small-break LOCA could raise the containment pressure and activate the containment spray system.

Response 15

The ABWR containment spray system is manually initiated. All essential equipment inside the containment is required to be qualified for the environmental conditions resulting from the initiation of the containment spray system.

Question 16*

This concern relates to the possibility that an operator may be included and perhaps even trained to isolate, where possible, a pipe break LOCA without realizing that it might be an unsafe action leading to high pressure, and short-term core bakeout. For example, if a BWR should experience a LOCA from a pressure boundary failure somewhere between the pump suction and discharge valve for either reactor recirculation pump, it would be possible for the operator to close these valves following the reactor blowdown to repressurize the reactor coolant system. Before such isolation should be permitted, it is first necessary to show by an appropriate analysis that the high pressure ECCS is adequate to reflood the uncovered core without assistance from the low pressure ECCS, which can no longer deliver flow because of the repressurization. Otherwise, such isolation action should be explicitly forbidden in the emergency operating instructions.

Table 1A-1 Responses to Questions Posed by Mr. C. Michelson [II.K.3(46)] (Continued)

Response 16

The ABWR does not have recirculation lines. However, there are other systems where it is possible for the operator to close these valves following the reactor blowdown to low pressure and thereby isolate the break. An example of this would be a break in the reactor water cleanup piping between the shutdown suction line valve and the containment boundary. In Reference 1A-2, the NRC concluded that, based on information provided by GE, break isolation is not a problem.

In order for the reactor vessel to repressurize following isolation of a line break, the isolation would have to occur before initiation of ADS due to a high drywell pressure in concurrence with low water Level 1 condition. Isolation of a line break prior to obtaining a high drywell pressure signal might occur for very small breaks (area << 9.3 cm^2), which may require several hundred seconds following the break to reach the high drywell pressure setpoint. In this case it has been shown in Reference 1A-3 that the high pressure systems (RCIC, HPCF and feedwater) are sufficient to maintain the water level above the top of the core.

If isolation of the break were to occur prior to reaching Level 1 but after the high drywell pressure signal, the vessel would pressurize to the SRV setpoint following isolation of the main steamlines and then oscillate as the SRVs cycle opened and closed. If no high pressure systems were available, the loss of mass out the SRVs would cause the level to continue dropping and result in automatic ADS actuation shortly after reaching Level 1. This would depressurize the vessel and allow the low pressure systems to begin injecting. This capability was demonstrated in NEDO-24708, explicitly to provide for manual depressurization in the event of low reactor water level with high pressure systems unable to maintain level for any reason.

In summary, in order to repressurize the vessel following break isolation, the isolation would have to occur prior to ADS blowdown. For these cases, high pressure systems would maintain inventory. If no high pressure system was available, the low pressure systems would control the vessel water level following automatic or manual vessel depressurization.

* Excerpt from Reference 1A-1

1AA Plant Shielding to Provide Access to Vital Areas and Protective Safety Equipment for Post-Accident Operation [II.B.2]

1AA.1 Introduction

General Electric has performed a review of the ABWR Standard Plant post-accident environment in response to NUREG-0737 Item II.B.2. This attachment discusses the results of that review.

1AA.2 Summary of Shielding Design Review

Several alternatives are potentially available to the designer to assure continued equipment availability and performance under post-accident conditions. One is to provide redundant systems and / or components which are qualified to operate in the expected environment. Another is to provide operator access to conduct the operations and to maintain the equipment. This latter alternative would generally be accompanied by appropriate shielding and in many cases would be difficult if not impossible to carry out.

GE has taken the first approach and furthermore has designed the plant so that most responses to transient conditions are automatic, including achieving and maintaining safe-shutdown conditions. The design basis for the ABWR Standard Plant is to require safety-related equipment to be appropriately environmentally qualified and operable from the control room. As a result of this design philosophy and as shown by this review, no changes are necessary to assure that personnel access is adequate or that safety equipment is not degraded because of post-accident operation.

As part of the design of the ABWR Standard Plant, it was necessary to establish the environmental conditions for qualification of safety-related equipment. A result of this design work was an environmental requirement establishing the integrated dose that the equipment must be able to withstand. These values are listed in Appendix 3I.

Another aspect of the review was the manner in which the safety-related equipment is arranged and operated during normal and abnormal operation and postulated accidents. The essence of the ABWR Standard Plant is to achieve and maintain a safe shutdown condition for all postulated accident conditions with all operator actions being conducted from outside the primary and secondary containment zones, principally from the control room.

The purpose of this review is first to verify that, where equipment access is required, it is reasonably accessible outside the primary and secondary containment zones. Secondly, the review should verify that inaccessible equipment is environmentally qualified and is operable from the control room.

The results of the review are:

(1) The period of interest begins with the plant in a safe shutdown condition. Thus, the various safety-related systems needed to achieve safe shutdown conditions have performed as expected, and only the engineered safety features systems (Chapter 6) and auxiliaries, as described later, are required to maintain this condition.

(2) Based upon the accident source terms of Regulatory Guides 1.3 and 1.7 and Standard Review Plan 15.6.5 including normal operations, the vital equipment exposures will be enveloped based upon the table below:

Area	**Gamma (Gy)**	**Beta (Gy)**
Primary Containment	$2x10^6$	$2x10^7$
ECCS Rooms	$4x10^5$	$8x10^7$
SGTS Area	$5x10^5$	$2x10^{-1}$

Each actual area will be environmentally qualified to the area specific envelope as defined in Tables 3I.3-9 through 3I.3-13 and 3I.3-19 through 3I.3-20.

(3) It is not necessary for operating personnel to have access to any place other than the control room, technical support center, post-accident sampling station, sample analysis area, and safety-related nitrogen supply bottles to operate the equipment of interest during the 100-day period. The control room, technical center and sample analysis area are designed to be accessible post-accident. The latter areas are considered accessible on a controlled exposure basis. Direct shine from the containment is less than 3.87E-06 c/kgh in the control room, technical support center, and counting facility and less than 1.29E-04 c/kgh in other vital areas in the Reactor Building.

(4) Access to radwaste is not required, but the Radwaste Building is accessible, since primary containment sump discharges are isolated and secondary containment sump pump power is shed at the onset of the accident. Thus, fission products are not transported to radwaste. The combustible gas control system is operated from the control room; the ABWR does not have a containment isolation reset control area or a manual ECCS alignment area. These functions are provided in the control room.

(5) Following an accident, access is available to electrical equipment rooms containing motor control centers and corridors in the upper Reactor Building (Subsection 12.3.6). This is based on radiation shine from the ECCS rooms and primary containment. While not necessary to maintain safe shutdown, such access can be useful in extending system functionality and plant recovery.

(6) The emergency power supplies (diesel generators) are accessible. However, access is not necessary, since the equipment is environmentally qualified.

1AA.3 Containment Description and Post Accident Operations

1AA.3.1 Description of Primary/Secondary Containment

The ABWR design includes many features to assure that personnel occupancy is not unduly limited and safety equipment is not degraded by post-accident radiation fields. These features are detailed in Tier 2 and only a brief summary description and Tier 2 reference are provided here for emphasis.

The configuration of the pressure suppression primary containment with the suppression pool maximizes the scrubbing action of fission products by the suppression pool. The particulate and halogen content of the primary containment atmosphere following an accident is thereby substantially reduced compared to the Regulatory Guide 1.3 source terms.

Primary containment leakage is limited to less than one half percent of the primary containment atmosphere per day. The surrounding secondary containment is kept at a negative pressure with respect to the environment permitting monitoring and treating all radioactive leakage from the primary containment.

The Standby Gas Treatment System (SGTS) operates automatically from the beginning of an accident to control the secondary containment pressure to -6.4 mm w.g. The large volume of this portion of the Reactor Building acts as a mixing chamber to dilute any primary containment leakage before processing by the SGTS and discharge to the environment. Discharge of radioactivity is thus controlled and reduced. Radioactivity content of secondary containment atmosphere is reduced with time by SGTS treatment as well as by decay. (However, prior removal of halogens by scrubbing in the suppression pool offsets the necessity of this treatment.)

Each ECCS pump and supporting equipment is located in an individual shielded, watertight compartment. Spread of radioactivity among compartments is thus limited. Radiation to the other equipment areas and corridors of the Reactor Building is limited to shine through the walls; there is no airborne radiation in these other areas. As these become accessible after an accident, any component failures can be repaired, thereby improving systems availability.

1AA.3.2 Vital Area and Systems

A vital area is any area which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident. Areas which must be considered as vital after an accident are the control room, technical support center, sampling station, sample analysis area and the HPIN nitrogen supply bottles.

The vital areas also include consideration (in accordance with NUREG- 0737, II.B.2) of the post-LOCA hydrogen control system, containment isolation reset control area, manual ECCS alignment area, motor control center and radwaste control panels. However, the ABWR design does not require a containment isolation reset control area or a manual ECCS alignment area, as these functions are available from the control room. Those vital areas which are normally areas of mild environment, allowing unlimited access, are not reviewed for access.

Essential systems specific to the ABWR to be considered post-accident are those for the ECCS, fission product and combustible gas control and the auxiliary systems necessary for their operation (i.e., instrumentation, control and monitoring, power, cooling water, and air cooling).

1AA.3.3 Post Accident Operation

Post-accident operations are those necessary to (1) maintain the reactor in a safe shutdown condition, (2) maintain adequate core cooling, (3) assure containment integrity, and (4) control radioactive releases within 10CFR100 guidelines.

Many of the safety-related systems are required for reactor protection or to achieve a safe shutdown condition. However, they are not necessarily needed once a safe shutdown condition is achieved. Thus, the systems considered herein are the engineered safety features (ESF) (Chapter 6) used to maintain the plant in a safe shutdown condition.

For purposes of this review, the plant is assumed to remain in the safe shutdown condition.

The basis for this position is that the foundation of plant safety is the provision of sufficient redundancy of systems and logic to assure that the plant is shut down and that adequate core cooling is maintained. Necessary shutdown and post-accident operations are performed from the control room, except for the post-accident sampling station, the sample analysis area, and two manual nitrogen reserve supply valves.

1AA.4 Design Review Bases

1AA.4.1 Radioactive Source Term and Dose Rates

The radioactive source term used is equivalent to the source terms recommended in Regulatory Guides 1.3 and 1.7 and Standard Review Plan 15.6.5 with appropriate decay times. Depressurized coolant is assumed to contain no noble gas. There is no leakage outside of secondary containment other than via the SGTS.

Dose rates for areas requiring continuous occupancy may be averaged over 30 days to achieve the desired <0.15 mSv/h.

Design dose rates for personnel in a vital area are such that the guidelines of General Design Criteria (GDC) 19 (i.e., <0.05 Sv whole body or its equivalent to any part of the body) are not exceeded for the duration of the accident, based upon expected occupancy and protection.

1AA.4.2 Accidents Used as the Basis for the Specified Radioactivity Release

Table 15.0-3 summarizes the various design basis accidents and associated potential for fuel rod failure. This chapter also provides the accident parameters. Of those accidents, only the DBA-LOCA may produce 100% failed fuel rods under NRC worst-case assumptions. The rod drop accident and fuel handling accident are the only other accidents postulated as leading to failed fuel rods with the potential consequence of radioactivity releases.

For the fuel handling accident, the reactor is either shutdown and cooled or is operating normally if the accident is in the spent fuel storage pool. Based on the conditions of Regulatory Guide 1.25, it is assumed that the airborne activity of the reactor building (Table 15.7-9) is released to the environment over a 2-hour period via a 99% iodine efficient SGTS. The total activity released to the environment is presented in Table 15.7-10 and the calculated exposure in Table 15.7-11. The exposures are within the guidelines of 10CFR100. Thus, recovery is possible well within the specified 100-day equipment qualification period. ECCS equipment is not affected by this accident and radiation in the ECCS area is not increased. This accident is not considered further.

The postulated control rod drop accident (Subsection 15.4.9) is one which occurs without a pipe break and so may require depressurization to attain long-term core cooling with the RHR System. Normally, this accident is terminated by a scram, and the plant is cooled and recovers. The performance of the separation-detection devices and the rod block interlocks virtually preclude the cause of a rod drop accident. This accident is not further considered.

The DBA-LOCA is the accident producing the limiting conditions of interest for this design review. In this accident the reactor is depressurized and reactor water mixes with suppression pool water in the process of keeping the fuel covered and cooled. Fission products are assumed to be essentially instantaneously released and mixed in the containment atmospheres and suppression pool-reactor water volumes.

1AA.4.3 Availability of Offsite Power

The availability of offsite power is not influenced by plant accident conditions. Loss of offsite power may be assumed as occurring coincident with the beginning of the accident sequence; however, continued absence of offsite power for the accident duration is not realistic. While restoration of offsite power is not a necessary condition for maintaining core cooling, its availability can permit operation of other plant systems which would not otherwise be permitted by emergency power restrictions (e.g., operation of the pneumatic air system, non-safety-related HVAC systems and other systems useful to plant cleanup and recovery).

Based on Table 19D.3-3, the probability for offsite power recovery is estimated to be very high in 8 hours. This is conservative, since the longest time for restoration of offsite power was six hours for the Pennsylvania-New Jersey-Maryland interconnection. The grid used as a basis for the probabilistic risk assessment is presented in Subsection 15D.3.

1AA.4.4 Radiation Qualification Conditions

The safety-related equipment requiring review for qualification is only that necessary for post-accident operations and for providing information for assuring post-accident control.

In 10CFR50, the long-term cooling capability is given as follows: "...decay heat shall be removed for the extended period of time required by the long lived radioactivity remaining in the core." A 100-day period has been selected as a sufficient extended period permitting site and facility response to terminate the event.

As part of the design review process, a set of reference conditions is necessary for comparing expected post-accident radiation exposures. Appendix 3I defines the environmental conditions for safety-related equipment zones for periods of 60 years normal operations, including anticipated tests and abnormal events, and six months following the DBA-LOCA. These conditions are upper bound envelopes used to establish the environmental design and qualification bases of safety-related equipment. In effect, these are specification values, and equipment will be qualified to meet or exceed these values.

Radiation sources in the secondary containment (especially the ECCS rooms of the Reactor Building) are the same as the Table 1AA-1 design basis values for water sources. For airborne radiation sources, the plant design basis of Table 1AA-1 for air is used. Primary containment leakage is assumed to occur in each of the individual secondary containment compartments. This leakage is limited by the fission product control systems (Subsection 6.5.3). As previously noted, no credit has been taken for the radio-halogen scrubbing, which is an inherent feature of the BWR.

1AA.5 Results of the Review

1AA.5.1 Systems Required Post-Accident

This section establishes the various systems equipment which are required to function following an accident along with their locations. The expected habitability conditions and access and control needs are identified for the required post-accident period.

1AA.5.1.1 Necessary Post-Accident Functions and Systems

Following an accident and assuming that immediate plant recovery is not possible, the following functions* are necessary:

(1) Reactivity control

(2) Reactor core cooling

(3) Reactor coolant system integrity

(4) Primary reactor containment integrity

(5) Radioactive effluent control

Reactivity control is a short-term function and is achieved when the reactor is shutdown. The remaining functions are achieved in the longer term post-accident period by use of:

(a) The Emergency Core Cooling System (ECCS) and their auxiliaries (for reactor core cooling)

(b) The Combustible Gas Control System (CGCS) and auxiliaries (for primary containment and reactor coolant system integrity)

(c) The fission product removal and control system and auxiliaries (for radioactive effluent control)

(d) Instrumentation and controls and power for accident monitoring and functioning of the necessary systems and associated habitability systems

* ANSI/ANS 4.5 Criteria for Accident Monitoring Functions in Light Water Reactors.

Tables 1AA-2 through 1AA-5 are generated to show:

(i) What major equipment and systems are required to function and thereby define the systems for review

(ii) The redundant equipment locations by divisional isolated room or area and containment or building

1AA.5.1.2 Emergency Core Cooling Systems and Auxiliaries

Table 1AA-2 lists various systems related to cooling the fuel under post-accident conditions as described in Section 6.3 and Subsection 9.4.5.2 (HVAC). This table shows ECCS equipment and equipment coolers in an ECCS room. Instrumentation transmitters are in adjoining areas. The required power and cooling water in the same division are described in Subsection 1AA.5.1.5. All perform together to provide an ECCS function.

The Automatic Depressurization System (ADS) function is described in Subsection 1.2.2.2.2.4. A postulated non-break or small break accident could require continued need for the depressurization function until the RHR System is placed in the shutdown reactor cooling mode. In the case of a non-break or a small break accident, the majority of the fission products would be released via the safety/relief valves to the suppression pool and hence to the containment, rather than direct mixing through the supersession pool vents, as would occur following a DBA-LOCA. In either case, the distribution of fission products is assumed to be the same as for the DBA-LOCA even though, realistically, a significant portion of halogens and solid fission products would be retained in the reactor pressure vessel. Thus, the results as they apply to the ADS are very conservative. The pneumatic nitrogen supply for the ADS and other containment valves is included in Table 1AA-3 as a portion of the combustible gas control. The hand-operated nitrogen reserve supply valves P54-F017C and D are accessible outside the secondary containment, if needed, to mitigate a large leak.

The high pressure core flooder (HPCF) and the low pressure flooder loop (LPFL) functions are described in Subsections 1.2.2.5.2 and 1.2.2.5.1.1, respectively. The cooling function can also satisfy the containment cooling function in that, by cooling suppression pool water, which is the source of water flowing to the reactor, the containment source of heat is also removed. The wetwell/drywell sprays are described in Subsection 1.2.2.5.1.3.

The fuel pool cooling function (Subsection 1.2.2.7.2) is also included on the basis that a recently unloaded fuel batch could require continued cooling during the post-accident period. The equipment is environmentally qualified, so access is not required and redundancy is included in system components.

The locations of selected associated valves and instrument transmitters are included. These do not represent all of this type of equipment which is environmentally qualified, safety-related, or included in the systems of Table 3.2-1. It does however, represent principal components which are needed to operate, generally during post accident operations. For example, most ECCS valves are normally open, and only a pump discharge valve needs to open to direct water to the reactor. Similarly, the instrument transmitters shown are those which would provide information on long-term system performance post-accident. Control room instrumentation is not listed, since it is all in an accessible area where no irradiation degradation would be expected. Passive elements such as thermocouples and flow sensors are not listed although they are environmentally qualified. The components listed under main steam (B21) are those for ECCS function or monitoring reactor vessel level. Suppression pool level is included with the HPCF instrumentation.

1AA.5.1.3 Combustible Gas Control Systems and Auxiliaries

Flammability control in the primary containment is achieved by an inert atmosphere during all plant operating modes except operator access for refueling and maintenance and a recombiner system to control oxygen produced by radiolysis. The high pressure nitrogen (HPIN) gas supply is described in Subsection 1.2.2.12.13. The Containment Atmospheric Monitoring System (CAMS) measures and records containment oxygen / hydrogen concentrations under post-accident conditions. It is automatically initiated by detection of a LOCA (Subsection 7.6.1.6). Table 1AA-3 lists the combustible gas control principal components and their locations.

1AA.5.1.4 Fission Product Removal and Control Systems and Auxiliaries

Engineered Safety Feature (ESF) filter systems are the Standby Gas Treatment System (SGTS) and the control building Outdoor Air Cleanup System. Both consist of redundant systems designed for accident conditions and are controlled from the control room. The SGTS filters the gaseous effluent from the primary and secondary containment when required to limit the discharge of radioactivity to the environment. The system function is described in Subsection 1.2.2.15.4.

A portion of the Control Building heating ventilating and air-conditioning (HVAC) provides detection and limits the introduction of radioactive material and smoke into the control room. This portion is described Subsection 9.4.1.1.3.

The CAMS described in the previous section also measures and records containment area radiation under post-accident conditions. A post-accident sampling system (PASS) obtains containment atmosphere and reactor water samples for chemical and radiochemical analysis in the laboratory. Delayed sampling, shielding, remote operated valves and sample transporting casks are utilized to reduce radiation exposure. The samples are manually transported between the PASS room in the Reactor Building and the analysis laboratory in the service building. The system is described in

Subsection 9.3.2.3.1. Table 1AA-4 lists the fission product removal control components and locations.

1AA.5.1.5 Instrumentation and Control, Power, and Habitability Systems and Auxiliaries

Most of the post-accident instrumentation and control system equipment is listed with the applicable equipment in Tables 1AA-2, 1AA-3 and 1AA-4. The remaining instrumentation and control equipment is included with the power and habitability systems equipment listed in Table 1AA-5. Instrumentation is consistent with the post accident phase variables monitored by the Post-Accident Monitoring (PAM) System listed in Table 7.5.2.

Standby AC power is supplied by three diesel generators in separate electrical divisions (Subsection 1.2.2.13.13). The diesel generators, switchgear and motor control centers are included in the unit Class 1E AC power system described in Subsection 1.2.2.13.14.1. Storage batteries are the standby power source for the unit Class 1E DC power system described in Subsection 1.2.2.13.12.2. The safety system logic and control power system is described in Subsection 1.2.2.13.14.1.

Habitability systems ensure that the operator can remain in the control room and take appropriate action for post-accident operations. The control building includes all the instrumentation and controls necessary for operating the systems required under post-accident conditions.

The control room, control and reactor building HVAC essential equipment are a portion of the plant environmental control of temperature, pressure, humidity and airborne contamination described in Subsection 1.2.2.16.5 (1), (4), (5), (7) and (8). HVAC units controlling the local room environments are included with respective equipment in Tables 1AA-2, 1AA-3 and 1AA-4. The major HVAC equipment and locations are listed in Table 1AA-5.

The Reactor Building Cooling Water (RBCW) System provides cooling water to designated equipment in the Reactor Building (including containment) as described in Subsection 1.2.2.12.3. The HVAC Emergency Cooling Water (HECW) System provides chilled water to designated equipment in the control building as described in Subsection 1.2.2.12.6.

Table 1AA-1 Radiation Source Comparison

Activity Group	% Core Inventory Released		
	R.G.1.3	R.G.1.7	Plant Design Basis
Air			
Noble Gases	100	100	100*
Halogens	25	—	25*
All Remaining	—	—	—
Water			
Noble Gases	0	—	0
Halogens	—	50	50†
All Remaining	—	1	1†

* Uniformly mixed within the primary containment boundary

† Uniformly mixed in the suppression pool and reactor coolant

Table 1AA-2 Post-Accident Emergency Core Cooling Systems and Auxiliaries

Equipment	MPL	Location
ADS & Transmitters		
SR Valve	B21-F010A,C,F,H,L,N,R,T	Upper Drywell (PC)
SR Accumulator	B21-A004A,C,F,H,L,N,R,T	Upper Drywell (PC)
Rx Water Level (ADS,RHR)	B21-LT003A thru H	Instrument Rack Rm. (SC)
Rx Water Level (HPCF)	B21-LT001A,B,C,D	Instrument Rack Rm. (SC)
Rx Pressure (RHR)	B21-PT301A,B,C,D	Instrument Rack Rm. (SC)
DW Pressure (HPCF, RHR)	B21-PT025A,B,C,D	Instrument Rack Rm. (SC)
HPCF		
Pumps	E22-C001B,C	HPCF Rm. B,C (SC)
SP Suction Valve	E22-F006B,C	HPCF Rm. B,C (SC)
Rx Injection Valve	E22-F003B,C	Valve Rm. B,C (SC)
CST Suction Valve	E22-F001B,C	Valve Rm. B,C (SC)
Essential HVH (HVAC)	U41-D102,106	HPCF Rm. B,C (SC)
CST Water Level	P13-LT001A,B,C,D	HPCF Rm. B,C (SC)
Flow	E22-FT008B1,B2,C1,C2	By HPCF Rm. B,C (SC)
Suction Pressure	E22-PT002,003; B,C	By HPCF Rm. B,C (SC)
Injection Pressure	E22-PT006,007; B,C	By HPCF Rm. B,C (SC)
LPCF		
Pump	E11-C001A,B,C	RHR Rm. A,B,C (SC)
Heat Exchanger	E11-B001A,B,C	RHR Rm. A,B,C (SC)
RCW Discharge Valve	P21-F013A,B,C	RHR Rm. A,B,C (SC)
SP Suction Valve	E11-F001A,B,C	RHR Rm. A,B,C (SC)
Rx Injection Valve	E11-F005A,B,C	Valve Rm. A,B,C (SC)
Rx Return Valve	E11-F010,011,012;A,B,C	Valve Rm. A,B,C (SC)
DW Spray Valve	E11-F017,018;B,C	Valve Rm. B,C (SC)
WW Spray Valve	E11-F019B,C	Valve Rm. B,C (SC)
FPC Supply Valve	E11-F015B,C	Valve Rm. B,C (SC)
FPS Supply Valve	E11-F101,102,103	Valve Rm. B,C (SC)
Essential HVH (HVAC)	U41-D103,104,105	RHR Rm. (SC)
Flow	E11-FT008A1,B1,C1	By RHR Rm. A,B,C (SC)
Flow	E11-FT008A2,B2,C2	By RHR Rm. A,B,C (SC)

Table 1AA-2 Post-Accident Emergency Core Cooling Systems and Auxiliaries (Continued)

Equipment	MPL	Location
RCW Flow	P21-FT008A,B,C	By RHR Rm. A,B,C (SC)
Hx I/O Temperature	E11-TT006,007;A,B,C	By RHR Rm. A,B,C (SC)
Discharge Pressure	E11-PT004A thru G	By RHR Rm. A,B,C (SC)
DW Temperature	T31-TT/SSA051,053	Inst. Rack Rm. (SC)
DW/WW Pressure Ratio	T31-PT055A,B	Inst. Rack Rm. (SC)
WW Pressure	T31-PT056A,B	Inst. Rack Rm. (SC)
DW Pressure	T31-PT054	Inst. Rack Rm. (SC)
FPCS		
Pump	G41-C001A,B	FPC Pump Rm. (SC)
Heat Exchanger	G41-B001A,B	FPC Hx Rm. (SC)
Pump Discharge Valve	G41-F021A,B	FPC Valve Rm. (SC)
Essential HVH (HVAC)	U41-D107, 108	FPC Valve Rm. (SC)
Flow	G41-FT006A,B	By FPC Pump Rm. (SC)
Suction Pressure	G41-PT003A,B	By FPC Pump Rm. (SC)
Skimmer ST Level	G41-LT020	Refueling Floor(SC)

(PC)—Primary Containment
(SC)—Secondary Containment

Table 1AA-3 Post-Accident Combustible Gas Control Systems and Auxiliaries

Equipment	MPL	Location
HPIN		
Nitrogen Storage Bottles	P54-A001A Thru V	By Valve Rm (RB)
Supply Pressure	P54-PT002A, B, 004, 005	By Valve Rm (RB)
FCS		
Recombiner & Auxiliaries	T49-A001A,B	(PC)
RHR Cooling/Isol. Valve	T49-F008,010; A,B	(PC)(SC)
Flow	T49-FT002,004; A,B	Inst. Rack Rm. A,B (SC)
Pressure	T49-PT003A,B	Inst. Rack Rm. A,B (SC)
CAMS		
Hydrogen, Oxygen Elements	D23-H_2, O_2 Rack A,B	CAMS Rm. A,B (SC)
Gas Measurement	D23-Gas Cal. Rack A,B	CAMS Rm. A,B (SC)
Gas Elements	D23-Gas Cal. Rack A,B	CAMS Rm. A,B (SC)
DW Gas Valve	D23-F004A,B	CAMS Rm. A,B (SC)
WW Gas Valve	D23-F006A,B	CAMS Rm. A,B (SC)
Essential HVH (HVAC)	U41-D113,114	CAMS Rm. A,B (SC)
Gas Supply	D23-Gas Cyl. Rack A,B	CAMS Rm. A,B (RB)

(PC)—Primary Containment
(SC)—Secondary Containment
(RB)—Reactor Building outside (Secondary Containment)

Table 1AA-4 Post-Accident Fission Product Removal and Control Systems and Auxiliaries

Equipment	MPL	Location
SGTS		
Exhaust Fan	T22-C001B, C	Fan/Dryer Rm. (SC)
Charcoal Filter	T22-D002C,D001B	Filter Train Rm. (SC)
PC Inlet Valve	T22-F002A,B	Fan/Dryer Rm. (SC)
SC Inlet Valve	T22-F001A,B	Fan/Dryer Rm. (SC)
Stack Outlet Valve	T22-F004A,B	Filter Train Rm. (SC)
PC (DW,WW) Isolation Valves	T31-F004,006,008	Valve Rm. (SC)
Essential HVH (HVAC)	U41-D111,112	SGTS HVH Rm. (SC)
Radiation (Ion/Scint.)	D11-RE002,011;A,B	SGTS Monitor Rm. (RB)
Sampling Rack	H22-P250	By SGTS (SC)
Flow	T22-FT018B,C	By Filter Train. Rm. (RB)
Filter Moisture	T22-MT011B,C & 012B,C	By Filter Train. Rm. (RB)
CR HVAC		
Emerg. Recirculation Fan	U41-C603B	CR HVAC Rm. A,B (CB)
Emerg. Charcoal Filter Unit	U41-B A,B	CR HVAC Rm. A,B (CB)
Air Intake Isolation Valves	U41-F A,B	CR HVAC Rm. A,B (CB)
PASS		
Conditioning/Holding Rack	P91	(SC)
Sampling/Casks Rack	P91	PASS Rack Rm. (RB)
LPCF Supply Valve	E11-F045,046; A	(SC)
DW/WW Gas (CAMS) Valve	D23	(SC)
Control Panel (PT,TT)	H22	PASS Rack Rm. (RB)
Chemical Radiological Analysis		Laboratory (SB)
Stack		
Radiation (Ion/Scint.)	D11-RE041,043; A,B	Stack (RB)
Monitor Racks, Control Rod	H21,H22	Stack Monitoring Rm.(RB)

(CB)—Control Building
(SC)—Secondary Containment
(RB)—Reactor Building outside (Secondary Containment)
(SB)—Service Building

Table 1AA-5 Post-Accident Instrumentation and Controls, Power and Habitability Systems and Auxiliaries

Equipment	MPL	Location
Instrumentation & Controls		
Post-Accident I&C	H11-Post-Accident	Control & Panel Rms. (CB)
Power		
DC Supply	R42-Storage Batteries	Battery Rm. (CB)
ESF HV&LV Switchgear	R22-Post-Accident	Emerg. Electric Rm. A,B,C (RB)
ESF Motor Control Center	R24-Post-Accident	Emerg. Electric Rm. A,B,C (RB)
Diesel Generator & Auxiliaries	R43-DG A,B,C	DG Rm. A,B,C (RB)
DG Motor Control Center	R43-P001A,B,C	DG MCC Rm. A,B,C (RB)
Supply Fan (HVAC)	U41-C201A,E,204B,F 207C, G	DG Supply Fan Rm. A,B,C (RB)
Exhaust Fan (HVAC)	U41-202A,E,205B,F 207C, G	DG Exhaust Fan Rm. A,B,C (RB)
Essen. Fresh Air Fan (HVAC)	U41-203A,E,206B,F 209C,G	DG Essen. Fan Rm. A,B,C (RB)
RCW Discharge Valve	P21-F055A thru F	DG Rm. A,B,C (RB)
Control Panel (H21)	R43-P002,003C,004;A,B,C	DG Control Pnl. Rm. A,B,C (RB)
CB HVAC		
Supply Fan	U41-C606B,F,608C,G	E/HVAC Rm. A,B,C (CB)
Exhaust Fan	U41-C607B,F,609C,G	E/HVAC Rm. A,B,C (CB)
MCR Supply Fan	U41-C601B,F,604A,E	CR HVAC Rm. A,B (CB)
MCR Exhaust Fan	U41-C602B,F,605A,E	CR HVAC Rm. A,B (CB)
RCW		
Pump	P21-C001A thru G	Pump Rm. A,B,C (CB)
Hx Return Valve	P21-F004D, E, F, A, B, C, D, G, H, J	Hx Rm. A,B,C (CB)
Non-Post-Accident Supply Valve	P21-F074A,B,C	(RB)
Non-Post-Accident Return Valve	P21-F082A,B,C	(RB)
Flow	P21-FT006A,B,C	By Pump Rm. A,B,C (CB)
Pressure	P21-PT004A,B,C	By Pump Rm. A,B,C (CB)
Surge Tank Level	P21-LT013A,B,C	By Surge Tank A,B,C (RB)
HECW		
Pump	P25-C001A,B,C,E,F	Chiller Rm. A,B,C (CB)

Table 1AA-5 Post-Accident Instrumentation and Controls, Power and Habitability Systems and Auxiliaries (Continued)

Equipment	MPL	Location
Refrigerator	P25-D001A,B,C,E,F	Chiller Rm. A,B,C (CB)
Pressure Control Valve	P25-F012 B,C	HVAC Rm. A,B,C (CB)
Temperature Control Valve	P25-F005 B,C	HVAC Rm. A,B,C (CB)
Temperature Control Valve	P25-F016 A,B,C	HVAC Rm. A,B,C (CB)
Temperature Control Valve	P25-F022 A,B,C	(RB)
RCW Temp. Control Valve	P21-F025 A,B,C,E,F	(CB)
Instrument Air		
Compressor	P52-C001,002	Inst.Air Rm. (RB)

(RB)—Reactor Building outside (Secondary Containment)
(CB)—Control Building

1B Not Used

1C ABWR Station Blackout Considerations

1C.1 Introduction

This appendix describes (a) how the ABWR Design addresses Station Blackout (SBO) Events; (b) how the ABWR Design complies with 10CFR50.63 SBO requirements; and (c) where supporting documentation to these conformances exist in Tier 2.

1C.2 Discussion

1C.2.1 Station Blackout (SBO) Definition

For the ABWR design the definitions of Station Blackout, Alternate AC (AAC) Power Source, and Safe Shutdown given in 10CFR50.02 are provided below:

- Station Blackout

  "Station blackout means the complete loss of alternating current (AC) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., the loss of offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency AC power system). Station blackout does not include the loss of available AC power to buses fed by station batteries through inverters or by alternate AC sources as defined in this section, nor does it assume a concurrent single failure or design basis accident."

- Alternate AC Power Source

  "Alternate AC source means an alternating current (AC) power source that is available to and located at or nearby a nuclear power plant and meets the following requirements:

  (1) Is connectable to but not normally connected to the offsite or onsite emergency AC power systems

  (2) Has minimum potential for common mode failure with offsite power or the onsite emergency AC power sources

  (3) Is available in a timely manner after the onset of station blackout

  (4) Has sufficient capacity and reliability for operation of all systems required for coping with station blackout and for the time required to bring and maintain the plant in safe shutdown (non-design basis accident)"

- Safe Shutdown (SSD)

  "Safe shutdown (non-design basis accident (non-DBA)) for station blackout means bringing the plant to those shutdown conditions specified in plant technical specifications as Hot Standby or Hot Shutdown, as appropriate…"

1C.2.2 Plant SBO Design Basis

1C.2.2.1 General SBO Design Basis

- The ABWR design will mitigate station blackout events as defined in Subsection 1C.2.2.
- The ABWR design will comply with 10CFR50.63 requirements relative to the loss of all alternating current power sources.
- The ABWR design will include and utilize an Alternate AC (AAC) power source to comply with 10CFR50.63 requirements and the recommendations for ALWRs, as defined by the NRC in SECY 90-016.
- The ABWR design will be consistent with Regulatory Guide 1.155 and NUMARC 87-00 guidelines relative to an AAC power source.
- The ABWR design AAC power source will supplement and compliment the current offsite AC power connections, the onsite normal AC power sources (the unit auxiliary and reserve auxiliary transformers), the onsite emergency AC power sources (DGs) and the onsite DC power sources.

1C.2.2.2 Specific SBO Design Basis

- The ABWR AAC power source will be a combustion turbine generator (CTG).
- The normal design function of the CTG will be to act as a standby, non-safety-related power source for the plant investment protection (PIP) non-safety-related loads during loss of preferred power (LOPP) events.
- The CTG will be capable of being manually configured to provide power to a selected safety-related emergency bus within 10 minutes during SBO events.
- The CTG will automatically start, accelerate to required speed, reach required voltage and frequency and be ready to accept PIP loads within two minutes of the receipt of its start signal.
- The CTG will be a diverse, self contained unit (including its auxiliaries) and will be independent of the plant preferred and emergency power sources.

- The target reliability of the GTG will be >0.95, as calculated by NSAC-108 methodology.
- The CTG will have capacity to supply the required safe shutdown loads.
- The CTG will be housed in a Uniform Building Code structure which is protected from adverse site weather related conditions.
- The CTG design will minimize potential for single point failure vulnerability with onsite emergency power sources.
- Adequate pneumatic pressure and water makeup sources will be available throughout the SBO duration.
- The ABWR design will confine the SBO duration to 10 minutes or less with the use of the AAC power source.
- The CTG will be controllable locally or from the MCR.
- Provisions will be made to facilitate the orderly restoration of offsite and onsite power source during the SBO event.
- Special quality assurance and control practices will be applied to the CTG.
- Special equipment requirements will be applied to the CTG support components.
- The CTG will utilize a separate fuel oil storage tank and transfer system from that of the onsite emergency power sources.
- The CTG will operate during the SBO event without external AC power sources.
- The standby function of the CTG will be to mitigate LOPP or SBO events.
- Dual manually operated circuit breakers will separate the CTG from the onsite emergency power buses.
- The AAC power source will utilize the available station and/or internal batteries for breaker control and initial CTG starting functions.
- The CTG Fuel Oil Supply will be periodically inspected and the oil analyzed.
- The CTG operation will be subject to plant operation, maintenance and testing procedures.
- All operator actions required during SBO events will be demonstrated by training exercises and will be according to appropriate plant procedures.

- CTG power will be used to restore various selected plant environmental control components (HVAC, chillers, etc.) as soon as possible.
- The CTG will not normally be used to provide power connected to the plant loads.
- The CTG will be capable of being inspected, tested, and maintained.
- The CTG capabilities will be demonstrated prior to shipment, during initial pre-operational test, and periodically during power operation.
- Required plant core cooling and containment integrity during the SBO duration (10 minutes) will not depend on any AC power sources.

1C.2.3 Plant SBO Safety Analysis

1C.2.3.1 Plant Event Evaluations

1C.2.3.1.1 Plant Normal Operation

The normal configuration of the onsite AC power distribution system and its individual power sources are described in Subsections 8.2.1 and 8.3.1. The CTG (AAC) system attributes and its interconnections are described in Subsection 9.5.11 and in Subsection 8.3.1, respectively. Both are shown on Figure 8.3-1.

The normal and alternate preferred AC power sources supply safety-related and non-safety-related loads. Power to these loads are supplied from the unit auxiliary transformers (UATs) units and the reserve auxiliary transformer (RAT).

The CTG is designed to supply standby power to the non-Class 1E 6.9 kV buses which carry the plant investment protection (PIP) loads. The CTG automatically starts on detection of under voltage on the PIP buses. When the CTG is ready to assume load, if the voltage is still deficient, power automatically transfers to the CTG (refer to Figure 8.3-1).

The CTG can also supply standby power to the non-Class 1E 6.9 kV power generation buses which supply feedwater and condensate pumps. These buses normally receive power from the unit auxiliary transformers and supply power to the plant investment protection (PIP) buses through a cross-tie. The cross-tie automatically opens on loss of power but may be manually reclosed if it is desired to operate a condensate pump from the combustion turbine generator or the reserve auxiliary transformer which are connectable to the PIP buses. This arrangement allows the powering of load groups of non-Class 1E equipment in addition to the Class 1E divisions which may be used to supply water to the reactor vessel (refer to Figure 8.3-1).

1C.2.3.1.2 LOPP Events

The ABWR onsite emergency power sources during LOPP events are the diesel generator (DG) units. These units and their system responses are discussed in Subsection 8.3.1.1.8. However, the CTG is available to provide backup emergency power during LOPP to safety-related loads by manual reconfiguration of the CTG and the loads.

1C.2.3.1.3 SBO Events

The CTG is the AC power source during an SBO event. The CTG can supply 6.9 kV Class 1E buses through the realignment of pre-selected breakers during SBO events. The CTG will reach operational speed and voltage in 2 minutes and will be available for bus connection within 10 minutes. Upon a LOPP, the CTG is automatically started and configured to non-safety-related PIP loads. Plant operators using appropriate procedures will reconfigure any of the 6.9 kV Class 1E buses to accept CTG power. Refer to Tier 2 Subsections 8.3.1.1.7 and 9.5.11.

1C.2.3.1.4 Other Operational Capabilities

The CTG can be used for postulated prolonged SBO scenarios.

Up to the limits of its capacity, the CTG can be connected to any combination of Class 1E and non-Class 1E buses to supply loads in excess of the minimum required for safe shutdown.

The ABWR design provides for local and main control room operation of the CTG. Communication is available between the CTG area and the main control room.

1C.2.3.2 Alternative AC Power Source Evaluation

The alternate AC power source (1) is a combustion turbine generator, (2) is provided with an immediate fuel supply that is separate from the fuel supply for other onsite emergency AC power systems, (3) fuel will be sampled and analyzed consistent with applicable standards, (4) is capable of operating during and after a station blackout without any AC support systems powered from the preferred power supply or the blacked-out units Class 1E power sources (5) is designed to power all of the PIP and / or Class 1E shutdown loads necessary within 10 minutes of the onset of the station blackout, such that the plant is capable of maintaining core cooling and containment integrity (6) will be protected from design basis weather events (except seismic and tornado missiles) to the extend that there will be no common mode failures between offsite preferred sources and the combustion turbine generator power source, (7) will be subject to quality assurance guidelines commensurate with its importance to SBO, (8) will have sufficient capacity and capability to supply one division of Class 1E loads, (9) will have sufficient capacity and capability to supply the required non-Class 1E loads used for a safe shutdown, (10) will undergo factory testing to demonstrate its ability to reliably start, accelerate to required speed and voltage and supply power within two minutes, (11) will not normally supply power to nuclear safety-related equipment except under specific conditions, (12) will not be a single point single failure detriment to onsite emergency AC power sources, and (13) will be subject to site acceptance testing; periodic preventative maintenance, inspection, testing; operational reliability assurance program goals.

Based on the above, the ABWR design for the AAC power supply complies with 10CFR50.63, with Regulatory Guide 1.155 and with NUMARC 87-00 and meets the SBO rule.

1C.2.4 Plant Conformance With SBO Requirements

A brief review of the general ABWR design conformance with various SBO requirements and guidelines is given below. A more complete in-depth and specific review of each of the SBO regulatory requirements or guidelines is given in the enclosed tables (refer to Tables 1C-1 through 1C-3).

1C.2.4.1 10CFR50.63 Requirements

The ABWR complies with the 10CFR50.63 requirements. Special attention was given to the regulation definition of the SBO event, the event conditions, and the requirement for safe shutdown status. The ABWR utilizes the AAC power source option and provides an evaluation of the requirements/compliances in Table 1C-1.

1C.2.4.2 New ALWR Requirements (SECY 90-016)

A review of the new ALWR SBO requirements in SECY 90-016 recommendations was conducted. The ABWR design is in compliance with the ALWR recommendations.

1C.2.4.3 Regulatory Guide 1.155 Guideline Requirements

A review of the ABWR CTG design relative to Sections 3.3.5, 3.3.6, 3.3.7, 3.4 and Appendix A and B of RG 1.155 was conducted. CTG design fully complies with the cited requirements. The use of the CTG as an AAC power source in the ABWR design eliminates the need for a SBO coping analyses by limiting the SBO duration to 10 minutes or less. No operator action is required within the initial ten minutes (refer to Table 1C-2).

1C.2.4.4 NUMARC 87-00 Guidelines

A review of the ABWR CTG design relative to the NUMARC SBO guidelines, Subsections 7.1.1 and 7.1.2 and Appendices A and B was conducted. The ABWR design with CTG is consistent with the NUMARC guidelines (refer to Table 1C-3).

1C.2.5 Other SBO Considerations

Several other SBO considerations are identified below for special compliance or consideration.

1C.2.5.1 Plant Technical Specifications

Surveillance and operational requirements are needed for the CTG in order to assure its reliability or maintainability. However, these will be part of the COL applicant maintenance, testing, and inspection procedures. These procedures will not be part of technical specifications.

1C.2.5.2 Design Interface Requirements

The CTG has a limited number of design interface requirements. Fuel oil is initially supplied from a local tank, and then transferred from a fuel oil storage tank, both of which are independent of the DG fuel oil tanks. A seven (7) day oil supply for the CTG sufficient for shutdown loads will be available onsite. The local CTG I&C is powered by the unit itself or supplied from station batteries. Other auxiliary functions are an integral part of the CTG unit.

1C.2.5.3 Station Blackout Procedures

Appropriate procedures will include the use of the CTG and are the COL applicant’s responsibility. The procedures will consider specific instructions for operation actions responses, timing and related matters during SBO events. The operator actions will include power source switching, load shedding, etc. See Subsection 1C.4.1 for COL license information requirements.

1C.2.5.4 Equipment Qualification, Testing and Reliability

The CTG will be qualified (as a non-Class 1E AAC power source) for its intended duties and service. Qualification testing, equipment inspections, and reliability data will be made available.

1C.2.5.5 Periodic Surveillance, Testing, Inspection and Maintenance

Operational reliability assurance program (ORAP) requirements will be established for the CTG.

1C.2.5.6 Power and Control Cable Routing

The CTG power and control cable routing is physically and electrically separated from other power sources to the extent practical. A suggested routing is shown in Figure 8.2-1.

1C.2.5.7 Plant Battery Recharging

The CTG is capable of recharging the plant batteries during SBO scenarios while supplying safe shutdown loads.

1C.2.5.8 Plant HVAC Restoration Capabilities

The CTG is capable of restoring environmental control components during the SBO duration while supplying the safe shutdown loads.

The Main Control Room environment will not exceed its design basis temperature even during a prolonged SBO event. With the CTG available in ten minutes, MCR HVAC can be restored.

1C.2.5.9 Circuit Breaker Operation

During the realignment of the CTG from non-safety-related buses to safety-related buses, at least two breakers will need to be manually closed. One of these breakers is Class 1E, and is controlled by the Class 1E battery power within the same division. The other breaker is non-Class 1E, and is controlled by the non-Class 1E battery.

The current SBO requirement that at least one emergency bus be powered within ten minutes is achieved by the manual operation of the two breakers between the CTG and the selected emergency bus (see Figure 8.3-1).

In order to maintain a minimum number of direct connections between the CTG and any of the three Class 1E emergency buses, only one Class 1E bus has its supply breaker racked in. It can therefore be controlled directly from the main control room. The other emergency buses have their supply breakers racked out, and therefore, require local operator action to rack in the breakers before main control room operation is available.

1C.2.5.10 CTG – Physical Protection Considerations

The CTG is housed in a building (separate from the building which contains the DGs) above the design flood levels. The building is designed to protect the CTG from site related weather conditions.

1C.3 Conclusions

In summary:

- The ABWR design will utilize a combustion turbine generator (CTG) as its Alternate AC (AAC) power source in complying with 10CFR50.63 SBO.
- The ABWR design complies with 10CFR50.63 and RG 1.155 and is consistent with NUMARC 87-000 guidelines.

- The ABWR design can successfully prevent or mitigate the consequences of an SBO event.

1C.4 COL License Information

1C.4.1 Station Blackout Procedures

The COL applicant shall provide procedures for SBO events including the use of the CTG as described in Subsection 1C.2.5.3.

1C.5 References

1C-1 SECY-90-016, “Evolutionary LWR Certification Issues and Their Relationship to Current Regulatory Requirements”, January 12, 1990.

1C-2 Letter J. Taylor to S. Chilk, “Evolutionary LWR Certification Issues and Their Relationship to Current Regulatory Requirements”, June 26, 1990.

1C-3 10CFR50.63, “Loss of All Alternating Current Power (Station Blackout-SBO)”, July 21, 1988.

1C-4 RG-1.155, “Station Blackout”, July 1988.

1C-5 NUMARC 87-00, “Guidelines and Technical Bases for NUMARC Initiative Addressing Station Blackout at LWRs” Plus Supplemental Questions and Answers, January 4, 1990.

1C-6 10CFR50.02, Definitions.

Table 1C-1 ABWR Design Compliance with 10CFR50.63 Regulations

Requirements	Compliance
§ 10CFR50-63 Loss of all alternating current power.	
§ 50.63 Loss of all alternating current power.	
(a) Requirements	
(1) Each light-water-cooled nuclear power plant licensed to operate must be able to withstand for a specified duration and recover from a station blackout as defined in § 50.2. The specified station blackout duration shall be based on the following factors:	The ABWR design will utilize an alternate AC (AAC) power source to mitigate and recover from station blackout events (defined in 50.2). The AAC power source will be a combustion turbine generator (CTG). The CTG will be totally independent from offsite preferred and onsite Class 1E sources. A ten (10) minute interval is used as the ABWR design basis for the SBO event duration. The AAC power source provides a diverse power source to the plant.
(i) The redundancy of the onsite emergency AC power sources	The ABWR design CTG will have sufficient capacity and capabilities to power the necessary reactor core coolant, control and protective systems including station battery and other auxiliary support loads needed to bring the plant to a safe and orderly shutdown condition (defined in 50.2). The CTG supplied will be rated at a minimum of 9 MWe and be capable of accepting shutdown loads within 10 minutes. The current plant onsite emergency power sources include three (3) independent and redundant DG divisions which are designed to supply approximately 5 MWe within 1 minute. Additionally, the plant has been designed to accommodate AC power source losses for a period up to 8 hours. The AAC limits the SBO event to 10 minutes.
(ii) The reliability of the onsite emergency AC power sources	The current onsite emergency AC power sources will have the following reliability: DGs…0.975 The CTG will have the following reliability: CTG…0.95 The above values are used in the ABWR-PRA analysis.

Table 1C-1 ABWR Design Compliance with 10CFR50.63 Regulations (Continued)

Requirements	Compliance
(iii) The expected frequency of loss of offsite power	The expected frequency of loss of offsite power assumed was 0.1 events/yr.
(iv) The probable time needed to restore offsite power	The offsite power is expected to be restored within 8 hours.
(2) The reactor core and associated coolant, control, and protection systems, including station batteries and any other necessary support systems, must provide sufficient capacity and capability to ensure that the core is cooled and appropriate containment integrity is maintained in the event of a station blackout for the specified duration. The capability for coping with a station blackout of specified duration shall be determined by an appropriate coping analysis. Utilities are expected to have the baseline assumptions, analyses, and related information used in their coping evaluations available for NRC review.	The AAC power source is capable of providing the necessary core, containment and equipment services (e.g. makeup and cooling water, I&C power, etc.) to bring the reactor to hot shutdown and then to cold shutdown conditions. The AAC will limit the SBO duration to 10 minutes. The current plant design assures that during the 10-minute interval, the plant core, containment and other safety functions will be maintained without the use or need for AC power. However, the AAC can operate indefinitely. A seven (7) day supply of oil sufficient for shutdown loads is available on site. Subsequent oil deliveries will be provided.

Table 1C-1 ABWR Design Compliance with 10CFR50.63 Regulations (Continued)

Requirements	Compliance
(b) Limitation of scope (c) Implementation (1) Information Submittal. For each light-water-cooled nuclear power plant licensed to operate after the effective date of this amendment, the licensee shall submit the information defined below to the Director by 270 days after the date of license issuance. (i) A proposed station blackout duration to be used in determining compliance with paragraph (a) of this section, including a justification for the selection based on the four factors identified in paragraph (a) of this section	In addition to the discussion under (a) above, the following is noted. The ABWR design SBO duration time considerations are consistent with RG1.155 and NUMARC-87-00. Upon loss of offsite power (LOPP) and upon the subsequent loss of all on site AC emergency power sources (three independent and redundant DGs), the CTG can be manually connected to any one of the three safety-related (Class 1E) busses by closing two circuit breakers. The alternative AC (AC) power source will automatically start, and within 2 minutes be up to required speed and voltage. It will then automatically connect to selected PIP buses (non-Class 1E) loads. During the first 10 minutes, the reactor will have automatically tripped, the main steam isolation valves (MSIVs) closed, and the RCIC actuated. The RCIC system will automatically control reactor coolant level. Any necessary relief valve operation will also be automatic. Within the 10 minute SBO interval, none of the above actions will require AC power or manual operator actions. The reconfiguration of the CTG to pick up the Class 1E buses will require manual closure of two circuit breakers from the control room. Upon restoration of power to the safety bus(es), the remaining safe shutdown loads will be energized.
(ii) A description of the procedures that will be implemented for station blackout events for the duration determined in paragraph (c)(1)(i) of this section and for recovery therefrom	Appropriate plant procedures will be developed by the COL applicant for the ABWR design. These procedures be integrated/coordinated with the plant EOPs, using the EOP methodology. Procedures will consider instructions for operator actions, responses, timing, and related matters during the SBO event.

Table 1C-1 ABWR Design Compliance with 10CFR50.63 Regulations (Continued)

Requirements	Compliance
(iii) A list of modifications to equipment and associated procedures, if any, necessary to meet the requirements of paragraph (a) of this section, for the specified station blackout duration determined in paragraph (c)(1)(i) of this section, and a proposed schedule for implementing the stated modifications	Modifications to equipment and procedures is not applicable since the use of an AAC source and other SBO considerations are included in the ABWR design.
(2) Alternate AC source: The alternate AC power source(s), as defined in § 50.2, will constitute acceptable capability to withstand station blackout provided an analysis is performed which demonstrates that the plant has this capability from onset of the station blackout until the alternate AC source(s) and required shutdown equipment are started and lined up to operate. The time required for startup and alignment of the alternate AC power source(s) and this equipment shall be demonstrated by test. Alternate AC source(s) serving a multiple unit site where onsite emergency AC source are not shared between units must have, as a minimum, the capacity and capability for coping with a station blackout in any of the units. At sites where onsite emergency AC sources are shared between units, the alternate AC source(s) must have the capacity and capability as required to ensure that all units can be brought to and maintained in safe shutdown (non-DBA) as defined in § 50.2. If the alternate AC source(s) meets the above requirements and can be demonstrated by test to be available to power the shutdown buses within 10 minutes of the onset of station blackout, then no coping analysis is required.	The ABWR CTG will be automatically initiated upon the loss of power to the PIP buses. The CTG will achieve required speed and voltage within 2 minutes. The CTG will be manually connected to safe shutdown buses within 10 minutes. These equipment capabilities will be demonstrated 1) by the manufacturer's component tests, 2) by the CTG initial startup tests and 3) periodically by the COL applicant as part of his operational reliability assurance program. The ABWR design is a single unit plant arrangement design. The CTG AAC source is available to power shutdown loads within 10 minutes as described above. Therefore, no coping analysis is required. In addition, the ABWR is designed with an 8-hour battery to accommodate station blackout without the need for AC power. Also, the three independent emergency diesel generator systems will accommodate one DG out of service, plus a single failure, with the remaining DG capable of bringing the plant to safe shutdown.
(3) Regulatory Assessment:	
(4) Implementation Schedule: (53 FR 23215, June 21, 1988)	

Table 1C-2 ABWR Design Compliance with Regulatory Guide 1.155

Requirements	Compliance
Regulatory Guide 1.155—Station Blackout	
Regulatory Position	
3.3.5 If an AAC power source is selected specifically for satisfying the requirements for station blackout, the design should meet the following criteria:	The ABWR AAC power source is not normally connected to the preferred or the onsite emergency AC power system. Two open circuit breakers—one Class 1E and the other non-Class 1E—separate the CTG from the safety-related emergency buses.
1. The AAC power source should not normally be directly connected to the preferred or the blacked-out unit's onsite emergency AC power system.	The AAC power source is also not normally connected to any of the preferred AC power sources or their associated non-safety-related buses. A non-Class 1E circuit breaker separates the CTG from the PIP buses.
2. There should be a minimum potential for common cause failure with the preferred or the blacked-out unit's onsite emergency AC power sources. No single-point vulnerability should exist whereby a weather-related event or single active failure could disable any portion of the blacked-out unit's onsite emergency AC power sources or the preferred power sources and simultaneously fail the AAC power source.	The ABWR design minimizes the potential for a) common cause failures between the preferred sources and the onsite emergency power sources; b) common cause failures between onsite emergency power sources themselves; c) common cause failures between onsite power sources and the AAC power source; and d) common cause failures between preferred sources and the AAC power source. The design also precludes interactions between preferred, onsite emergency, and AAC power systems resulting from weather related events or single failures such that a single point vulnerability will not simultaneously fail both the AAC power source and the onsite emergency or offsite preferred power source(s). This is accomplished by having onsite emergency and the AAC power sources inside weather protected buildings and by maintaining adequate separation between the four power sources. None of the four standby power sources share emergency buses or loads, auxiliary services or instrumentation and controls prior to the recovery actions from the SBO event. These power sources are physically, electrically, mechanically and environmentally separated.

Table 1C-2 ABWR Design Compliance with Regulatory Guide 1.155 (Continued)

Requirements	Compliance
3. The AAC power source should be available in a timely manner after the onset of station blackout and have provisions to be manually connected to one or all of the redundant safety buses as required. The time required for making this equipment available should not be more than 1 hour as demonstrated by test. If the AAC power source can be demonstrated by test to be available to power the shutdown buses within 10 minutes of the onset of station blackout, no coping analysis is required.	The ABWR AAC design power source will be automatically started and reach rated speed and voltage and be available to supply PIP loads within 2 minutes, and safety-related loads within 10 minutes for any loss of preferred offsite power sources (LOPP). The design has provisions to assure the timely manual interconnection between the AAC (CTG) and any one or more of the safety-related shutdown buses. The ABWR AAC design will be demonstrated by test to show that it can be connected to safety-related buses within 10 minutes. Therefore, no coping analysis is required.
4. The AAC power source should have sufficient capacity to operate the systems necessary for coping with a station blackout for the time required to bring and maintain the plant in safe shutdown.	The ABWR AAC power source is rated at 9 MWe, which is more than sufficient capacity to operate the necessary safe shutdown loads which are less than 5 MWe.
5. The AAC power system should be inspected, maintained, and tested periodically to demonstrate operability and reliability. The reliability of the AAC power system should meet or exceed 95% as determined in accordance with NSAC-108 (Reference 11) or equivalent methodology.	The ABWR design includes previsions to demonstrate the operability and reliability of the AAC power source. The CTG will be subject to surveillance inspection, testing and maintenance in accordance with the manufacturer's requirements, the COL applicant's maintenance program and with operational reliability assurance program requirements. The CTG will meet or exceed a reliability goal of 0.95 in accordance with NSAC-108 or equivalent methodology.

Table 1C-2 ABWR Design Compliance with Regulatory Guide 1.155 (Continued)

Requirements	Compliance
3.3.6 If a system or component is added specifically to meet the recommendations on station blackout duration in Regulatory Position 3.1, system walk downs and initial tests of new or modified systems or critical components should be performed to verify that the modifications were performed properly. Failures of added components that may be vulnerable to internal or external hazards within the design basis (e.g., seismic events) should not affect the operation of systems required for the design basis accident.	The ABWR design includes the CTG as the AAC power source for SBO mitigation. A test program will be conducted by the manufacturer/equipment vendor to verify the major equipment performance objectives (e.g., start time, rated speed and voltage times, stable voltage outputs, etc.). These tests will be conducted prior to CTG installation at the plant site. Prior to plant operation, the AAC power source will be subject to pre-operational testing to demonstrate that the CTG will perform its intended function. Periodically, the AAC power source will be tested to assure that the reliability/availability goals are being met and maintained. The ABWR design safety evaluations take into account potential plant disturbances that could affect AAC power source reliability. These disturbances could occur as a result of internal and external hazards (e.g., floods, fires and harsh environs, respectively). The adverse effects on AAC power source components due to operational hazards will not affect the operations of safety-related systems required for the design basis events. The effects caused by or upon the AAC power source due to operational events (internal and external hazards) are limited since the AAC power source components are physically, mechanically and essentially electrically isolated from the design basis engineered safety features and other power generation systems and components. Design bases accident events may result in the potential degradation of the AAC power source. However, the resulting effects of the AAC will not diminish the current safety system responses and the current event outcomes.

Table 1C-2 ABWR Design Compliance with Regulatory Guide 1.155 (Continued)

Requirements	Compliance
3.3.7 The system or component added specifically to meet the recommendations on station blackout duration in Regulatory Position 3.1 should be inspected, maintained, and tested periodically to demonstrate equipment operability and reliability.	The ABWR design AAC power source will be capable of being tested, inspected and maintained on a periodic basis. The CTG location in the Turbine Building provides easy access to the unit. The access and environmental conditions in the CTG area allow physical surveillance, easy maintenance, and testing. The CTG will be periodically started, brought up to speed and voltage, and connected to the PIP buses. The CTG will be subject to periodic test in order to verify the operability and reliability goals in the plant operational reliability assurance program (ORAP).
3.4 Procedures and Training To Cope with Station Blackout Procedures* and training should include all operator actions necessary to cope with a station blackout for at least the duration determined according to Regulatory Position 3.1 and to restore normal long-term core cooling/decay heat removal once AC power is restored.	Appropriate plant procedures will be developed by the COL applicant for the ABWR design. These procedures will be integrated/coordinated with the plant EOPs, using the EOP methodology. Procedures will consider instructions for operator actions, responses, timing, and related matters during the SBO event.

* Procedures should be integrated with plant-specific technical guidelines and emergency procedures developed using the emergency operating procedure upgrade program established in response to Supplement 1 of NUREG-0737 (Reference 12). The task analysis portion of the emergency operating procedure upgrade program should include an analysis of instrumentation adequacy during a station blackout.

Table 1C-2 ABWR Design Compliance with Regulatory Guide 1.155 (Continued)

	Requirements	Compliance
3.5	Quality Assurance and Specification Guidance for Station Blackout Equipment that is Not Safety-Related Appendices A and B provide guidance on quality assurance (QA) activities and specifications respectively for non-safety-related equipment used to meet the requirements of § 50.63 and not already covered by existing QA requirements in Appendix B or R of Part 50. Appropriate activities should be implemented from among those listed in these appendices depending on whether the non-safety equipment is being added (new) or is existing. This QA guidance is applicable to non-safety systems and equipment for meeting the requirements of § 50.63 of 10CFR50. The guidance on QA and specifications incorporates a lesser degree of stringency by eliminating requirements for involvement of parties outside the normal line organization. NRC inspections will focus on the implementation and effectiveness of the quality controls described in Appendices A and B. Additionally, the equipment installed to meet the station blackout rule must be implemented such that it does not degrade the existing safety-related systems. This is to be accomplished by making the non-safety-related equipment as independent as practicable from existing safety-related systems. The non-safety systems identified in Appendix B are acceptable to the NRC staff for responding to a station blackout.	The ABWR AAC power source design addresses the quality assurance and equipment specification guidance indicated in Appendices A and B of this guide. The specific responses to Appendices A and B are presented in the following sections in this table.

Table 1C-2 ABWR Design Compliance with Regulatory Guide 1.155 (Continued)

Requirements	Compliance
Appendix A – Quality Assurance	
The QA guidance provided here is applicable to non-safety systems and equipment used to meet the requirements of § 50.63 and not already explicitly covered by existing QA requirements in 10CFR50 in Appendix B or R. Additionally, non-safety equipment installed to meet the station blackout rule must be implemented so that it does not degrade the existing safety-related systems. This is accomplished by making the non-safety equipment as independent as practicable from existing safety-related systems. The guidance provided in this section outlined an acceptable QA program for non-safety equipment used for meeting the station blackout rule and not already covered by existing QA requirements. Activities should be implemented from this section as appropriate depending on whether the equipment is being added (new) or is existing.	The ABWR AAC power source design is in compliance with the following QA guidelines in 10CFR50.63 as indicated below:
1. Design Control and Procurement Document Control	
Measures should be established to ensure that all design-related guidances used in complying with § 50.63 are included in design and procurement documents, and that deviations therefrom are controlled.	The COL applicant's QA program will comply with this requirement.
2. Instructions, Procedures, and Drawings	
Inspections, tests, administrative controls, and training necessary for compliance with § 50.63 should be prescribed by documented instructions, procedures, and drawings and should be accomplished in accordance with these documents.	The COL applicant's QA program will comply with this requirement.
3. Control of Purchased Material, Equipment, and Services	
Measures should be established to ensure that purchased material, equipment, and services conform to the procurement documents.	The COL applicant's QA program will comply with this requirement.

Table 1C-2 ABWR Design Compliance with Regulatory Guide 1.155 (Continued)

Requirements	Compliance
Appendix A – Quality Assurance	
4. Inspection	
A program for independent inspection of activities required to comply with § 50.63 should be established and executed by (or for) the organization performing the activity to verify conformance with documented installation drawings and test procedures for accomplishing the activities.	The COL applicant's QA program will comply with this requirement.
5. Testing and Test Control	
A test program should be established and implemented to ensure that testing is performed and verified by inspection and audit to demonstrate conformance with design and system readiness requirements. The tests should be performed in accordance with written test procedures; test results should be properly evaluated and acted on.	The COL applicant's QA program will comply with this requirement.
6. Inspection, Test, and Operating Status	
Measures should be established to identify items that have satisfactorily passed required tests and inspections.	The COL applicant's QA program will comply with this requirement.
7. Nonconforming Items	
Measures should be established to control items that do not conform to specified requirements to prevent inadvertent base or installation.	The COL applicant's QA program will comply with this requirement.
8. Corrective Action	
Measures should be established to ensure that failures, malfunctions, deficiencies, deviations, defective components, and nonconformances are promptly identified, reported, and corrected.	The COL applicant's QA program will comply with this requirement.

Table 1C-2 ABWR Design Compliance with Regulatory Guide 1.155 (Continued)

Requirements	Compliance
Appendix A – Quality Assurance	
9. Records	
Records should be prepared and maintained to furnish evidence that the criteria enumerated above are being met for activities required to comply with § 50.63.	The COL applicant's QA program will comply with this requirement.
10. Audits	
Audits should be conducted and documented to verify compliance with design and procurement documents, instructions, procedures, drawings, and inspection and test activities developed to comply with § 50.63.	The COL applicant's QA program will comply with this requirement.

Table 1C-2 ABWR Design Compliance with RG 1.155 (Continued)

Requirements		Compliance
Appendix B—Guidance Regarding Systems/Components		
	Alternate AC Sources	**ABWR AAC Power Source**
Safety-Related Equipment (Compliance with IEEE-279)	Not required, but the existing Class 1E electrical systems must continue to meet all applicable safety-related criteria.	Existing onsite emergency power sources, buses and loads will continue to meet all applicable safety-related criteria.
Redundancy	Not required.	—
Diversity from Existing EDGs	See Regulatory Position 3.3.4 of this guide.	The ABWR design will utilize a AAC diverse power source from that of the EDGs. A qualified combustion turbine generator will be used as the AAC.
Independence from Existing Safety-Related Systems	Required if connected to Class 1E buses. Separation to be provided by 2 circuit breakers in series (1 Class 1E at the Class 1E bus and 1 non-Class 1E).	Two breakers separate the onsite emergency power buses from the CTG. One breaker is Class 1E and the breaker closest to the CTG is non-Class 1E (see Figure 8.3-1).
Seismic Qualification	Not required.	—
Environmental Consideration	If normal cooling is lost, needed for station blackout event only and not for design basis accident (DBA) conditions. Procedures should be in place to affect the actions necessary to maintain acceptable environmental conditions for the required equipment. See Regulatory Position 3.2.4.	The use of the ACC power source will assure that the plant equipment/environment cooling loss will be limited to 10 to 60 minutes (SBO duration). Normal plant cooling loads will be restored after shutdown loads are reestablished. Temperature rise conditions will be limited to minutes rather than hours
Capacity	Specified in § 50.63 and Regulatory Position 3.3.4.	The AAC power source is capable of powering more than the minimum required shutdown loads.
Quality Assurance	Indicated in Regulatory Position 3.5.	The ABWR design will be subjected to the quality assurance standards cited in Appendix A.

Table 1C-2 ABWR Design Compliance with RG 1.155 (Continued)

Requirements		Compliance
Appendix B—Guidance Regarding Systems/Components		
	Alternate AC Sources	**ABWR AAC Power Source**
Technical Specification for Maintenance, Limiting Condition, FSAR, etc.	Should be consistent with the Interim Commission Policy Statement on Technical Specifications (Federal Register Notice 52 FR 3789) as applicable.	The AAC power source operational and test requirements will be defined by the Plant Maintenance Program and the ORAP. They will also be consistent with the Interim Commission Policy Statement on Tech Specs.
Instrumentation and Monitoring	Must meet system functional requirements.	The AAC power source instrumentation, controls and monitoring will be of such number, type and quality to assure that the CTG reliability goals are met.
Single Failure	Not required.	—
Common Cause Failure (CCF)	Design should, to the extent practicable, minimize CCF between safety-related and non-safety-related equipment.	The AAC power source will be physically, mechanically and electrically independent of the offsite and onsite power systems.

Table 1C-2 ABWR Design Compliance with RG 1.155 (Continued)

Requirements		Compliance
Appendix B—Guidance Regarding Systems/Components		
	Water Source (Existing Condensate Storage Tank or Alternate)	**SBO Recovery with AAC Power Source**
Safety-Related Equipment (Compliance with IEEE-279)	Not required, but the existing Class 1E systems must continue to meet all applicable safety-related criteria.	The ABWR design Condensate Storage Tank will provide primary makeup water via the RCIC or HPCF. The suppression pool will serve as the secondary water source. The AAC powered RCWS and RSWS pumps will provide heat removal service to the plant systems including chillers and HVAC cooling subsystems.
Redundancy	Not required.	—
Diversity	Not required.	—
Independence from Existing Safety-Related Systems	Ensure that the existing safety functions are not compromised, including the capability to isolate components, subsystems, or piping, if necessary.	The loss of all AC power (SBO) will automatically cause reactor scram, MSIV closure, and initiation of the RCIC. The AAC power source will re-energize the lost shutdown loads (emergency makeup water, heat removal and HVAC services) due to the SBO condition within ten (10) to 60 minutes. The condensate storage tank will used during the first ten minutes and throughout the hot shutdown transition period. A significant amount of water is available from the CST (e.g. 2271 m^3). After restoration of power via AAC other plant makeup and cooling water sources will be made available.
Seismic Qualification	Not required.	—
Environmental Consideration	Need for station blackout event only and not for DBA conditions. See Regulatory Position 3.2.4. Procedures should be in place to effect the actions necessary to maintain acceptable environmental conditions for required equipment.	The AAC power source does not need plant service or cooling water for operation. It's a self (air) cooled, self-lubricated and self-controlled machine.

Table 1C-2 ABWR Design Compliance with RG 1.155 (Continued)

Requirements		Compliance
Appendix B—Guidance Regarding Systems/Components		
	Water Source (Existing Condensate Storage Tank or Alternate)	**SBO Recovery with AAC Power Source**
Capacity	Capability to provide sufficient water for core cooling in the event of a station blackout for the specified duration to meet § 50.63 and this regulatory guide.	The Condensate Storage Tank (CST) is capable of providing at least 8 hours of makeup water without replenishment. With the use of the AAC power sources other water sources are readily available for makeup, heat removal, and plant equipment cooling.
Quality Assurance	As indicated in Regulatory Position 3.5.	The ABWR design's immediate response to an SBO event does utilize a non-safety makeup water source (the CST). The AAC power source will allow the use of non-safety water sources.
Technical Specifications for Maintenance, Surveillance, Limiting Conditions, FSAR, etc.	Should be consistent with the Interim Commission Policy Statement on Technical Specifications (Federal Register Notice 52 FR 3789) as applicable.	No additional non-safety-related water sources are required during the duration of the 10- to 60-minute SBO event. Use of other sources during cold shutdown activities is optional.
Instrumentation and Monitoring	Must meet system functional requirements.	The makeup water source instrumentation and controls, used during the SBO duration, are safety-related and divisionally separated.
Single Failure	Not required.	—
Common Cause Failure (CCF)	Design should, to the extent practicable, minimize CCF between safety-related and non-safety-related systems.	The primary makeup water source (Condensate Storage Tank) and the secondary makeup water source (Suppression Pool), utilized during the 10 minute SBO duration, are physically, mechanically and environmentally separated from one another.

Table 1C-2 ABWR Design Compliance with RG 1.155 (Continued)

Requirements		Compliance
Appendix B—Guidance Regarding Systems/Components		
	Instrument Air (Compressed Air System)	**SBO Recovery with AAC Power Source**
Safety-Related Equipment (Compliance with IEEE-279)	Not required, but the existing Class 1E systems must continue to meet all applicable safety-related criteria.	Use of Plant Instrument Air/Compressed Air Systems during the 10 minute SBO duration is not required. Plant air systems availability is restored after 10 minutes by the AAC power source. Safety-related SRV nitrogen gas sources are available during the SBO event and are independent of non-safety air systems.
Redundancy	Not required.	—
Diversity	Not required.	—
Independence from Existing Safety-Related Systems	Ensure that the existing safety functions are not compromised, including the capability to isolate components, subsystems, or piping, if necessary.	Air systems are not required to operate during the SBO duration. The CTG unit does not depend on an air starter system nor air supplied services. The CTG does have a self-contained intake and exhaust system. This is provided by the machine power sources itself.
Seismic Qualification	Not required.	—
Environmental Consideration	Needed for station blackout event only and not for DBA conditions. See Regulatory Position 3.2.4. Procedures should be in place to effect the actions necessary to maintain acceptable environmental conditions for required equipment.	The CTG does not require special air or environmental control services before, during or after the SBO event.

Table 1C-2 ABWR Design Compliance with RG 1.155 (Continued)

Requirements		Compliance
Appendix B—Guidance Regarding Systems/Components		
	Water Source (Existing Condensate Storage Tank or Alternate)	**SBO Recovery with AAC Power Source**
Capacity	Sufficient compressed air to components, as necessary, to ensure that the core is cooled and appropriate containment integrity is maintained for the specified duration of station blackout to meet § 50.63 and Regulatory Guide 1.155.	Air service may be utilized later in the SBO recovery stage to reconfigure plant system to normal operation alignments.
Quality Assurance	As indicated in Regulatory Position 3.3.	Non-safety-related air systems are not utilized during the 10 minute SBO duration.
Technical Specifications for Maintenance, Surveillance, Limiting Conditions, FSAR, etc.	Should be consistent with the Interim Commission Policy Statement on Technical Specifications (Federal Register Notice 52 FR 3789) as applicable.	The CTG does not require air start services. The unit is started by a self-contained diesel engine starting system.
Instrumentation and Monitoring	Must meet system functional requirements.	Plant air system instrumentation, control and monitoring is not required during the 10 minute SBO duration.
Single Failure	Not required.	—
Common Cause Failure (CCF)	Design should, to the extent practicable, minimize CCF between safety-related and non-safety-related systems.	—

Table 1C-2 ABWR Design Compliance with RG 1.155 (Continued)

Requirements		Compliance
Appendix B—Guidance Regarding Systems/Components		
	Water Delivery System (Alternative to Auxiliary Feedwater System, RCIC System, or Isolation Condenser Makeup)	**SBO Recovery with AAC Power Source**
Safety-Related Equipment (Compliance with IEEE-279)	Not required, but the existing Class 1E systems must continue to meet all applicable safety-related criteria.	The ABWR AAC power source design response during the 10 minute SBO duration does not require additional water makeup sources beyond the CST and/or the Suppression Pool. Later in the SBO recovery sequence, the ABWR will utilize the normal plant water systems by powering selective divisions with the AAC power source (e.g. reactor service water and reactor cooling water systems).
Redundancy	Not required.	—
Diversity	Not required.	—
Independence from Existing Safety-Related Systems	Ensure that the existing safety functions are not compromised, including the capability to isolate components, subsystems, or piping, if necessary.	The powering of the normal plant water sources by the AAC power source during SBO will not be inconsistent or contrary with their current DBA design basis.
Seismic Qualification	Not required.	—
Environmental Consideration	Need for station blackout event only and not for DBA conditions. See Regulatory Position 3.2.4. Procedures should be in place to effect the actions necessary to maintain acceptable environmental conditions for required equipment.	The use of the normal plant cooling water systems will not require prior equipment environment controls or cooling. Their operation will be provided concurrently with the powering of water makeup sources.

Table 1C-2 ABWR Design Compliance with RG 1.155 (Continued)

Requirements		Compliance
Appendix B—Guidance Regarding Systems/Components		
	Water Delivery System (Alternative to Auxiliary Feedwater System, RCIC System, or Isolation Condenser Makeup)	**SBO Recovery with AAC Power Source**
Capacity	Capability to provide sufficient water for core cooling in the event of a station blackout for the specified duration to meet § 50.63 and this regulatory guide.	The emergency water makeup sources include the condensate storage tank and the suppression pool inventory. The normal plant water makeup sources (component and service water, etc.) are in addition to other alternative core and containment makeup sources (e.g., feedwater, fire pumps, makeup water systems, etc.) all of these systems can supply makeup or cooling water.
Quality Assurance	As indicated in Regulatory Position 3.5.	The plant normal makeup water systems are subject to quality assurance evaluations (e.g. CST and the SP).
Technical Specifications for Maintenance, Surveillance, Limiting Conditions, FSAR, etc.	Should be consistent with the Interim Commission Policy Statement on Technical Specifications (Federal Register Notice 52 FR 3789) as applicable.	Emergency water makeup systems are subject to Technical Specifications requirements.
Instrumentation and Monitoring	Must meet system functional requirements.	Instrumentation and controls for normal plant makeup water systems are qualified for their functional services.
Single Failure	Not required.	—
Common Cause Failure (CCF)	Design should, to the extent practicable, minimize CCF between safety-related and non-safety-related systems.	The use of additional plant water makeup systems (post SBO) will not degrade the operation or reliability of the necessary makeup systems (RCIC, HPCF, etc.). The CTG has sufficient capacity to power necessary shutdown loads and selective other safety and non-safety loads needed for water makeup.

Table 1C-3 ABWR Design Compliance with NUMARC 87-00 Guidelines

Requirements	Compliance
7.0 Coping Evaluations	
7.1.1 Coping Methods	
For purposes of this assessment, coping methods are separated into two different approaches. The first is referred to as the “AC-Independent” approach. In this approach, plants rely on available process steam, DC power, and compressed air to operate equipment necessary to achieve safe shutdown conditions (i.e., Hot Standby or Hot Shutdown, as appropriate) until offsite or emergency AC power is restored. A second approach is called the “Alternate AC” approach. This method is named for its use of equipment that is capable of being electrically isolated from the preferred offsite and emergency onsite AC power sources. Station blackout coping using the Alternate AC power approach would entail a short period of time in an AC-Independent state (up to one hour) while the operators initiate power from the backup source. Once power is available, the plant would transition to the Alternate AC state and provide decay heat removal until offsite or emergency AC power becomes available. The AC power sources used in the Alternate AC power approach would be subject to the Appendix B criteria including electrical isolation requirements in order to assure their availability in the event of a station blackout. Appendix A provides a definition of Alternate AC power sources. Appendix B provides detailed acceptance criteria for an Alternate AC power source.	The ABWR design utilizes the “Alternate AC (AAC)” approach as defined in Appendix A. The AAC power source will be available to be connected to the core inventory makeup and decay heat removal loads within ten (10) minutes. The AAC power source is capable of being electrically isolated from the preferred offsite and emergency onsite AC power sources and complies with the Appendix B criteria including electrical isolation requirements.

Table 1C-3 ABWR Design Compliance with NUMARC 87-00 Guidelines (Continued)

Requirements	Compliance
7.1.2 Coping Duration AC-Independent plants must meet the requirements of this methodology for at least four hours (or at least two hours for plants in both emergency AC group A and offsite power group P1). Plants using an Alternate AC power source must assess their ability to cope for one hour. However, if an Alternate AC power source can be shown by test to be available within 10 minutes of the onset of station blackout, then no coping assessment is required. Available within 10 minutes means that circuit breakers necessary to bring power to safe shutdown buses are capable of being actuated in the control room within that period.	ABWR design will demonstrate by test that the AAC CTG is capable of being available within ten (10) minutes of the onset of a SBO event and therefore no formal coping evaluation is necessary or required. All actions during the 10 minute period are safety-related and automatic. The ABWR design provides the operator in the main control room with the means to reconfigure the electrical distribution system including circuit breakers, and to connect the AAC power source to the necessary shutdown buses and loads within the ten (10) minute interval.

Table 1C-3 ABWR Design Compliance with NUMARC 87-00 Guidelines (Continued)

Requirements	Compliance
Appendix A — Definitions	
This appendix defines the terminology used throughout the guide.	The ABWR AAC power source design will meet the following requirements:
ALTERNATE AC POWER SOURCE. An alternating current (AC) power source that is available to and located at or nearby a nuclear power plant and meets the following requirements: (i) Is connectable to but not normally connected to the preferred or onsite emergency AC power systems (ii) Has minimal potential for common cause failure with offsite power or the onsite AC power sources (iii) Is available in a timely manner after the onset of station blackout (iv) Has sufficient capacity and reliability for operation of all systems necessary for coping with a station blackout and for the time required to bring and maintain the plant in safe shutdown (Hot Shutdown or Hot Standby, as appropriate) (v) Is inspected, maintained, and tested periodically to demonstrate operability and reliability as set forth in Appendix B	(i) The design is connectable to (but not normally connected to) the preferred or onsite emergency AC power sources. Two normally open breakers separate the AAC CTG from the safety-related onsite emergency power buses. A single normally open breaker separates the AAC CTG from the non-safety-related PIP buses (preferred power) (see Figure 8.3-1). (ii) The ABWR design has a minimal potential for common cause failure between preferred power or onsite AC power sources. The ABWR AAC power source is a diverse power supply to the normal onsite emergency DGs. The AAC power supply is totally independent of the preferred and onsite power sources. The AAC power source automatically starts and is available for loading in two minutes. The AAC power supply is connectable to a Class 1E bus through the actuation of two (2) manual operated circuit breakers. The AAC power source is normally electrically, physically, mechanically, and environmentally isolated from the preferred and onsite power sources. The AAC power source is normally used during LOPP and SBO events. However, the CTG can be used for a number of operational services (e.g. maintenance backup, etc.). (iii) The ABWR AAC power source is available in a timely manner after the onset of a SBO event. The AAC power source automatically starts on LOPP, attains required speed and voltage within two (2) minutes, and is capable of being connected to shutdown loads within ten (10) minutes.

Table 1C-3 ABWR Design Compliance with NUMARC 87-00 Guidelines (Continued)

Requirements	Compliance
	(iv) The ABWR AAC power source is rated a minimum of 9 MWe. The shutdown loads are less than 5 MWe. The CTG reliability is 0.95. The ABWR is expected to be in hot shutdown condition in twenty four (24) hours, and in cold shutdown condition in ninety-six (96) hours. The CTG, is designed to run indefinitely under SBO conditions at rated load. A seven-day fuel supply is available on the site for the CTG. (v) The ABWR AAC power source will be capable of being inspected, maintained and tested periodically to demonstrate its operability and reliability to guidelines set forth in Appendix B.
REQUIRED COPING DURATION. The time between the onset of station blackout and the restoration of offsite AC power to safe shutdown buses.	The ABWR AAC power source design does not require a formal SBO coping analysis. The AAC power source will be available to supply shutdown loads within ten (10) minutes. The current design requirements associated with DBA events assure that the plant will be able to cope with a ten (10) minute SBO event.
SAFE SHUTDOWN. For the purpose of this procedure safe shutdown is the plant conditions defined in plant technical specifications as Hot Standby or Hot Shutdown, as appropriate.	The ABWR design will assure safe shutdown plant conditions as defined by the Plant Technical Specifications and the definition in 10CFR50.63.
STATION BLACKOUT. Means the complete loss of alternating current (AC) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of onsite emergency AC power system). Station Blackout does not include the loss of available AC power to buses fed by station batteries through inverters or by Alternate AC power sources as defined in this appendix, nor does it assume a concurrent single failure or a design basis accident. At a multi-unit site, station blackout is assumed to occur in only one unit unless the emergency AC power sources are totally shared between the units.	The ABWR design accommodates the SBO definition and the other definitions defined in 10CFR50.63. The ABWR design utilizes the current available station batteries throughout the event. The station batteries will be recharged as necessary by the AAC power source.

Table 1C-3 ABWR Design Compliance with NUMARC 87-00 Guidelines (Continued)

Requirements	Compliance
Appendix B—Alternate AC Power Criteria	
This appendix describes the criteria that must be met by a power supply in order to be classified as an Alternate AC power source. The criteria focus on ensuring that station blackout equipment is not unduly susceptible to dependent failure by establishing independence of the AAC system from the emergency and non-Class 1E AC power systems.	
AAC Power Source Criteria	
B.1 The AAC system and its components need not be designed to meet Class 1E or safety system requirements. If a Class 1E EDG is used as an Alternate AC power source, this existing Class 1E EDG must continue to meet all applicable safety-related criteria.	The ABWR AAC power source is a non-safety-related CTG.
B.2 Unless otherwise provided in this criteria, the AAC system need not be protected against the effects of: (a) Failure or misoperation of mechanical equipment, including (i) fire, (ii) pipe whip, (iii) jet impingement, (iv) water spray, (v) flooding from a pipe break, (vi) radiation, pressurization, elevated temperature or humidity caused by high or medium energy pipe break, and (vii) missiles resulting from the failure of rotating equipment or high energy systems (b) Seismic events	The ABWR AAC power source is housed in a Uniform Building Code Building (Turbine Building). The AAC power source is physically, mechanically, electrically and environmentally separated from the preferred and onsite power sources. The AAC power source is protected from normal plant and site environmental perturbations (e.g., wind, temperature, etc.).
B.3 Components and subsystems shall be protected against the effects of likely weather-related events that may initiate the loss of offsite power event. Protection may be provided by enclosing AAC components within structures that conform with the Uniform Building Code, and burying exposed electrical cable run between buildings (i.e., connections between the AAC power source and the shutdown buses).	The ABWR AAC power source is protected against the effects of weather-related events that may initiate the loss of offsite power events. The AAC power source is located above the maximum flood level in the Turbine Building. The power and control cables from the CTG to the shutdown buses are routed separately from the offsite preferred power and control cables to the shutdown buses in the Reactor Building. The Turbine Building design basis capabilities will provide adequate protection for the enclosed equipment in compliance with their equipment design basis requirements.

Table 1C-3 ABWR Design Compliance with NUMARC 87-00 Guidelines (Continued)

Requirements	Compliance
B.4 Physical separation of AAC components from safety-related components or equipment shall conform with the separation criteria applicable for the unit's licensing basis.	The ABWR AAC power source design maintains physical separation between safety-related components or equipment and the CTG by adhering to applicable separation criteria used in the plant licensing basis.
Connectability to AC Power Systems	
B.5 Failure of AAC components shall not adversely affect Class 1E AC power systems.	The ABWR AAC power source design and its associated components failures will not adversely affect Class 1E AC power systems. Class 1E AC power system failures will not affect AAC power source operability.
B.6 Electrical isolation of AAC power shall be provided through an appropriate isolation device. If the AAC source is connected to Class 1E buses, isolation shall be provided by two circuit breakers in series (one Class 1E breaker at the Class 1E bus and one non-Class 1E breaker to protect the source).	The ABWR AAC power source is electrically isolated from the Class 1E power sources by two (2) circuit breakers in series (one Class 1E at the Class 1E buses and one non-Class 1E breaker at the CTG bus). Power to the breakers will be from appropriate DC sources.
B.7 The AAC power source shall not normally be directly connected to the preferred or onsite emergency AC power system for the unit affected by the blackout. In addition, the AAC system shall not be capable of automatic loading of shutdown equipment from the blacked-out unit unless licensed with such capability.	The ABWR AAC power source will not normally be connected to the preferred or onsite emergency AC power system. However, the COL applicant may use the CTG for other services (e.g. maintenance backup, etc.). The AAC power system will not automatically connect to or load any shutdown equipment on safety-related emergency buses. The AAC power source will automatically start upon occurrence of a LOPP event. It is connected automatically to the non-safety-related Plant Investment Protection (PIP) buses. It is capable of being manually connected to safety-related buses. It is also capable of being manually connected to non-safety power generation loads, condensate pumps, etc.).
Minimum Potential for Common Cause Failure	
B.8 There shall be minimal potential for common cause failure of the AAC power source(s). The following system features provide assurance that the minimal potential for common cause failure has been adequately addressed.	The ABWR AAC power source design contains a number of design and operational features which provide assurance of minimal potential for common cause failure.

Table 1C-3 ABWR Design Compliance with NUMARC 87-00 Guidelines (Continued)

Requirements	Compliance
(a) The AAC power system shall be equipped with a DC power source that is electrically independent from the blacked-out unit's preferred and Class 1E power system.	The AAC power system is equipped with sufficient plant or self-contained non-Class 1E DC power supplies (separate from the Class 1E DC power supplies) to facilitate successful operation. During normal operation, the plant electrical distribution systems will provide charging power to the plant battery systems.
(b) The AAC power system shall be equipped with an air start system, as applicable, that is independent of the preferred and the blacked-out unit's preferred and Class 1E power supply.	The AAC power system is equipped with a self-contained, independent diesel engine hydraulic starting system. This starter is designed for SBO conditions. The entire starter assembly is mounted on the same skid with the CTG.
(c) The AAC power system shall be provided with a fuel oil supply, as applicable, that is separate from the fuel oil supply for the onsite emergency AC power system. A separate day tank supplied from a common storage tank is acceptable provided the fuel oil is sampled and analyzed consistent with applicable standards prior to transfer to the day tank.	The AAC power supply is equipped with a fuel system separate from that of the DGs. An external fuel supply transfer system will also be provided. A seven (7) day supply of oil for use by the CTG to achieve safe shutdown is available on site. The CTG oil storage and transfer system is physically and mechanically independent of the DG oil storage and transfer system.
(d) If the AAC power source is an identical machine to the emergency onsite AC power source, active failures of the emergency AC power source shall be evaluated for applicability and corrective action taken to reduce subsequent failures.	The ABWR AAC power source is an independent and diverse power supply from the onsite emergency DG power sources. The AAC power source is a combustion turbine generator.
(e) No single point vulnerability shall exist whereby a likely weather-related event or single active failure could disable any portion of the onsite emergency AC power sources or the preferred power sources, and simultaneously fail the AAC power source(s).	The ABWR of the AAC power source design precludes single point vulnerabilities, weather-related events effects, or single active failures that could disable any portion of the onsite emergency AC power sources or the preferred power sources and simultaneously fail the AAC power source. The AAC power source is physically, mechanically, electrically and environmentally separated from the other plant power systems (e.g. circuit breaker separation, separate oil supplies, separate auto start circuits, etc.).

Table 1C-3 ABWR Design Compliance with NUMARC 87-00 Guidelines (Continued)

Requirements	Compliance
(f) The AAC power system shall be capable of operating during and after a station blackout without any support systems powered from the preferred power supply, or the blacked-out unit's Class 1E power source affected by the event.	The ABWR AAC power source design does not require preferred or onsite AC power sources to support the operation of the CTG unit. The CTG and its auxiliary support systems are maintained in their standby status by normal plant power sources. Upon reaching design speed and voltage, the CTG operation is supported by a self-powered internal control package. This package assures continued operation without external power or auxiliary service needs.
(g) The portions of the AAC power system subjected to maintenance activities shall be tested prior to returning the AAC power system to service.	The ABWR AAC power source is capable of being tested and will be periodically tested: (i) To demonstrate its reliability and its availability (ii) To demonstrate that it can be connected to shutdown buses within ten (10) minutes from the MCR (iii) To demonstrate the operability after maintenance has been performed on the CTG
Availability After Onset of Station Blackout	
B.9 The AAC power system shall be sized to carry the required shutdown loads for the required coping duration determined in Section 3.2.5, and be capable of maintaining voltage and frequency within limits consistent with established industry standards that will not degrade the performance of any shutdown system or component. At a multi-unit site, except for 1/2 shared or 2/3 emergency AC power configurations, an adjacent unit's Class 1E power source may be used as an AAC power source for the blacked-out unit if it is capable of powering the required loads at both units.	The ABWR AAC power source is designed to provide reliable power to shutdown loads during and after the SBO duration. The CTG will maintain supply voltage and frequency within the limits currently required during normal operation, and during loading transients, etc.

Table 1C-3 ABWR Design Compliance with NUMARC 87-00 Guidelines (Continued)

Requirements	Compliance
Capacity and Reliability	
B.10 Unless otherwise governed by technical specifications, the AAC power source shall be started and brought to operating conditions that are consistent with its function as an AAC source at intervals not longer than three months, following manufacturer's recommendations or in accordance with plant-developed procedures. Once every refueling outage, a timed start (within the time period specified under blackout conditions) and rated load capacity test shall be performed.	The ABWR AAC power source will be started and brought to operating conditions consistent with manufacturer's recommendations, the plant ORAP, or in accordance with specific plant developed procedures. This is a COL applicant interface item. The AAC power source is capable of being started and connected to the preferred power source for load capacity testing. The COL applicant will provide testing procedures based on plant specific ORAP objectives.
B.11 Unless otherwise governed by technical specifications, surveillance and maintenance procedures for the AAC system shall be implemented considering manufacturer's recommendations or in accordance with plant-developed procedures.	Plant specific surveillance and maintenance procedures based on the appropriate manufacturer's/vendor's recommendations, operational reliability assurance programs, plant maintenance effectiveness programs and plant operational requirements will be provided by the COL applicant.
B.12 Unless otherwise governed by technical specifications, the AAC system shall be demonstrated by initial test to be capable of powering required shutdown equipment within one hour of a station blackout event.	The ABWR AAC power source design will be tested to demonstrate that the CTG is capable of powering shutdown equipment within 10 minutes of the SBO event.

Table 1C-3 ABWR Design Compliance with NUMARC 87-00 Guidelines (Continued)

Requirements	Compliance
B.13 The Non-Class 1E AAC system should attempt to meet the target reliability and availability goals specified below, depending on normal system state. In this content, reliability and availability goals apply to the overall AAC system rather than individual machines, where a system may comprise more than one AAC power source. (a) Systems Not Normally Operated (Standby Systems) System reliability should be maintained at or below 0.95 per demand, as determined in accordance with NSAC-108 methodology (or equivalent). (b) Systems Normally operated (Online Systems) Availability: AAC systems normally online should attempt to be available to its associated unit at least 95% of the time the reactor is operating. Reliability: No reliability targets or standards are established for online systems.	The ABWR AAC power source satisfies the following reliability and availability goal: System reliability will be maintained at or above 0.95 per demand as determined in accordance with NSAC-108 methodology or its equivalent. Periodic testing and maintenance, to assure this reliability, will be performed.