3.0 Design of Structures, Components, Equipment and Systems

3.1 Conformance with NRC General Design Criteria

3.1.1 Summary Description

This section contains an evaluation of the principal design criteria of the ABWR Standard Plant as measured against the NRC General Design Criteria for Nuclear Power Plants, 10CFR50 Appendix A. The general design criteria, which are divided into six groups with the last criterion numbered 64, are intended to establish minimum requirements for the principal design criteria for nuclear power plants.

The NRC General Design Criteria were intended to guide the design of all water-cooled nuclear power plants; separate BWR-specific criteria are not addressed. As a result, the criteria are subject to a variety of interpretations. For this reason, there are some cases where conformance to a particular criterion is not directly measurable. In these cases, the conformance of the ABWR design to the interpretation of the criteria is discussed. For each criterion, a specific assessment of the plant design is made and a complete list of references is included to identify where detailed design information pertinent to that criterion is treated in this document.

Based on the contents herein, the design of the ABWR design fully satisfies and is in compliance with the NRC General Design Criteria.

3.1.2 Evaluation Against Criteria

3.1.2.1 Group I—Overall Requirements

3.1.2.1.1 Criterion 1—Quality Standards and Records

3.1.2.1.1.1 Criterion 1 Statement

Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance (QA) program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

3.1.2.1.1.2 Evaluation Against Criterion 1

Safety-related and non-safety-related structures, systems, and components are identified on Table 3.2-1. The total QA program is described in Chapter 17 and is applied to the safety-related items. The quality requirements for non-safety-related items are controlled by the QA program described in Chapter 17 in accordance with the functional importance of the item. The intent of the QA program is to assure sound engineering in all phases of design and construction through conformity to regulatory requirements and design bases described in the license application. In addition, the program assures adherence to specified standards of workmanship and implementation of recognized codes and standards in fabrication and construction. It also includes the observance of proper preoperational and operational testing and maintenance procedures, as well as the documentation of the foregoing by keeping appropriate records. The total QA program is responsive to and in conformance with the intent of the quality-related requirements of 10CFR50 Appendix B.

Structures, systems, and components are identified in Section 3.2 with respect to their location, service and their relationship to the safety-related or non-safety-related function to be performed. Recognized codes and standards are applied to the equipment per the safety classifications to assure meeting the required safety-related function.

Documents are maintained which demonstrate that all the requirements of the QA program are being satisfied. This documentation shows that appropriate codes, standards, and regulatory requirements are observed, specified materials are used, correct procedures are utilized, qualified personnel are provided, and the finished parts and components meet the applicable specifications for safe and reliable operation. These records are available so that any desired item of information is retrievable for reference. These records will be maintained during the life of the operating licenses.

The detailed QA program is in conformance with the requirements of Criterion 1.

For further discussion, see the following sections:

Chapter/Section	Title
1.2	General Plant Description
3.2	Classification of Structures, Components, and Systems

3.1.2.1.2 Criterion 2—Design Bases for Protection Against Natural Phenomena

3.1.2.1.2.1 Criterion 2 Statement

Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems and components shall reflect:

(1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated.

(2) Appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena.

(3) The importance of the safety functions to be performed.

3.1.2.1.2.2 Evaluation Against Criterion 2

Since the ABWR design is designated as a standard plant, the design bases for safety-related (Subsection 3.1.2.1.1.2) structures, systems, and components, cannot accurately reflect the most severe of the natural phenomena that have been historically reported for each possible site and their surrounding areas. However, the envelope of site-related parameters which blankets the majority of potential sites in the contiguous United States is defined in Chapter 2. The design bases for these structures, systems, and components reflect this envelope of natural phenomena, including appropriate combinations of the effects of normal and accident conditions with this envelope. The design bases is not required to meet the requirements of Criterion 2.

Detailed discussions of the various phenomena considered and design criteria developed are presented in the following sections:

Chapter/Section	Title
2.0	Summary of Site Characteristics
3.2	Classification of Structures, Components, and Systems
3.3	Wind and Tornado Loadings
3.4	Water Level (Flood) Design
3.5	Missile Protection
3.7	Seismic Design
3.8	Design of Seismic Category I Structures
3.9	Mechanical Systems and Components

Chapter/Section	Title
3.10	Seismic Qualifications of Seismic Category I Instrumentation and Electrical Equipment
3.11	Environmental Qualification of Safety-Related Mechanical and Electrical Equipment
7.1	Table 7.1-2

3.1.2.1.3 Criterion 3—Fire Protection

3.1.2.1.3.1 Criterion 3 Statement

Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat-resistant materials shall be used whenever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire fighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

3.1.2.1.3.2 Evaluation Against Criterion 3

Fires in the plant are prevented or mitigated by the use of non-combustible and heat-resistant materials such as metal cabinets, metal wireways, high melting point insulation, and flame-resistant markers for identification wherever practicable.

Cabling is suitably rated and cable tray loading is designed to avoid objectionable internal heat buildup. Cable trays are suitably separated to avoid the loss of redundant channels of protective cabling if a fire occurs. The arrangement of equipment in reactor protection channels provides physical separation to limit the effects of a fire.

Combustible supplies, such as logs, records, manuals, etc., are limited in such areas as the control room to amounts required for current operation, thus limiting the effect of a fire or explosion.

The plant Fire Protection System includes the following provisions:

(1) Automatic fire detection equipment in those areas where fire danger is greatest.

(2) Extinguishing services which include automatic actuation with manual override as well as manually-operated fire extinguishers.

The design of the Fire Protection System meets the requirements of Criterion 3. For further discussion, see the following sections:

Chapter/Section	Title
3.8.2.6	Materials, Quality Control and Special Construction Techniques
7	Instrumentation and Control Systems
8	Electric Power
9.5	Fire Protection System
Appendix 9A	Fire Hazard Analysis
13	Conduct of Operations

3.1.2.1.4 Criterion 4—Environmental and Dynamic Effects Design Bases

3.1.2.1.4.1 Criterion 4 Statement

Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents (LOCAs). These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the NRC demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

3.1.2.1.4.2 Evaluation Against Criterion 4

Essential (see introduction to Section 3.6) structures, systems, and components are designed to accommodate the dynamic effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, and postulated pipe failure accidents, including LOCAs.

These structures, systems, and components are appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failure. The effects of missiles from sources external to the ABWR Standard Plant are also considered. Design requirements specify the time which each must survive the extreme environmental conditions following a LOCA. The design of these structures, systems, and components meets the requirements of Criterion 4.

Subsection 3.6.3 identifies the requirements for the piping that is to be excluded from postulation of pipe ruptures for design of the plant against dynamic effects from the associated pipe ruptures.

For further discussion, see the following sections:

Chapter/Section	Title
2.0	Summary of Site Characteristics
3.3	Wind and Tornado Loadings
3.4	Water Level (Flood) Design
3.5	Missile Protection
3.6	Protection Against Dynamic Effects Associated with the Postulated Rupture of Piping
3.8	Design of Seismic Category I Structures
3.11	Environmental Qualification of Safety-Related Mechanical and Electrical Equipment
5.2	Integrity of Reactor Coolant Pressure Boundary
6	Engineered Safety Features
7	Instrumentation and Control Systems
8	Electric Power

3.1.2.1.5 Criterion 5—Sharing of Structures, Systems, and Components

3.1.2.1.5.1 Criterion 5 Statement

Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair the ability to perform the safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

3.1.2.1.5.2 Evaluation Against Criterion 5

Since the ABWR design is for a single-unit station, this criterion is not applicable.

3.1.2.2 Group II—Protection by Multiple Fission Product Barriers

3.1.2.2.1 Criterion 10—Reactor Design

3.1.2.2.1.1 Criterion 10 Statement

The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

3.1.2.2.1.2 Evaluation Against Criterion 10

The reactor core components consist of fuel assemblies, control rods, incore ion chambers, neutron sources, and related items. The mechanical design is based on conservative application of stress limits, operating experience, and experimental test results. The fuel is designed to provide integrity over a complete range of power levels, including transient conditions. The core is sized with sufficient heat transfer area and coolant flow to ensure that fuel design limits are not exceeded under normal conditions or anticipated operational occurrences.

The Reactor Protection System (RPS) is designed to monitor certain reactor parameters, sense abnormalities, and scram the reactor, thereby preventing fuel design limits from being exceeded when trip points are exceeded. Scram trip setpoints are selected on operating experience and by the safety design basis. There is no case in which the scram trip setpoints allow the core to exceed the thermal-hydraulic safety limits. Power for the RPS is supplied by four independent uninterruptible AC power supplies. An alternate power source and battery are available for each bus. The reactor will scram on loss of power or hydraulic pressure.

An analysis and evaluation has been made of the effects upon core fuel following adverse plant operating conditions. The results of abnormal operational transients (Chapter 15) show that the minimum critical power ratio (MCPR) does not fall below the transient MCPR limit, thereby satisfying the transient design basis.

The reactor core and associated coolant, control, and protection systems are designed to assure that the specified acceptable fuel design limits are not exceeded during conditions of normal or abnormal operation and, therefore, meet the requirements of Criterion 10.

For further discussion, see the following sections:

Chapter/Section	Title
1.2	General Plant Description
4.2	Fuel Design System
4.3	Nuclear Design

4.4	Thermal and Hydraulic Design
5.4.1	Reactor Recirculation System
5.4.6	Reactor Core Isolation Cooling System
5.4.7	Residual Heat Removal System
7.2	Reactor Protection System
7.3	Engineered Safety Feature Systems—Instrumentation and Control
15	Accident Analyses

3.1.2.2.2 Criterion 11—Reactor Inherent Protection

3.1.2.2.2.1 Criterion 11 Statement

The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

3.1.2.2.2.2 Evaluation Against Criterion 11

The reactor core is designed to have a reactivity response that regulates or damps changes in power level and spatial distributions of power production to a level consistent with safe and efficient operation.

The inherent dynamic behavior of the core is characterized in terms of:

(1) Fuel temperature or Doppler coefficient.

(2) Moderator void coefficient.

(3) Moderator temperature coefficient.

The combined effect of these coefficients in the power range is termed the "power coefficient."

Doppler reactivity feedback occurs simultaneously with a change in fuel temperature and opposes the power change that caused it; it contributes to system stability. Since the Doppler reactivity opposes load changes, it is desirable to maintain a large ratio of moderator void coefficient to Doppler coefficient for optimum load-following capability. The BWR has an inherently large moderator-to-Doppler coefficient ratio which permits use of coolant flow rate for load following.

In a BWR, the moderator void coefficient is of importance during operation at power. Nuclear design requires the void coefficient inside the fuel channel to be negative. The negative void reactivity coefficient provides an inherent negative feedback during power transients. Because of the large negative moderator coefficient of reactivity, the BWR has a number of inherent advantages, such as:

(1) The use of coolant flow as opposed to control rods for load following.

(2) The inherent self-flattening of the radial power distribution.

(3) The ease of control.

(4) The spatial xenon stability.

The reactor is designed so that the moderator temperature coefficient is small and positive in the cold condition; however, the overall power reactivity coefficient is negative. Typically, the power coefficient at full power is about –0.04 Δk/k/ ΔP/P at the beginning of life and about –0.03 Δk/k/ ΔP/P at 10,000 MW·d/t. These values are well within the range required for adequate damping of power and spatial xenon disturbances.

The reactor core and associated coolant system are designed so that in the power operating range, prompt inherent dynamic behavior tends to compensate for any rapid increase in reactivity in accord with Criterion 11.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
4.3	Nuclear Design
4.4	Thermal and Hydraulic Design
7.2	Reactor Protection System, Instrumentation and Control
7.3	Engineered Safety Feature Systems, Instrumentation and Control
7.7	Control Systems not Required for Safety
15	Accident Analyses

3.1.2.2.3 Criterion 12—Suppression of Reactor Power Oscillations

3.1.2.2.3.1 Criterion 12 Statement

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

3.1.2.2.3.2 Evaluation Against Criterion 12

The reactor core is designed to ensure that no power oscillation will cause fuel design limits to be exceeded. The power reactivity coefficient is the composite simultaneous effect of the fuel temperature or Doppler coefficient, moderator void coefficient, and moderator temperature coefficient to the change in power level. It is negative and well within the range required for adequate damping of power and spatial xenon disturbances. Analytical studies indicate that for large BWRs, under-damped, unacceptable power distribution behavior could only be expected to occur with power coefficients more positive than about $-0.01\ \Delta k/k/\Delta P/P$. Operating experience has shown large BWRs to be inherently stable against xenon induced power instability. The large negative coefficients provide:

(1) Good load following with well-damped behavior and little undershoot or overshoot in the heat transfer response.

(2) Load following with recirculation flow control.

(3) Strong damping of spatial power disturbances.

The RPS design provides protection from excessive fuel cladding temperatures and protects the reactor coolant pressure boundary (RCPB) from excessive pressures which threaten the integrity of the system. Local abnormalities are sensed, and, if protection system limits are reached, corrective action is initiated through an automatic scram. High integrity of the protection system is achieved through the combination of logic arrangement, trip channel redundance, power supply redundancy, and physical separation.

The reactor core and associated coolant, control, and protection systems are designed to suppress any power oscillations which could result in exceeding fuel design limits. These systems assure that Criterion 12 is met.

For further discussions, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
4.3	Nuclear Design
4.4	Thermal and Hydraulic Design
7.2	Reactor Protection System—Instrumentation and Control
7.3	Engineered Safety Feature Systems—Instrumentation and Control
7.7	Control Systems not Required for Safety
15	Accident Analyses

3.1.2.2.4 Criterion 13—Instrumentation and Control

3.1.2.2.4.1 Criterion 13 Statement

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for (1) normal operation, (2) anticipated operational occurrences, and (3) accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the RCPB, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

3.1.2.2.4.2 Evaluation Against Criterion 13

The neutron flux in the reactor core is monitored by five subsystems. The Startup Range Neutron Monitor (SRNM) Subsystem measures the flux from startup through 15% power (into the power range). The power range is monitored by many detectors which make up the Local Power Range Monitor (LPRM) Subsystem. The output from these detectors is used in many ways. The output of selected core-wide sets of detectors is averaged to provide a core-average neutron flux. This output is called the Average Power Range Monitor (APRM) Subsystem. The Flow Rate Subsystem (FRS) provides the control and reference signal for the APRM core flow-rate dependent trips. The Automated Traversing Incore Probe (ATIP) Subsystem provides a means for calibrating the LPRM Subsystem. Both the SRNM and APRM Subsystems generate scram trips to the RPS. They also generate rod-block trips.

The RPS protects the fuel barriers and the nuclear process barrier by monitoring plant parameters and causing a reactor scram when predetermined setpoints are exceeded. Separation of the scram and normal rod control function prevents failures in the reactor manual control circuity from affecting the scram circuitry.

To provide protection against the consequences of accidents involving the release of radioactive materials from the fuel and RCPB, the Leak Detection and Isolation System (LDS) initiates automatic isolation of appropriate pipelines whenever monitored variables exceed preselected operational limits.

The leakage limits for the Reactor Coolant System (Subsection 3.1.2.2.6.2) are established so that appropriate action can be taken to ensure the integrity of the RCPB. The monitored leakage rates are classified as identified and unidentified, which corresponds, respectively, to the flow to the equipment drain and floor drain sumps. The permissible total leakage rate limit to these sumps is based upon the makeup capabilities of various reactor component systems. High pump fill-up rate and pump-out rate are alarmed in the main control room. The unidentified leakage rate (Chapter 5) is less than the value that has been conservatively calculated to be a minimum leakage from a crack large enough to propagate rapidly, but which still allows time for identification and corrective action before integrity of the process barrier is threatened.

The Process Radiation Monitoring System monitors radiation levels of various processes and provides trip signals to the RPS and LDS whenever pre-established limits are exceeded.

Adequate instrumentation has been provided to monitor system variables in the reactor core, RCPB, and reactor containment. Appropriate controls have been provided to maintain the variables in the operating range and to initiate the necessary corrective action in the event of abnormal operational occurrence or accident.

Additional information on the instrumentation and controls is given in Chapter 7.

3.1.2.2.5 Criterion 14—Reactor Coolant Pressure Boundary

3.1.2.2.5.1 Criterion 14 Statement

The RCPB shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, rapidly propagating failure, and gross rupture.

3.1.2.2.5.2 Evaluation Against Criterion 14

The piping and equipment pressure parts within the RCPB (as defined by Section 50.2 of 10CFR50) are designed, fabricated, erected, and tested in accordance with 10CFR50.55a to provide a high degree of integrity throughout the plant lifetime. Section 3.2 classifies systems and components within the RCPB as Quality Group A. The design requirements and codes and standards applied to this quality group ensure high integrity in keeping with the safety-related function.

In order to minimize the possibility of brittle fracture within the RCPB, the fracture toughness properties and the operating temperature of ferritic materials are controlled to ensure adequate toughness. Section 5.2 describes the methods utilized to control toughness properties of the RCPB materials. Materials are to be impact tested in accordance with ASME Boiler and

Pressure Vessel (B&PV) Code Section III, where applicable. Where RCPB piping penetrate the containment, the fracture toughness temperature requirements of the RCPB materials apply.

Piping and equipment pressure parts of the RCPB are assembled and erected by welding unless applicable codes permit flanged or screwed joints. Welding procedures are employed which produce welds of complete fusion free of unacceptable defects. All welding procedures, welders, and welding machine operators used in producing press-containing welds are qualified in accordance with the requirements of ASME B&PV Code Section IX for the materials to be welded. Qualification records, including the results of procedure and performance qualification tests and identification symbols assigned to each welder, are maintained.

Section 5.2 contains the detailed material and examination requirements for the piping and equipment of the RCPB prior to and after its assembly and erection. Leakage testing and surveillance is accomplished as described in the evaluation against Criterion 30 of the General Design Criteria.

The design, fabrication, erection, and testing of the RCPB assure an extremely low probability of failure of abnormal leakage, thus satisfying the requirements of Criterion 14.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
3	Design of Structures, Components, Equipment, and Systems
5.2	Integrity of Reactor Coolant Pressure Boundary
5.3	Reactor Vessel
5.4.1	Reactor Recirculation System
15	Accident Analyses
17	Quality Assurance

3.1.2.2.6 Criterion 15—Reactor Coolant System Design

3.1.2.2.6.1 Criterion 15 Statement

The Reactor Coolant System (RCS) and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences (AOO).

3.1.2.2.6.2 Evaluation Against Criterion 15

The RCS, as identified in Section 5.1, consists mainly of the Nuclear Steam Supply Systems (NSSS), comprised of the reactor vessel and appurtenances, the Reactor Recirculation System (RCS) and the Nuclear Boiler System (NBS) including the main steamlines, feedwater lines and pressure-relief discharge system; the Reactor Core Isolation Cooling (RCIC) System; the Residual Heat Removal (RHR) System; and the Reactor Water Cleanup (CUW) System.

The auxiliary, control, and protection systems associated with the RCS act to provide sufficient margin to assure that the design conditions of the RCPB are not exceeded during any condition of normal operation, including anticipated operational occurrences. As described in the evaluation of Criterion 13, instrumentation is provided to monitor essential variables to ensure that they are within prescribed operating limits. If the monitored variables exceed their predetermined settings, the auxiliary, control, and protection systems automatically respond to maintain the variables and systems within allowable design limits.

An example of the integrated protective action scheme which provides sufficient margin to assure that the design conditions of the RCPB are not exceeded is the automatic initiation of the pressure relief system of the NBS upon receipt of an overpressure signal. To accomplish overpressure protection of the reactor pressure vessel system and RCPB, a number of pressure-operated relief valves are provided that can discharge steam from the main steamlines to the suppression pool. The pressure relief system also provides for automatic depressurization of the RCS in the event of an LOCA in which the vessel is not depressurized by the accident. The depressurization of the RCS in this situation allows operation of the low-pressure emergency core cooling systems to supply enough cooling water to adequately cool the core. In a similar manner, other auxiliary, control, and protection systems provide assurance that the design conditions of the RCPB are not exceeded during any conditions of normal operation, including AOOs.

The application of appropriate codes and standards and high quality requirements to the RCPB and the design features of its associated auxiliary, control, and protection systems assure that the requirements of Criterion 15 are satisfied.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
3	Design of Structure, Components, Equipment, and Systems
5.2.2	Overpressurization Protection
5.2.5	RCPB and Core Cooling Systems Leakage Detection

Chapter/Section	Title
5.3	Reactor Vessel
5.4.1	Reactor Recirculation System
7.3	Engineered Safety Feature Systems—Instrumentation and Control
15	Accident Analyses

3.1.2.2.7 Criterion 16—Containment Design

3.1.2.2.7.1 Criterion 16 Statement

Reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

3.1.2.2.7.2 Evaluation Against Criterion 16

The Primary Containment System consists of the following major structures and components:

(1) A leaktight primary containment vessel (PCV) enclosing the reactor pressure vessel, the RCPB, and other branch connections of the reactor primary coolant system. The PCV is a cylindrical steel-lined reinforced concrete structure with a removable steel head and has upper and lower drywell zones, diaphragm floor (D/F) and annular suppression chamber (or wetwell zone) under upper drywell separated by the D/F.

(2) A suppression pool containing a large amount of water used to rapidly condense steam from a reactor vessel blowdown or from a break in a major pipe.

(3) Associated containment penetrations and isolation devices.

The drywell and wetwell zones condense the steam and contain fission product releases from the postulated design bases accident (i.e., the double-ended rupture of the largest pipe in the primary coolant system). The leaktight PCV prevents the release of fission products to the environment.

The secondary containment boundary of the reactor building, which completely encloses and structurally integrates the PCV, provides additional radiation shielding to protect operating personnel and the public and also protects the PCV from weather and external missiles.

Temperature and pressure in the PCV are limited following an accident by using the RHR System to condense steam in the containment atmosphere and to cool the suppression pool water.

The design of the containment systems meets the requirements of Criterion 16.

For further discussion, see the following sections.

Chapter/Section	Title
1.2	General Plant Description
3.8.2	Steel Components of the Reinforced Concrete Containment
6.2	Containment Systems
15	Accident Analyses

3.1.2.2.8 Criterion 17—Electric Power Systems

3.1.2.2.8.1 Criterion 17 Statement

An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that:

(1) Specified acceptable fuel design limits and design conditions of the RCPB are not exceeded as a result of anticipated operational occurrences.

(2) The core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The onsite electric power supplies, including the batteries, and the onsite electric distribution system shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located to minimize to the extent practical the likelihood of simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current (AC) power supplies and the other offsite electric power circuit to assure that specified acceptable fuel design limits and design conditions of the RCPB are not exceeded. One of these circuits shall be designed to be available within a few seconds following a LOCA to assure that the core cooling, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of or coincident with (1) the loss of power generated by the nuclear power unit, (2) the loss of power from the transmission network, or (3) the loss of power from the onsite electric power supplies.

3.1.2.2.8.2 Evaluation Against Criterion 17

3.1.2.2.8.2.1 Onsite Electric Power System

There are three independent AC load groups provided to assure independence and redundancy of equipment function. These meet the safety requirements, assuming a single failure, since:

(1) Each load group is independently capable of isolation from the offsite power sources.

(2) Each load group has separate circuits to independent power sources.

For each of the three AC load groups, there are independent batteries which furnish DC load and control power for the corresponding divisions. An additional battery furnishes DC load and control power for the safety system logic and control (SSLC) Division IV bus.

The reactor protection instrumentation is powered from four independent AC/DC power sources.

The onsite electric power systems are designed to meet the requirements of Criterion 17. For further discussion, see the following sections:

Chapter/Section	Title
1.2	General Plant Description
3.10	Seismic Qualification of Seismic Category I Instrumentation and Electrical Equipment
3.11	Environmental Qualification of Safety-Related Mechanical and Electrical Equipment
8.3	Onsite Power Systems

3.1.2.2.8.2.2 Offsite Electric Power System

A part of the design of the offsite power systems is out of the scope of the ABWR design. A description of the offsite power system and the scope split between the ABWR Standard Plant design and the COL applicant design is defined in Subsection 8.2.1.1 and 8.2.1.2. The ABWR Standard Plant interfaces requirements are addressed in Subsection 8.2.3.

3.1.2.2.9 Criterion 18—Inspection and Testing of Electric Power Systems

3.1.2.2.9.1 Criterion 18 Statement

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to periodically test:

(1) The operability and functional performance of the components of the systems such as onsite power sources, relays, switches, and buses.

(2) The operability of the systems as a whole and, under conditions as close to design as practical, the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.

3.1.2.2.9.2 Evaluation Against Criterion 18

The important power supply buses and associated normal preferred, alternate, and standby AC power supplies are arranged for periodic inspection and testing of each load group independently. The testing procedure includes a bus transfer from normal preferred power supply to alternate preferred power supply, and simulates a loss of preferred power (LOPP) signal or a LOCA signal to start the diesel generator bringing it to operating condition. Full load testing of the diesel generator can be performed by manually synchronizing the generator to the normal preferred power supply. These tests are performed periodically to prove the engineered safety system operability.

Design of the standby power systems provides testability in accordance with the requirements of Criterion 18. For further discussion, see the following sections:

Chapter/Section	Title
8.3	Onsite Power Systems
14	Initial Test Program

3.1.2.2.10 Criterion 19—Control Room

3.1.2.2.10.1 Criterion 19 Statement

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room shall be provided:

(1) With a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown.

(2) With a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

3.1.2.2.10.2 Evaluation Against Criterion 19

The control room contains the following equipment: (1) controls and necessary surveillance equipment for operation of the plant functions such as the reactor and its auxiliary systems, (2) engineered safety features, (3) turbine generator, (4) steam and power conversion systems, and (5) station electrical distribution boards.

The control room is located in a Seismic Category I Control Building. Safe occupancy of the control room during abnormal conditions is provided for in the design. Adequate shielding is provided to maintain tolerable radiation levels in the control room in the event of a design basis accident for the duration of the accident.

The control building ventilation system has redundant equipment and provides radiation detectors and smoke detectors with appropriate alarms and interlocks. The control room intake air can be filtered through high-efficiency particulate air/absolute (HEPA) and charcoal filters.

The control room is continuously occupied by qualified operating personnel under all operating and accident conditions. In the unlikely event that the control room must be vacated and access is restricted, instrumentation and controls are provided outside the control room which can be utilized to safely perform a hot shutdown and a subsequent cold shutdown of the reactor.

The control room design meets the requirements of Criterion 19.

For further discussion, see the following sections:

Chapter/Section	Title
1.2	General Plant Description
3.8.4	Other Seismic Category I Structures
7	Instrumentation and Control Systems

Chapter/Section	Title
7.4.1.4 and 7.4.2.4	Remote Shutdown System— Instrumentation and Controls
6.4	Habitability Systems
9.4.1	Control Building Ventilation System
9.5.1	Fire Protection System
12.3.2	Shielding
12.3.3	Ventilation

3.1.2.3 Group III—Protection and Reactivity Control System

3.1.2.3.1 Criterion 20—Protection System Functions

3.1.2.3.1.1 Criterion 20 Statement

The protection system shall be designed

(1) To initiate automatically the operation of appropriate systems, including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences.

(2) To sense accident conditions and initiate the operation of systems and components important to safety.

3.1.2.3.1.2 Evaluation Against Criterion 20

The Reactor Protection System (RPS) is designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the RCPB barrier. Fuel damage is prevented by initiation of an automatic reactor shutdown if monitored variables of nuclear steam supply systems (Subsection 3.1.2.2.6.2) exceed pre-established limits of anticipated operational occurrences. Scram trip settings are selected and verified to be far enough above or below operating levels to provide proper protection but not be subject to spurious scrams. The RPS includes the ride through power sources, sensors, transmitters, bypass circuity, and switches that signal the control rod system to scram and shut down the reactor. The scrams initiated by neutron monitoring system variables, nuclear steam supply systems (NSSS) high pressure, high suppression pool temperature, turbine stop valve closure, turbine control valve fast closure, and reactor vessel low-water level prevent fuel damage following abnormal operational transients. Specifically, these process parameters initiate a scram in time to prevent the core from exceeding thermal hydraulic safety limits during abnormal operational transients. Response by the RPS is prompt and the total scram time is short. Control rod scram motion starts in about 290 milliseconds after the high flux setpoint is exceeded.

A fully withdrawn control rod traverses 60% of its full stroke in sufficient time to assure that acceptable fuel design limits are not exceeded.

In addition to the RPS, which provides for automatic shutdown of the reactor to prevent fuel damage, protection systems are provided to sense accident conditions and initiate automatically the operation of other systems and components important to safety. Systems such as the Emergency Core Cooling System (ECCS) are initiated automatically to limit the extent of fuel damage following a LOCA. Other systems automatically isolate the reactor vessel or the containment to prevent the release of significant amounts of radioactive materials from the fuel and the RCPB. The controls and instrumentation for the ECCS and the isolation systems are initiated automatically when monitored variables exceed pre-selected operational limits.

The design of the protection system satisfies the functional requirements as specified in Criterion 20.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
4.6	Functional Design of Reactivity Control Systems
5.2.2	Overpressurization Protection
5.4.5	Main Steamline Isolation System
6.3	Emergency Core Cooling System
7.2	Reactor Protection System
7.3.1.1 and 7.3.2.1	Emergency Core Cooling Systems— Instrumentation and Control
7.3.1.1.2 and 7.3.2.2	Leak Detection and Isolation System—Instrumentation and Controls

Chapter/Section	Title
7.6.1.2 and 7.6.2.2	Process Radiation Monitoring and System—Instrumentation and Controls
15	Accident Analyses

3.1.2.3.2 Criterion 21—Protection System Reliability and Testability

3.1.2.3.2.1 Criterion 21 Statement

The protection system shall be designed for functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that:

(1) No single failure results in loss of the protection function.

(2) Removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated.

The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

3.1.2.3.2.2 Evaluation Against Criterion 21

RPS design provides assurance that, through redundancy, each channel has sufficient reliability to fulfill the single-failure criterion. No single component failure, intentional bypass maintenance operation, calibration operation, or test to verify operational availability impairs the ability of the system to perform its intended safety function. Additionally, the system design assures that when a scram trip point is exceeded, there is a high scram probability. However, should a scram not occur, other monitored components scram the reactor if their trip points are exceeded. There is sufficient electrical and physical separation between channels and between logics monitoring the same variable to prevent environmental factors, electrical transients, and physical events from impairing the ability of the system to respond correctly.

The RPS includes design features that permit inservice testing. This ensures the functional reliability of the system should the reactor variable exceed the corrective action setpoint.

The RPS initiates an automatic reactor shutdown if the monitored plant variables exceed preestablished limits. This system is arranged as four separately powered divisions.

Each division has a logic which can produce an automatic trip signal. The logic scheme is a two-out-of-four arrangement. The RPS can be tested during reactor operation. Manual scram testing is performed by operating one of the four manual scram controls; this tests one division. The total test verifies the ability to de-energize the scram pilot valve solenoids. Indicating lights verify that the actuator contacts have opened. This capability for a thorough testing program significantly increases reliability.

Control rod drive operability can be tested during normal reactor operation. Rod position indicators and in core neutron detectors are used to verify control rod movement. Each control rod can be withdrawn one step and then reinserted to the original position without significantly perturbing the nuclear steam supply systems at most power levels. One control rod is tested at a time. Control rod mechanism overdrive demonstrates rod-to-drive coupling integrity. Hydraulic supply subsystem pressures can be observed on control room instrumentation. More importantly, the HCU scram accumulator level is continuously monitored.

The main steamline isolation valves may be tested during full reactor operation. Individually, they can be closed to 90% of full-open position without affecting the reactor operation. If reactor power is reduced sufficiently, the isolation valves may be fully closed. During refueling operation, valve leakage rates can be determined.

The RHR System testing can be performed during normal operation. Main system pumps can be evaluated by taking suction from the suppression pool and discharging through test lines back to the suppression pool. System design and operating procedures also permit testing the supply valves of the three RHR lines. The lower pressure flooder mode can be tested after reactor shutdown. Each active component of the ECCS provided to operate in a design basis accident is designed to be operable for test purposes during normal operation of the nuclear system.

The high functional reliability, redundancy, and inservice testability of the protection system satisfy the requirements specified in Criterion 21.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
4.6	Functional Design of Reactivity Control Systems
5.4.5	Main Steamline Isolation Valve System
5.4.7	Residual Heat Removal System
6.2	Containment Systems
6.3	Emergency Core Cooling Systems

Chapter/Section	Title
7.2	Reactor Protection System
7.3.1.1 and 7.3.2.1	Emergency Core Cooling Systems—Instrumentation and Control
7.3.1.1.2 and 7.3.2.2	Leak Detection and Isolation System—Instrumentation and Controls
7.6.1.2 and 7.6.2.2	Process Radiation Monitoring—Instrumentation and Controls
15	Accident Analyses

3.1.2.3.3 Criterion 22—Protection System Independence

3.1.2.3.3.1 Criterion 22 Statement

The protection system shall be designed to assure that the effects of natural phenomena and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

3.1.2.3.3.2 Evaluation Against Criterion 22

Components of protection systems are designed so that the mechanical, thermal and radiological environment resulting from any accident situation in which the components are required to function do not interfere with the operation of that function.

The redundant sensors are electrically and physically separated. Only circuits of the same division are run in the same raceway. Data communication signals are carried by fiber optic medium to assure control signal isolation.

The RPS is designed to permit maintenance and diagnostic work while the reactor is operating without restricting the plant operation or hindering the output of safety functions. The flexibility in design afforded the protection system allows operational system testing by the use of an independent input for each actuator logic. When an essential monitored variable exceeds its scram trip point, it is sensed by four independent sensors each located in a separate instrumentation channel. A bypass of any single channel is permitted for maintenance operation, calibration operation, test, etc. This leaves three channels per monitored variable, each of which is capable of initiating a scram. Only two actuator logics must trip to initiate a scram. Thus, the two-out-of-four arrangement assures that a scram occurs as a monitored variable exceeds its scram setting.

The protection system meets the design requirements for functional and physical independence as specified in Criterion 22.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
4.6	Functional Design of Reactivity Control Systems
5.4.5	Main Steamline Isolation Valve System
5.4.7	Residual Heat Removal System
6.3	Emergency Core Cooling Systems
7.2	Reactor Protection System
7.3.1.1 and 7.3.2.1	Emergency Core Cooling System—Instrumentation and Controls
7.3.1.2 and 7.3.2.2	Leak Detection and Isolation System—Instrumentation and Controls
7.6.1.2 and 7.6.2.2	Process Radiation Monitoring— Instrumentation and Controls
15	Accident Analyses

3.1.2.3.4 Criterion 23—Protection System Failure Modes

3.1.2.3.4.1 Criterion 23 Statement

The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

3.1.2.3.4.2 Evaluation Against Criterion 23

The RPS is designed to fail into a safe state. Use of an independent channel for each actuator logic allows the system to sustain any logic channel failure without preventing other sensors monitoring the same variable from initiating a scram. Any two-out-of-four logic channel trips initiate a scram. Intentional bypass for maintenance or testing causes the scram logic to revert to two-out-of-three. A failure of any one RPS input or subsystem component produces a trip in one channel. This condition is insufficient to produce a reactor scram, but the system is ready to perform its protective function upon trip of another channel.

The environmental conditions in which the instrumentation and equipment of the RPS must operate were considered in establishing the component specifications. Instrumentation specifications are based on the worst expected ambient conditions in which the instruments must operate.

The failure modes of the RPS are such that it fails into a safe state as required by Criterion 23.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
7.2	Reactor Protection System

3.1.2.3.5 Criterion 24—Separation of Protection and Control Systems

3.1.2.3.5.1 Criterion 24 Statement

The protection system shall be separated from control systems to the extent that failure of any single control system component or channel or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited to assure that safety is not significantly impaired.

3.1.2.3.5.2 Evaluation Against Criterion 24

There is separation between the RPS and the process control systems. Logic channels and actuator logics of the RPS are not used directly for automatic control of process systems. Sensor outputs may be shared, but each signal is optically isolated before entering a redundant or non-safety channel interface. Therefore, failure in the controls and instrumentation of process systems cannot induce failure of any portion of the protection system. Scram reliability is designed into the RPS and hydraulic control unit for the control rod drive. The scram signal and mode of operation override all other signals.

The systems that isolate the containment and reactor pressure vessel are designed so that any one failure, maintenance operation, calibration operation, or test to verify operational availability does not impair the functional ability of the isolation systems to respond to essential variables.

Process radiation monitoring is provided on process liquid and gas lines that may serve as discharge routes for radioactive materials. Four instrumentation channels are used to prevent an inadvertent scram and isolation as a result of instrumentation malfunctions. The output trip signals from each channel are combined in such a way that two channels must signal high radiation to initiate scram and main steam isolation.

The protection system is separated from control systems as required in Criterion 24.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
4.6	Functional Evaluation of Reactivity Control Systems
6.3	Emergency Core Cooling Systems
7.2	Reactor Trip System
7.3.1.1 and 7.3.2.1	Emergency Core Cooling System— Instrumentation and Controls
7.3.1.1.2 and 7.3.2.2	Leak Detection and Isolation System—Instrumentation and Controls
7.6.1.2 and 7.6.2.2	Process Radiation Monitoring—and Instrumentation and Controls
7.7.1.2 and 7.7.2.2	Rod Control and Information System—Instrumentation and Controls

3.1.2.3.6 Criterion 25—Protection System Requirements for Reactivity Control Malfunctions

3.1.2.3.6.1 Criterion 25 Statement

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control system such as accidental withdrawal (not ejection or dropout) of control rods.

3.1.2.3.6.2 Evaluation Against Criterion 25

The RPS provides protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the RCPB or the primary containment vessel pressure boundary. Any monitored variable which exceeds the scram setpoint will initiate an automatic scram and not impair the remaining variables from being monitored, and if one channel fails, the remaining portions of the RPS shall function.

The Rod Control and Information System (RCIS) is designed so that no single failure can negate the effectiveness of a reactor scram. The circuitry for the RCIS is completely independent of the circuitry controlling the scram valves. This separation of the scram and normal rod control functions prevents failures in the RCIS circuitry from affecting the scram circuitry. Because only two control rods are controlled by an individual hydraulic control unit (HCU), a failure that results in continued energizing of an insert solenoid valve on an HCU can affect only two control rods. The effectiveness of a reactor scram is not impaired by the malfunctioning of any one HCU or two control rods.

The design of the protection system assures that specified acceptable fuel limits are not exceeded for any single malfunction of the reactivity control systems as specified in Criterion 25.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
4.3	Nuclear Design
4.4	Thermal and Hydraulic Design
4.6	Functional Design of Reactivity Control Systems
7.2	Reactor Trip System

Chapter/Section	Title
7.7.1.2 and 7.7.2.2	Rod Control and Information System—Instrumentation and Controls
15	Accident Analyses

3.1.2.3.7 Criterion 26—Reactivity Control System Redundancy and Capability

3.1.2.3.7.1 Criterion 26 Statement

Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure that acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

3.1.2.3.7.2 Evaluation Against Criterion 26

Two independent reactivity control systems utilizing different design principles are provided. The normal method of reactivity control employs control rod assemblies which contain boron carbide (B_4C) powder. Positive insertion of these control rods is provided by means of the control rod drive electrical and hydraulic system. The control rods are capable of reliably controlling reactivity changes during normal operation (e.g., power changes, power shaping, xenon burnout, normal startup and shutdown) via operator-controlled insertions and withdrawals. The control rods are also capable of maintaining the core within acceptable fuel design limits during anticipated operational occurrences via the automatic scram function. The unlikely occurrence of a limited number of stuck rods during a scram will not adversely affect the capability to maintain the core within fuel design limits.

The circuitry for manual insertion or withdrawal of control rods is completely independent of the circuitry for reactor scram. This separation of the scram and normal rod control functions prevents failures in the reactor manual-control circuitry from affecting the scram circuitry. Two sources of energy (accumulator pressure and electrical power to the motors of the fine motion control rod drives, FMCRDs) provide needed control rod insertion performance over the entire range of reactor pressure (i.e., from operating conditions to cold shutdown). The design of the control rod system includes appropriate margin for malfunctions such as stuck rods in the unlikely event that they do occur. Control rod withdrawal sequences and patterns are selected prior to operation to achieve optimum core performance and, simultaneously, low individual rod worths. The operating procedures to accomplish such patterns are supplemented by the rod pattern control system, which prevents rod withdrawals yielding a rod worth greater than permitted by the preselected rod withdrawal pattern. Because of the carefully planned and regulated rod withdrawal sequence, prompt shutdown of the reactor can be achieved with the insertion of a small number of the many independent control rods. In the event that a reactor scram is necessary, the unlikely occurrence of a limited number of stuck rods will not hinder the capability of the control rod system to render the core subcritical.

A standby liquid control system containing a neutron-absorbing sodium pentaborate solution is the independent backup system. This system has the capability to shut the reactor down from full power and maintain it in a subcritical condition at any time during the core life. The reactivity control provided to reduce reactor power from rated power to a shutdown condition with the control rods withdrawn in the power pattern accounts for the reactivity effects of xenon decay, elimination of steam voids, change in water density due to the reduction in water temperature, Doppler effect in uranium, change in the neutron leakage from boiling to cold, and change in the rod worth as boron affects the neutron migration length.

The control rod system is capable of holding the reactor core subcritical under cold conditions, even when the number of control rods of highest worth controlled by an hydraulic control unit is assumed to be stuck in the fully withdrawn position. This shutdown capability of the control rod system is made possible by designing the fuel with burnable poison (Gd_2O_3) to control the high reactivity of fresh fuel.

The redundancy and capabilities of the reactivity control systems for the ABWR satisfy the requirements of Criterion 26.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
4.6	Functional Design of Reactivity Control Systems
7.3	Engineered Safety Feature Systems
7.4.1.2 and 7.4.2.2	Standby Liquid Control System—Instrumentation and Controls
7.7.1.2 and 7.7.2.2	Rod Control and Information System— Instrumentation and Controls

3.1.2.3.8 Criterion 27—Combined Reactivity Control Systems Capability

3.1.2.3.8.1 Criterion 27 Statement

The reactivity control systems shall be designed to have a combined capability in conjunction with poison addition by the emergency core cooling systems of reliably controlling reactivity changes to assure that, under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained.

3.1.2.3.8.2 Evaluation Against Criterion 27

There is no credible event applicable to the ABWR which requires combined capability of the control rod system and poison additions. The ABWR design is capable of maintaining the reactor core subcritical, including allowance for a pair of stuck rods controlled by a hydraulic control unit (HCU), without addition of any poison to the reactor coolant. The primary reactivity control system for the ABWR during postulated accident conditions is the control rod system. Abnormalities are sensed, and, if protection system limits are reached, corrective action is initiated through automatic insertion of control rods. High integrity of the protection system is achieved through the combination of logic arrangement, actuator redundancy, power supply redundancy, and physical separation. High reliability of reactor scram is further achieved by separation of scram and manual control circuitry, individual HCU controlling a pair of control rods, and fail-safe design features built into the rod drive system. Response by the RPS is prompt and the total scram time is short.

In the unlikely event that more than one control rod fails to insert and the core cannot be maintained in a subcritical condition by control rods alone as the reactor is cooled down subsequent to initial shutdown, the Standby Liquid Control System (SLCS) can be actuated to insert soluble boron into the reactor core. The SLCS has sufficient capacity to ensure that the reactor can always be maintained subcritical; and, hence, only decay heat will be generated by the core which can be removed by the Residual Heat Removal (RHR) System, thereby ensuring that the core will always be coolable.

The design of the reactivity control systems assures reliable control of reactivity under postulated accident conditions with appropriate margin for stuck rods. The capability to cool the core is maintained under all postulated accident conditions; thus, Criterion 27 is satisfied.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
4.3	Nuclear Design
4.4	Thermal and Hydraulic Design

Chapter/Section	Title
4.6	Functional Design of Reactivity Control System
7.2	Reactor Trip System
7.4.1.2 and 7.4.2.2	Standby Liquid Control System—Instrumentation and Controls
15	Accident Analyses

3.1.2.3.9 Criterion 28—Reactivity Limits

3.1.2.3.9.1 Criterion 28 Statement

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither:

(1) Result in damage to the RCPB greater than limited local yielding, nor

(2) Sufficiently disturb the core, its support structures, or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means), rod dropout, steamline rupture, changes in reactor coolant temperature and pressure, and cold water addition.

3.1.2.3.9.2 Evaluation Against Criterion 28

The control rod system design incorporates appropriate limits on the potential amount and rate of reactivity increase. Control rod withdrawal sequences and patterns are selected to achieve optimum core performance and low individual rod worths. The Rod Pattern Control System (RPCS) prevents withdrawal other than by the preselected rod withdrawal pattern. The RPCS function assists the operator with an effective backup control rod monitoring routine that enforces adherence to established startup, shutdown, and low-power level operation control rod procedures.

The control rod mechanical design incorporates a passive brake and hydraulic inlet check valve which individually, prevents rapid rod ejection. The brake spring force holds the rod position if there is a break in the FMCRD primary pressure boundary. The check valve prevents rod ejection if there is a failure of the scram insert line. Normal rod movement and the rod withdrawal rate is limited through the fine motion control motor.

The accident analysis (Chapter 15) evaluates the postulated reactivity accidents, as well as abnormal operational transients in detail. Analyses are included for rod dropout, steamline rupture, changes in reactor coolant temperature and pressure, and cold water addition. The initial conditions, assumptions, calculational models, sequences of events, and anticipated results of each postulated occurrence are covered in detail. The results of these analyses indicate that none of the postulated reactivity transients or accidents results in damage to the RCPB. In addition, the integrity of the core, its support structures or other reactor pressure vessel internals is maintained so that the capability to cool the core is not impaired for any of the postulated reactivity accidents described in the accident analysis.

The design features of the reactivity control system which limit the potential amount and rate of reactivity increase ensure that Criterion 28 is satisfied for all postulated reactivity accidents.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
3.9.4	Control Rod Drive System
3.9.5	Reactor Pressure Vessel Internals
4.3	Nuclear Design
4.5.3	Control Rod Drive Housing Supports
4.6	Functional Design of Reactivity Control Systems
5.2.2	Overpressurization Protection
5.3	Reactor Vessel
5.4.4	Main Steamline Flow Restrictors
5.4.5	Main Steamline Isolation Valves
7.7.1.2 and 7.7.2.2	Rod Control and Information System—Instrumentation and Controls
15	Accident Analyses

3.1.2.3.10 Criterion 29—Protection Against Anticipated Operational Occurrences

3.1.2.3.10.1 Criterion 29 Statement

The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

3.1.2.3.10.2 Evaluation Against Criterion 29

The high functional reliability of the reactor protection (trip) system and reactivity control system is achieved through the combination of logic arrangement, redundancy, physical and electrical independence, functional separation, fail-safe design, and inservice testability. These design features are discussed in detail in Criteria 21, 22, 23, 24, and 26.

An extremely high reliability of timely response to anticipated operational occurrences is maintained by a thorough program of inservice testing and surveillance. Active components can be tested or removed from service for maintenance during reactor operation without compromising the protection or reactivity control functions even in the event of a subsequent single failure. Components important to safety, such as control rod drives, MSIVs, RHR pumps, RCIC, etc., are testable during normal reactor operation. Functional testing and calibration schedules are developed using available failure rate data, reliability analysis, and operating experience. These schedules represent an optimization of protection and reactivity control system reliability by considering the failure probabilities of individual components and the reliability effects during individual component testing on the portion of the system not undergoing test. The capability for inservice testing ensures the high functional reliability of protection and reactivity control systems if a reactor variable exceeds the corrective action setpoint.

The capabilities of the protection and reactivity control systems to perform their safety functions in the event of anticipated operational occurrences satisfy the requirements of Criterion 29.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
5.4.5	Main Steamline Isolation Valve System
5.4.6	Reactor Core Isolation Cooling
5.4.7	Residual Heat Removal System

Chapter/Section	Title
6.2	Containment Systems
6.3	Emergency Core Cooling Systems
7.2	Reactor Trip System
7.3	Engineered Safety Feature Systems
15	Accident Analyses

3.1.2.4 Group IV—Fluid Systems

3.1.2.4.1 Criterion 30—Quality of Reactor Coolant Pressure Boundary

3.1.2.4.1.1 Criterion 30 Statement

Components which are part of the RCPB shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

3.1.2.4.1.2 Evaluation Against Criterion 30

By utilizing conservative design practices and detailed quality control procedures, the pressure-retaining components of the RCPB are designed and fabricated to retain their integrity during normal and postulated accident conditions (Subsection 3.1.2.2.5.2). Accordingly, components which comprise the RCPB are designed, fabricated, erected, and tested in accordance with recognized industry codes and standards listed in Chapter 5 and Table 3.2-1. Further, product and process quality planning is provided as described in Chapter 17 to assure conformance with the applicable codes and standards, and to retain appropriate documented evidence verifying compliance. Because the subject matter of this criterion deals with aspects of the RCPB, further discussion on this subject is treated in the response to Criterion 14.

Means are provided for detecting leakage in the Reactor Coolant System (RCS). The Leak Detection and Isolation System (LDS) consists of sensors and instruments to detect, annunciate, and, in some cases, isolate the RCPB from potential hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and pressure changes, increased frequency of sump pump operation, and by measuring fission product concentration. In addition to these means of detection, large leaks are detected by changes in flow rates in process lines, and changes in reactor water level. The allowable leakage rates have been based on the predicted and experimentally determined behavior of cracks in pipes, the ability to make up the RCS the normally expected background leakage due to equipment design, and the detection capability of the various sensors and instruments. The total leakage rate limit is established so that, in the absence of normal AC power with loss of feedwater supply, makeup capabilities are provided by the RCIC System. While the LDS provides protection from small leaks, the ECCS network provides protection for the complete range of discharges from ruptured pipes. Thus, protection is provided for the full spectrum of possible discharges.

The RCPB and the LDS are designed to meet requirements of Criterion 30.

For further discussion, see the following sections:

Chapter/Section	Title
1.2.1	Principal Design Criteria
3.2	Classification of Structures, Components, and Systems
5.2.2	Overpressurization Protection
5.2.5	Detection of Reactor Coolant Leakage Through RCPB
5.3	Reactor Vessel
5.4.1	Reactor Recirculation Pumps
7.3.1.1.2 and 7.3.2.2	Leak Detection and Isolation System—Instrumentation and Controls
7.7.1.1	Reactor Vessel Instrumentation
17	Quality Control System

3.1.2.4.2 Criterion 31—Fracture Prevention of Reactor Coolant Pressure Boundary

3.1.2.4.2.1 Criterion 31 Statement

The RCPB shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions:

(1) The boundary behaves in a nonbrittle manner.

(2) The probability of rapidly propagating fracture is minimized.

The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining:

(1) Material properties.

(2) The effects of irradiation on material properties.

(3) Residual, steady-state, and transient stresses.

(4) Size of flaws.

3.1.2.4.2.2 Evaluation Against Criterion 31

Brittle fracture control of pressure-retaining ferritic materials is provided to ensure protection against nonductile fracture. To minimize the possibility of brittle fracture failure of the reactor pressure vessel, the reactor pressure vessel is designed to meet the requirements of ASME Code Section III.

The nil-ductility transition (NDT) temperature is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. The NDT temperature increases as a function of neutron exposure at integrated neutron exposures greater than about 1×10^{17} nvt with neutron of energies in excess of 1.6022E-13J.

The reactor assembly design provides an annular space from the outermost fuel assemblies to the inner surface of the reactor vessel that serves to attenuate the fast neutron flux incident upon the reactor vessel wall. This annular volume contains the core shroud and reactor coolant. Assuming plant operation at rated power and availability of 100% for the plant life time, the neutron fluence at the inner surface of the vessel causes a slight shift in the transition temperatures. Expected shifts in transition temperature during design life as a result of environmental conditions, such as neutron flux, are considered in the design. Operational limitations assume that NDT temperature shifts are accounted for in the reactor operation.

The RCPB is designed, maintained, and tested to provide adequate assurance that the boundary will behave in a non-brittle manner throughout the life of the plant. Therefore, the RCPB is in conformance with Criterion 31.

For further discussion, see the following sections:

Chapter/Section	Title
3	Design of Structures, Components, Equipment and Systems
5.2	Integrity of Reactor Coolant Pressure Boundary
5.3	Reactor Vessel

3.1.2.4.3 Criterion 32—Inspection of Reactor Coolant Pressure Boundary

3.1.2.4.3.1 Criterion 32 Statement

Components which are part of the RCPB shall be designed to permit: (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity and (2) an appropriate material surveillance program for the reactor pressure vessel.

3.1.2.4.3.2 Evaluation Against Criterion 32

The reactor pressure vessel design and engineering effort include provisions for inservice inspection. Removable plugs in the reactor shield wall and/or removable panels in the insulation provide access for examination of the vessel and its appurtenances. Also, removable insulation is provided on the reactor coolant system safety/relief valves, and on the main steam and feedwater systems extending out to and including the first isolation valve outside containment. Inspection of the RCPB is in accordance with ASME B&PV Code Section XI. Section 5.2 defines the Inservice Inspection Plan, access provisions, and areas of restricted access.

Vessel material surveillance samples will be located within the reactor pressure vessel. The program will include specimens of the base metal, weld metal, and heat-affected zone metal.

The plant testing and inspection program ensure that the requirements of Criterion 32 will be met.

For further discussion, see the following sections:

Chapter/Section	Title
3.9	Mechanical Systems and Components
5.2	Integrity of Reactor Coolant Pressure Boundary

3.1.2.4.4 Criterion 33—Reactor Coolant Makeup

3.1.2.4.4.1 Criterion 33 Statement

A system to supply reactor coolant makeup for protection against small breaks in the RCPB shall be provided. The system safety function shall assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the RCPB and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that, for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.

3.1.2.4.4.2 Response to Criterion 33

Means are provided for detecting reactor coolant leakage. The LDS consists of sensors and instruments to detect, annunciate, and, in some cases, isolate the RCPB from potential hazardous leaks before predetermined limits are exceeded. Small leaks are detected by temperature and pressure changes, increased frequency of sump pump operation, and by measuring fission product concentration. In addition to these means of detection, large leaks are detected by changes in flow rates in process lines and changes in reactor water level. The allowable leakage rates have been based on predicted and experimentally determined behavior of cracks in pipes, the ability to make up reactor coolant leakage, the normally expected background leakage due to equipment design, and the detection capability of the various sensors and instruments. The total leakage rate limit is established so that, in the absence of normal AC power containment with a loss of feedwater supply, makeup capabilities are provided by the RCIC System.

The plant is designed to provide ample reactor coolant makeup for protection against small leaks in the RCPB for anticipated operational occurrences and postulated accident conditions. The design of these systems meets the requirements of Criterion 33.

For further discussion, see the following sections:

Chapter/Section	Title
5.2.5	Detection of Reactor Coolant Leakage Through Reactor Coolant Pressure Boundary
5.4.6	Reactor Core Isolation Cooling System
6.3	Emergency Core Cooling Systems
7.3.1.1.2 and 7.3.2.2	Leak Detection and Isolation System—Instrumentation and Controls

3.1.2.4.5 Criterion 34—Residual Heat Removal

3.1.2.4.5.1 Criterion 34 Statement

A system to remove residual heat shall be provided. The safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the RCPB are not exceeded.

Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that, for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), the system safety function can be accomplished, assuming a single failure.

3.1.2.4.5.2 Evaluation Against Criterion 34

The RHR System provides the means to remove decay heat and residual heat from the nuclear steam supply systems (NSSS) so that refueling and servicing of NSSS can be performed.

The major equipment of the RHR System consists of heat exchangers, main system pumps, and service water pumps. The equipment is connected by associated valves and piping, and the controls and instrumentation are provided for proper system operation.

Three independent loops are located in separate protected areas.

Both normal AC power and the auxiliary onsite power system provide power adequate to operate all the auxiliary loads necessary for plant operation. The power sources for the plant auxiliary power system are sufficient in number and of such electrical and physical independence that no single failure will prevent auxiliary systems from supporting two of the three RHR divisions.

The plant auxiliary buses supplying power to engineered safety features and RPS s and those auxiliaries required for safe shutdown are connected by appropriate switching to standby diesel-driven generators located in the plant. Each power source, up to the point of its connection to the auxiliary power buses, is capable of complete and rapid isolation from any other source.

Loads important to plant operation and safety are split and diversified between switchgear sections, and means are provided for detection and isolation of system faults.

The plant layout is designed to effect physical separation of essential bus sections, standby generators, switchgear, interconnections, feeders, power centers, motor control centers, and other system components.

Full capacity standby diesel generators are provided to supply a source of electrical power which is self-contained within the ABWR Standard Plant and is not dependent on external sources of supply. The standby generators produce AC power at a voltage and frequency compatible with the normal bus requirements for essential equipment within the plant. Each of the diesel generators has sufficient capacity to start and carry the essential loads it is expected to drive.

The RHR System is adequate to remove residual heat from the reactor core to assure fuel and RCPB design limits are not exceeded. Redundant reactor coolant circulation paths are available to and from the vessel and RHR System. Redundant onsite electric power systems are provided. The design of the RHR System, including its power supply, meets the requirements of Criterion 34.

For further discussion, see the following sections:

Chapter/Section	Title
5.4.7	Residual Heat Removal System
6.3	Emergency Core Cooling System
7.3.1.1.2 and 7.3.2.1	Emergency Core Cooling System—Instrumentation and Controls
8.2	Onsite Power Systems
9.2	Water Systems
15	Accident Analyses

3.1.2.4.6 Criterion 35—Emergency Core Cooling

3.1.2.4.6.1 Criterion 35 Statement

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any LOCA at a rate such that:

(1) Fuel and clad damage that could interfere with continued effective core cooling is prevented.

(2) Clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that, for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), the system safety function can be accomplished, assuming a single failure.

3.1.2.4.6.2 Evaluation Against Criterion 35

The Emergency Core Cooling System (ECCS) consists of the following:

(1) High Pressure Core Flooder (HPCF) System

(2) Reactor Core Isolation Cooling (RCIC)System

(3) Low Pressure Flooder (LPFL) mode of the Residual Heat Removal System (RHR)

(4) Automatic Depressurization System (ADS)

The ECCS is designed to limit fuel cladding temperature over the complete spectrum of possible break sizes in the RCPB, including a complete and sudden circumferential rupture of the larger pipe connected to the reactor vessel.

The HPCF System consists of two subsystems, each having a single motor-driven pump, system piping, valves, controls, and instrumentation. The RCIC System consists of similar equipment except that it is a single system and the pump delivering high pressure flow is driven by steam turbine. The HPCF and RCIC Systems assure that the reactor core is adequately cooled to prevent excessive fuel clad temperatures for breaks in the NSSS which do not result in rapid depressurization of the reactor vessel. The HPCF or RCIC System continues to operate when reactor vessel pressure is below the pressure at which the RHR/LPFL System operation maintains core cooling. A source of water is available from either the condensate storage pool or the suppression pool.

The ADS functions to reduce the reactor pressure so that flow from RHR/LPFL enters the reactor vessel in time to cool the core and prevent excessive fuel clad temperature. The ADS uses several of the Nuclear Boiler System SRVs to relieve the high pressure steam to the suppression pool.

The HPCF System consists of a centrifugal pump that can be powered by normal auxiliary power of the standby AC power system, a core flooder sparger in the reactor vessel above the core, piping and valves to convey water from the condensate storage pool or the suppression pool to the sparger, and associated controls and instrumentation. In case of low-water level in the reactor vessel or high pressure in the drywell, the HPCF System automatically injects water into the vessel in time and at a sufficient flow rate to cool the core and prevent excessive fuel temperature.

In case of low-water level in the reactor or high pressure in the drywell, the LPFL mode of operation of the RHR System pumps and injects water into the reactor vessel in time to flood the core and prevent excessive fuel temperature. The RHR System is described in Subsection 3.1.2.4.5.2. Protection provided by RHR/LPFL extends to a small break where the ADS has operated to lower the reactor vessel pressure.

Results of the performance of the ECCS for the entire spectrum of liquid line breaks are discussed in Subsection 6.3.3. Peak cladding temperatures are well below the 1204°C design basis.

Also provided in Subsection 6.3.3 is an analysis to show that the ECCS conforms to the 10CFR50 Appendix K. This analysis shows complete compliance with the Criterion 35 with the following results:

(1) Peak clad temperatures are well below the 1204°C NRC acceptability limit.

(2) The amount of fuel cladding reacting with steam is nearly an order of magnitude below the 1% acceptability limit.

(3) The clad temperature transient is terminated while core geometry is still amenable to cooling.

(4) The core temperature is reduced and the decay heat can be removed for an extended period of time.

The redundancy and capability of the onsite electrical power systems for the ECCS are represented in the evaluation against Criterion 34.

The ECCS is adequate to prevent fuel and clad damage which could interfere with effective core cooling and to limit clad metal water reaction to a negligible amount. The design of the ECCS, including the power supply, meets the requirements of Criterion 35.

For further discussion, see the following sections:

Chapter/Section	Title
5.4.7	Residual Heat Removal System
5.4.6	Reactor Core Isolation Cooling System
6.3	Emergency Core Cooling System
7.3.1.1.2 and 7.3.2.1	Emergency Core Cooling System—Instrumentation and Controls
8.3	Onsite Power Systems
9.2	Water Systems
15	Accident Analyses

3.1.2.4.7 Criterion 36—Inspection of Emergency Core Cooling System

3.1.2.4.7.1 Criterion 36 Statement

The ECCS shall be designed to permit appropriate periodic inspection of important components, such as flooder rings in the reactor pressure vessel, water injection nozzles, and piping, to assure the integrity and capability of the system.

3.1.2.4.7.2 Evaluation Against Criterion 36

The ECCS discussed in Criterion 35 includes inservice inspection considerations. The flooder spargers within the vessel are accessible for inspection during each refueling outage. Removable plugs in the reactor shield and/or panels in the insulation is provided on the ECCS piping up to and including the first isolation valve outside the drywell. Inspection of the ECCS is in accordance with the intent of ASME Code Section XI. Subsection 5.2.4 defines the Inservice Inspection Plan, access provisions, and areas of restricted access.

During plant operations, the pumps, valves, piping, instrumentation, wiring, and other components outside the containment can be visually inspected at any time. Components inside the containment can be inspected when the containment is open for access. When the reactor vessel is open for refueling or other purposes, the spargers and other internals can be inspected. Portions of the ECCS which are part of the RCPB are designed to specifications for inservice inspection to detect defects which might affect the cooling performance. Particular attention will be given to the reactor nozzles, and core flooder spargers. The design of the reactor vessel and internals for inservice inspection and the plant testing and inspection program ensures that the requirements of Criterion 36 will be met.

For further discussion, see the following sections:

Chapter/Section	Title
3.9.5	Reactor Pressure Vessel Internals
5.2.4	Inservice Inspection and Testing of Reactor Coolant Pressure Boundary
5.3	Reactor Vessel
6.3	Emergency Core Cooling Systems
6.6	Inservice Inspection of Class 2 and 3 Components

3.1.2.4.8 Criterion 37—Testing of Emergency Core Cooling System

3.1.2.4.8.1 Criterion 37 Statement

The ECCS shall be designed to permit appropriate periodic pressure and functional testing to assure:

(1) The structural and leaktight integrity of its components.

(2) The operability and performance of the active components of the system.

(3) The operability and performance of the active components of the system.The operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

3.1.2.4.8.2 Evaluation Against Criterion 37

The ECCS consists of the HPCF System, RCIC System, LPFL mode of the RHR System, and the ADS. Each of these systems is provided with sufficient test connections and isolation valves to permit appropriate periodic pressure testing to assure the structural and leaktight integrity of its components.

Each of the ECCS is designed to permit periodic testing to assure the operability and performance of the active components of each system.

The pumps and valves of these systems will be tested periodically to verify operability. Flow rate tests will be conducted on the HPCF System, RCIC System, and RHR/LPFL System.

The ECCS will be subjected to tests to verify the performance of the full operational sequence that brings each system into operation. The operation of the associated cooling water systems is discussed in the evaluation of Criterion 46. It is concluded that the requirements of Criterion 37 are met.

For further discussion, see the following sections:

Chapter/Section	Title
5.2.2	Overpressurization Protection
6.3	Emergency Core Cooling Systems
7.3.1.1	Emergency Core Cooling Systems—Instrumentation and Controls
8.3.1	AC Power Systems
16	Technical Specifications

3.1.2.4.9 Criterion 38—Containment Heat Removal

3.1.2.4.9.1 Criterion 38 Statement

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any LOCA and maintain them at acceptable low levels.

Suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that, for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available), the system safety function can be accomplished, assuming a single failure.

3.1.2.4.9.2 Evaluation Against Criterion 38

The containment heat removal function is accomplished by the suppression pool cooling mode of the RHR System. Following a LOCA, the suppression pool cooling (SPC) mode limits the temperature within the wetwell by recirculating the suppression pool water and removing heat via the RHR System heat exchangers. This subsystem is initiated automatically when suppression pool temperature increases to a preset level. Suppression pool cooling can also be initiated manually. If a LOCA signal is present, the RHR System will function in the core cooling (LPFL) mode.

Following a LOCA, wetwell and drywell spray mode of the RHR System condenses steam within the drywell and wetwell zones of the containment by spraying suppression pool water cooled through the heat exchangers. Wetwell/drywell spray is started manually. The drywell spray mode is initiated by operator action post-LOCA in the presence of high drywell pressure. The wetwell spray mode can be manually initiated in the control room, unless an overriding LOCA signal for the LPFL is present. The wetwell spray mode does not depend on the operation of the suppression pool cooling mode.

The redundancy and capability of the offsite and onsite electrical power systems for the RHR System is presented in the evaluation against Criterion 34.

For further discussion, see the following sections:

Chapter/Section	Title
5.4.7	Residual Heat Removal System
6.2.2	Containment Heat Removal Systems
7.3	Engineered Safety Features—Instrumentation and Controls
8.3.1	AC Power Systems
9.2	Water Systems
15	Accident Analyses

3.1.2.4.10 Criterion 39—Inspection of Containment Heat Removal System

3.1.2.4.10.1 Criterion 39 Statement

The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping, to assure the integrity and capability of the system.

3.1.2.4.10.2 Evaluation Against Criterion 39

Provisions are made to facilitate periodic inspections of active components and other important equipment of the containment heat removal systems. During plant operations, the pumps, valves, piping, instrumentation, wiring, and other components outside the containment can be visually inspected at any time and will be inspected periodically. Such components inside the containment will be tested and inspected during periodic outages. The testing frequencies of most components will be correlated with the component inspection.

The suppression pool is designed to permit appropriate periodic inspection. Space is provided outside the containment for inspection and maintenance.

The containment heat removal system is designed to permit periodic inspection of major components. This design meets the requirements of Criterion 39.

For further discussion, see the following sections:

Chapter/Section	Title
5.4.7	Residual Heat Removal System
6.2	Containment Systems
6.3	Emergency Core Cooling Systems
9.2	Water Systems

3.1.2.4.11 Criterion 40—Testing of Containment Heat Removal System

3.1.2.4.11.1 Criterion 40 Statement

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure:

(1) The structural and leaktight integrity of its components.

(2) The operability and performance of the active components of the system.

(3) The operability of the system as a whole, and, under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

3.1.2.4.11.2 Evaluation Against Criterion 40

The containment heat removal function is accomplished by a suppression pool cooling mode of the RHR System.

The RHR System is provided with sufficient test connections and isolation valves to permit periodic pressure and flow rate testing.

The pumps and valves of the RHR System will be operated periodically to verify operability. The cooling is initiated manually or automatically on sensed high temperature in the suppression pool, however, operation of the components is periodically verified. The operation of associated cooling water systems is discussed in the response to Design Criterion 46. It is concluded that the requirements of Criterion 40 are met.

For further discussion, see Subsection 6.2.2.

3.1.2.4.12 Criterion 41—Containment Atmosphere Cleanup

3.1.2.4.12.1 Criterion 41 Statement

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quantity of fission products released to the environment following postulated accidents and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features and suitable interconnections, leak detection, isolation, and containment capabilities to assure that, for onsite electric power system operation (assuming offsite power is not available) and its safety function can be accomplished, assuming a single failure.

3.1.2.4.12.2 Evaluation Against Criterion 41

The quantity and quality of fission products released into the environment following postulated accidents is controlled by the Standby Gas Treatment System (SGTS) that has the redundancy and capability to filter and treat the gaseous effluent from the primary and the secondary containment.

Equipment for purging and inerting is provided to control the oxygen concentration of the inert volume of the ABWR primary containment atmosphere so that a flammable condition will not be created during an accident.

These systems have design provisions to ensure that the safety function is accomplished, assuming a single failure. These systems meet the requirements of Criterion 41. For further discussion, see the following sections:

Chapter/Section	Title
1.2	General Plant Description
6.2.5	Combustible Gas Control in Containment
6.5.1	Engineered Safety Features Filter System
6.5.3	Fission Product Control Systems
7	Instrumentation and Control System
8	Electric Power
9.5.9	Suppression Pool Cleanup System
15	Accident Analyses

3.1.2.4.13 Criterion 42— Inspection of Containment Atmosphere Cleanup System

3.1.2.4.13.1 Criterion 42 Statement

The Containment Atmosphere Cleanup System shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping, to assure the integrity and capability of the systems.

3.1.2.4.13.2 Evaluation Against Criterion 42

Except for components located in the primary containment and steam tunnel, all components of the fission product control system can be inspected during normal plant operation at power. The components within the containment and steam tunnel may be inspected during refueling and maintenance outages.

The design of the system, therefore, meets the requirements of Criterion 42. For further discussion, see the following sections:

Chapter/Section	Title
1.2	General Plant Description
6.2.5	Combustible Gas Control in Containment
6.5.1	Engineered Safety Features Filter System
6.5.3	Fission Product Control Systems
6.6	Preservice and Inservice Inspection and Testing of Class 2 and 3 Components and Piping
7	Instrumentation and Control System
8	Electric Power
9.5.9	Suppression Pool Cleanup System

3.1.2.4.14 Criterion 43—Testing of Containment Atmosphere Cleanup System

3.1.2.4.14.1 Criterion 43 Statement

The Containment Atmosphere Cleanup System shall be designed to permit appropriate periodic pressure and functional testing to assure:

(1) The structural and leaktight integrity of its components.

(2) The operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves.

(3) The operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

3.1.2.4.14.2 Evaluation Against Criterion 43

All active components of the fission product control system can be tested during normal plan operation at power.

Complete system operation can be tested during reactor shutdown.

The design of the system, therefore, meets the requirements of Criterion 43. For further discussion, see the following sections:

Chapter/Section	Title
1.2	General Plant Description
6.2.5	Combustible Gas Control in Containment
6.5.3	Fission Product Control System
7	Instrumentation and Control System
8	Electric Power

3.1.2.4.15 Criterion 44—Cooling Water

3.1.2.4.15.1 Criterion 44 Statement

A system to transfer heat from structures, systems, and components important to safety to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power systems operation (assuming onsite power is not available), the system safety function can be accomplished, assuming a single failure.

3.1.2.4.15.2 Evaluation Against Criterion 44

The system provided to transfer heat from safety-related equipment to the ultimate heat sink is the Reactor Building Cooling Water (RCW) System.

This system is operable either from offsite power or from onsite emergency power and is designed with suitable redundancy, isolation capability, and separation such that no single failure prevents a safe plant shutdown.

The design of this system meets the requirements of Criterion 44.

For further discussion, see the following sections:

Chapter/Section	Title
1.2	General Plant Description
9.2	Water Systems

3.1.2.4.16 Criterion 45— Inspection of Cooling Water System

3.1.2.4.16.1 Criterion 45 Statement

The cooling water system shall be designed to permit appropriate periodic inspection of important components such as heat exchangers and piping to assure the integrity and capability of the system.

3.1.2.4.16.2 Evaluation Against Criterion 45

All important components in the ABWR Standard Plant, including important components in the Cooling Water Systems, are located in accessible locations to facilitate periodic inspection during normal plant operation. Suitable manholes, handholes, inspection ports, or other design and layout features are provided for this purpose.

These features meet the requirements of Criterion 45.

For further discussion, see the following sections:

Chapter/Section	Title
1.2	General Plant Description
9.2	Water Systems
14	Initial Test Program

3.1.2.4.17 Criterion 46—Testing of Cooling Water System

3.1.2.4.17.1 Criterion 46 Statement

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure:

(1) The structural and leaktight integrity of its components.

(2) The operability and the performance of the active components of the system.

(3) The operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

3.1.2.4.17.2 Evaluation Against Criterion 46

Redundancy and isolation are provided to allow periodic pressure and functional testing of the system as a whole including the functional sequence that initiates system operation. This also includes transfer between the offsite power supply and the onsite emergency diesel-generator power supply. At least one of the redundant systems is in service during normal plant operations.

The system design thus meets the requirements of Criterion 46.

For further discussion, see the following sections:

Chapter/Section	Section
1.2	General Plant Description
9.2	Water Systems
14	Initial Test Program
16	Technical Specifications

3.1.2.5 Group V—Reactor Containment

3.1.2.5.1 Criterion 50—Containment Design Basis

3.1.2.5.1.1 Criterion 50 Statement

The reactor containment structure, including access openings, penetrations, and the containment heat removal system, shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin shall reflect consideration of:

(1) The effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and as required by Section 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning.

(2) The limited experience and experimental data available for defining accident phenomena and containment responses.

(3) The conservatism of the calculational model and input parameters.

3.1.2.5.1.2 Evaluation Against Criterion 50

Design of the containment is based on the safe shutdown earthquake (SSE) postulated to occur at the site simultaneously with the design basis accident (DBA), which is defined as the worst LOCA pipe break having the consequences of maximum containment and drywell pressure and/or temperature. These conditions are coupled with the loss of offsite power.

The maximum pressure and temperature reached in the drywell and containment during this worst-case accident are shown in Chapter 6 to be well below the design pressure and temperature of the structures. This provides an adequate margin for uncertainties in potential energy sources.

The design of the containment system thus meets the requirements of Criterion 50.

For further discussion, see the following sections:

Chapter/Section	Title
3.7	Seismic Design
3.8	Design of Seismic Category I Structures
6.2.1	Containment Functional Design
6.2.2	Containment Heat Removal System
15	Accident Analyses

3.1.2.5.2 Criterion 51—Fracture Prevention of Containment Pressure Boundary

3.1.2.5.2.1 Criterion 51 Statement

The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions:

(1) Its ferritic materials behave in a nonbrittle manner.

(2) The probability of rapidly propagating fracture is minimized.

The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions and the uncertainties in determining:

(1) Material properties

(2) Residual, steady-state, and transient stresses

(3) Size of flaws

3.1.2.5.2.2 Evaluation Against Criterion 51

The primary containment vessel (PCV) is a reinforced concrete structure with ferritic parts (the removable head, personnel locks, equipment hatches and penetrations), which are made of material that has a nil-ductility transition temperature of at least 17°C below the minimum service temperature.

The PCV is enclosed by and is integrated with the reinforced concrete reactor building. The pre-operational test program and the QA program ensure the integrity of the containment and its ability to meet all normal operating and accident requirements.

The containment design thus meets the requirements of Criterion 51.

For further discussion, see the following sections:

Chapter/Section	Title
3.8	Design of Seismic Category I Structures
17	Quality Assurance

3.1.2.5.3 Criterion 52—Capability for Containment Leakage Rate Testing

3.1.2.5.3.1 Criterion 52 Statement

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

3.1.2.5.3.2 Evaluation Against Criterion 52

The containment system is designed and constructed and the necessary equipment is provided to permit periodic integrated leak-rate tests during the plant lifetime. The testing program is conducted in accordance with 10CFR50 Appendix J.

The testing provisions provided and the test program meet the requirements of Criterion 52.

For further discussion, see the following sections:

Chapter/Section	Title
3.8.2.7	Testing and Inservice Inspection Requirements
6.2.6.1	Containment Integrated Leakage Test Rate

3.1.2.5.4 Criterion 53—Provisions for Containment Testing and Inspection

3.1.2.5.4.1 Criterion 53 Statement

The reactor containment shall be designed to permit:

(1) Appropriate periodic inspection of all important areas such as penetrations.

(2) An appropriate surveillance program.

(3) Periodic testing at containment design pressure of the leaktightness of penetrations which have resilient seals and expansion bellows.

3.1.2.5.4.2 Evaluation Against Criterion 53

There are special provisions for conducting individual leakage rate tests on applicable penetrations. Penetrations are visually inspected and pressure tested for leaktightness at periodic intervals in accordance with 10CFR50 Appendix J.

The provisions made for protection testing meet the requirements of Criterion 53.

For further discussion, see the following sections:

Chapter/Section	Title
6.2.6.2	Containment Penetration Leakage Rate Test (Type B)
6.2.6.3	Containment Isolation Value Leakage Rate Test (Type C)

3.1.2.5.5 Criterion 54—Piping Systems Penetrating Containment

3.1.2.5.5.1 Criterion 54 Statement

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to periodically test the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

3.1.2.5.5.2 Evaluation Against Criterion 54

Piping systems penetrating containment are designed to provide the required isolation and testing capabilities. These piping systems are provided with test connections to allow periodic leak detection tests as necessary to determine if valve leakage is within acceptable limits.

The actuation test circuitry provides the means for testing isolation valve operability as necessary to determine if operability is within acceptable limits.

The design and provisions made for piping systems penetrating containment meet the requirements of Criterion 54.

3.1.2.5.6 Criterion 55—Reactor Coolant Pressure Boundary Penetrating Containment

3.1.2.5.6.1 Criterion 55 Statement

Each line that is part of the RCPB and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines such as instrument lines, are acceptable on some other defined basis, including either:

(1) One locked-closed isolation valve inside and one locked-closed isolation valve outside containment

(2) One automatic isolation valve inside and one locked-closed isolation valve outside containment

(3) One locked-closed isolation valve inside and one automatic isolation valve outside containment (a simple check valve may not be used as the automatic isolation valve outside containment)

(4) One automatic isolation valve inside and one automatic isolation valve outside containment (a simple check valve may not be used as the automatic isolation valve outside containment)

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

3.1.2.5.6.2 Evaluation Against Criterion 55

The RCPB as defined in 10CFR50 Section 50.2, consists of the reactor pressure vessel, pressure retaining appurtenances attached to the vessel, valves, and pipes which extend from the reactor pressure vessel up to and including the outermost isolation valves. The lines of the RCPB which penetrate the containment have suitable isolation valves capable of isolating the containment, thereby precluding any significant release of radioactivity.

The design of the isolation systems detailed in the following sections meets the requirements of Criterion 55.

For further discussion, see the following sections:

Chapter/Section	Title
5.2	Integrity of Reactor Coolant Pressure Boundary
6.2.4	Containment Isolation Systems
7	Instrumentation and Controls
15	Accident Analyses
16	Technical Specifications

3.1.2.5.7 Criterion 56—Primary Containment Isolation

3.1.2.5.7.1 Criterion 56 Statement

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked-closed isolation valve inside and one locked-closed isolation valve outside containment.

(2) One automatic isolation valve inside and one locked-closed isolation valve outside containment.

(3) One locked-closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.

3.1.2.5.7.2 Evaluation Against Criterion 56

The manner in which the containment isolation system meets this requirement is detailed in the following sections:

Chapter/Section	Title
6.2.4	Containment Isolation Systems
7	Instrumentation and Controls
15	Accident Analyses
16	Technical Specifications

3.1.2.5.8 Criterion 57—Closed System Isolation Valves

3.1.2.5.8.1 Criterion 57 Statement

Each line that penetrates primary reactor containment and is neither part of the RCPB nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked-closed, or capable of remote manual operation. This valve shall be outside the containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

3.1.2.5.8.2 Evaluation Against Criterion 57

Each line that penetrates containment and is not connected to the containment atmosphere and is not part of the RCPB has at least one isolation valve located outside containment.

Details demonstrating conformance with Criterion 57 are provided in the following section:

Chapter/Section	Title
6.2.4	Containment Isolation Systems

3.1.2.6 Group VI—Fuel and Reactivity Control

3.1.2.6.1 Criterion 60—Control of Releases of Radioactive Materials to the Environment

3.1.2.6.1.1 Criterion 60 Statement

The nuclear power unit design shall include means to suitably control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

3.1.2.6.1.2 Evaluation Against Criterion 60

The ABWR is designed so that releases of radioactive materials, in either gaseous, liquid, or solid form, are minimized. Gaseous releases come primarily from the turbine condenser offgas and the ventilation systems along with the flue gas from any incineration of dry active waste (DAW). Activity in the condenser offgas is minimized by the advanced fuel design utilized in the ABWR and the improved water chemistry which reduces the corrosion potential of the primary system water. Noble gas and any iodine activity that enters the Turbine Offgas System is held up by ambient temperature charcoal beds. The Offgas System itself has undergone significant improvement to increase reliability. All ventilation system releases are through the plant stack. The plant stack and the major streams feeding the plant stack are monitored by the process radiation monitoring system so that suitable action may be taken to avoid releases in excess of the relevant regulatory limits.

The Radwaste System processes liquid and solid wastes. Because of the overall improved system and equipment design, improved reliability and capacity factor, and improved operations, the generation of solid radwaste is expected to be significantly less than in current operating plants. Processes are installed in radwaste to fully treat and solidify solid wastes, as required by applicable state and federal regulations. In addition, the ABWR Radwaste System can be operated in a mode where all non-detergent and non-chemical waste streams are treated to allow maximum recycle to the primary system (condensate storage tank). This mode of operation would minimize releases via the liquid or discharge pathway at the expense of some increase in solid waste generated. The optimal balance is best established during operations and is significantly affected by the overall plant water balance.

The Radwaste System has significant holdup capacity, both in waste collection tanks and in sample tanks containing processed water. This holdup or surge capacity provides the utility flexibility in operations when deciding when and how to release effluents to the environment.

For further discussion, see the following Sections: 11.2 through 11.4, “Radioactive Waste Management” and 11.5, “Process Radiation Monitoring.”

3.1.2.6.2 Criterion 61—Fuel Storage and Handling and Radioactivity Control

3.1.2.6.2.1 Criterion 61 Statement

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed:

(1) With a capability to permit appropriate periodic inspection and testing of components important to safety

(2) With suitable shielding for radiation protection

(3) With appropriate containment, confinement, and filtering systems

(4) With a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal

(5) To prevent significant reduction in fuel storage coolant inventory under accident conditions

3.1.2.6.2.2 Evaluation Against Criterion 61

3.1.2.6.2.2.1 Fuel Storage and Handling System

Fuel storage pools have adequate water shielding for stored spent fuel. Adequate shielding for transporting fuel is also provided. Liquid level sensors are installed to detect low pool water level. Buildings are designed to meet Regulatory Guide 1.13 criteria. The fuel storage pools are designed with no penetrations below the water level that is needed for maintenance of adequate shielding at the operating floor and cooling. Check valves are used in pool circulation lines to prevent siphoning in the event of a break of such a line.

New fuel storage racks are located in the concrete fuel storage vault. No cooling or air filtering system is required. These storage racks preclude accidental criticality (see evaluation against Criterion 62). The new fuel storage racks do not require any special inspection and testing for nuclear safety purposes.

The fuel storage and handling system is designed to assure adequate safety under normal and postulated accident conditions. The design of these systems meets the requirements of Criterion 61.

Per Regulatory Guide 1.143, the substructure of the radwaste building is sufficient to contain the maximum liquid inventory expected to be in the building.

For further discussion, see the following sections:

Chapter/Section	Title
5.4.7	Residual Heat Removal System
6.2	Containment Systems
9.1	Fuel Storage and Handling
11	Radioactive Waste Management
12	Radiation Protection

3.1.2.6.3 Criterion 62—Prevention of Criticality in Fuel Storage and Handling

3.1.2.6.3.1 Criterion 62 Statement

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

3.1.2.6.3.2 Evaluation Against Criterion 62

Appropriate plant fuel handling and storage facilities are provided to preclude accidental criticality for new and spent fuel. Criticality in new and spent fuel storage is prevented by presence of fixed neutron absorbing material. Fuel elements are limited by rack design to only top-loaded fuel assembly positions. The new and spent fuel racks are Seismic Category I components.

New fuel is placed in dry storage in the top-loaded new fuel storage vault. This vault contains a drain to prevent the accumulation of water. Neutron absorbing material in the new fuel storage vault racks prevents an accidental critical array, even in the event the vault becomes flooded or subjected to seismic loadings.

The center-to-center new fuel assembly spacing limits the k_{eff} of the array to not more than 0.95 for new dry fuel.

The spent fuel is stored under water in the spent fuel pool. A full array of loaded spent fuel racks is designed to be subcritical, by at least 5% Δk. Neutron-absorbing material, as an integral part of the design, is employed to assure that the calculated k_{eff}, including biases and uncertainties, will not exceed 0.95 under all normal and abnormal conditions. The abnormal conditions accounted for are an earthquake, accidental dropping of equipment, or impact caused by the horizontal movement of fuel handling equipment without first disengaging the fuel from the hoisting equipment.

Refueling interlocks include circuitry which senses conditions of the refueling equipment and the control rods. These interlocks reinforce operational procedures that prohibit making the reactor critical. The fuel handling system is designed to provide a safe, effective means of transporting and handling fuel and is designed to minimize the possibility of mishandling or maloperation.

The presence of fixed neutron-absorbing material in the new and spent fuel storage and the design of fuel handling systems precludes accidental criticality in accordance with Criterion 62.

For further discussion, see the following section:

Chapter/Section	Title
9.1	Fuel Storage and Handling

3.1.2.6.4 Criterion 63—Monitoring Fuel and Waste Storage

3.1.2.6.4.1 Criterion 63 Statement

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas to:

(1) Detect conditions that may result in loss of residual heat removal capability and excessive radiation levels.

(2) Initiate appropriate safety actions.

3.1.2.6.4.2 Evaluation Against Criterion 63

The Fuel Pool Cooling and Cleanup (FPC) System removes decay heat from fuel storage pools. In addition, three loops of the RHR System can provide additional cooling of the spent fuel pool, as required. Fuel pool temperature and level are monitored as part of the FPC System. High pool temperature or low skimmer surge tank level would signal the need for providing additional cooling [e.g., adding a loop of RHR, or makeup water, e.g., from the makeup water system (condensate) connection, as appropriate]. Area radiation monitors are provided as part of the area radiation monitoring system which monitors the operating/refueling floor for high radiation levels.

Area radiation and tank and sump levels are monitored and alarmed to give indication of conditions which may result in excessive radiation levels in radioactive waste system areas. These systems satisfy the requirements of Criterion 63.

For further discussion, see the following sections:

Chapter/Section	Title
5.4.7	Residual Heat Removal System
9.1.3	Fuel Pool Cooling and Cleanup System
9.2.9	Makeup Water System (Condensate)
11	Radioactive Waste Management
12	Radiation Protection

3.1.2.6.5 Criterion 64—Monitoring Radioactivity Releases

3.1.2.6.5.1 Criterion 64 Statement

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of LOCA fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and from postulated accidents.

3.1.2.6.5.2 Evaluation Against Criterion 64

Means have been provided for monitoring radioactivity releases resulting from normal and anticipated operational occurrences and from postulated accidents. The following releases are monitored:

(1) Gaseous releases

(2) Liquid discharge

In addition, the containment atmosphere is monitored.

For further discussion of the same means and equipment used for monitoring reactivity releases, see the following sections:

Chapter/Section	Title
5.2.5	Reactor Coolant Pressure Boundary and Core Cooling Systems Leakage Detection
11	Radioactive Waste Management