CHAPTER 1 TABLE OF CONTENTS

1.0 INTRODUCTION AND SUMMARY

The Safety Analysis Report (FSAR) is submitted as required by 10 CFR 50.71(e) "Periodic Updating of Final Safety Analysis Report." The FSAR is based on the original Final Facility Description and Safety Analysis Report (FFDSAR) and a compilation of docketed material that affected the original FFDSAR content. The FFDSAR was submitted in support of the application by Wisconsin Electric Power Company to operate a nuclear power plant designated as Point Beach Nuclear Plant, Units 1 and 2. The FSAR and other docketed material remain as the licensing basis for the Point Beach Nuclear Plant. Unit 2 is located adjacent to Unit 1 on a site situated on Lake Michigan. Certain components of Units 1 and 2 are shared and described herein.

The Point Beach Units 1 and 2 reactors are pressurized light water moderated and cooled systems. Each unit was intitially designed to produce a reactor thermal output of 1518.5 MWt. All steam and power conversion equipment, including each turbine generator, was originally designed to permit generation of 523.8 MW of gross electrical power. Unit 1 achieved commercial operation in December 1970. Unit 2 achieved commercial operation in October 1972. Since being placed into commercial operation, each unit has undergone a LP Turbine retrofit modification that increases the unit design output to 537,960 kWe. In addition, a measurement uncertainty recapture (MUR) power uprate has been implemented for both units. The MUR uprate increased license reactor thermal power to 1540 MWt and turbine generator output to approximately 545MWe.

For Extended Power Uprate (EPU) operation, the reactor thermal power was increased to 1800 MWt, and the turbine generator output to approximately 640 MWe. For EPU, modifications were made to both unit's high pressure turbines, instrumentation and controls, and the associated steam, condensate, and feedwater paths.

The nuclear power plant incorporates two Westinghouse closed-cycle pressurized water nuclear steam supply systems and turbine-generator systems utilizing dry and saturated steam. Equipment includes systems for the processing of radioactive wastes, handling of fuel, electrical distribution, cooling, power generation structures, and all other on-site facilities required to provide a complete and operable nuclear power plant.

All plant safety systems, including containment and engineered safety features are designed and evaluated for operation at 1800 MWt power rating of the reactor. This power rating is used in the analysis of postulated accidents reported herein.

The remainder of Chapter 1 of this report summarizes the principal design features and safety criteria of the nuclear units. Also provided is a description of the Quality Assurance program which ensures compliance with standards. Chapter 2 contains a description and evaluation of the Point Beach Site and environs, supporting the suitability of that site for a nuclear plant of the size and type described.

Chapter 3 and Chapter 4 describe the reactors and the reactor coolant systems, Chapter 5 the containment and related systems, and Chapter 6 through Chapter 10 the emergency and other auxiliary systems. Chapter 11 describes the Radiological Protection aspects of the station. Chapter 12 describes the Company's program for organization and training of plant personnel. Chapter 13 contains an outline and description of the initial tests and operations associated with plant startup and the on-site Quality Assurance program. Chapter 14 is a safety evaluation summarizing the analyses that demonstrate the adequacy of the reactor protection system and the engineered safety features systems. The consequences of various postulated accidents are within the guidelines set forth in the Nuclear Regulatory Commission regulation 10 CFR 50.67.

Chapter 15 is a description of the Aging Management Program and Time Limited Aging Analysis.

Appendix T incorporates by reference the Technical Requirements Manual; a compilation of specifications relocated from the previous Technical Specifications in conjunction with the Improved Standard Technical Specification conversion. The appendices contain the additional analyses and initial licensing information.

The Technical Specifications for Point Beach designate safety limits, maximum safety system settings, minimum conditions for operation, and surveillance standards for the safe operation of the plant. Included with these specifications is a summary of the material presented in the Final Safety Analysis Report used as the bases for each specification. The Technical Specifications are provided in a separate volume.

Within the context of the FSAR, the following definitions apply:

1) Hot Shutdown – The reactor is in the hot shutdown condition when the reactor is subcritical, by an amount greater than or equal to Technical Requirements Manual (TRM) 2.1, Figure 2 and T_{avg} is at or greater than 540^{o}F.

2) Refueling Shutdown – The reactor is in the refueling shutdown condition when the reactor is subcritical by at least 5 percent $\Delta k/k$, and T_{avg} is less than or equal to 140^{o}F. A refueling shutdown refers to a shutdown to move fuel to and from the reactor core.

1.1 SITE AND ENVIRONMENT

The plant site is in east central Wisconsin on the west shore of Lake Michigan about 30 miles SE of Green Bay and about 90 miles NNE of Milwaukee. Cooling water is drawn from Lake Michigan. Farming is the predominant activity in this sparsely populated area of the state. The plant is situated in a productive dairy farming and vegetable canning region; however, it is industrialized to the south in Two Rivers and Manitowoc and to the west in the Fox River Valley.

Soil and subsurface layers contain high clay content which inhibit percolation and drainage to Lake Michigan. The site is well ventilated and not subject to severe persistent inversions. While tornadoes occur in the region, none has been reported to affect the lakeshore site directly. High winds (on the order of 108 mph) can be expected once in 100 years from storms.

Upper glacial till or underlying lake deposits on the site provide a suitable foundation for plant structures. A horizontal ground acceleration of 0.06 gravity and a vertical acceleration of 0.04 gravity are used for the earthquake design criteria, based on a report by John A. Blume and Associates. These accelerations are considered as acting simultaneously. Site soil and geological investigations were performed by Dames and Moore. Additional consultants in the site evaluation were Harza Engineering Company for hydrologic and hydraulic studies.

1.2 SUMMARY PLANT DESCRIPTION

Inherent to the design of closed-cycle reactors is the ability to significantly reduce the release of fission products to the environment. Four barriers exist between the fission product accumulation and the environment. These are the uranium dioxide fuel matrix, the fuel cladding, the reactor vessel and coolant loops, and the reactor containment. The consequences of a breach of the fuel cladding are greatly reduced by the ability of the uranium dioxide lattice to retain fission products. Escape of fission products through fuel cladding defect would be contained within the pressure vessel, loops, and auxiliary systems. Breach of these systems or equipment would release the fission products to the reactor containment where they would be retained. The reactor containment is designed to adequately retain these fission products under the most severe accident conditions, as analyzed in Section 14.

Several engineered safety features have been incorporated into the plant design to reduce the consequences of a loss-of-coolant accident. These safety features include a safety injection system. This system automatically delivers borated water to the reactor vessel for cooling the core under high and low reactor coolant pressure conditions. The safety injection system also serves to insert negative reactivity into the core in the form of borated water during an uncontrolled plant cooldown following a steam line break or an accidental steam release. Other safety features which have been included in the reactor containment design are a containment air recirculation cooling system which acts to effect a depressurization of the containment following a loss of coolant, and a containment spray system which acts to depressurize the containment and remove elemental iodine and particulates from the atmosphere by washing action. The containment spray system provides redundant backup by an alternate principle for the containment air recirculation cooling system.

1.2.1 STRUCTURES

The major structures on the site are the two reactor containments, one for each unit, and the following which are shared: Auxiliary building, pumphouse, turbine building (including the control room), emergency diesel generator building, and service buildings. The relationship of the Unit 2 containment to the Unit 1 containment is shown in Figure 1.2-1. General equipment and plant layout appear in Figure 1.2-2 through Figure 1.2-14.

The reactor containment is a steel-lined concrete cylinder with prestressed tendons in the walls and dome, anchored to a reinforced concrete foundation slab which is supported by steel H-piles driven to refusal in the underlying bedrock. The containment is designed to withstand the internal pressure accompanying a loss-of-coolant accident, is virtually leak-tight, and provides adequate radiation shielding for both normal operation and accident conditions.

Seismic Classification of Particular Structures and Equipment

Particular structures and equipment are classified according to seismic design. The definition of the three seismic classifications is given in Appendix A.5.

1.2.2 NUCLEAR STEAM SUPPLY SYSTEM

For each unit the nuclear steam supply system consists of a pressurized water reactor, reactor coolant system, and associated auxiliary fluid systems. The reactor coolant system is arranged as two closed reactor coolant loops connected in parallel to the reactor vessel, each containing a reactor coolant pump and a steam generator. An electrically heated pressurizer is connected to one of the loops.

The reactor core is composed of uranium dioxide pellets enclosed in ZIRLO® or Optimized ZIRLO™ High Performance Fuel Cladding Material with welded end plugs. The use of Optimized ZIRLO material was approved by NRC Safety Evaluation dated May 9, 2014 (Reference 5). The tubes are supported in assemblies by a spring clip grid structure. The mechanical control rods consist of clusters of stainless steel clad absorber rods which are inserted into ZIRLO guide tubes located within the fuel assembly. The core fuel is loaded during each refueling in accordance with a loading pattern designed and analyzed to achieve the desired thermal and nuclear characteristics. The steam generators are vertical U-tube units utilizing inconel tubes. Integral separating equipment reduces the moisture content of the steam at the steam generator outlet to 1/4% or less.

The reactor coolant pumps are vertical, single stage, centrifugal pumps equipped with controlled leakage shaft seals.

Auxiliary systems are provided to charge the reactor coolant system and to add makeup water, purify and degas reactor coolant water, provide chemicals for corrosion inhibition and reactor control, cool system components, remove residual heat when the reactor is shut down, cool the spent fuel storage pool, sample reactor coolant water, provide for emergency safety injection, and vent and drain the reactor coolant system.

1.2.3 REACTOR AND PLANT CONTROL

The reactor is controlled by a coordinated combination of chemical shim and mechanical control rods. The control system allows the plant to accept step load changes of 10% and ramp load changes of 5% per minute over the load range of 15 to 100% power under nominal operating conditions. It is also designed to sustain reactor operation following a rapid load decrease of 50% power at a rate up to 200%/minute with the steam and atmospheric dumps available.

Complete supervision of both the reactor and turbine generator is accomplished from the control room. Units 1 and 2 share the control room located as an integral part of the turbine hall. The control room layout, including location of control boards for each unit, is shown in Figure 7.5-1.

The control room for the combined plant is approximately 50'× 80'. Annunciators for alarms for the two units are on different control boards with the exception that safeguards and electrical system alarms are on common control boards. The Auxiliary Safety Instrumentation Panels (ASIPs) described in Chapter 7 are common panels with unit specific and common alarms.

The waste disposal control board is located in the auxiliary building. This board permits the auxiliary operator to control and monitor the processing of wastes from a central location in the same general area where equipment is located.

1.2.4 WASTE DISPOSAL SYSTEM

The waste disposal system, common to both units, provides all equipment necessary to collect, process, and prepare for disposal all potentially radioactive liquid, gaseous, and solid wastes produced as a result of reactor operation.

Liquid wastes are processed through a filtration and demineralization system. The processed liquid is sampled to determine residual activity and monitored during discharge to the lake via the condenser circulating water discharge to assure concentrations below 10 CFR 20 limits. Exhausted filtration and demineralization media is dewatered and packaged for shipping from the site for ultimate disposal in an authorized location.

Liquid wastes can also be processed through the blowdown evaporator. The condensate from the blowdown evaporator is sampled to determine residual activity and monitored during discharge to the lake via the condenser circulating water discharge to assure concentrations below 10 CFR 20 limits. The evaporator residues are pumped to the PAB Truck Bay for processing and shipping from the site for ultimate disposal in an authorized location.

Gaseous wastes are collected and stored until their radioactivity level is low enough so that discharge to the environment does not create radioactivity concentrations above 10 CFR 20 limits. Measures provided for the purpose of keeping releases of radioactive materials to unrestricted areas during normal reactor operations, including expected operational occurrences, as low as reasonably achievable are presented in Chapter 11 to this document.

1.2.5 FUEL HANDLING SYSTEM

Each reactor is refueled with equipment designed to handle spent fuel under water from the time it leaves either reactor vessel until it is placed in a cask for shipment from the site. Underwater transfer of spent fuel provides an optically transparent radiation shield, as well as a reliable source of coolant for removal of decay heat. This system also provides capability for receiving, handling, and storage of new fuel. Both the new fuel storage facility and the spent fuel storage facility are shared by the two units.

1.2.6 TURBINE AND AUXILIARIES

Each turbine is a tandem-compound, 3-element, 1,800 rpm unit. Four moisture separator reheater units are employed to dry and superheat the steam between the high and low pressure turbine cylinders.

Single-pass de-aerating, radial flow surface condensers, steam-jet air ejector, two 50% capacity condensate pumps, two 50% capacity motor-driven feedwater pumps, and five stages of feedwater heaters are provided. One steam-driven and one motor-driven auxiliary feedwater pump per unit are available to remove residual heat in case of a complete loss of auxiliary power.

1.2.7 ELECTRICAL SYSTEM

Each main generator is an 1,800 rpm, 3-phase, 60 cycle, hydrogen inner-cooled unit. Three single phase main step-up transformers on each unit deliver power to the 345 kV switchyard.

The Station Service System consists of auxiliary transformers, 4.16 kV switchgear, 480V motor control centers, and 125V DC and 120V AC equipment.

Emergency power is supplied by four emergency diesel generators. Each emergency diesel generator (DG) is capable of operating one train of post-accident containment cooling equipment as well as high head and low head safety injection pumps to ensure an acceptable post-loss-of-

coolant containment pressure transient. Sufficient power capacity is provided to safely shut down the unaffected unit at the same time adequate power is provided to the engineered safety features of the affected unit.

1.2.8 ENGINEERED SAFETY FEATURES SYSTEMS

The engineered safety features (ESF) systems provided for this plant have redundant components and power sources such that under the conditions of a hypothetical loss-of-coolant accident, the systems can, even when operating with partial effectiveness, maintain the integrity of the containment and keep the exposure of the public below the limits of 10 CFR 50.67.

The ESF systems provided are summarized below:

1. Each containment system provides a highly reliable, essentially leak-tight barrier against the escape of fission products. These provisions minimize leakage to the environment.

2. Each safety injection system (SI) provides borated water to cool the core by redundant injection into the cold legs of the reactor coolant loops and by discharging coolant over the top of the core via injection through the core deluge nozzles.

3. Each containment air recirculation cooling system (VNCC) provides a dynamic heat sink to cool the containment atmosphere under the conditions of a loss-of-coolant accident. The system utilizes the normal containment ventilation and cooling equipment.

4. Each containment spray system (SI) provides a spray of cool, chemically treated borated water to the containment atmosphere to provide removal of elemental iodine and particulates and works independent of the containment air recirculation cooling system to remove heat.

1.2.9 SHARED FACILITIES AND EQUIPMENT

Per GDC 4, Reactor Facilities may share systems or components if it can be shown that such sharing will not result in undue risk to the health and safety of the public.

Separate and similar systems and equipment are provided for each unit and are described in Appendix A.6. In these instances where some components of a system are shared by both units, only those components which are shared are shown. A functional evaluation of the components of the systems which are shared by the two units is provided in Appendix A.6 together with a short discussion on the operation of those items of shared equipment which are components of the engineered safety features system.

1.2.10 INDEPENDENT SPENT FUEL STORAGE INSTALLATION

The Point Beach Nuclear Plant site has an Independent Spent Fuel Storage Installation (ISFSI) that was built to accommodate dry storage containers of spent nuclear fuel from the spent fuel pool. The ISFSI was constructed because the national spent nuclear fuel disposal facility was not ready to accept spent fuel. Without removal of fuel from the spent fuel pool fuel storage racks,

the racks would have been full before the end of license life, resulting in premature shutdown of the plant. The ISFSI is shown on Figure 2.2-4.

Complete information on the licensing of the ISFSI may be found in Reference 1 and Reference 2.

REFERENCES:

1. Point Beach 10 CFR 72.212 and Certificate of Compliance Evaluation Report for VSC-24 System.

2. Point Beach 10 CFR 72.212 and Certificate of Compliance Evaluation Report for NUHOMS®-32PT System.

3. NRC Safety Evaluation 2011-0004, "Issuance of License Amendments Regarding Extended Power Uprate," dated May 3, 2011.

4. NRC Safety Evaluation 2011-0003, "Issuance of License Amendments Regarding Use of Alternate Source Term," dated April 14, 2011.

5. NRC Safety Evaluation, "Issuance of Amendment Regarding the Use of Optimized ZIRLO™ Fuel Rod Cladding Material," dated May 9, 2014.

Figure 1.2-1 CONTAINMENT LAYOUT PLAN EQUIPMENT ARRANGEMENT

[redacted]

Figure 1.2-2 EQUIPMENT LOCATION PLAN UNIT 1

[redacted]

Figure 1.2-3 EQUIPMENT LOCATION PLAN UNIT 1

[redacted]

Figure 1.2-4 UNIT-1 EQUIPMENT LOCATION PLAN

[redacted]

Figure 1.2-5 UNIT-1 EQUIPMENT LOCATION PLAN

[redacted]

Figure 1.2-6 EQUIPMENT LOCATION PLAN

[redacted]

Figure 1.2-7 UNIT 1 EQUIPMENT LOCATION - SECTIONS

[redacted]

Figure 1.2-8 MISCELLANEOUS SECTIONS UNIT 1

[redacted]

Figure 1.2-9 UNIT-2 EQUIPMENT LOCATION - PLAN

[redacted]

Figure 1.2-10 EQUIPMENT LOCATION PLAN UNIT 2

[redacted]

Figure 1.2-11 EQUIPMENT LOCATION PLAN UNIT 2

[redacted]

Figure 1.2-12 UNIT 2 EQUIPMENT LOCATION - PLAN

[redacted]

Figure 1.2-13 UNIT-2 EQUIPMENT LOCATION - PLAN

[redacted]

Figure 1.2-14 MISCELLANEOUS SECTIONS UNIT 2

[redacted]

Figure 1.2-15 GENERAL ARRANGEMENT - WASTE DISPOSAL SYSTEM MODIFICATIONS

[redacted]

1.3 GENERAL DESIGN CRITERIA

The general design criteria define the principal criteria and safety objectives for the design of this plant. A complete set of these GDCs are stated explicitly in Table 1.3-1. Table 1.3-1 also identifies other locations in this report that repeat specific GDCs.

Regarding the origin of these criteria, the Atomic Energy Commission (AEC) published proposed GDCs for public comment in 1967. The Atomic Industrial Forum (AIF) reviewed these proposed criteria and recommended changes. The Point Beach GDCs documented in this FSAR are similar in content to the AIF version of the Proposed 1967 GDCs.

Appendix A of 10 CFR 50 contains a different set of GDCs which were published in 1971 (After Point Beach construction permits were issued). Note that the GDCs found in 10 CFR 50 Appendix A differ both in numbering and content from the GDCs adopted herein for PBNP.

The parenthetical numbers following the section headings indicate the numbers of the proposed General Design Criterion (GDC).

1.3.1 OVERALL PLANT REQUIREMENTS (GDC 1- GDC 5)

All systems and components of the facility are classified according to their importance. The original classification system at PBNP used designators called Class I, Class II and Class III. Those items vital to safe shutdown and isolation of the reactor, or whose failure might cause or increase the severity of an accident or result in an uncontrolled release of excessive amounts of radioactivity were designated Class I. Class I systems and components were considered essential to the protection of the health and safety of the public. Those items important to reactor operation, but not essential to safe shutdown and isolation of the reactor or control of the release of substantial amounts of radioactivity were designated Class II. Those items not related to reactor operation or safety were designated Class III.

Subsequent evaluation of the equipment classification system pursuant to NRC Generic Letter 83-28 resulted in the definition of safety-related functions and the related classification criteria described in more detail in the Quality Assurance Program section of the FSAR (1.4).

These safety classifications are: Safety-Related, Augmented Quality, and Non-Safety-Related. After the adoption of these classifications pursuant to Generic Letter 83-28, PBNP systems and components were reclassified accordingly. Although there may be some commonality between the original Class I category and the Safety-Related category, it is important to note that these classifications are defined differently and represent different time periods of plant operation. Quality standards of material selection, design, fabrication, and inspection conform to the applicable provisions of recognized codes and good nuclear practice.

All systems and components designated Seismic Class I are designed so that there is no loss of function in the event of the maximum hypothetical ground acceleration acting in the horizontal and vertical directions simultaneously. The working stress for both Seismic Class I and Seismic Class II items is kept within code allowable values for the design earthquake. Similarly, measures are taken in the plant design to protect against high winds, flooding, and other natural phenomena.

The containments and Seismic Class I portions of the Auxiliary Building, the turbine hall, the pumphouse, and the diesel generator building are designed to withstand the effects of a tornado. The design criteria of the containment and the Class I portions of the auxiliary and turbine buildings to withstand the effects of a tornado, including wind force, pressure differential, and missile impingement are described in Bechtel Topical Report B-TOP-3, "Design Criteria for Nuclear Power Plants Against Tornadoes." Design criteria for the diesel generator building are described in FSAR Reference D. The design of the pumphouse to withstand tornados and tornado missiles is described in Section 9.6. Seismic design criteria are described in FSAR Reference A.5.

The design basis for tornado missile protection of systems and components is that it is possible to shut the plant down and keep it in hot shutdown during and after the passage of a tornado. The equipment needed for this event remains operable if (Reference 1):

a) Critical items are housed in structures capable of withstanding tornado winds, depressurization, and missiles;

OR

b) the separation provided between redundant systems or components is such that reasonable assurance exists that a single missile cannot render both systems or components inoperable; and large structures, such as facade, auxiliary building superstructure, turbine buildings, etc., are so designed that they will not collapse and fall on redundant components or systems.

Reference Sections:

Section Title	Chapter
SITE AND ENVIRONMENT; METEOROLOGY, SEISMOLOGY	2.0
REACTOR COOLANT SYSTEM; DESIGN BASIS (RCS)	4.1
CONTAINMENT SYSTEM STRUCTURE; DESIGN BASIS (CONT)	5.1
ELECTRICAL SYSTEM; DESIGN BASES	8.0
FUEL HANDLING SYSTEM (FH)	9.4
SERVICE WATER SYSTEM (SW)	9.6
CLASS I DESIGN CRITERIA FOR VESSELS, AND STRUCTURES	Reference A.5
DIESEL GENERATOR PROJECT	Reference D

Refer to the Fire Protection Evaluation Report (FPER) for design philosophy and specifics concerning fire protection at Point Beach Nuclear Plant.

A complete set of as-built facility plant and system diagrams, including arrangement plans and structural plans, and records of initial tests and operation are maintained throughout the life of the plant. A set of all the quality assurance data generated during fabrication and erection of the essential components of the plant, as defined by the quality assurance program, is retained.

Reference Sections:

Section Title	Chapter
RECORDS	12.5
SITE SURVEILLANCE AND TESTING PROGRAMS	13

1.3.2 PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS (GDC 6-GDC 10)

Each reactor core, with its related control and protection system, is designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The core design, together with reliable process and decay heat removal systems, provides for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations.

Each reactor control and protection system is designed to actuate a reactor trip for any anticipated combination of plant conditions, when necessary, to ensure a minimum Departure from Nucleate Boiling (DNB) ratio equal to or greater than the limit value.

Reference Sections:

Section Title	Chapter
REACTOR, DESIGN BASIS	3.1
INSTRUMENTATION AND CONTROL, Protective Systems (RP)	7.2
SAFETY ANALYSIS	14.0

The design of the reactor core and related protection systems ensures that power oscillations which could cause fuel damage in excess of acceptable limits are not possible or can be readily suppressed. The potential for possible spatial oscillations of power distribution for these cores has been reviewed. It is concluded that low frequency xenon oscillations may occur in the axial dimension, and part length control rods were initially provided to suppress these oscillations. Experience has demonstrated that full length rods are effective in controlling these oscillations and the part length control rods have been removed. The core has been stable with respect to xenon oscillations in the X-Y dimension.

Out-of-core instrumentation is provided to obtain necessary information concerning power distribution. This instrumentation is adequate to enable the operator to monitor and control xenon induced oscillations.

The moderator temperature and overall power coefficient in the power operating range is maintained negative by inclusion of burnable poison shims, as necessary, dependent on a particular core reload.

Reference Sections:

Section Title	Chapter
REACTOR DESIGN, NUCLEAR DESIGN AND EVALUATION	3.0
PRIMARY SYSTEM PIPE RUPTURE	14.3

Each reactor coolant system, in conjunction with its control and protective provisions, is designed to accommodate the system pressures and temperatures attained under all expected modes of plant operation or anticipated system interactions, and maintain the stresses within applicable code stress limits.

The materials of construction of the pressure boundary of the reactor coolant system are protected, by control of coolant chemistry, from corrosion phenomena which might otherwise reduce the system structural integrity during its service lifetime.

System conditions resulting from anticipated transients or malfunctions are monitored, and appropriate action is automatically initiated to maintain the required cooling capability and to limit system conditions to a safe level.

The system is protected from overpressure by means of pressure relieving devices, as required by Section III of the ASME Boiler and Pressure Vessel Code.

Sections of the system that can be isolated are provided with overpressure relieving devices discharging to closed systems, such that the system allowable pressure within the protected section is not exceeded

Reference Section:

Section Title	Chapter
DESIGN BASIS, REACTOR COOLANT SYSTEM (RCS)	4.1

The containment design pressure and temperature exceeds the peak pressure and temperature occurring as the result of the complete blowdown of the reactor coolant through any pipe rupture of the reactor coolant system up to and including the hypothetical severance of a reactor coolant pipe.

The penetration for the main steam, feedwater, blowdown, and sample lines are designed so that the penetration is stronger than the piping system and the vapor barrier will not be breached due to a hypothesized pipe rupture. All lines connected to the reactor coolant system that penetrate the vapor barrier are also anchored in the loop compartment shield walls and are each provided with at least one valve between the anchor and the coolant system. These anchors are designed to withstand the thrust moment and torque resulting from a hypothesized rupture of the attached pipe or the loads induced by the maximum hypothetical earthquake.

All isolation valves are supported to withstand, without impairment of valve operability, the loading of the design basis accident or maximum hypothetical seismic conditions

Reference Section:

Section Title	Chapter
CONTAINMENT SYSTEM STRUCTURE (CONT)	5.1

1.3.3 NUCLEAR AND RADIATION CONTROLS (GDC 11 - GDC 18)

The plant is equipped with a control room which contains the controls and instrumentation necessary for operation of both reactors and turbine generators under normal and accident conditions.

Sufficient shielding, distance, ventilation-purification, and containment integrity are provided to assure that control room personnel shall not be subjected to doses under postulated accident conditions during occupancy of, ingress to, and egress from the control room which, in the aggregate, would exceed 5 rem total effective dose equivalent (TEDE), or its equivalent to any part of the body, for the duration of the accident.

For each unit, instrumentation and controls essential to avoid undue risk to the health and safety of the public are provided to monitor and maintain neutron flux, primary coolant pressure, flow rate, temperature, and control rod positions within prescribed operating ranges.

Other instrumentation and control systems are provided to monitor and maintain within prescribed operating ranges the temperatures, pressures, flows, and levels in the reactor coolant systems, steam systems, containments, and other auxiliary systems. The quantity and types of instrumentation provided are adequate for safe and orderly operation of all systems and processes over the full operating range of the plant.

The operational status of each reactor is monitored from the control room. When the reactor is subcritical, the neutron source multiplication is continuously monitored and indicated by proportional counters located in instrument wells in the primary shield adjacent to the reactor vessel. Neutron sources can be installed in the core, if necessary, during startup to provide a minimum count rate for verifying operation of the source detector channels. Any appreciable increase in the neutron source multiplication, including that caused by the maximum physical boron dilution rate, is slow enough to give ample time to start corrective action (boron dilution stop and/or emergency boron injection) to prevent the core from becoming critical inadvertently.

Means for showing the relative reactivity status of each reactor is provided by control bank positions displayed in the control room. Periodic samples of coolant boron concentration are taken. The variation in concentration during core life provides a further check on the reactivity status of the reactor, including core depletion.

Instrumentation and controls provided for the protective systems are designed to trip the reactors when necessary to prevent or limit fission product release from the cores and to limit energy release; to signal containment isolation; and to control the operation of engineered safety features equipment.

During reactor operation in the startup and power modes, redundant safety limit signals will automatically actuate two reactor trip breakers which are in series with the rod drive mechanism coils. The action would interrupt rod drive power and initiate reactor trip

Reference Section:

Section Title	Chapter
INSTRUMENTATION AND CONTROL	7.0

If the reactor protection system receives signals which are indicative of an approach to an unsafe operating condition, the system actuates alarms, prevents control rod out motion, initiates load cutback, and/or opens the reactor trip breakers.

The basic reactor tripping philosophy is to define an allowable region of power and coolant temperature conditions. This allowable range is defined by the primary tripping functions, the overpower high ΔT trip, overtemperature high ΔT trip, and the nuclear overpower trip. The operating region below these trip settings is designed so that no combination of power, temperatures, and pressure could result in a Departure from Nucleate Boiling Ratio (DNBR) less than the limit value. Additional tripping functions such as a high pressurizer water level trip, loss of flow trip, steam and feedwater flow mismatch trip, steam generator low-low level trip, turbine trip, safety injection trip, nuclear source and intermediate range trips, and manual trip are provided to back up the primary tripping functions for specific accident conditions and mechanical failures.

Rod stops from nuclear overpower, overpower ΔT and overtemperature ΔT deviation are provided to prevent abnormal power conditions which could result from excessive control rod withdrawal initiated by a malfunction of the reactor control system or by operator violation of administrative procedures.

Reference Sections:

Section Title	Chapter
ENGINEERED SAFETY FEATURES (ESF)	6.0
REACTOR PROTECTION SYSTEM (RPS)	7.2

Positive indication in the control room of leakage of coolant from the reactor coolant systems to the containments is provided by equipment which permits continuous monitoring of the containment air activity and humidity, and is provided by the runoff from the condensate collecting pans under the cooling coils of the containment air recirculation units. The basic design criterion is the detection of deviations from normal containment environmental conditions including air particulate activity, radiogas activity, humidity, condensate and floor drain runoff, and in addition, in the case of gross leakage, the liquid inventory in the process systems and containment sump.

The containment atmosphere, the plant vents, the containment service water discharges, the condenser air ejectors, the steam generator blowdown effluents, and the Waste Disposal System liquid effluent are monitored for radioactivity concentration during all normal operations, anticipated transients, and accident conditions.

For the case of leakage from the reactor containment under accident conditions, the plant area radiation monitoring system supplemented by portable survey equipment provides adequate monitoring of releases during an accident.

Monitoring and alarm instrumentation are provided for fuel and waste storage and handling areas to detect inadequate cooling and to detect excessive radiation levels. Radiation monitors are provided to maintain surveillance over the release of radioactive gases and liquids.

Controlled ventilation systems remove gaseous radioactivity from the atmosphere of the fuel storage and waste treating areas of the auxiliary building and discharge it to the atmosphere via the vents. Radiation monitors are in continuous service in these areas to actuate high-activity alarms on the control board annunciator, as described in Chapter 11.0.

Reference Sections:

Section Title	Chapter
ENGINEERED SAFETY FEATURES (ESF)	6.0
AUXILIARY COOLANT SYSTEM (CC, SF, SW)	9.0
RADIATION PROTECTION (RM)	11.0

1.3.4 RELIABILITY AND TESTABILITY OF PROTECTION SYSTEMS (GDC 19 - GDC 26)

Upon a loss of power to the gripper coils, the rod cluster control (RCC) assemblies are released and fall by gravity into the core. The reactor internals, fuel assemblies, RCC assemblies, and drive system components are designed as Safety-Related equipment. The RCC assemblies are fully guided through the fuel assembly and for the maximum travel of the control rod into the guide tube. Furthermore, the RCC assemblies are never fully withdrawn from their guide tube thimbles in the fuel assembly while in the core. As a result of these design safeguards and the flexibility designed into the RCC assemblies, abnormal loading and misalignments can be sustained without impairing operation of the RCC assemblies.

Protection channels are designed with sufficient redundancy for individual channel calibration and test to be made during operation without degrading the reactor protection system. Bypass removal of one trip circuit is accomplished by placing that channel in a partial-tripped mode, i.e., a two-out-of-three trip matrix becomes a one-out-of-two trip matrix. Testing does not cause a trip unless a trip condition exists in a channel not being tested. The trip signal furnished by the remaining channels is unimpaired by testing.

In the reactor protection system (RP) of each unit, two reactor trip breakers are provided to interrupt power to the RCCA drive mechanisms.

The breaker main contacts are connected in series (with the power supply) so that opening either breaker interrupts power to all RCC assemblies permitting them to fall by gravity into the core. Each trip breaker is opened through an undervoltage or shunt trip coil. Each protection channel actuates two separate trip logic trains, one for each reactor trip breaker. The protection system is thus inherently safe in the event of a loss of rod control power.

Channel independence is carried throughout the system extending from the sensor to the relay actuating the protective function. The protective and control functions are combined only through isolation devices. A failure in the control circuit does not affect the protection channel.

The power supplied to the channels is fed from four 120 volt instrument buses for each unit. Each instrument bus is supplied from an inverter. The inverters are supplied from the common 125 volt DC buses. Each of the four DC buses are connected to a plant battery.

The initiation of the engineered safety features provided for loss-of-coolant accidents; e.g., high head safety injection and residual heat removal pumps, and containment spray systems, is accomplished from redundant signals derived from reactor coolant system and containment instrumentation. The initiation signal for containment spray comes from coincidence of two sets of two-out-of-three high containment pressure signals. On loss of voltage of a safety features equipment bus, the diesel generator aligned to that bus will be automatically started and connected to the bus. Automatic safety injection actuation actuates containment isolation.

The components of the protection system are designed and laid out so that the mechanical and thermal environment accompanying any emergency situation in which the components are required to function does not interfere with that function.

Each protection channel in service at power is capable of being calibrated and tripped independently by simulated signals to verify its operation without tripping the plant.

Each reactor trip circuit is designed so that trip occurs when the circuit is de-energized.

Therefore, an open circuit or loss of channel power causes the system to go into its trip mode. In a two-out-of-three circuit, the three channels are equipped with separate primary sensors and each channel is energized from independent electrical buses.

Redundancy in emergency power is provided in that there are four diesel generator sets capable of supplying separate 4.16 kV buses. One complete set of safety features equipment for both units is, therefore, capable of being independently supplied from either one of the two diesels associated with a train, or both diesels each supplying safety features equipment for one unit.

Diesel engine cranking is accomplished by a Diesel Air Starting System (DA) supplied solely for the associated diesel generator. The undervoltage relay scheme is designed so that loss of power does not prevent the relay scheme from functioning properly.

The ability of the diesel generator sets to start within the prescribed time and to carry load can periodically be checked. The diesel generator breaker is not closed automatically after starting during this testing. The generator may be manually synchronized to 4.16 kV bus for loading.

Reference Section:

Section Title	Chapter
INSTRUMENTATION AND CONTROL; PROTECTION SYSTEMS (RPS)	7.2

1.3.5 REACTIVITY CONTROL (GDC 27 - GDC 32)

In addition to the reactivity control achieved by the RCC assemblies as detailed in Chapter 7.0, reactivity control is provided by the chemical and volume control system which regulates the concentration of boric acid solution neutron absorber in the reactor coolant system. The system is designed to prevent uncontrolled or inadvertent reactivity changes which might cause system parameters to exceed design limits. The reactivity control systems provided are capable of making and holding the core subcritical from any cold shutdown, hot shutdown, or hot operating condition, including those resulting from power changes.

The RCC assemblies are divided into categories comprising control and shutdown groups. One control group of RCC assemblies is used to compensate for short term reactivity changes at power such as those produced due to variations in reactor power requirements or in coolant temperature. The chemical shim control is used to compensate for the more slowly occurring changes in reactivity throughout core life such as those due to fuel depletion, fission product buildup and decay, and load follow.

The shutdown groups are provided to supplement the control groups of RCC assemblies to make the reactor at least 1% subcritical (K_{eff} = 0.99) following trip from any credible operating condition to the hot, zero power condition assuming the most reactive RCC assembly remains in the fully withdrawn position.

Any time that the plant is at power, the quantity of boric acid retained in the boric acid storage tanks or the refueling water storage tank (RWST) and ready for injection will always exceed that quantity required for normal cold shutdown of both units.

For each unit, boric acid may be pumped from the boric acid storage tanks by one of two boric acid transfer pumps (or via gravity feed from the RWST) to the suction of one of three charging pumps which inject boric acid into the reactor coolant. Any charging pump and any boric acid transfer pump can be operated from diesel generator power on loss of offsite power. Boric acid can be injected by one charging pump supplied by one boric acid transfer pump to take the reactor to hot shutdown, with no rods inserted, in less than 120 minutes. In 120 additional minutes, enough boric acid can be injected to compensate for xenon decay. If two charging pumps are available, the time is reduced. Additional boric acid injection is employed if it is desired to bring the reactor to cold shutdown conditions.

The reactor protection systems are designed to limit reactivity transients to DNBR $\geq$ the limit value due to any single malfunction in the deboration controls.

Limits, which include considerable margin, are placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals so as to lose capability to cool the core.

The rod cluster drive mechanisms are wired into preselected groups, and are normally prevented from being withdrawn in other than their respective groups. The control and shutdown rod drive mechanisms are of the magnetic latch type and the coil actuation is programmed to provide

variable speed rod travel. The maximum insertion rate is analyzed in the detailed plant analysis assuming two of the highest worth groups to be accidentally withdrawn at maximum speed, yielding reactivity insertion rates of the order of 6 x 10^{-4} Δk/sec, which is well within the capability of the reactor protection circuits to prevent core damage.

Reference Sections:

Section Title	Chapter
REACTOR DESIGN BASIS	3.1
PROTECTION SYSTEMS (RPS)	7.2
REGULATING SYSTEMS (RDC)	7
CHEMICAL AND VOLUME CONTROL SYSTEM (CV)	9.3

1.3.6 REACTOR COOLANT PRESSURE BOUNDARY (GDC 33 - GDC 36)

The reactor coolant boundary is shown to be capable of accommodating, without rupture, the static and dynamic loads imposed as a result of a sudden reactivity insertion such as a rod ejection. The operation of the reactor is such that the severity of an ejection accident is inherently limited. Since RCC assemblies are used to control load variations only and boron dilution is used to compensate for core depletion, only the RCC assemblies in the controlling groups are inserted in the core at power, and at full power these rods are only partially inserted. Rod insertion alarms are provided as an aid to the operator to ensure that this condition is met.

By using the flexibility in the selection of control rod groupings, radial locations, and position as a function of load, the design limits the maximum fuel temperature for the highest worth ejected rod to a value which precludes any resultant damage to the primary system pressure boundary from possible excessive pressure surges.

The failure of a rod mechanism housing causing a rod cluster to be rapidly ejected from the core is evaluated as a theoretical, though not a credible accident. While limited fuel damage could result from this hypothetical event, the fission products are confined to the reactor coolant system and the reactor containment.

The reactor coolant pressure boundary is designed to reduce, to an acceptable level, the probability of a rapidly propagating type failure.

The fracture toughness of the materials in the beltline region of the reactor vessel will decrease as a result of fast neutron irradiation induced embrittlement. Fracture toughness will decrease with increasing the reference nil ductility temperature (RT_{NDT}) which increases as a function of several factors, including accumulated fast neutron fluence. This change in material properties is factored into the operating procedures such that the reactor coolant system pressure is limited with respect to RCS temperature during plant heatup, cooldown, and normal operation. These limits are determined in accordance with the methods of analysis and the margins of safety of Appendix G of ASME Code Section XI and are included in the Point Beach Pressure Temperature Limits Report (PTLR).

The design of the reactor vessel and its arrangement in the system permits accessibility during the service life to the entire internal surfaces of the vessel and to the following external zones of the vessel: the flange seal surface, the flange O.D. down to the cavity seal ring, the closure head and the nozzle to reactor coolant piping welds. The reactor arrangement within the containment provides sufficient space for inspection of the external surfaces of the reactor coolant piping, except for the length of pipe within the primary shielding concrete.

To define permissible operating conditions, a pressure range is established which is bounded by a lower limit for pump operation and an upper limit that satisfies the criteria of ASME Code Section XI, Appendix G, “Protection Against Nonductile Failure.” The criteria of Appendix G of the ASME Code also ensures that the reactor vessel temperature for normal operation is maintained such that brittle fracture is not considered to be credible.

Monitoring of the RT_{NDT} of the beltline region plates, forgings, weldments and associated heat affected zone materials is performed in accordance with ASTM E 185-82 (Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels). In addition to the required tension and Charpy impact specimens, the Point Beach material surveillance program also includes fracture toughness specimens. Additional samples of reactor vessel plate and forging materials have been retained and catalogued and are available for future testing, as needed.

The measured shift in RT_{NDT} of the beltline region materials with irradiation are used to establish plant specific values of shift in accordance with the regulatory guidance of NRC Regulatory Guide 1.99, Rev. 2, “Radiation Embrittlement of Reactor Vessel Materials.” Where credible data is not available for specific weld or base metals, Regulatory Guide 1.99 provides trend curves for the shift in RT_{NDT} based on fast neutron fluence, material form (base or weld metal), and the weight-percent of copper and nickel of the reactor vessel steel. A margin term is also added to the shift to obtain conservative, upper-bound values of the adjusted RT_{NDT} for use in the evaluations required by Appendix G to 10 CFR 50. See Section 15.4.1 for the discussion of the fracture toughness methodology evaluation reviewed and approved by the NRC for License Renewal for Unit 2. (NRC SE dated 12/2005, NUREG-1839)

As a supplement to the plant specific material surveillance program for Point Beach, additional surveillance data is available through participation in the Babcock & Wilcox Owners Group Master Integrated Reactor Vessel Surveillance Program. This integrated program includes weld metals used in the construction of the Point Beach reactor vessels that are not included in the plant specific surveillance program for Point Beach.

Reference Sections:

Section Title	Chapter
REACTOR COOLANT SYSTEM (RCS)	4.1
RCS SYSTEM DESIGN AND OPERATION	4.0
SYSTEM DESIGN EVALUATION	4.3
VESSEL RT_{NDT}	4.0

1.3.7 ENGINEERED SAFETY FEATURES (GDC 37 - GDC 65)

The design, fabrication, testing, and inspection of the core, reactor coolant pressure boundary, and their protection systems give assurance of safe and reliable operation under all anticipated normal, transient, and accident conditions.

However, engineered safety features are provided in the facility to back up the safety provided by these components. These engineered safety features have been designed to cope with any size reactor coolant pipe break up to and including the circumferential rupture of any pipe assuming unobstructed discharge from both ends, and to cope with any steam or feedwater line break up to and including the main steam or feedwater headers. The concurrent, total loss of all offsite power is assumed with these accidents.

The release of fission products from the reactor fuel is limited by the Safety Injection System which, by cooling the core and limiting the fuel cladding temperature, keeps the fuel in place and substantially intact with its heat transfer geometry preserved and limits the metal-water reaction to an insignificant amount.

The basic criteria for loss-of-coolant accident evaluations (discussed in Chapter 6.0) are: no cladding melting, Zircaloy-water reactions will be limited to an insignificant amount and the core geometry is to remain essentially in place and intact so that effective cooling of the core will not be impaired. The Zircaloy-water reactions will be limited to an insignificant amount so that the accident:

1. Does not interfere with the emergency core cooling function to limit cladding temperatures.

2. Does not produce H_2 in an amount that when burned would cause the containment pressure to exceed the design value.

For any rupture of a steam pipe and the associated uncontrolled heat removal from the core, the emergency core cooling system adds shutdown reactivity so that with a stuck rod, loss of off-site power, and minimum engineered safety features, there is no consequential damage to the primary system and the core remains in place and intact. With no stuck rod, no off-site power, and all equipment operating at design capacity, there is insignificant cladding rupture.

The safety injection system (SI) consists of high and low head centrifugal pumps driven by electric motors, and passive accumulator tanks which are self energized and which act independently of any actuation signal or power source.

The release of fission products from the containment is limited in three ways:

1. Blocking the potential leakage paths from the containment. This is accomplished by:

    a. A steel-lined, concrete reactor containment with testable penetrations and liner weld channels.

    b. Isolation of process lines by the containment isolation system which imposes double barriers for each line which penetrates the containment.

2. Reducing the fission product concentration in the containment atmosphere. This is accomplished by spraying chemically treated borated water which removes airborne elemental iodine and particulates by washing action.

3. Reducing the containment pressure and thereby limiting the driving potential for fission product leakage by cooling the containment atmosphere using the following independent systems:

   a. Containment Spray System (SI)

   b. Containment Air Recirculation Cooling System (VNCC)

A comprehensive program of plant testing is formulated for all equipment systems and system control vital to the functioning of engineered safety features. The program consists of performance tests of individual pieces of equipment in the manufacturer's shop, integrated tests of the system as a whole, and periodic tests of the actuation circuitry and mechanical components to assure reliable performance upon demand throughout the plant lifetime. In the event that one of the components should require maintenance as a result of failure to perform during the test according to prescribed limits, the necessary corrections or minor maintenance will be made and the unit retested.

The plant is supplied with normal, standby, and emergency power sources as follows:

1. The normal source of auxiliary power for safeguards equipment is the off-site power source. Power is supplied via the high- and low-voltage unit station auxiliary transformers.

2. Four diesel generator sets are connected to the emergency buses to supply power in the event of loss of all other AC auxiliary power. Each of the diesel engine electric generator sets is capable of supplying automatically the engineered safety features load required for an acceptable post-blowdown containment pressure transient for any loss-of-coolant accident, and shutdown of the other unit.

3. Emergency power supply for vital instruments, for control, and for emergency lighting is supplied from the 125V DC station batteries.

The emergency bus electrical power arrangement and logic network provides the capability to manually transfer component loads to another diesel following the failure of one diesel generator unit to start.

For such engineered safety features as are required to ensure safety in the event of such an accident or equipment failure, protection from these dynamic effects or missiles is considered in the layout of plant equipment and missile barriers.[1]

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1. The licensing requirement for protection of plant equipment against the dynamic effects associated with Loss of Coolant Accidents from postulated pipe ruptures is no longer applicable. This was an original design and licensing basis requirement, and the description has been retained because some missiles resulting from other postulated events (RCP flywheel failure, CRDM ejection, etc.) remain. See the discussion of GDC 40 in Section 4.1 for further details.

Layout and structural design specifically protect injection paths leading to unbroken reactor coolant loops against damage as a result of the maximum reactor coolant pipe rupture. Injection lines penetrate the main compartment walls which act as missile barriers. The injection headers are located in the missile-protected area between the compartment walls and the containment outside wall. Individual injection lines are connected to the injection header, pass through the compartment walls, and then connect to the loops. Movement of the injection line, associated with rupture of a reactor coolant loop, is accommodated by line flexibility and by the design of the pipe supports such that no damage outside the missile barrier is possible.[1]

Each engineered safety feature provides sufficient performance capability to accommodate any single failure of an active component and still function in a manner to avoid undue risk to the health and safety of the public.

Under the hypothetical accident conditions, the containment air recirculation cooling system and the containment spray system are designed and sized to rapidly reduce the containment pressure following blowdown. Either of the two spray pumps is capable of providing the necessary iodine and particulate removal.

All active components of the safety injection system (with the exception of injection line isolation valves) and the containment spray system are located outside the containment and not subjected to containment accident conditions.

Instrumentation, motors, cables, and penetrations located inside the containment are selected to meet the most adverse accident conditions to which they may be subjected. These items are either protected from containment accident conditions or are designed to withstand, without failure, exposure to the worst combination of temperature, pressure, and humidity expected during the required operational period.

The reactor is maintained subcritical following a primary system pipe rupture accident. Introduction of borated cooling water into the core results in a net negative reactivity addition. The control rods insert and remain inserted.

The delivery of cold safety injection water to the reactor vessel following accidental expulsion of reactor coolant does not cause further loss of integrity of the reactor coolant system boundary.

Design provisions are made to facilitate access to the critical parts of the reactor vessel internals, injection nozzles, pipes, valves, and safety injection pumps for visual or boroscopic inspection for erosion, corrosion, and vibration wear evidence; and for non-destructive inspection where such techniques are desirable and appropriate.

The design provides for periodic testing of active components of the Safety Injection System for operability and functional performance. If required, the Safety Injection System flow path can be tested during plant operation up to the valves inside the containment using the minimum flow test line. The safety injection (SI) pumps and the residual heat removal (RH) pumps can also be tested during plant operation using the full flow test lines provided. The residual heat removal pumps are also used every time the residual heat removal loop is put into operation.

An integrated system test can be performed when the residual heat removal loop is in service. This test does not introduce flow into the reactor coolant system, but does demonstrate the operation of the valves, pump circuit breakers, and automatic circuitry upon initiation of safety injection.

The accumulators and the safety injection piping up to the final isolation valve is maintained full of borated water at refueling water concentration while the plant is in operation. Flow in each of the high head injection header lines and in the main flow line for the residual heat removal pumps is monitored by a flow indicator.

The design provides for capability to test initially, to the extent practical, the full operational sequence up to the design conditions for the safety injection system to demonstrate the state of readiness and capability of the system. These functional tests provide information to confirm valve operating times, pump motor starting times, the proper automatic sequencing of load addition to the diesel generators, and delivery rates of injection water to the reactor coolant system.

The following general criteria are followed to assure conservatism in computing the required containment structural load capacity:

1. In calculating the containment pressure, rupture sizes up to and including a double-ended severance of reactor coolant pipe are considered.
2. In considering post-accident pressure effects, various malfunctions of the emergency systems are evaluated. Contingent mechanical or electrical failures are assumed to disable one of the diesel generators, such that only two of the four fan-cooler units and one of the two containment spray units operate.
3. The pressure and temperature loadings obtained by analyzing various loss-of-coolant accidents, when combined with operating load and maximum wind or seismic forces, do not exceed the load-carrying capacity of the structure, its access opening, or penetrations.

Discharge of reactor coolant through a double-ended rupture of the main loop piping, followed by operation of only those engineered safety features which can run simultaneously with power from an emergency on-site diesel generator results in a sufficiently low radioactive materials leakage from the containment structure such that there is no undue risk to the health and safety of the public.

The reinforced concrete containment is not susceptible to a low temperature brittle fracture. The containment liner is enclosed within the containment and thus is not exposed to the temperature extremes of the environs. The containment ambient temperature during operation is expected to be well above the NDT temperature +30 °F for the liner material. Containment penetrations which can be exposed to the environment are also designed to the NDT +30 °F criterion.

Isolation valves are provided as necessary for all fluid system lines penetrating the containment to assure at least two barriers for redundance against leakage of radioactive fluids to the environment in the event of a loss-of-coolant accident. These barriers, in the form of isolation valves or closed systems, are defined on an individual line basis. In addition to satisfying containment isolation criteria, the valving is designed to facilitate normal operation and maintenance of the systems and to ensure reliable operation of other engineered safety features.

After completion of the containment structure and installation of all penetration and weld channels, an initial integrated leakage rate test was conducted at the peak calculated accident pressure and maintained for a minimum of 24 hours to verify that the leakage rate was not greater

than 0.4% by weight of the containment volume per day. The Absolute Method was used, and the test continued at a reduced pressure to provide a leak rate versus pressure characteristic curve. Weld channels and double penetrations were not pressurized during this test. A leak rate test at the peak calculated accident pressure using the same method as the initial leak rate test can be performed during the unit shutdown. The allowable leakage rate has since been reduced to 0.2% per day.

Most penetrations are designed with double seals to permit test pressurization of the interior of the penetration. To accomplish this, a supply of clean, dry, compressed air is connected to the penetrations raising the internal pressure to the peak calculated accident pressure. Leakage from the system is checked by either direct flow measurement of the input air, or measurement of the pressure loss. In the event excessive leakage is discovered, penetration groups can then be checked separately.

Capability is provided to the extent practical for testing the functional operability of valves and associated apparatus during periods of reactor shutdown.

Initiation of containment isolation employs coincidence circuits which allow checking of the operability and calibration of one channel at a time.

The main steam and feedwater barriers and isolation valves in systems which connect to the Reactor Coolant System are hydrostatically tested to measure leakage. The main steam isolation valves (MSIVs) can be tested periodically for operability during the life of the plant.

Design provisions are made to the extent practical to facilitate access for periodic visual inspection of important components of the containment air recirculation cooling and containment spray systems. The containment pressure reducing systems are designed to the extent practical so that the spray pumps, spray injection valves, spray nozzles, and additive injection valves can be tested periodically and after any component maintenance for operability and functional performance. Permanent test lines for all the containment spray loops are located so that all components up to the isolation valves at the containment may be tested. These isolation valves are checked separately.

The air test lines, for checking that spray nozzles are not obstructed, connect downstream of the isolation valves. Air flow through the nozzles is monitored by use of a smoke generator or tell-tale devices.

Capability is provided to test initially, to the extent practical, the operational startup sequence beginning with transfer to alternate power sources and ending with near design conditions for the containment spray, including the transfer to the alternate emergency diesel generator power source.

Reference Sections:

Section Title	Chapter
CONTAINMENT SYSTEMS (CONT)	5.0
ENGINEERED SAFETY FEATURES (ESF)	6.0
ELECTRICAL SYSTEM	8.0

1.3.8 FUEL AND WASTE STORAGE SYSTEMS (GDC 66 - GDC 69)

The new fuel storage area is designed so it is impossible to insert assemblies in locations other than those in the new fuel racks. However, the spent fuel storage rack design does not prevent placing assemblies in areas outside the spent fuel storage racks. The minimum spent fuel pool boron concentration specified in Technical Specifications 3.7.11 ensures the K_{eff} storage limit of 0.95 is maintained under postulated accident conditions (Reference TS Bases 3.7.11). Administrative controls ensure fuel is stored in accordance with requirements of criticality analyses discussed in FSAR Section 9.4, Fuel Handling System.

During reactor vessel head removal, and while loading and unloading fuel from the reactor, the boron concentration is maintained at not less than that required to shut down the core to a $K_{eff} = 0.95$. This shutdown margin maintains the core at $K_{eff} < 0.99$ even if all control rods are withdrawn from the core. Periodic checks of refueling water boron concentration ensure the proper shutdown margin.

The design of the fuel handling equipment incorporated built-in interlocks and safety features, the use of detailed refueling instructions, and observance of minimum operating conditions provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety.

The refueling water provides a reliable and adequate cooling medium for spent fuel transfer. Heat removal is accomplished with an auxiliary cooling system.

Adequate shielding for radiation protection is provided during reactor refueling by conducting all spent fuel transfer and storage operations under water. This permits visual control of the operation at all times while maintaining the ability to alert personnel should radiation levels increase during fuel movement. Low and high spent fuel pool water level are alarmed in the control room and corrective action is initiated as necessary. Shielding is provided for waste handling and storage facilities to permit operation within requirements of 10 CFR 20.

The reactor cavity, refueling canal, and spent fuel storage pool are reinforced concrete structures with a seam-welded stainless steel plate liner. These structures are designed to withstand the anticipated earthquake loadings as Seismic Class I structures so the liner will prevent leakage.

Gamma radiation is monitored continuously at various locations in the auxiliary building. A high level signal is alarmed locally and is annunciated in the control room.

Auxiliary shielding for the waste disposal system and its storage components is designed to limit the dose rate levels.

All waste handling and storage facilities are contained and equipment designed so accidental releases directly to the atmosphere are monitored and will not exceed the guidelines of 10 CFR 20, Subpart D. Refer also to Chapter 11.0.

Reference Sections:

Section Title	Chapter
FUEL HANDLING SYSTEM (FH)	9.4
WASTE DISPOSAL SYSTEM (WG, WL, WS)	11.0

Section Title	Chapter
RADIATION PROTECTION (RM)	11.0
SPENT FUEL COOLING & FILTRATION (SF)	9.9

1.3.9 PLANT EFFLUENTS (GDC 70)

Liquid, gaseous, and solid waste disposal facilities are designed so discharge of effluents and off-site shipments are in accordance with applicable governmental regulations.

Radioactive fluids entering the waste disposal system are collected in sumps and tanks until determination of subsequent treatment can be made. They are sampled and analyzed to determine the quantity of radioactivity, with an isotopic identification, if necessary. Before discharge, radioactive fluids are processed as required and then released under controlled conditions. The system design and operation are characteristically directed toward minimizing releases to unrestricted areas. Discharge streams are appropriately monitored and safety features are incorporated to preclude releases in excess of the limits of 10 CFR 20.

Radioactive gases are pumped by compressors through a manifold to one of the gas decay tanks where they are held for a suitable period of time for decay. Cover gases in the nitrogen blanketing system are re-used to minimize gaseous wastes. During normal operation, gases are discharged intermittently at a controlled rate from these tanks through the monitored plant vent.

Liquid wastes are processed to remove radioactive materials. Filter cartridges, the spent resins from the demineralizers, and the concentrates from the evaporators are packaged and stored on-site until shipped off-site for disposal.

Since Alternate Source Term was implemented, the basis for reactor accident dosage level guidelines is 10 CFR 50.67.

Reference Sections:

Section Title	Chapter
WASTE DISPOSAL SYSTEM (WG, WL, WS)	11.0

1.3.10 RESOLUTION OF SYSTEMATIC EVALUATION PROGRAM ISSUES

In 1977, the NRC initiated the Systematic Evaluation Program (SEP) to review the designs of 51 older operating nuclear power plants. Point Beach Units 1 and 2 were listed among the 51 plants. In Phase I of the SEP, the NRC staff defined 137 issues for which the regulatory requirements had changed enough over time to warrant an evaluation of those plants licensed before the issuance of the Standard Review Plan. In Phase II of the SEP, the NRC staff compared the design of 10 of the 51 older plants to the Standard Review Plan issued in 1975. Based on these reviews, the NRC staff identified 27 issues of the original 137 that required some corrective action at one or more of the 10 plants which were reviewed. The staff referred to the issues on this smaller list as the SEP “lessons learned” issues, and concluded that these issues would generally apply to operating plants that received operating licenses before the Standard Review Plan was issued in 1975.

The NRC staff placed each SEP issue into one of the following categories: (1) issues that have been completely resolved (i.e. necessary corrective actions had been identified by the NRC staff, transmitted to licensees and implemented by licensees); (2) issues which are of such low safety significance so as to require no further regulatory action; (3) issues which are unresolved, but for which the NRC staff had identified existing regulatory programs that cover the scope of the technical concerns and whose implementation would resolve the specific SEP issues; and (4) issues which were unresolved, and regulatory actions to resolve the issues had not been identified (Reference 2).

The NRC staff concluded that there were six category 1 or 2 issues that were considered resolved, twenty category 3 issues that would be adequately addressed by ongoing programs, and one category 4 issues that would be resolved using the established generic issues resolution process (Reference 3).

Table 1.3-2 lists the SEP Category 3 and 4 issues that have been resolved on the Point Beach docket using information submitted in support of the Individual Plant Examination of External Events (IPEEE; Reference 4). SEP issues not listed in the table were either dropped by the NRC (e.g. Item 1.6, Turbine Missiles), were resolved by other means (e.g. Item 3.2, Service and Cooling Water under Generic Letters 89-13 and 91-13), or else no specific docketed resolution was identified (e.g. Item 3.4, Isolation of High and Low Pressure Systems).

1.3.11 RESOLUTION OF OTHER ISSUES ADDRESSED BY THE INDIVIDUAL PLANT EXAMINATION OF EXTERNAL EVENTS

In addition to several SEP issues, the IPEEE addressed several other Generic Safety Issues (GSIs) identified by the NRC in Generic Letter 88-20 Supplement 4. The NRC subsequently reviewed and accepted the information contained in the IPEEE submittal, and closed the associated open GSIs based on that information (Reference 5). In addition, during the NRC review of the Extended Power Uprate (EPU) License Amendment Request, the NRC revisited the IPEEE information (Reference 6), thereby incorporating it by reference into the EPU license bases.

Table 1.3-3 lists the Generic Safety Issues that were resolved for Point Beach by the IPEEE submittals and review.

1.3.12 REFERENCES

1. Letter E-R-206 dated October 2, 1969, from R. Salvatori of Westinghouse PWR Systems Division Reliability Group to F. Konchar, Point Beach Project, Point Beach Criteria.

2. SECY-90-343, “Status of the Staff Program to Determine How the Lessons Learned from the Systematic Evaluation Program Have Been Factored Into the Licensing Bases of Operating Plants,” October 4, 1990.

3. NRC Generic Letter 95-04, “Final Disposition of the Systematic Evaluation Program Lessons-Learned Issues,” April 28, 1995.

4. Point Beach Letter VPNPD-95-056, “Generic Letter 88-20, Supplement 4 Summary Report on Individual Plant Examination of External Events for Severe Accident Vulnerabilities,” June 30, 1995.

5. NRC Staff Evaluation Report on Individual Plant Examination of External Events Submittal for Point Beach Units 1 and 2, dated Sept 15, 1999 (ML112030452, SER 1999-0003).

6. NRC Safety Evaluation Report, "Point Beach Nuclear Plant Units 1 and 2 - Issuance of License Amendments Regarding Extended Power Uprate," May 3, 2011 (ML11045159, SER 2011-0004).

Table 1.3-1 POINT BEACH GENERAL DESIGN CRITERIA

CRITERION	DESCRIPTION	FSAR LOCATION(S)
1	Quality Standards Those systems and components of reactor facilities which are essential to the prevention, or the mitigation of the consequences, of nuclear accidents which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required.	4.1, 5.1
2	Performance Standards Those systems and components of reactor facilities which are essential to the prevention or to the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that enable such systems and components to withstand, without undue risk to the health and safety of the public, the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind, or heavy ice. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.	4.1, 5.1, 8.0
3	Fire Protection A reactor facility shall be designed to ensure that the probability of events such as fires and explosions and the potential consequences of such events will not result in undue risk to the health and safety of the public. Noncombustible and fire resistant materials shall be used throughout the facility wherever necessary to preclude such risk, particularly in area containing critical portions of the facility such as containment, control room, and components of engineered safety features.	5.1, 9.10
4	Sharing of Systems Reactor facilities may share systems or components if it can be shown that such sharing will not result in undue risk to the health and safety of the public.	6.1
5	Records Requirement The reactor licensee shall be responsible for assuring the maintenance throughout the life of the reactor of records of the design, fabrication, and construction of major components of the plant essential to avoid undue risk to the health and safety of the public.	4.1, 5.1

Table 1.3-1 POINT BEACH GENERAL DESIGN CRITERIA

CRITERION	DESCRIPTION	FSAR LOCATION(S)
6	Reactor Core Design The reactor core with its related controls and protection systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated.	3.1, 7.1
7	Suppression of Power Oscillations The design of the reactor core with its related controls and protection systems shall ensure that power oscillations, the magnitude of which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed.	3.1, 7.1
8	THIS GDC DOES NOT APPEAR IN THE FSAR	N/A
9	Reactor Coolant Pressure Boundary The reactor coolant pressure boundary shall be designed, fabricated, and constructed so as to have an exceedingly low probability of gross rupture or significant uncontrolled leakage throughout its design lifetime.	4.1
10	Reactor Containment The containment structure shall be designed (a) to sustain, without undue risk to the health and safety of the public, the initial effects of gross equipment failures, such as a large reactor coolant pipe break, without loss of required integrity, and (b) together with other engineered safety features as may be necessary, to retain for as long as the situation requires, the functional capability of the containment to the extent necessary to avoid undue risk to the health and safety of the public.	5.1
11	Control Room The facility shall be provided with a control room from which actions to maintain safe operational status of the plant can be controlled. Adequate radiation protection shall be provided to permit continuous occupancy of the control room under any credible post-accident condition or as an alternative, access to other areas of the facility as necessary to shut down and maintain safe control of the facility without excessive radiation exposures of personnel.	7.1
12	Instrumentation and Control Systems Instrumentation and controls shall be provided as required to monitor and maintain within prescribed operating ranges essential reactor facility operating variables.	7.1
13	Fission Process Monitors and Controls Means shall be provided for monitoring or otherwise measuring and maintaining control over the fission process through-out core life under all conditions that can reasonable be anticipated to cause variations in reactivity of the core.	7.1
14	Core Protection Systems Core protection systems, together with associated equipment, shall be designed to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits.	7.1
15	Engineered Safety Features Protection Systems Protection systems shall be provided for sensing accident situations and initiating the operation of necessary engineered safety features.	7.1

Table 1.3-1 POINT BEACH GENERAL DESIGN CRITERIA

CRITERION	DESCRIPTION	FSAR LOCATION(S)
16	Monitoring Reactor Coolant Leakage Means shall be provided to detect significant uncontrolled leakage from the reactor coolant pressure boundary.	4.1, 6.5
17	Monitoring Radioactivity Releases Means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released from normal operations, from anticipated transients, and from accident conditions. An environmental monitoring program shall be maintained to confirm that radioactivity releases to the environs of the plant have not been excessive.	6.5, 11.5
18	Monitoring Fuel and Waste Storage Areas Monitoring and alarm instrumentation shall be provided for fuel and waste storage and associated handling areas for conditions that might result in loss of capability to remove decay heat and to detect excessive radiation levels.	11.5
19	Protection Systems Reliability Protection systems shall be designed for high functional reliability and inservice testability necessary to avoid undue risk to the health and safety of the public.	7.1
20	Protection Systems Redundancy and Independence Redundancy and independence designed into protection systems shall be sufficient to assure that no single failure or removal from service of any component or channel of such a system will result in loss of the protection function. The redundancy provided shall include, as a minimum, two channels of protection for each protection function to be served.	7.1
21	THIS GDC DOES NOT APPEAR IN THE FSAR	N/A
22	THIS GDC DOES NOT APPEAR IN THE FSAR	N/A
23	Protection Against Multiple Disability for Protection Systems The effects of adverse conditions to which redundant channels or protection systems might be exposed in common, either under normal conditions or those of an accident, shall not result in loss of the protection function or shall be tolerable on some other basis.	7.1
24	THIS GDC DOES NOT APPEAR IN THE FSAR	N/A
25	Demonstration of Functional Operability of Protection Systems Means shall be included for suitable testing of the active components of protection systems while the reactor is in operation to determine if failure or loss of redundancy has occurred.	7.1
26	Protection Systems Failure Analysis Design The protection systems shall be designed to fail into a safe state or into a state established as tolerable on a defined basis if conditions such as disconnection of the system, loss of energy (e.g., electrical power, instrument air), or adverse environments (e.g., extreme heat or cold, fire, steam, or water) are experienced.	7.1
27	Redundancy of Reactivity Control Two independent [reactivity] control systems, preferably of different principles, shall be provided.	3.1, 6.2, 9.3

Table 1.3-1 POINT BEACH GENERAL DESIGN CRITERIA

CRITERION	DESCRIPTION	FSAR LOCATION(S)
28	Reactivity Hot Shutdown Capability The reactivity control system provided shall be capable of making and holding the core subcritical from any hot standby or hot operating condition.	3.1, 9.3
29	Reactivity Shutdown Capability One of the reactivity control systems shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margin should assure subcriticality with the most reactive control rod fully withdrawn.	3.1, 9.3, 7.1
30	Reactivity Hold-down Capability The reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public.	3.1, 9.3
31	Reactivity Control Systems Malfunction The reactor protection system shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding fuel damage limits.	3.1, 7.1, 9.0
32	Maximum Reactivity Worth of Control Rods Limits, which include reasonable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to lose capability of cooling the core.	3.1
33	Reactor Coolant Pressure Boundary Capability The reactor coolant pressure boundary shall be capable of accommodating without rupture the static and dynamic loads imposed on any boundary component as a result of an inadvertent and sudden release of energy to the coolant. As a design reference, this sudden release shall be taken as that which would result from a sudden reactivity insertion such as rod ejection (unless prevented by positive mechanical means), rod dropout, or cold water addition.	4.1
34	Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention The reactor coolant pressure boundary shall be designed and operated to reduce to an acceptable level the probability of rapidly propagating type failures. Consideration is given (a) to the provisions for control over service temperature and irradiation effects which may require operational restrictions, (b) to the design and construction of the reactor pressure vessel in accordance with applicable codes, including those which establish requirements for absorption of energy within the elastic strain energy range and for absorption of energy by plastic deformation and (c) to the design and construction of reactor coolant pressure boundary piping and equipment in accordance with applicable codes.	4.1
35	THIS GDC DOES NOT APPEAR IN THE FSAR	N/A

Table 1.3-1 POINT BEACH GENERAL DESIGN CRITERIA

CRITERION	DESCRIPTION	FSAR LOCATION(S)
36	Reactor Coolant Pressure Boundary Surveillance Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance of critical areas by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming with current applicable codes shall be provided.	4.1
37	Engineered Safety Features Basis for Design Engineered safety features shall be provided in the facility to back up the safety provided by the core design, the reactor coolant pressure boundary, and their protection systems. Such engineered safety features shall be designed to cope with any size reactor coolant piping break up to and including the equivalent of a circumferential rupture of any pipe in that boundary, assuming unobstructed discharge from both ends.	6.1
38	Reliability and Testability of Engineered Safety Features All engineered safety features shall be designed to provide such functional reliability and ready testability as is necessary to avoid undue risk to the health and safety of the public.	6.1
39	Emergency Power An emergency power source shall be provided and designed with adequate independency, redundancy, capacity, and testability to permit the functioning of the engineered safety features and protection systems required to avoid undue risk to the health and safety of the public. This power source shall provide this capacity assuming a failure of a single active component.	8.0
40	Missile Protection Adequate protection for those engineered safety features, the failures of which could cause an undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures other than a rupture of the Reactor Coolant System piping. An original design basis for protection of equipment against the dynamic effects of a rupture of the Reactor Coolant System piping is no longer applicable.	4.1, 6.1
41	Engineered Safety Features Performance Capability Engineered safety features, such as the emergency core cooling system and the containment heat removal system, shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public.	6.1, 9.0
42	Engineered Safety Features Components Capability Engineered safety features shall be designed so that the capability of these features to perform their required function is not impaired by the effects of a loss-of-coolant accident to the extent of causing undue risk to the health and safety of the public.	6.1
43	Accident Aggravation Prevention Protection against any action of the engineered safety features which would accentuate significantly the adverse after- effects of a loss of normal cooling shall be provided.	6.1

Table 1.3-1 POINT BEACH GENERAL DESIGN CRITERIA

CRITERION	DESCRIPTION	FSAR LOCATION(S)
44	Emergency Core Cooling System Capability An emergency core cooling system with the capability for accomplishing adequate emergency core cooling shall be provided. This core cooling system and the core shall be designed to prevent fuel and clad damage that would interface with the emergency core cooling function and to limit the clad metal-water reaction to acceptable amounts for all sizes of breaks in the reactor coolant piping up to the equivalent of a double-ended rupture of the largest pipe. The performance of such emergency core cooling system shall be evaluated conservatively in each area of uncertainty.	6.2
45	Inspection of Emergency Core Cooling System Design provisions shall, where practical, be made to facilitate inspection of physical parts of the emergency core cooling system, including reactor vessel internals and water injection nozzles.	6.2
46	Testing of Emergency Core Cooling System Components Design provisions shall be made so that components of the emergency core cooling system can be tested periodically for operability and functional performance.	6.2
47	Testing of Emergency Core Cooling System Capability shall be provided to test periodically the operability of the emergency core cooling system up to a location as close to the core as is practical.	6.2
48	Testing of Operational Sequence of Emergency Core Cooling System Capability shall be provided to test initially, under conditions as close as practical to design, the full operational sequence that would bring the emergency core cooling system into action, including the transfer to alternate power sources.	6.2
49	Reactor Containment Design Basis The reactor containment structure, including openings and penetrations, and any necessary containment heat removal systems, shall be designed so that the leakage of radioactive materials from the containment structure under conditions of pressure and temperature resulting from the largest credible energy release following a loss-of-coolant accident, including the calculated energy from metal-water or other chemical reactions that could occur as a consequence of failure of any single active component in the emergency core cooling system, will not result in undue risk to the health and safety of the public.	5.1
50	NDT Requirement for Containment Material The selection and use of containment materials shall be in accordance with applicable engineering codes.	5.1
51	THIS GDC DOES NOT APPEAR IN THE FSAR	N/A
52	Containment Heat Removal Systems Where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure, this system shall perform its required function, assuming failure of any single active component.	6.3, 9.0
53	Containment Isolation Valves Penetrations that require closure for the containment function shall be protected by redundant valving and associated apparatus.	5.2

Table 1.3-1 POINT BEACH GENERAL DESIGN CRITERIA

CRITERION	DESCRIPTION	FSAR LOCATION(S)
54	Initial Containment Leakage Rate Testing Containment shall be designed so that integrated leakage rate testing can be conducted at the peak pressure calculated to result from the design basis accident after completion and installation of all penetrations and the leakage rate shall be measured over a sufficient period of time to verify its conformance with required performance.	5.7
55	Periodic Containment Leakage Rate Testing The containment shall be designed so that an integrated leakage rate can be periodically determined by test during plant lifetime.	5.7
56	Provisions for Testing of Penetrations Provisions shall be made to the extent practical for periodically testing penetrations which have resilient seals or expansion bellows to permit leak tightness to be demonstrated at the peak pressure calculated to result from occurrence of the design basis accident.	5.7
57	Provisions for Testing of Isolation Valves Capability shall be provided to the extent practical for testing functional operability of valves and associated apparatus essential to the containment function for establishing that no failure has occurred and for determining that valve leakage does not exceed acceptable limits.	5.7
58	Inspection of Containment Pressure Reducing Systems Design provisions shall be made to the extent practical to facilitate the periodic physical inspection of all important components of the containment pressure reducing systems, such as pumps, valves, spray nozzles and sumps.	6.3, 6.4
59	Testing of Containment Pressure Reducing Systems Components The containment pressure reducing systems shall be designed, to the extent practical, so that active components, such as pumps and valves, can be tested periodically for operability and required function performance.	6.3, 6.4
60	Testing of Containment Spray Systems A capability shall be provided to the extent practical to test periodically the delivery capability of the containment spray system at a position as close to the spray nozzles as is practical.	6.4
61	Testing of Operational Sequence of Containment Pressure-Reducing Systems A capability shall be provided to test initially under conditions as close as practical to the design and the full operational sequence that would bring the containment pressure reducing systems into action, including the transfer to alternate power sources.	6.3, 6.4
62	Inspection of Air Cleanup Systems Design provisions shall be made to the extent practical to facilitate physical inspection of all critical parts of containment air cleanup systems, such as ducts, filters, fans, and damper.	6.4
63	Testing of Air Cleanup Systems Components Design provisions shall be made to the extent practical so that active components of the air cleanup systems, such as fans and dampers, can be tested periodically for operability and required functional performance.	6.4

Table 1.3-1 POINT BEACH GENERAL DESIGN CRITERIA

CRITERION	DESCRIPTION	FSAR LOCATION(S)
64	Testing Air Cleanup Systems A capability shall be provided, to the extent practical, for on-site periodic testing and surveillance of the air cleanup systems to ensure (a) filter bypass paths have not developed, and (b) filter and trapping materials have not deteriorated beyond acceptable limits.	6.4
65	Testing of Operational Sequence of Air Cleanup Systems A capability shall be provided to test initially under conditions, as close to design as practical, the full operational sequence that would bring the air cleanup systems into action, including the transfer to alternate power sources and the design air flow delivery capability.	6.4
66	Prevention of Fuel Storage Criticality Criticality in the new and spent fuel storage pits shall be prevented by physical systems or processes. Such means as geometrically safe configurations shall be emphasized over procedural controls.	9.4
67	Fuel and Waste Storage Decay Heat Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities and to waste storage tanks that could result in radioactivity release which would result in undue risk to the health and safety of the public.	9.4, 9.9
68	Fuel and Waste Storage Radiation Shielding Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities.	9.4, 11.6
69	Protection Against Radioactivity Release from Spent Fuel and Waste Storage Provisions shall be made in the design of fuel and waste storage facilities such that no undue risk to the health and safety of the public could result from an accidental release of radioactivity.	9.4, 11.5
70	Control of Releases of Radioactivity to the Environment The facility design shall include those means necessary to maintain control over the plant radioactive effluents, whether gaseous, liquid, or solid. Appropriate holdup capacity shall be provided for retention of gaseous, liquid, or solid effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10 CFR 20 requirements, for both normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10 CFR 100 dosage level guidelines for potential reactor accidents of exceedingly low probability of occurrence.	11.1, 11.2, 11.3,

Table 1.3-2 SEP CATEGORY 3 AND 4 ISSUES RESOLVED BY IPEEE

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Issue	SEP Issue #	References
Settlement of Foundations & Buried Equipment	1.1	Reference 2, Reference 3, Reference 4, and Reference 5
Dam Integrity & Site Flooding	1.2	Reference 2, Reference 3, Reference 4, and Reference 5
Site Hydrologic Characteristics & Capability to Withstand Flooding	1.3	Reference 2, Reference 3, Reference 4, and Reference 5
Industrial Hazards	1.4	Reference 2, Reference 3, Reference 4, and Reference 5
Tornado Missiles	1.5	Reference 2, Reference 3, Reference 4, and Reference 5
Severe Weather Effects on Structures	2.1	Reference 2, Reference 3, and Reference 5
Design Codes, Criteria, and Load Combinations for Structures	2.2	Reference 2, Reference 3, and Reference 5
Seismic Design of Structures, Systems and Components	2.4	Reference 2, Reference 3, and Reference 5

Table 1.3-3 ADDITIONAL GENERIC SAFETY ISSUES RESOLVED BY IPEEE

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Issue	GSI Designation	References
Shutdown Decay Heat Removal Requirements	USI A-45	Reference 4 and Reference 5
Potential Seismic Interaction Involving the Movable In-Core Flux Mapping System Used in Westinghouse Plants	GSI-131	Reference 4 and Reference 5
Design for Probable Maximum Precipitation	GSI-103	Reference 4 and Reference 5
Fire Risk Scoping Study Issues	[no designation]	Reference 4 and Reference 5
Effects of Fire Protection System Actuation on Safety-Related Equipment	GSI-57	Reference 4 and Reference 5
Fire-Induced Alternate Shutdown/Control Room Panel Interactions	GSI-147	Reference 4 and Reference 5
Smoke Control and Manual Fire-Fighting Effectiveness	GSI-148	Reference 4 and Reference 5
Systematic Evaluation Program (SEP)	GSI-156	Reference 4 and Reference 5
Multiple System Responses Program (MSRP)	GSI-172	Reference 4 and Reference 5

1.4 QUALITY ASSURANCE PROGRAM

NextEra Energy Point Beach, LLC nuclear plant operational and support activities are conducted under NextEra Energy Quality Assurance Topical Report (QATR), FPL-1. FPL-1 is the top-level policy document that establishes the manner in which quality is to be achieved and presents NextEra Energy's overall philosophy regarding achievement and assurance of quality. The QATR responds to and satisfies the requirements of Appendix B of 10 CFR Part 50 (Reference 1).

In addition to the commitments identified in the QATR, Point Beach also has the following commitment to Regulatory Guide (RG) 1.54 dated June 1973:

PBNP is committed to follow the position of RG 1.54 (1973), Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants, which endorses and supplements ANSI N101.4-1972, Quality Assurance for Protective Coatings Applied to Nuclear Facilities, for activities that affect quality and occur during the operational phase, and that are comparable in nature and extent to related activities occurring during construction.

Procedures and programmatic controls ensure that the applicable requirements for the procurement, application, inspection, and maintenance of Service Level I coatings in containment are implemented. The surface preparation, application and surveillance during installation of Service Level I coatings used for new applications or repair/replacement activities inside containment meet the applicable portions of RG 1.54 and ANSI N101.4-1972.

Point Beach was built and licensed prior to RG 1.54 being issued, and, as such, does not conform fully to all aspects of ANSI N101.4-1972 and RG 1.54. The original coatings inside containment were applied without the documentation and/or testing necessary to be considered Service Level I coatings. These original coatings are considered acceptable based on WCAP-7198-L and the evaluation in Section 5.6.2.4.

Relatively small amount of coatings applied by vendors on supplied equipment, miscellaneous structural supports, and small areas of touch-up on qualified Service Level I coatings may not be Service Level I coatings. With the exception of isolated minor touch-up repairs (i.e., less than 1 ft^2), all coating repairs, maintenance, and applications inside containment are required to be performed with Service Level I coatings.

For details of the Quality Assurance requirements for the Aging Management Programs implemented in accordance with 10 CFR 54, see Chapter 15. Records necessary to document compliance with the provisions of 10 CFR 54 will be retained for the term of the renewed operating license (Reference 2).

1.4.1 REFERENCES

1. FPL letter to the NRC; “Quality Assurance Topical Report (QATR FPL-1), Revisions 1 and 2,” dated June 27, 2008.

2. NUREG-1839, Safety Evaluation Report Related to the License Renewal of the Point Beach Nuclear Plant, Units 1 and 2, Docket Nos. 50-266 and 50-301, dated December 2005.

1.5 FACILITY SAFETY CONCLUSIONS

The safety of the public and plant operating personnel and reliability of plant equipment and systems have been the primary considerations in the plant design. The approach taken in fulfilling the safety consideration is three-fold. First, careful attention has been given to the design to prevent the release of radioactivity to the environment under conditions which could be hazardous to the health and safety of the public. Second, the plant has been designed so as to provide adequate protection for plant personnel wherever a potential radiation hazard exists. Third, reactor systems and controls have been designed with a great degree of redundancy and fail-safe characteristics.

Based on the over-all design of the plant including its safety features, the analyses of the possible incidents and of hypothetical accidents, and the operational history of the Point Beach Nuclear Plant, it is concluded that Point Beach Nuclear Plant Units 1 and 2 can be operated without undue hazard to the health and safety of the public.

On April 16, 1970, by letter to the Chairman of the U. S. Atomic Energy Commission, the Advisory Committee on Reactor Safeguards (ACRS) reported its completed review of the operating license application for Point Beach Nuclear Plant Units 1 and 2 (Reference 1). The ACRS concluded that subject to satisfactory completion of construction and pre-operational testing, and given due regard for those items mentioned in the letter, Point Beach Nuclear Plant Units 1 and 2 can be operated at power levels up to 1518.5 MWt for each unit without undue risk to the health and safety of the public. Similarly, the U.S. Atomic Energy Commission, in its Safety Evaluation Report for the Point Beach Nuclear Plant Units 1 and 2 dated July 15, 1970 (Reference 2), concluded that, “There is reasonable assurance (i) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission set forth in 10 CFR Chapter 1.” On 11/29/2002 a License Amendment Request, increasing Thermal Power to 1540 MWt, was approved by the NRC (Reference 3). The basis of the change was the implementation of a 10 CFR 50, Appendix K uprate based on a reduction in power measurement uncertainty.

In December, 2005, the NRC issued NUREG-1839, “Safety Evaluation Report Related to the License Renewal of the Point Beach Nuclear Plant, Units 1 and 2.” Based on the license renewal application, the NRC staff concluded that the requirements of 10 CFR 54.29(a) had been met and that all open items and confirmatory items have been resolved. The renewed licenses are applicable for 20 years beyond the expiration date of midnight, October 5, 2010 for Unit 1 and midnight, March 8, 2013 for Unit 2.

On May 3, 2011, the NRC approved a License Amendment Request increasing core thermal power to 1800 MWt (Reference 4). This power increase and associated changes to the operating license, Technical Specifications, and licensing basis is defined as an Extended Power Uprate (EPU).

1.5.1 REFERENCES

1. Advisory Committee on Reactor Safeguards letter to the U. S. Atomic Energy Commission, dated April 16, 1970.

2. U. S. Atomic Energy Commission letter to Wisconsin Electric Power Company, dated July 15, 1970

3. NRC letter to NMC, “Point Beach Nuclear Plant, Units 1 and 2 - Issuance of Amendments Re: Measurement of Uncertainty Recapture Power Uprate (TAC NOS. MB4956 and MB4957),” dated November 29, 2002.

4. NRC Safety Evaluation 2011-0004, “Issuance of License Amendments Regarding Extended Power Uprate,” dated May 3, 2011