Nuclear Power Plant Design and Seismic Safety Considerations

Anthony Andrews
Specialist in Energy and Defense Policy

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Summary

Since the March 11, 2011, earthquake and tsunami that devastated Japan's Fukushima Daiichi nuclear power station, the seismic criteria applied to siting commercial nuclear power plants operating in the United States have received increased attention; particularly the Nuclear Regulatory Commission's (NRC's) 2010 reassessment of seismic risks at certain plant sites.

Commercial nuclear power plants operating in the United States vary considerably, as most were custom-designed and custom-built. Boiling water reactors (BWRs) directly generate steam inside the reactor vessel. Pressurized water reactors (PWRs) use heat exchangers to convert the heat generated by the reactor core into steam outside of the reactor vessel. U.S. utilities currently operate 104 nuclear power reactors at 65 sites in 31 states; 69 are PWR designs and the 35 remaining are BWR designs.

One of the most severe operating conditions for a reactor is a loss of coolant accident (LOCA), which can lead to a reactor core meltdown. The emergency core cooling system (ECCS) provides core cooling to minimize fuel damage by injecting large amounts of cool, borated water into the reactor coolant system following a pipe rupture or other water loss, and (secondarily) to provide extra neutron poisons to ensure the reactor remains shut down. The ECCS must be sized to provide adequate make-up water to compensate for a break of the largest diameter pipe in the primary system (i.e., the so-called "double-ended guillotine break" (DEGB)). However, the NRC considers the DEGB to be an extremely unlikely event. Nevertheless, even unlikely events can occur, as the combined tsunami and magnitude 9.0 earthquake that struck Fukushima Daiichi proves.

U.S. nuclear power plants have designs based on Deterministic Seismic Hazard Analysis (DSHA). Since then, Probabilistic Seismic Hazard Analysis (PSHA) has been adopted as a more comprehensive approach in engineering practice. Consequently, the NRC is reassessing the probability of seismic core damage at existing plants.

In 2008, the U.S Geological Survey (USGS) released an update of the National Seismic Hazard Maps (NSHM). USGS notes that the 2008 hazard maps differ significantly from the 2002 maps in many parts of the United States, and generally show 10%-15% reductions in spectral and peak ground acceleration across much of the Central and Eastern United States (CEUS), and about 10% reductions for spectral and peak horizontal ground acceleration in the Western United States (WUS). Seismic hazards are greatest in the WUS, particularly in California, Oregon, and Washington, as well as Alaska and Hawaii.

In 2010, NRC published its GI-199 Safety/Risk Assessment; a two-stage assessment of the implications of USGS updated probabilistic seismic hazards analysis in the CEUS on existing nuclear power plants sites. NRC does not rank nuclear plants by seismic risk. NRC's objective in GI-199 was to evaluate the need for further investigations of seismic safety for operating reactors in the CEUS. The data evaluated in the assessment suggest that the probability for earthquake ground motion above the seismic design basis for some nuclear plants in the CEUS, although still low, is larger than previous estimates. In late March 2011, NRC announced that it had identified 27 nuclear reactors operating in the CEUS that would receive priority earthquake safety reviews.

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Background on Seismic Standards

The seismic design criteria applied to siting commercial nuclear power plants operating in the United States received increased attention following the March 11 earthquake and tsunami that devastated Japan's Fukushima Daiichi nuclear power station. Since the events, some in Congress have begun to question whether U.S plants are vulnerable to a similar threat, particularly in light of the Nuclear Regulatory Commission's (NRC's) ongoing reassessment of seismic risks at certain plant sites.[1]

Commercial nuclear power plants operating in the United States use light water reactor designs, but vary widely in design and construction. Light water reactors use ordinary water as a neutron moderator and coolant, and uranium fuel enriched in fissile uranium-235.[2] Designs fall into either pressurized water reactor (PWR) or boiling water reactor (BWR) categories. Both have reactor cores (the source of heat) consisting of arrays of uranium fuel bundles capable of sustaining a controlled nuclear reaction.[3] U.S. commercial nuclear power plants incorporate safety features intended to ensure that, in the event of an earthquake, the reactor core would remain cooled, the reactor containment would remain intact, and radioactive releases would not occur from spent fuel storage pools. The Nuclear Regulatory Commission (NRC) defines this as the "safe-shutdown condition."

When utilities began building nuclear power plants in the 1960s-1970s era, they typically hired an architect/engineering firm, then contracted with a reactor manufacturer ("nuclear vendors") to build the nuclear steam supply system (NSSS), consisting of the nuclear core, reactor vessel, steam generators and pressurizer (in PWRs), and control mechanisms—representing about 10% of the plant investment.[4] The balance of the plant (BOP) consisted of secondary cooling systems, feed-water systems, steam systems, control room, and generator systems. At the time, the four vendors who offered designs for nuclear reactor systems were Babcock & Wilcox, Combustion Engineering, General Electric, and Westinghouse. About 12 architect/engineering firms were available to design the balance of the plant. Each architect/engineer had its own preferred approach to designing the balance of plant systems. In addition, plant site-conditions varied due to the different meteorological, seismic, and hydrological conditions. The custom design-and-build industry approach resulted in problems verifying the safety of individual plants and in transferring the safety lessons learned from one reactor to another.

The previous design approach to withstanding earthquakes had relied on Deterministic Seismic Hazard Analysis (DSHA). Any new plant design is to consider Probabilistic Seismic Hazard Analysis (PSHA), which has been widely adopted in engineering practice. Deterministic analysis attempts to quantify the worst-case scenario based on the combination of earthquake sources at a site's location that results in the strongest ground-motion potentially generated.[5] In other words,

[1] This report does not discuss the risk from earthquake-caused tsunamis, as associated with the catastrophic damage to the Fukushima plants.

[2] Heavy water reactors, such as Canada's CANDU reactor, use water enriched with a heavier hydrogen isotope and natural uranium for fuel, which contains less than 3.5% uranium-235.

[3] For further background uranium fuel, see CRS Report RL34234, *Managing the Nuclear Fuel Cycle: Policy Implications of Expanding Global Access to Nuclear Power*, coordinated by Mary Beth Nikitin.

[4] Office of Technology Assessment, *Nuclear Power Plant Standardization: Light Water Reactors*, NTIS order #PB81-213589, April 1981, p. 11.

[5] Julian J. Bommer, Norman A. Abrahamson, and Fleur O. Strasser, et al., "The Challenge of Defining Upper Bounds (continued...)

the deterministic assessment focuses on a single earthquake event to determine the finite probability of occurrence. PSHA is a methodology that estimates the likelihood that various levels of earthquake-caused ground motion will be exceeded at a given location in a given future time period.[6] Due to possible uncertainties in geoscience data and in the models used to estimate ground motion from earthquakes, multiple model interpretations are often possible. This has led to disagreement among experts, which in turn has led to disagreement on the selection of ground motion magnitudes for the design at a given site. PSHA traditionally quantified ground motion based on peak ground acceleration (PGA).[7] Today, the preferred parameter is Response Spectral Acceleration (SA), which gives the maximum acceleration of an oscillating structure such as a building or power plant.

In its 2010 study (GI-199), the NRC concludes that deterministic assessments (DSHA) do not necessarily mean that the seismic design basis for the Safe Shutdown Earthquake (SSE) condition was, or is, deficient in some fashion.[8] The design approach to developing loadings on power plant piping and equipment systems relies on the SSE condition. Existing nuclear plants designs include considerable safety margins that enable them to withstand "deterministic" or "scenario earthquake" ground motions that accounted for the largest earthquakes expected in the area around the plant.[9] The NRC study found that some plant sites might have an increased probability, albeit relatively small, of exceeding their design basis ground motion. NRC considers that the probabilities of seismic core damage occurring are lower than its guidelines for taking immediate action, but has determined that some plants' performance should be reassessed based on updated seismic hazards.

This report presents some of the general design concepts of operating nuclear power plants in order to discuss design considerations for seismic events. This report does not attempt to conclude whether one design is inherently safer or less safe than another plant. Nor does it attempt to conclude whether operating nuclear power plants are at any greater or lesser risk from earthquakes given recent updates to seismic data and seismic hazard maps.

Nuclear Power Plant Designs

Currently, 104 nuclear power plants currently operate at 65 sites in 31 states; 69 are PWR designs and the 35 remaining are BWR designs.

(...continued)

on Earthquake Ground Motions," *Seismological Research Letters*, vol. 75, no. 1 (February 2004).

[6] R. J. Budnitz, G. Apostolakis, and D. M. Boore, *Recommendations for Probabilistic Seismic Hazard Analysis: Guidance on Uncertainty and Use of Experts: Main Report*, U.S. Nuclear Regulatory Commission, Nureg/CR-6372, Lawrence Berkeley National Laboratory, CA, April 1997, http://www.nrc.gov/reading-rm/doc-collections/nuregs/contract/cr6372/vol1/index.html#pub-info.

[7] Edward (Ned) H. Field, *Probabilistic Seismic Hazard Analysis (PSHA) - A Primer*, http://www.relm.org/tutorial_materials.

[8] U.S. Nuclear Regulatory Commission, *Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States Existing Plants - Safety/Risk Assessment*, Generic Issue 199 (GI-199), August 2010.

[9] U.S. NRC, NRC frequently asked questions related to the March 11, 2011 Japanese Earthquake and Tsunami, March 2011, http://www.nrc.gov.

The more numerous PWR plants include Babcock & Wilcox, Combustion Engineering, and Westinghouse designs. The BWR plants all use a General Electric design. **Table 1** summarizes the various reactor types. The sections that follow discuss them further.

Table 1. Reactor Type, Vendor, and Containment

Reactor Type	Vendor	Containment Type	No. of Plants
PWR	Babcock & Wilcox 2-Loop Lower	Dry, Ambient Pressure	7
	Combustion Engineering	Dry, Ambient Pressure	11
	Combustion Engineering System 80	Large Dry, Ambient Pressure	3
	Westinghouse 2-Loop	Dry, Ambient Pressure	6
	Westinghouse 3-Loop	Dry, Ambient Pressure	7
	Westinghouse 3-Loop	Dry, Sub-atmospheric	6
	Westinghouse 4-Loop	Dry, Ambient Pressure	18
	Westinghouse 4-Loop	Dry, Sub-atmospheric	1
	Westinghouse 4-Loop	Wet, Ice Condenser	9
	Westinghouse 4-Loop	Dry, Ambient Pressure	1
			69
BWR	General Electric Type 2	Wet, Mark I	2
	General Electric Type 3	Wet, Mark I	6
	General Electric Type 4	Wet, Mark I	15
	General Electric Type 4	Wet, Mark II	4
	General Electric Type 5	Wet, Mark II	4
	General Electric Type 6	Wet, Mark III	4
			35

Source: U.S. NRC.

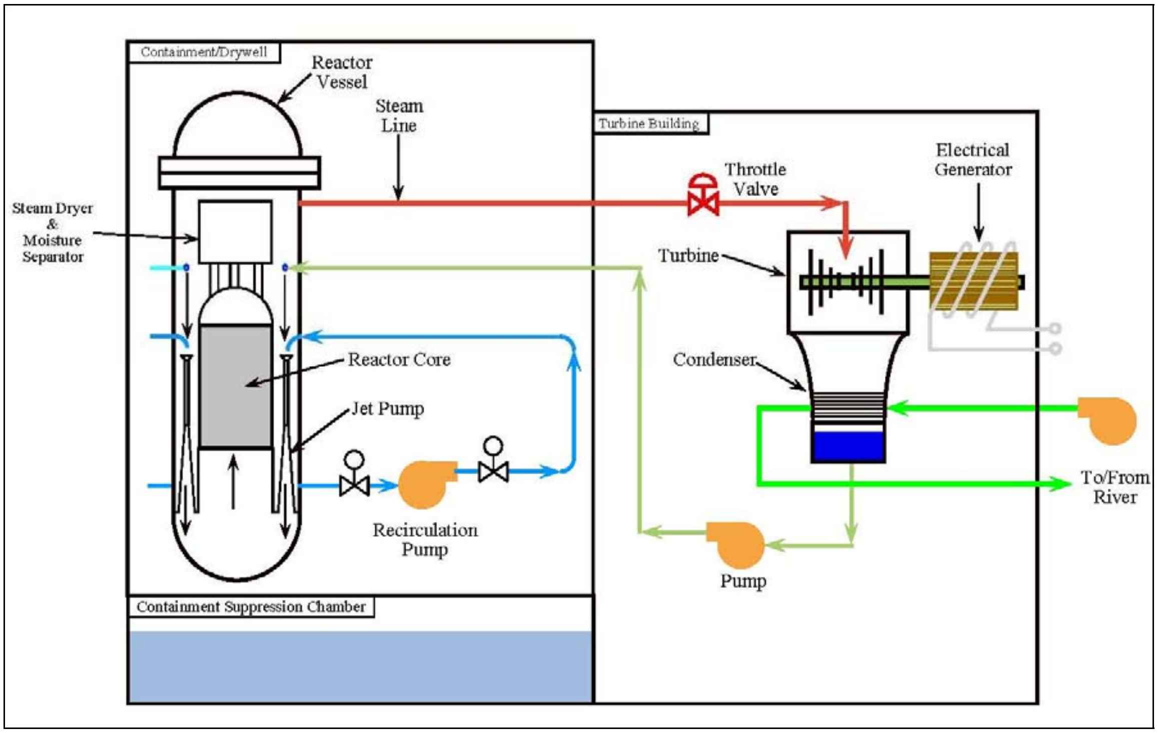
Boiling Water Reactor (BWR) Systems

A boiling water reactor generates steam directly inside the reactor vessel as water flows upward through the reactor's core (see **Figure 1**).[10] The water also cools the reactor core, and the reactor operator is able to vary the reactor's power by controlling the rate of water flow through the core with recirculation pumps and jet pumps. The generated steam flows out the top of the reactor vessel through pipelines to a combined high-pressure/low-pressure turbine-generator. After the exhausted steam leaves the low-pressure turbine, it runs through a condenser/heat exchanger that cools the steam and condenses it back to water. A series of pumps return the condensed water back to the reactor vessel. The heat exchanger cycles cooling water through a cooling tower, or takes in and discharges water with a lake, river, or ocean. The water that flows through the reactor, steam turbines, and condenser is a closed loop that never contacts the outside environment under normal operating conditions. Reactors of this design operate at temperatures of approximately 570º F and pressures of 1,000 pounds per square inch (psi) atmospheric.

[10] U.S. Nuclear Regulatory Commission, *Reactor Concepts Manual*, *Boiling Water Reactor Systems*, http://www.nrc.gov/reading-rm/basic-ref/teachers/03.pdf - 2005-10-17.

Figure 1. Boiling Water Reactor (BWR) Plant

Generic Design Features



Source: U.S. Nuclear Regulatory Commission, *Reactor Concepts Manual, Boiling Water Reactor Systems, 2005.*

Safe-Shutdown Condition

During normal operation, reactor cooling relies on the water that enters the reactor vessel and the generated steam that leaves. During safe shutdown, the core continues to generate heat by radioactive decay and generates steam.[11] Under this condition, the steam bypasses the turbine and diverts directly to the condenser to cool the reactor. When the reactor vessel pressure decreases to approximately 50 psi, the shutdown-cooling mode removes residual heat by pumping water from the reactor recirculation loop through a heat exchanger and back to the reactor via the recirculation loop. The recirculation loop design limits the number of pipes that penetrate the reactor vessel.

Loss of Coolant Accident

The most severe operating condition that a reactor design must contend with is a loss of coolant accident (LOCA). In the absence of coolant, the uncovered reactor core continues to generate heat through fission. The resulting heat buildup can damage the fuel or fuel cladding and lead to a fuel "meltdown." Under such a condition, an emergency core cooling system (ECCS) provides water

[11] During the sustained chain reaction in an operating reactor, the U-235 splits into highly radioactive fission products, while the U-238 is partially converted to plutonium-239 by neutron capture, some of which also fissions. Further neutron capture creates other radioactive elements. The process of radioactive decay transforms an atom to a more stable element through the release of radiation—alpha particles (two protons and two neutrons), charged beta particles (positive or negative electrons), or gamma rays (electromagnetic radiation).

to cool the reactor core. The ECCS is an independent high-pressure coolant injection system that requires no auxiliary electrical power, plant air systems, or external cooling water systems to provide makeup water under small and intermediate loss of coolant accidents. A low-pressure ECCS sprays water from the suppression pool into the reactor vessel and on top of the fuel assemblies.[12] The ECCS must also be sized to provide adequate makeup water to compensate for a break of the largest diameter pipe in the primary system (i.e., the so-called "double-ended guillotine break" (DEGB)). However, the NRC views the DEGB as an extremely unlikely event (likely to occur only once per 100,000 years of reactor operation).[13]

BWR Design Evolution

Currently, General Electric Type 2 through Type 6 BWRs operate in the United States (**Table 1**). BWRs are inherently simpler designs than other light water reactor types. Since they heat water and generate steam directly inside the reactor vessel, there are fewer components.

Table 2. BWR Design Evolution

Model	Year Introduced	Design Feature	Typical Plants
BWR/1	1955	Natural circulation	Dresden 1
		First internal steam separation	Big Rock Point
		Isolation condenser	Humboldt Bay
		Pressure Suppression Containment	
BWR/2	1963	Large direct cycle	Oyster Creek
BWR/3/4	1965/1966	First jet pump application	Dresden 2
		Improved Emergency Core Cooling System (ECCS); spray and flood	Browns Ferry
		Reactor Core Isolation Cooling, (RCIC) system	
BWR/5	1969	Improved ECCS systems	LaSalle
		Valve recirculation flow control	9 Mile Point 2
BWR/6	1972	Improved jet pumps and steam separators	Clinton
		Reduced fuel duty: 13.4 kW/ft, 44 kW/m	Grand Gulf
		Improved ECCS performance	Perry
		Gravity containment flooder	
		Solid-state nuclear system protection system (Option, Clinton only)	
		Compact control room option	

Source: M. Ragheb, Chapter 3, *Boiling Water Reactors*, https://netfiles.uiuc.edu/mragheb/www/NPRE%20402%20ME%20405%20Nuclear%20Power%20Engineering/Boiling%20Water%20Reactors.pdf.

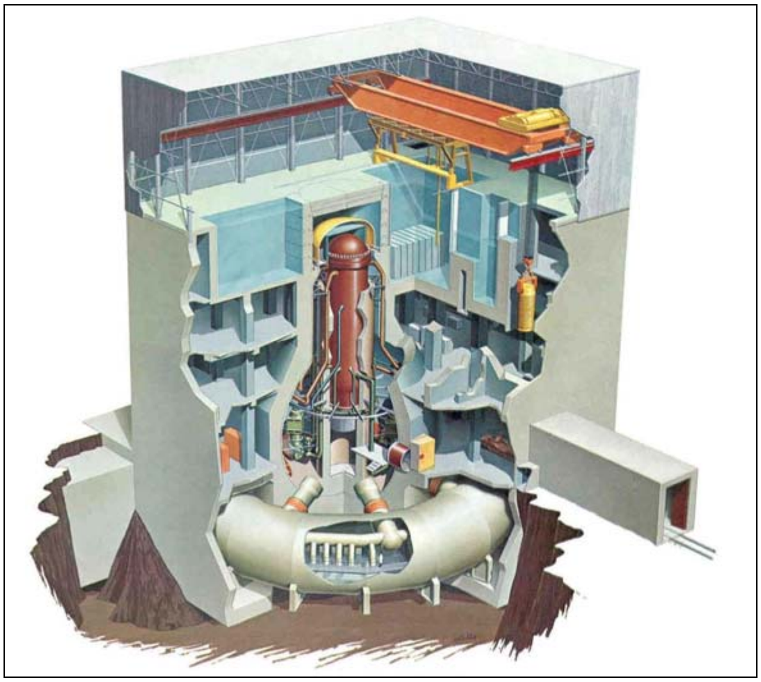
Note: All BWR/1 plants that operated in the United States have been decommissioned.

[12] The NRC regulates the design, construction, and operation requirements of the ECCS under 10 C.F.R. 50.46, "Acceptance criteria for emergency core cooling systems for light-water nuclear reactors"; Appendix K to 10 C.F.R. Part 50, "ECCS Evaluation Models"; and Appendix A to 10 C.F.R. Part 50, "General Design Criteria [GDC] for Nuclear Power Plants" (e.g., GDC 35, "Emergency Core Cooling").

[13] N.C. Chokshi, S.K. Shaukat, and A.L. Hiser, et al., *Seismic Considerations for the Transition Break Size,* U.S. Nuclear Regulatory Commission, NUREG 1903, Brookhaven National Laboratory, February 2008.

Figure 2. GE BWR / Mark I Containment Structure

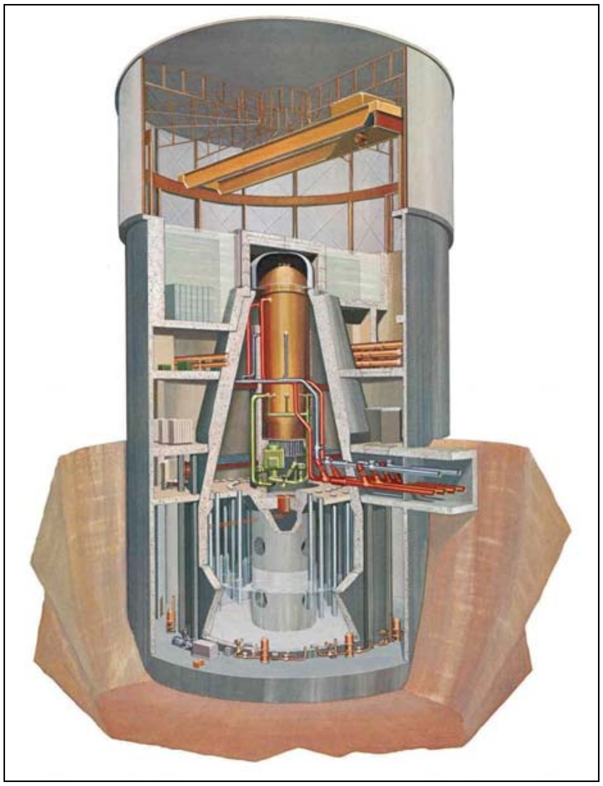
Showing Torus Suppression Pool



Source: General Electric, in *NRC Boiling Water Reactor (BWR) Systems*, http://www.nrc.gov/reading-rm/basic-ref/teachers/03.pdf.

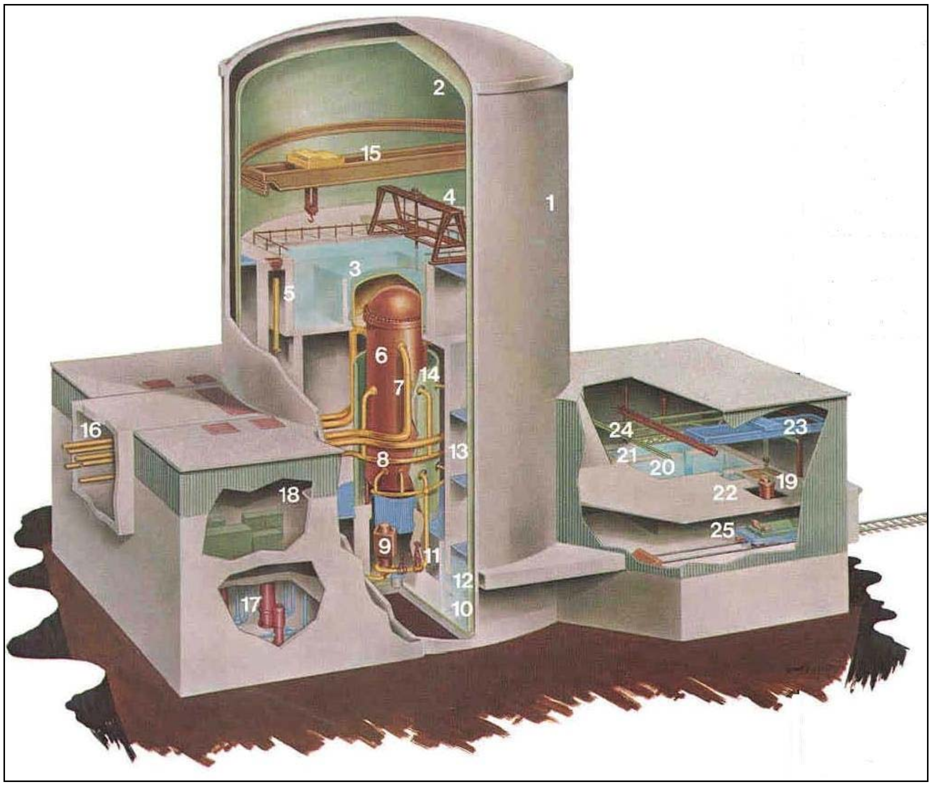
Note: Japan's Fukushima Daiichi plants use designs similar to this.

Figure 3. General Electric Mark II Containment Structure



Source: General Electric, in NRC *Boiling Water Reactor (BWR) Systems*, http://www.nrc.gov/reading-rm/basic-ref/teachers/03.pdf.

Figure 4. General Electric Mark III Containment Structure



Source: General Electric, in *NRC Boiling Water Reactor (BWR) Systems*, http://www.nrc.gov/reading-rm/basic-ref/teachers/03.pdf.

Notes:

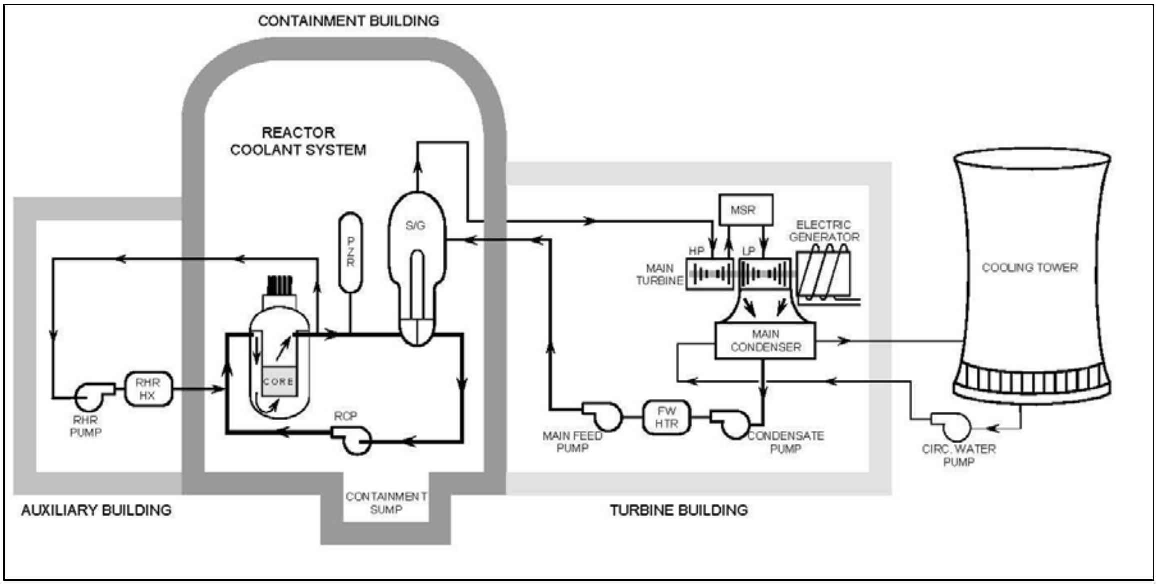
Reactor Building	Auxiliary Building	Fuel Building
1. Shield Building	16. Steam Line Channel	19. Spent Fuel Shipping cask
2. Free Standing Steel Containment	17. RHR System	20. Fuel Storage Pool
3. Upper Pool	18. Electrical Equipment Room	21. Fuel Transfer Pool
4. Refueling Platform		22. Cask Loading Pool
5. Reactor Water Cleanup		23. Cask Handling Crane
6. Reactor Vessel		24. Fuel Transfer Bridge
7. Steam Line		25. Fuel Cask Skid on Railroad Car
8. Feed-water Line		
9. Recirculation Loop		
10. Suppression Pool		
11. Weir Wall		
12. Horizontal Vent		
13. Dry Well		
14. Shield Wall		
15. Polar Crane		

Pressurized Water Reactor Systems

A pressurized water reactor (PWR) generates steam outside the reactor vessel, unlike a BWR design. A primary system (reactor cooling system) cycles superheated water from the core to a heat exchanger/steam generator. A secondary system then transfers steam to a combined high-pressure/ low-pressure turbine generator (**Figure 5**).[14] Steam exhausted from the low-pressure turbine runs through a condenser that cools and condenses it back to water. Pumps return the cooled water back to the steam generator for reuse. The condenser cools the steam leaving the turbine-generator through a third system by flowing past a heat-exchanger that recycles cooling water through a cooling tower, or takes in and discharges water with a lake, river, or ocean. Unlike a BWR design, the cooling water that flows through the reactor core never contacts the turbine-generator. Nor does reactor cooling water contact the environment under normal operating conditions.

Figure 5. Pressurized Water Reactor (PWR) Plant

Generic Design Features



Source: U.S. Nuclear Regulatory Commission, *Reactor Concepts Manual*, *Boiling Water Reactor Systems, 2005*.

Notes: PIZ – Pressurizer; S/G – Steam generator

To keep the reactor operating under ideal conditions, a pressurizer keeps water and steam pressure under equilibrium conditions. The pressurizer is part of the reactor coolant system, and consists of electrical heaters, pressure sprays, power-operated relief valves, and safety valves. For example, if pressure rises too high, water spray cools the steam in the pressurizer; or if pressure is too low, the heaters increase steam pressure. The cause of the pressure deviation is normally associated with a change in the temperature of the reactor coolant system.

PWR Design Configurations

All PWR systems consist of the same major components, but arranged and designed differently. For example, Westinghouse has built plants with two, three, or four primary coolant loops, depending upon the power output of the plant.

[14] U.S. NRC, *Reactor Concepts Manual, Pressurized Water Reactor Systems,* http://www.nrc.gov/reading-rm/basic-ref/teachers/04.pdf - 2005-10-17.

- Two-loop Westinghouse reactors have two steam generators, two reactor coolant pumps, a pressurizer, and 121 fuel assemblies; electrical output is approximately 500 megawatts. Six currently operate.[15]
- Three-loop Westinghouse reactors have three steam generators, three reactor coolant pumps, a pressurizer, and 157 fuel assemblies; output ranges from 700 to more than 900 megawatts. Thirteen currently operate.[16]
- Four-loop Westinghouse reactors have four steam generators, four reactor coolant pumps, a pressurizer, and 193 fuel assemblies; output ranges from 950 to 1,250 megawatts.[17] Twenty-nine currently operate.

The seven operating Babcock & Wilcox reactors have two once-through steam generators, four reactor coolant pumps, and a pressurizer.[18] These reactors have 177 fuel assemblies and produce approximately 850 megawatts of electricity.

The 14 operating Combustion Engineering reactors have two steam generators, four reactor coolant pumps, and a pressurizer.[19] They produce from less than 500 to more than 1,200 megawatts.

Safe Shutdown Condition

During normal operation, a PWR does not generate steam directly. For cooling, it transfers heat via the reactor primary coolant to a secondary coolant in the steam generators. There, the secondary coolant water is boiled into steam and sent to the main turbine to generate electricity. Even after shutdown (when the moderated uranium fission is halted), the reactor continues to produce a significant amount of heat from decay of uranium fission products (decay heat). The decay heat is sufficient to cause fuel damage if the core cooling is inadequate. Auxiliary feed-water systems and the steam dump systems work together to remove the decay heat from the reactor. If a system for dumping built-up steam is not available or inoperative, atmospheric relief valves can dump the steam directly to the atmosphere. Under normal operating conditions, water flowing through the secondary system does not contact the reactor core; dumped-steam does not present a radiological release.

Loss of Coolant Accident

The most severe operating condition that reactor designs must contend with is the loss of coolant accident (LOCA); the extreme case represented by the double-ended guillotine break (DEGB) of

[15] The two-loop units in the United States are Ginna, Kewaunee, Point Beach 1 and 2, and Prairie Island 1 and 2.

[16] The three-loop units in the United States are Beaver Valley 1 and 2, Farley 1 and 2, H. B. Robinson 2, North Anna 1 and 2, Shearon Harris 1, V. C. Summer, Surry 1 and 2, and Turkey Point 3 and 4.

[17] The four-loop units in the United States are Braidwood 1 and 2, Byron 1 and 2, Callaway, Catawba 1 and 2, Comanche Peak 1 and 2, D. C. Cook 1 and 2, Diablo Canyon 1 and 2, Indian Point 2 and 3, McGuire 1 and 2, Millstone 3, Salem 1 and 2, Seabrook, Sequoyah 1 and 2, South Texas Project 1 and 2, Vogtle 1 and 2, Watts Bar 1, and Wolf Creek.

[18] The Babcock & Wilcox units in the United States are Arkansas 1, Crystal River 3, Davis Besse, Oconee 1, 2, and 3, and Three Mile Island 1.

[19] The Combustion Engineering units in the United States are Arkansas 2, Calvert Cliffs 1 and 2, Fort Calhoun, Millstone 2, Palisades, Palo Verde 1, 2, and 3, San Onofre 2 and 3, Saint Lucie 1 and 2, and Waterford 3.

large diameter pipe systems. In the event of a LOCA, the reactor's emergency core cooling system (ECCS) provides core cooling to minimize fuel damage by injecting large amounts of cool, borated water into the reactor coolant system from a storage tank. The borated water stops the fission process by absorbing neutrons, and thus aids in shutting down the reactor.

The ECCS on the PWR consist of four separate systems: the high-pressure injection (or charging) system, the intermediate pressure injection system, the cold leg accumulators, and the low-pressure injection system (residual heat removal). The high pressure injection system provides water to the core during emergencies in which reactor coolant system pressure remains relatively high (such as small breaks in the reactor coolant system, steam break accidents, and leaks of reactor coolant through a steam generator tube to the secondary side). The intermediate pressure injection system is designed to accommodate emergency conditions under which the primary pressure stays relatively high; for example, small to intermediate size primary breaks. The cold leg accumulators operate without electrical power by using a pressurized nitrogen gas bubble on the top of tanks that contain large amounts of borated water. The low-pressure injection system removes residual heat by injecting water from the refueling water storage tank into the reactor coolant system during large breaks (which would cause very low reactor coolant-system pressure).

Containment Structure Designs

All U.S. reactors are surrounded by a primary containment structure that is designed to minimize releases of radioactive material into the environment. The PWR primary containment structure must surround all the components of the primary cooling system, including the reactor vessel, steam generators, and pressurizer. BWR primary containments typically are smaller, because there are no steam generators or pressurizers.

Containments must be strong enough to withstand the pressure created by large amounts of steam that may be released from the reactor cooling system during an accident. The largest containments are designed to provide sufficient space for steam released by an accident to expand and cool to keep pressure within the design parameters of the structure. Smaller containments, such as those for BWRs, require pressure suppression systems to condense much of the released steam into water. Smaller PWR containments also may include pressure suppression systems, such as ice condensers.[20]

To further limit the leakage from the containment structure following an accident, a steel liner that covers the inside surface of the containment building acts as a vapor-proof membrane to prevent any gas from escaping through any cracks that may develop in the concrete of the containment structure. Two systems act to reduce temperature and pressure within the containment structure: a fan cooler system that circulates air through heat exchangers, and a containment spray system.

All U.S. PWR designs include a containment system with Multiple Engineered Safety Features (ESFs).[21] A dry containment system consists of a steel shell surrounded by a concrete biological

[20] Kazys Almenas and R. Lee, *Nuclear Engineering: An Introduction* (Berlin: Springer-Verlag, 1992), pp. 507-514.

[21] M. Ragheb, *Containment Structures* (2011). University of Illinois Champaign-Urbana, https://netfiles.uiuc.edu/mragheb/www/NPRE%20457%20CSE%20462%20Safety%20Analysis%20of%20Nuclear%20Reactor%20Systems/Containment%20Structures.pdf.

shield that protects the reactor against outside elements, for example, debris driven by hurricane winds or an aircraft strike.[22] The outer shield is not designed as a barrier against the release of radiation. Although the concrete structures in existing plants act as insulators against uncontrolled releases of radioactivity to the environment, they will fail if the ESFs fail in their function. Some containment building design features are summarized in **Table 3**.

Table 3. Containment Building Design Parameters

Containment Type, plant	Parameter	Technical Specification
SP-1, Zion	Containment capability pressure	149 psia[a]
	Upper bound spike pressure	107 psia
	Early failure physically unreasonable best estimate pressure rise, including heat sinks	10 psi/hour
	Time to failure, best estimate with unlimited water in cavity	16 hours
SP-2, Surry	Containment capability pressure	134 psia
	Upper bound spike pressure	107 psia
	Time to failure, early failure physically unreasonable best estimate with dry cavity	Several days
SP-3, Sequoyah	Containment capability pressure	65 psia, 330 °F
	Upper bound loading pressure	70-100 psia
	Lower bound loading pressure	50-70 psia
	Thermal loads	500-700 °F
	Early failure	Quite likely
SP-4, Browns Ferry	Containment capability pressure	132 psia, 330 °F
	Upper bound loading pressure	132 psia in 40 minutes
	Lower bound loading pressure	132 psia in 2 hours
	Thermal loads	500-700 °F
	Early failure	Quite likely
SP-6, Grand Gulf	Containment capability pressure	75 psia
	Upper bound loading pressure	30 psia
	Wall heat flux	1,000 to 10,000 Btu/hr-square foot
	Penetration seal temperature	345 °F
	Pressurization failure from diffusion flames	Unreasonable
	Seal failure	Unlikely
SP-15, Limerick	Containment capability pressure	155 psia, 330 °F
	Upper bound loading pressure	145 psia in 2-3 hours
	Lower bound loading pressure	100 psia in 3 hours
	Thermal loads	500-700 °F
	Early failure	Rather unlikely

Source: U.S. NRC, *General Studies of Nuclear Reactors; BWR Type Reactors; Containment; Reactor Accidents; Leaks; PWR Type Reactors; Accidents; Reactors; Water Cooled Reactors; Water Moderated Reactors, NUREG-1037, 1985,* as cited by M. Ragheb UICU.

[22] NRC regulations require that new reactors be designed to withstand the impact of large commercial aircraft and that existing plants develop strategies to mitigate the effects of large aircraft crashes. See CRS Report RL34331, *Nuclear Power Plant Security and Vulnerabilities*, by Mark Holt and Anthony Andrews.

Notes: NUREG-1037 was never released, but draft versions were apparently circulated.

a. psia = pounds per square inch atmospheric.

The NRC Containment Performance Working Group studied containment buildings in 1985 to estimate their potential leak rates as a function of increasing internal pressure and temperature associated with severe accident sequences involving significant core damage.[23] It indentified potential leak paths through containment penetration assemblies (such as equipment hatches, airlocks, purge and vent valves, and electrical penetrations) and their contributions to leakage from for the containment. Because the group lacked reliable experimental data on the leakage behavior of containment penetrations and isolation barriers at pressures beyond their design conditions, it relied on an analytical approach to estimate the leakage behavior of components found in specific reference plants that approximately characterize the various containment types.

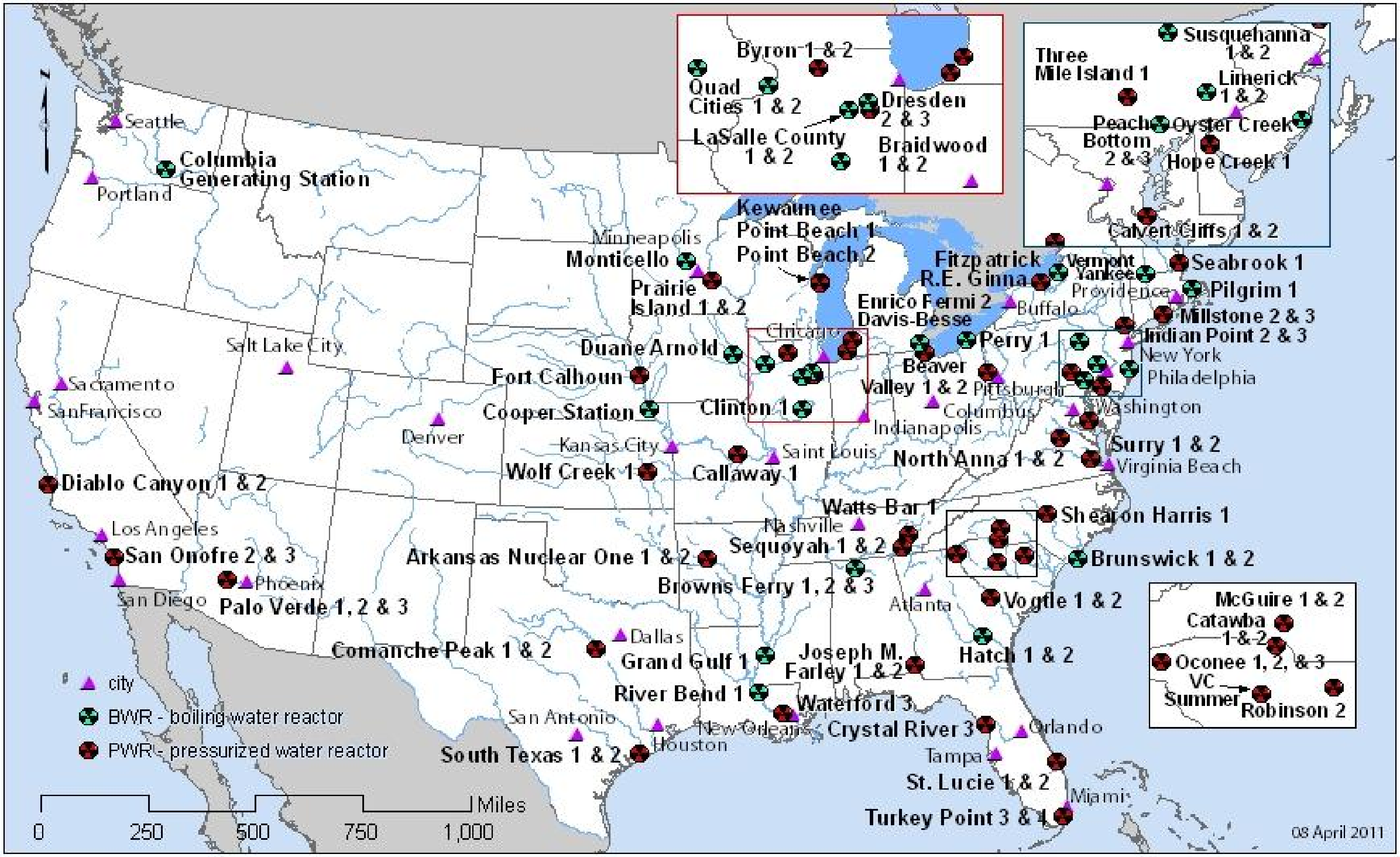
Nuclear Power Plants Operating in the United States

The locations of all 104 nuclear power plants operating in the United States are shown on the map in **Figure 6**.

[23] U.S. Nuclear Regulatory Commission, *General Studies of Nuclear Reactors; BWR Type Reactors; Containment; Reactor Accidents; Leaks; PWR Type Reactors; Accidents; Reactors; Water Cooled Reactors; Water Moderated Reactors*, NUREG-1037, May 1, 1985.

Figure 6. Commercial Nuclear Power Plants Operating in the United States

One hundred and four (104) Operating Reactors



Source: Prepared by the Library of Congress Geography and Maps Division for CRS using U.S. NRC Find Operating Nuclear Reactors by Location or Name, http://www.nrc.gov/info-finder/reactor/index.html#AlphabeticalList.

Notes:

Unit	Type	MW	Vendor	St.	Lic.	Unit	Type	MW	Vendor	St.	Lic.	Unit	Type	MW	Vendor	St.	Lic.
Arkansas Nuclear 1	PWR	843	B&W	AK	1974	Grand Gulf 1	BWR	1,297	GET6	MS	1984	Point Beach 1	PWR	512	W2L	WI	1970
Arkansas Nuclear 2	PWR	995	CE	AK	1974	Hatch 1	BWR	876	GET4	GA	1974	Point Beach 2	PWR	514	W2L	WI	1973
Beaver Valley 1	PWR	892	W3L	PA	1976	Hatch 2	BWR	883	GET4	GA	1978	Prairie Island 1	PWR	551	W2L	MN	1874
Beaver Valley 2	PWR	846	W3L	PA	1987	Robinson 2	PWR	710	W3L	SC	1970	Prairie Island 2	PWR	545	W2L	MN	1974
Braidwood 1	PWR	1,178	W4L	IL	1987	Hope Creek 1	BWR	1,061	GET4	NJ	1986	Quad Cities 1	BWR	867	GET3	IL	1972
Braidwood 2	PWR	1,152	W4L	IL	1988	Indian Point 2	PWR	1,023	W4L	NY	1973	Quad Cities 2	BWR	869	GET3	IL	1972
Browns Ferry 1	BWR	1,065	GET4	AL	1973	Indian Point 3	PWR	1,025	W4L	NY	1975	R. E. Ginna	PWR	498	W2L	NY	1969
Browns Ferry 2	BWR	1,104	GET4	AL	1974	Joseph M. Farley 1	PWR	851	W3L	AL	1977	River Bend 1	BWR	989	GET6	LA	1985
Browns Ferry 3	BWR	1,115	GET4	AL	1976	Joseph M. Farley 2	PWR	860	W3L	AL	1981	Salem 1	PWR	1,174	W4L	NJ	1976
Brunswick 1	BWR	938	GET4	NC	1976	Kewaunee	PWR	556	W2L	WI	1973	Salem 2	PWR	1,130	W4l	NJ	1981
Brunswick 2	BWR	937	GET4	NC	1974	LaSalle County 1	BWR	1,118	GET5	IL	1982	San Onofre 2	PWR	1,070	CE	CA	1982
Byron 1	PWR	1,164	W4L	IL	1985	LaSalle County 2	BWR	1,120	GET5	IL	1983	San Onofre 3	PWR	1,080	CE	CA	1992
Byron 2	PWR	1,136	W4L	IL	1987	Limerick 1	BWR	1,134	GET4	PA	1985	Seabrook 1	PWR	1,295	W4L	NH	1990
Callaway 1	PWR	1,236	WFL	MO	1984	Limerick 2	BWR	1,134	GET4	PA	1989	Sequoyah 1	PWR	1,148	W4L	TN	1980
Calvert Cliffs 1	PWR	873	CE	MD	1974	McGuire 1	PWR	1,100	W4L	NC	1981	Sequoyah 2	PWR	1,126	W4L	TN	1981
Calvert Cliffs 2	PWR	862	CE	MD	1976	McGuire 2	PWR	1,100	W4L	NC	1983	Shearon Harris 1	PWR	900	W3L	NC	1986
Catawba 1	PWR	1,129	W4L	SC	1985	Millstone 2	PWR	884	CE	CT	1975	South Texas 1	PWR	1,410	W4L	TX	1988
Catawba 2	PWR	1,129	W4L	SC	1986	Millstone 3	PWR	1,227	W4L	CT	1986	South Texas 2	PWR	1,410	W4L	TX	1989
Clinton 1	BWR	1,065	GET6	IL	1987	Monticello	BWR	579	GET3	MN	1970	St. Lucie 1	PWR	839	CE	FL	1976
Columbia Gen. St.	BWR	1,190	GET5	WA	1984	Nine Mile Pt .1	BWR	621	GET2	NY	1974	St. Lucie 2	PWR	839	CE	FL	1983
Comanche Peak 1	PWR	1,200	W4L	TX	1990	Nine Mile Pt. 2	BWR	1,140	GET5	NY	1987	Surry 1	PWR	799	W3L	VA	1972
Comanche Peak 2	PWR	1,150	W4L	TX	1993	North Anna 1	PWR	981	W3L	VA	1978	Surry 2	PWR	799	W3l	VA	1973
Cooper Station	BWR	830	GET4	NE	1974	North Anna 2	PWR	973	W3L	VA	1980	Susquehanna 1	BWR	1,149	GET4	PA	1982
Crystal River 3	PWR	838	B&WLL	FL	1976	Oconee 1	PWR	846	B&WLL	SC	1973	Susquehanna 2	BWR	1,140	GET4	PA	1984
Davis-Besse	PWR	893	B&WLL	OH	1977	Oconee 2	PWR	846	B&WLL	SC	1973	Three Mile Isl. 1	PWR	786	B&WLL	PA	1974
Diablo Canyon 1	PWR	1,151	W4L	CA	1984	Oconee 3	PWR	846	B&WLL	SC	1974	Turkey Point 3	PWR	720	W3L	FL	1972
Diablo Canyon 2	PWR	1149	W4L	CA	1985	Oyster Creek	BWR	619	GET2	NJ	1991	Turkey Point 4	PWR	720	W3l	FL	1973
Donald C. Cook 1	PWR	1,009	W4L	MI	1974	Palisades	PWR	778	CE	MI	1971	VC Summer	PWR	966	W3l	SC	1982
Donald C. Cook 2	PWR	1,060	W4L	MI	1977	Palo Verde 1	PWR	1,335	CES80	AZ	1985	Vermont Yankee	BWR	510	GET4	VT	1972
Dresden 2	BWR	867	GET3	IL	1991	Palo Verde 2	PWR	1,335	CES80	AZ	1986	Vogtle 1	PWR	1,109	W4L	GA	1987
Dresden 3	BWR	867	GET3	IL	1971	Palo Verde 3	PWR	1,335	CES80	AZ	1987	Vogtle 2	PWR	1,127	W4L	GA	1989
Duane Arnold	BWR	640	GET4	IA	1974	Peach Bottom 2	BWR	1,112	GET4	PA	1973	Waterford 3	PWR	1,250	CE	LA	1985
Fermi 2	BWR	1,122	GET4	MI	1985	Peach Bottom 3	BWR	1,112	GET4	PA	1974	Watts Bar 1	PWR	1,123	W4l	TN	1996
Fitzpatrick	BWR	852	GET4	NY	1974	Perry 1	BWR	1,261	GET6	OH	1986	Wolf Creek 1	PWR	1,166	W4L	KS	1985
Fort Calhoun	PWR	500	CE	NE	1973	Pilgrim 1	BWR	685	GET3	MA	1972						

Notes: No commercial nuclear power plants operate in Alaska or Hawaii. B&W: Babcock & Wilcox 2-Loop Lower; CE: Combustion Engineering; CE80: Combustion Engineering System 80; W2L Westinghouse 2-Loop; W3L Westinghouse 3-Loop; W4L Westinghouse 4-Loop; GET2: General Electric Type 2; GET3: General Electric Type 3; GET4: General Electric Type 4; GET5: General Electric Type 5; GET6: General Electric Type 6.

Plant Seismic Siting Criteria

Earthquakes occur when stresses in the earth exceed the strength of a rock mass, creating a fault or mobilizing an existing fault.[24] The fault can slip laterally (a strike/slip fault, such as the San Andreas Fault), move vertically (a thrust or reverse fault, such as the fault that caused the March 11 Japanese earthquake), or move in some combination of the two. The fault's sudden release sends seismic shock waves through the earth that have two primary characteristics: amplitude—a measure of the peak wave height, and period—the time interval between the arrival of successive peaks or valleys.[25] The seismic wave's arrival causes ground motion. The ground motion intensity depends on three factors: the distance from the source (also known as focus or epicenter), the amount of energy released (magnitude of the earthquake), and the type of soil or rock at the site.

The shallower the earthquake's focus, the stronger the waves will be when they reach the surface. Generally, the intensity of ground shaking diminishes with increasing distance from the earthquake focus. The earthquake's magnitude (*M*) is measured on a logarithmic scale (sometimes referred to as the Richter scale), thus an *M* 7.0 earthquake has amplitude that is ten times larger than an M 6.0, but releases 31.5 times more energy than an M 6.0 earthquake. Sites with deep, soft soils or loosely compacted fill will experience stronger ground motion than sites with stiff soils, soft rock, or hard rock.

Refer to **Appendix A** of this report for additional discussion on magnitude, Richter scale, and intensity. For more detailed information about earthquake hazards, refer to CRS Report RL33861, *Earthquakes: Risk, Detection, Warning, and Research*, by Peter Folger.

General Design Criteria

For nuclear power plants granted construction permits during the 1960s and 1970s, a design approach emerged for considering seismic loads based on site-specific investigations of local and regional seismology, geology and geotechnical engineering.[26] The 1973 publication of 10 C.F.R. 100, *Appendix A—Seismic and Geologic Siting Criteria for Nuclear Power Plants*, included the concept of a "safe shutdown earthquake" (SSE), which is discussed in a later section of this report.

General design criteria for nuclear power plants require that structures and components important to safety be designed to withstand the effects of earthquakes, tornados, hurricanes, floods, tsunamis, and seiche[27] waves without losing the capability to perform their safety function. These "safety-related" structures, systems, and components are those necessary to assure:

1. the integrity of the reactor coolant pressure boundary,

[24] The Applied Technology Council (ATC) and the Structural Engineers Association of California (SEAOC), *Briefing Paper 1 Building Safety and Earthquakes Part A: Earthquake Shaking and Building Response*, Redwood City, CA, http://www.atcouncil.org/.

[25] The wave's frequency is the inverse of the period (1/*s*), and is expressed as the number of wave cycles per second (termed Hertz or *Hz*).

[26] U.S. Nuclear Regulatory Commission, *Evaluation of the Seismic Design Criteria in ASCE/SEI Standard 43-05 for Application to Nuclear Power Plants*, NUREG/CR-6926, Brookhaven National Laboratory, NY, March 2007.

[27] Standing waves, or waves that move vertically but not horizontally. Seiche waves can be triggered by earthquakes, strong winds, tides, and other causes.

2. the capability to shut down the reactor and maintain it in a safe condition, or
3. the capability to prevent or mitigate the consequences of accidents, which could result in potential offsite exposures.

Refer to this report's section on "Nuclear Power Plant Designs" for some discussion of safety-related components.

The language in 10 C.F.R. 100, *Appendix A*, notes that the seismic criteria are based on limited geophysical and geologic information, available at the time, on faults and earthquake occurrences, and that the information would be revised when more information became available. The information is based on a review of historical records and a site investigation. Ultimately, the investigation provides the basis for determining a "safe shutdown earthquake," alternately referred to as the "design basis earthquake," defined as the maximum vibratory ground motion for which certain structures, systems, and components are designed to remain functional. Under an "operating basis earthquake," the reactor could continue operation without undue risk to the safety of the public.

The NRC subsequently published a series of Regulatory Guides in support of *Appendix A* of 10 C.F.R. 100. These guides provide technical information, procedures, and design criteria that are beyond the scope of this report.

- Regulatory Guide 1.60, *Design Response Spectra of Nuclear Power Reactors* (1973), provides ground design response spectral shapes for horizontal and vertical ground movements developed from a statistical analysis of response spectra of past Western United States (WUS) strong-motion earthquakes collected from a variety of different site conditions, primarily at deep soil sites.
- Regulatory Guide 1.165, *Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion* (1997) ,provided procedures for (1) conducting geological, geophysical, seismological, and geotechnical investigations, (2) identifying and characterizing seismic sources, (3) conducting probabilistic seismic hazard analysis (PSHA), and (4) determining the safe shutdown earthquake for satisfying the requirements of 10 C.F.R. 100.23. The guide evolved out of investigations into seismic hazard estimates for nuclear power plant sites operating in the Central and Eastern United States (CEUS).
- NUREG/CR-6926, *Evaluation of the Seismic Design Criteria in ASCE/SEI Standard 43-05 for Application to Nuclear Power Plants* (2007), provided seismic design criteria for safety-related structures, systems, and components in a broad spectrum of nuclear facilities.[28]

Site Investigations

The site investigations required under 10 C.F.R. 100, *Appendix A*, starts with a review of pertinent literature and progresses to field investigations. The required investigations include:

[28] Based on a review by the American Society of Civil Engineers/Structural Engineering Institute (ASCE/SEI) Standard 43-05 - *Seismic Design Criteria for Structures, Systems, and Components in Nuclear Facilities.*

- Vibratory Ground Motion—examines lithology, stratigraphy, structural geology, underlying tectonic structures, physical earthquake evidence, engineering properties of underlying soil and rock, historically reported earthquakes, earthquake epicenters within 200 miles of site, faults within 200 miles.
- Surface Faulting—evaluates lithology, stratigraphy, structural geology, underlying tectonic structures, evidence of fault offsets, nearby faults greater than 1,000 feet in length, records of earthquakes associated with faults greater than 1,000 feet in length, epicenters of earthquakes with faults greater than 1,000 feet in length.
- Seismically Induced Floods and Water Waves ─ looks at reports or evidence of distantly and locally generated waves or tsunamis which have or could have affected the site, and evidence for seismically induced floods and water waves that have or could have affected the site.

Safe Shutdown Earthquake Condition

The NRC defines the Safe Shutdown Earthquake as the maximum earthquake potential for which certain structures, systems, and components, important to safety, are designed to sustain and remain functional.[29] During an earthquake, ground motion sets up vibrations in a nuclear power plant's foundation and structure. In simple terms, the vibrations represent the back-and-forth acceleration of an object (the distance moved is the amplitude). Vibration, or horizontal ground acceleration, is measured in terms of the earth's gravitational acceleration constant (g) for structural design purposes.[30] These vibrations place additional loads and displacements on the nuclear power plant's structure, equipment and piping systems. The additional loading must be accounted for in the structural design of the piping systems supports.

Various plant structures, depending upon their elevation above the foundation, vibrate at different frequencies during an earthquake. Low frequency vibrations in the range of 1 to 10 Hz (cycles per second) are particularly problematic for a wide range of structures because such structures are often susceptible to damaging resonance at those frequencies. These accelerations and the corresponding shaking frequencies are used in the probabilistic seismic hazard analysis (PSHA) discussed in this report's "Background on Seismic Standards" section. The full seismic spectrum can be characterized by two intervals: peak ground acceleration (PGA) and spectral acceleration (SA) averaged between 5 and 10 Hz. PGA has been widely used to develop nuclear power plant "fragility estimates" and represents the performance of nuclear plant structures, systems, and components (SSCs) that are sensitive to inertial effects.

The maximum vibratory accelerations of the Safe Shutdown Earthquake must take into account the characteristics of the underlying soil material in transmitting the earthquake-induced motions at the various locations of the plant's foundation. A multiple degree-of-freedom analysis is used to simulate the effect of the earthquake on the piping systems.

Experimental and empirical seismic data have provided insights into the behavior of different structures under various acceleration and shaking conditions. One conclusion reached regarded

[29] http://www.nrc.gov/reading-rm/basic-ref/glossary/safe-shutdown-earthquake.html

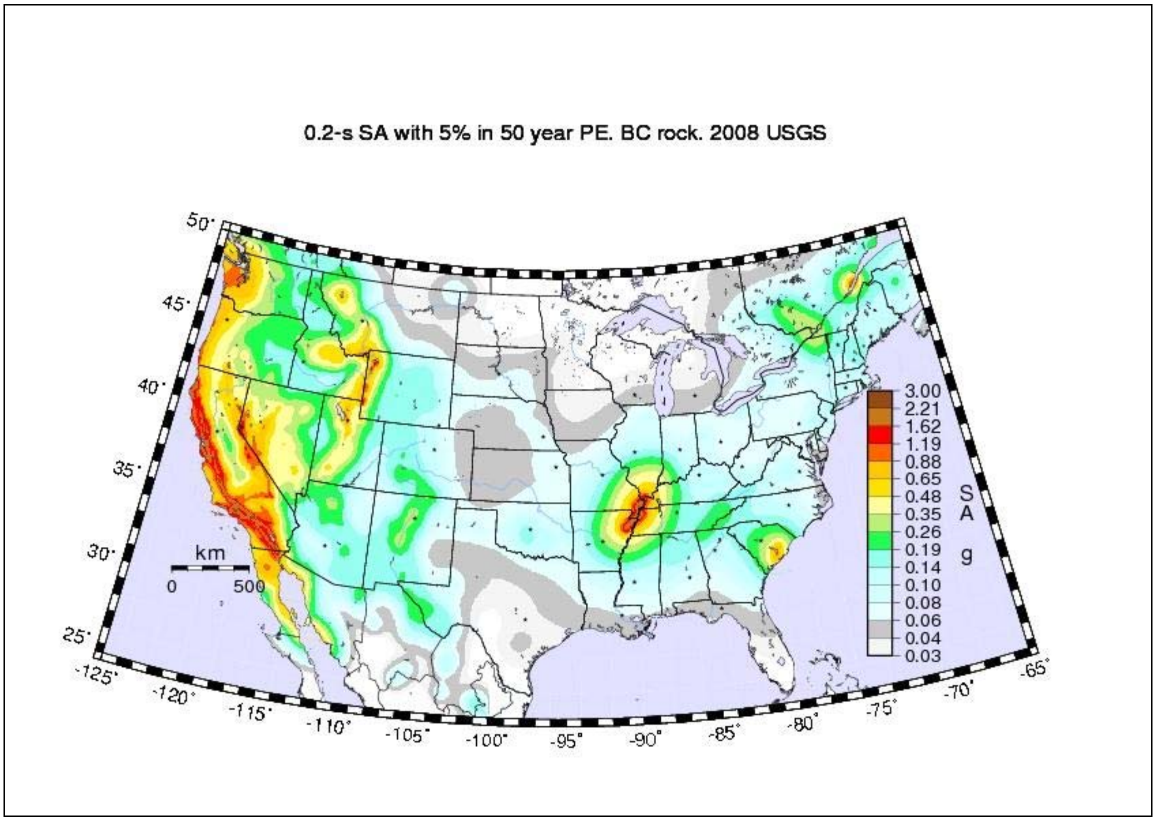
[30] Gravitation acceleration g = 32 feet/second/second (ft/second^2).

the performance of welded steel piping at power plants during strong motion earthquakes. Relatively small numbers of failures occurred when peak ground accelerations remained below 0.5g.[31] Other types of structures would exhibit different behaviors, and engineers design the various plant structures to withstand a certain severity of earthquake specific to each plant site.

The example of **Figure 7** shows areas susceptible to shaking of a frequency of 5 Hz having a 5% probability of occurring at least once within 50 years.[32] The map shows the strength of the expected acceleration (in g) for areas experiencing such an earthquake. The darker colors on the map indicate areas of strongest shaking.

Figure 7. Spectral Acceleration 5 Hz

Return Period of 5% in 50 Years



Source: USGS National Seismic Hazard maps, USGS Open-File Report 2008-1128, 2008, http://earthquake.usgs.gov/hazards/.

Notes: Areas that are susceptible to shaking at a frequency of 5 Hz with a 5% probability of occurring at least once within 50 years. The strength of the expected acceleration is expressed in terms of earth's gravitational

[31] N.C. Chokshi, S.K. Shaukat, and A.L. Hiser, et al., *Seismic Considerations for the Transition Break Size,* U.S. NRC, NUREG-1903, February 2008, pp. 29-30.

[32] This collection of USGS seismic hazard maps includes probabilistic ground motion maps for Peak Ground Acceleration (PGA), 1Hz (1.0 second SA), and 5Hz (0.2 second SA). (Refer to the report section on "Safe Shutdown Earthquake" for a discussion of spectral acceleration.) Some additional spectral accelerations (SA) are also included for central and southern California. Most figures correspond to the 2% in 50-year probability of exceedance, but there are a few figures for the 10% in 50 year and the 5% in 50-year probability of exceedance as well a range of accelerations and associated probabilities.

acceleration constant (g) for areas experiencing such an earthquake. The darker colors on the map indicate areas of strongest shaking.

National Seismic Hazard Maps

In 2008, the U.S. Geological Survey (USGS) released an update of the National Seismic Hazard Maps (NSHM).[33] The purpose of the maps is to show the likelihood of a particular severity of shaking within a specified time-period. The Seismic Hazard maps are the basis for seismic design provisions of building codes to allow buildings, highways, and critical infrastructure to withstand earthquake shaking without collapse. The NRC requires that every nuclear plant be designed for site-specific ground motions that are appropriate for their site locations. In addition, the NRC has specified a minimum ground motion level to which nuclear plants must be designed. (See discussion above on design criteria.)

The USGS revises the NHSM every six years to reflect newly published earthquake data to update building code seismic design provisions. USGS notes that the 2008 hazard maps differ significantly from the 2002 maps in many parts of the United States:

> The new maps generally show 10- to 15-percent reductions in acceleration across much of the Central and Eastern United States [CEUS] for 0.2-s [second] and 1.0-s spectral acceleration and peak horizontal ground acceleration for 2-percent probability of exceedance in 50 years. The new maps for the Western United States [WUS] indicate about 10-percent reductions for 0.2-s spectral acceleration and peak horizontal ground acceleration and up to 30-percent reductions in 1.0-s spectral acceleration at similar hazard levels.[34]

In the Central and Eastern United States (CEUS), the New Madrid Seismic Zone and the Charleston area in southeast South Carolina comprise the dominant seismic hazard (at 2% probability of exceedance in 50 years). Seismically active portions of eastern Tennessee and some portions of the northeast also contribute to the seismic hazard. The hazard at the 2% probability of exceedance in 50 years level is typically a factor of two to four times higher than the 10% probability of exceedance in 50 years values in the seismically active portions of the CEUS.

Seismic hazards are greatest in the Western United States (WUS), particularly in California, Oregon, and Washington, as well as Alaska and Hawaii. The hazard at the 2% probability of exceedance in 50 years level is typically a factor of 1.5 to 2 times higher than the 10% in 50 years values in coastal California and from 2 to 3.5 higher across the rest of the WUS.

CRS has mapped the proximity of plant sites to seismic hazards based on the USGS National Seismic Hazard Map for the United States in **Figure 8**. This map displays quantitative information about seismic ground motion hazards as horizontal ground acceleration (*g*) of a particle at ground level moving horizontally during an earthquake.

CRS has also mapped the proximity of plant sites to Quaternary period faults based on the USGS Quaternary Fault and Fold Database of the United States in **Figure 9**. The USGS Database has information on faults and associated folds in the United States that are believed to be sources of

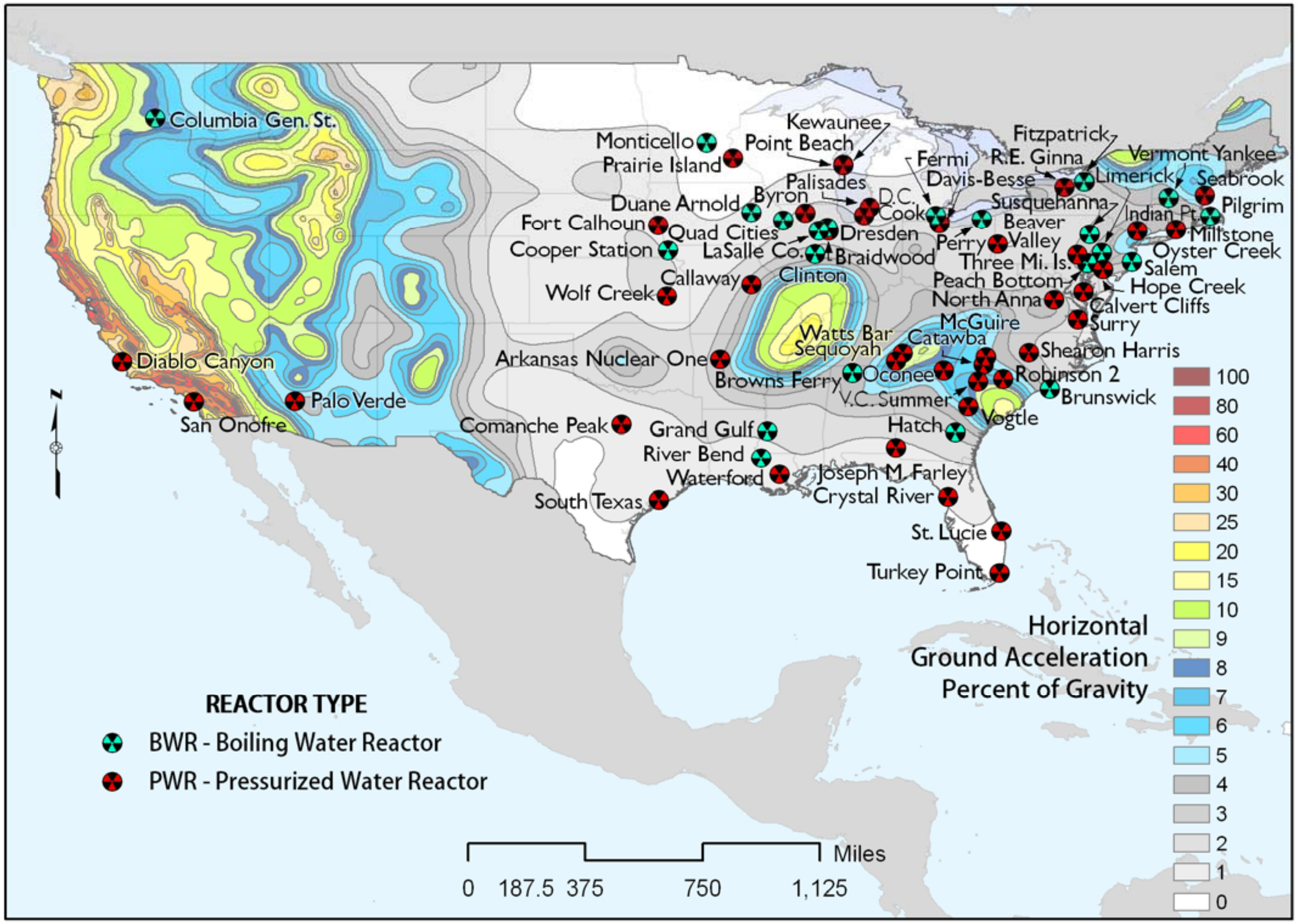
[33] Mark D. Petersen, Arthur D. Frankel, and Stephen C. Harmsen, et al., *Documentation for the 2008 Update of the United States National Seismic Hazard Maps*, U.S Geological Survey, Open-File Report 2008-1128, 2008, http://earthquake.usgs.gov/hazards/.

[34] Ibid.

greater than magnitude 6 earthquakes during the past 1,600,000 years — the Quaternary period of the geologic time scale. The map is not a prediction of an earthquake event.

Figure 8. Operating Nuclear Power Plant Sites and Seismic Hazard

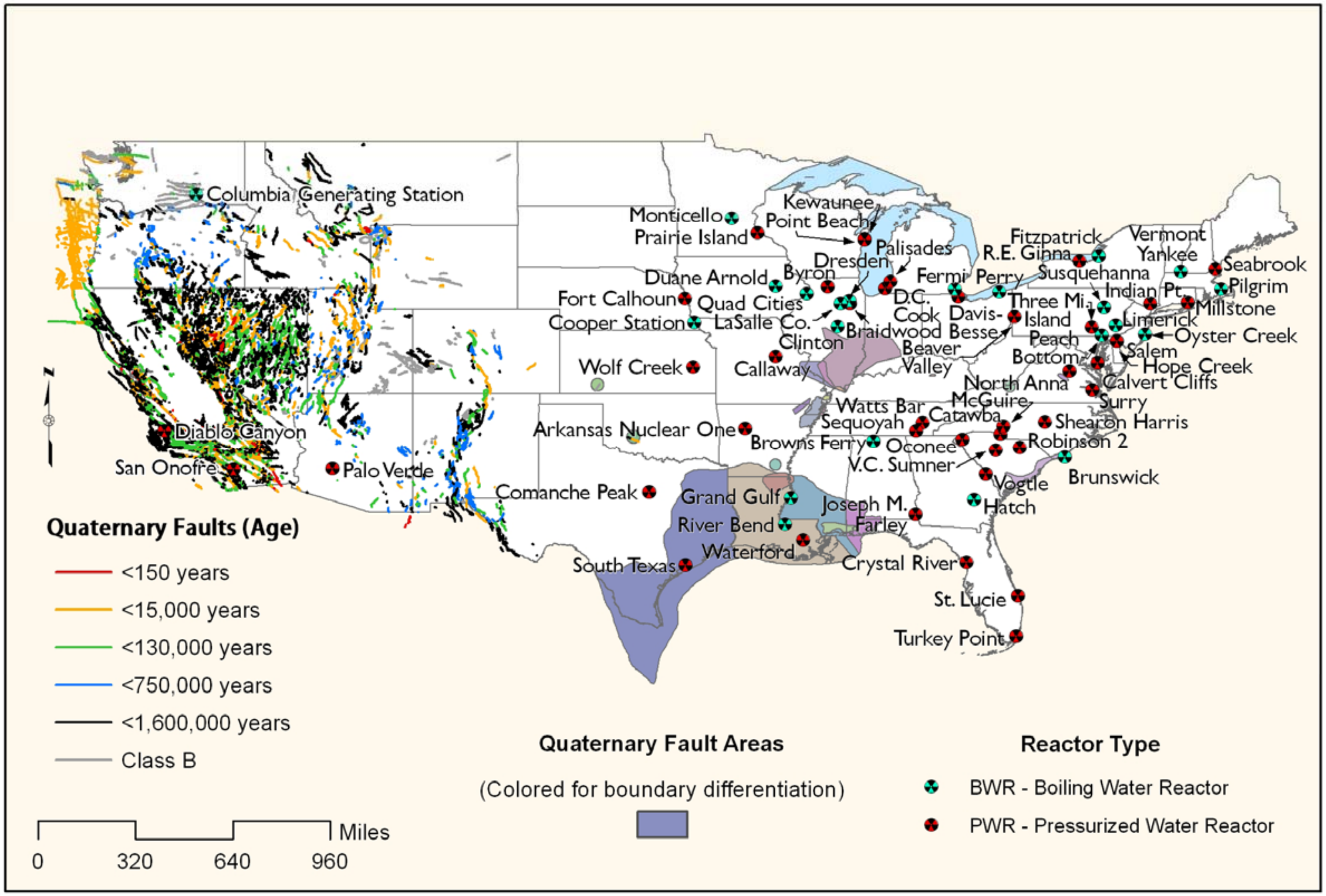
Seismic hazard expressed as horizontal ground acceleration (shown as a percent of gravity)



Source: Background map USGS Seismic Hazard Map for the United States, prepared for CRS by the Library of Congress Geography and Maps Division.

Notes: This map displays quantitative information about seismic ground motion hazards as horizontal ground acceleration (in terms of gravitational acceleration) of a particle at ground level moving horizontally during an earthquake. This map is not a prediction of an earthquake event. The NRC does not rank nuclear plants by seismic risk. No commercial nuclear power plants operate in either Alaska or Hawaii.

Figure 9. Operating Nuclear Power Plant Sites and Mapped Quaternary Faults



Source: CRS and the USGS Quaternary Fault and Fold Database of the United States.

Notes: To map the proximity of plant sites to faults, CRS referred to the USGS Quaternary Fault and Fold Database of the United States. This is information on faults and associated folds in the United States that are believed to be sources of greater than moment magnitude 6 (M>6) earthquakes during the Quaternary (the past 1,600,000 years). This map is not a prediction of an earthquake event. No commercial nuclear power plants operate in either Alaska or Hawaii.

NRC Priority Earthquake Safety Review

The NRC has required that each nuclear plant be built to certain structural specifications based on the earthquake susceptibility of each plant site, but some of those design specifications may be re-evaluated in light of new seismic analysis in the United States. In 2010 the NRC published GI-199 Safety/Risk Assessment, a two-stage assessment that determines the implications of USGS updated probabilistic seismic hazards in the Central and Eastern U.S. (CEUS) on existing nuclear power plant sites.[35] The assessment first evaluated the change in seismic hazard with respect to previous estimates at individual NPPs, and then estimated the change in Seismic Core Damage Frequency (SCDF) resulting from change in the seismic hazard. Seismic core damage frequency is the probability of damage to the reactor core (fuel rods) resulting from a seismic initiating event. It does not imply either a core meltdown or the loss of containment, which would be required for radiological release to occur. The seismic hazard at each plant site depends on the unique seismology and geology surrounding the site. Consequently, the report separately determined the implications of updated probabilistic seismic hazard for each of the 96 operating NPPs in the CEUS.[36]

The NRC does not rank nuclear plants by seismic risk. NRC's objective in the GI-199 Safety/Risk Assessment was to evaluate the need for further investigations of seismic safety for operating reactors in the CEUS. The data evaluated in the assessment suggest that the probability for earthquake ground motion above the seismic design basis for some nuclear plants in the CEUS, although still low, is larger than previous estimates. In late March 2011, the NRC announced that it had identified 27 nuclear reactors operating in the CEUS that would receive priority earthquake safety reviews.[37] Those 27 reactors are listed in **Table 4**.

Table 4. Operating Nuclear Power Plants Subject to Earthquake Safety Reviews

Plant	St.	Type	Plant	St.	Type	Plant	St.	Type
Crystal River 3	FL	PWR	North Anna 1 & 2	VA	PWR	Sequoyah 1 & 2	TN	PWR
Dresden 2 & 3	IL	BWR	Oconee 1, 2 & 3	SC	PWR	Seabrook	NH	PWR
Duane Arnold	IA	BWR	Perry 1	OH	BWR	V.C. Summer	SC	PWR
Joseph M. Farley 1 & 2	AL	PWR	Peach Bottom 2 & 3	PA	BWR	Watts Bar 1	TN	PWR
Indian Point 2 & 3	NY	PWR	River Bend 1	LA	BWR	Wolf Creek	KS	PWR
Limerick 1 & 2	PA	BWR	Saint Lucie 1 & 2	FL	PWR			

Source: The Energy Daily.

Note: The NRC has not announced a schedule for completing the seismic reviews at the time of this report.

[35] U.S. Nuclear Regulatory Commission, *Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States Existing Plants—Safety/Risk Assessment,* Generic Issue 199 (GI-199), August 2010.

[36] Ibid.

[37] George Lobsenz, "NRC Task Force To Review Safety: 27 Reactors Are Seismic Priorities," *The Energy Daily*, March 24, 2011.

Recent Legislative Activities

Within a few days following Japan's nuclear crisis, Democrats on the House Energy and Commerce Committee requested a hearing on U.S. Nuclear Power Plant Safety and Preparedness.[38]

On March 17, 2011, the Senate Committee on Homeland Security and Governmental Affairs held a hearing on Catastrophic Preparedness that looked at technologies and emergency procedures used in the event of a large-scale earthquake or other natural disaster.[39] On April 6, 2011, the Subcommittee on Oversight and Investigations of the House Energy and Commerce Committee held a hearing the U.S. Government Response to the Nuclear Power Plant Incident in Japan.[40] On April 7, 2011, the Subcommittee on Technology and Innovation of the House Science, Space, and Technology Committee held a hearing on Earthquake Risk Reduction.[41]

Several bills have been introduced in the 112th Congress that are relevant to either nuclear power plant safety of earthquake hazard assessment.

S. 646, the Natural Hazards Risk Reduction Act of 2011, would amend the Earthquake Hazards Reduction Act of 1977 (42 U.S.C. 7704) to add program activities to research and develop effective methods, tools, and technologies to reduce the risk posed by earthquakes, and authorize the United States Geological Survey to conduct research and other activities necessary to characterize and identify earthquake hazards, assess earthquake risks, monitor seismic activity, and provide real-time earthquake information.

H.R. 1379, the Natural Hazards Risk Reduction Act of 2011, would also amend the Earthquake Hazards Reduction Act of 1977 (42 U.S.C. 7704) to research and develop effective methods, tools, and technologies to reduce the risk posed by earthquakes to the built environment, especially to lessen the risk to existing structures and lifelines.

H.R. 1268, the Nuclear Power Licensing Reform Act of 2011, would amend Section 103 of the Atomic Energy Act of 1954 (42 U.S.C. 2133), subsection c, by adding at the end the following: 'Any such renewal shall be subject to the same criteria and requirements that would be applicable for an original application for initial construction, and the Commission shall ensure that any changes in the size or distribution of the surrounding population, or seismic or other scientific data not available at time of original licensing, have not resulted in the facility being located at a site at which a new facility would not be allowed to be built.

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[38] House Committee on Energy & Commerce Democrats, Committee Democrats Request Hearing on U.S. Nuclear Power Plant Safety and Preparedness, http://democrats.energycommerce.house.gov/index.php?q=news/committee-democrats-request-hearing-on-us-nuclear-power-plant-safety-and-preparedness.

[39] Senate Committee on Homeland Security & Governmental Affairs, Catastrophic Preparedness: How Ready is FEMA for the Next Big Disaster? http://hsgac.senate.gov/public/index.cfm?FuseAction=Hearings.Hearing&Hearing_ID=a42880b1-22fc-4890-b82c-dd2a369e2aa2

[40] House Energy & Commerce Committee, The U.S. Government Response to the Nuclear Power Plant Incident in Japan, http://energycommerce.house.gov/hearings/hearingdetail.aspx?NewsID=8420.

[41] House Committee on Science, Space, and Technology, Subcommittee Reviews Status of U.S. Earthquake Preparedness, http://science.house.gov/press-release/subcommittee-reviews-status-us-earthquake-preparedness.

H.R. 1242, the Nuclear Power Safety Act of 2011, would amend the Atomic Energy Act to revise regulations to ensure that nuclear facilities licensed under the act can withstand and adequately respond to an earthquake, tsunami (for a facility located in a coastal area), strong storm, or other event that threatens a major impact to the facility; a loss of the primary operating power source for at least 14 days; and a loss of the primary backup operating power source for at least 72 hours.

Appendix A. Magnitude, Intensity, and Seismic Spectrum

Earthquake magnitude is a measure of the strength of the earthquake as determined from seismographic observations. Magnitude is essentially an objective, quantitative measure of the size of an earthquake.[42] The magnitude can be expressed in various ways based on seismographic records (e.g., Richter Local Magnitude, Surface Wave Magnitude, Body Wave Magnitude, and Moment Magnitude). Currently, the most commonly used magnitude measurement is the Moment Magnitude (M) which is based on the strength of the rock that ruptured, the area of the fault that ruptured, and the average amount of slip.[43] Moment is a physical quantity proportional to the slip on the fault times the area of the fault surface that slips; it is related to the total energy released in the earthquake. The moment can be estimated from seismograms (and from geodetic measurements). The Moment Magnitude provides an estimate of earthquake size that is valid over the complete range of magnitudes, a characteristic that was lacking in other magnitude scales, such as the Richter scale.

Because of the logarithmic basis of the scale, each whole number increase in magnitude represents a tenfold increase in measured amplitude; as an estimate of energy, each whole number step in the magnitude scale corresponds to the release of about 31 times more energy than the amount associated with the preceding whole number value.

The Richter magnitude scale was developed in 1935 by Charles F. Richter of the California Institute of Technology and was based on the behavior of a specific seismograph that was manufactured at that time. The instruments are no longer in use and therefore the Richter magnitude scale is no longer used in the technical community. However, the Richter Scale is a term that is so commonly used by the public that scientists generally just answer questions about "Richter" magnitude by substituting moment magnitude without correcting the misunderstanding.

The intensity of an earthquake is a qualitative assessment of effects of the earthquake at a particular location. The intensity assigned is based on observed effects on humans, on human-built structures, and on the earth's surface at a particular location. The most commonly used scale in the United States is the Modified Mercalli Intensity (MMI) scale, which has values ranging from I to XII in the order of severity. MMI of I indicates an earthquake that was not felt except by a very few, whereas MMI of XII indicates total damage of all works of construction, either partially or completely. While an earthquake has only one magnitude, intensity depends on the effects at each particular location.

Greater magnitude earthquakes are generally associated with greater lengths of fault ruptures.[44] A fault break of 100 miles might be associated with an M8 earthquake, while a break of several miles might generate an M6 earthquake. The length of the fault break, however, is not directly proportional to the energy released. The induced amplitude of acceleration (g) does increase with

[42] US NRC, *NRC frequently asked questions related to the March 11, 2011 Japanese Earthquake and Tsunami*.

[43] USGS, *Measuring Earthquakes*, http://earthquake.usgs.gov/learn/faq/?categoryID=2&faqID=23.

[44] H. Bolton Seed, I. M. Idriss, and Fred. W. Kiefer, "Characteristics of Rock Motions During Earthquakes," *Journal of Soil Mechanics and Foundation Division, Proceedings of the American Society of Civil Engineers*, September 1969, pp. 1199-1217.

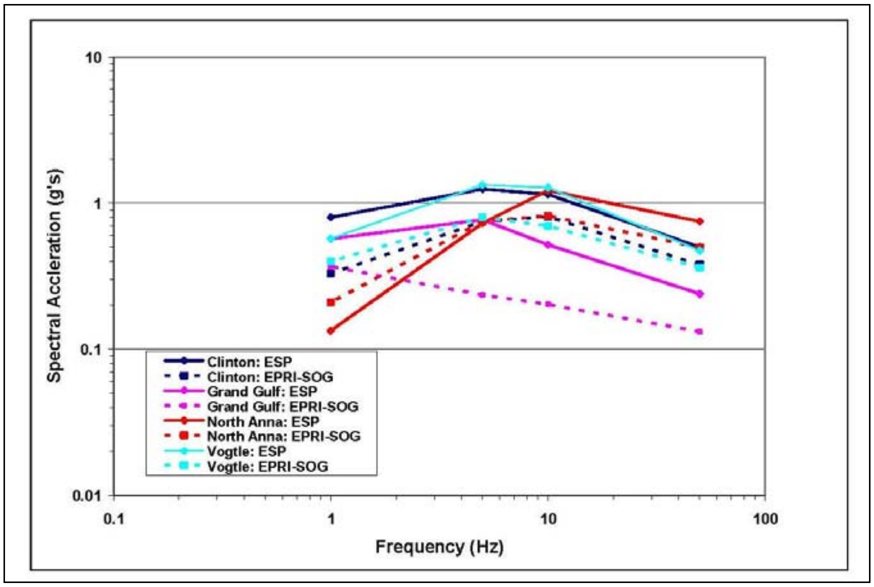
increasing magnitude (M). Various methods have been developed to relate the magnitude of an earthquake to the amplitude of acceleration it induces, and different methods may result in significant variations in results.

The seismic spectrum can be characterized by two intervals—peak ground acceleration (PGA) and spectral acceleration averaged between 5 and 10 Hz (SAAvg5-10). PGA has been widely used to develop fragility estimates and represents the performance of nuclear plant structures, systems, and components (SSCs) that are sensitive to inertial effects.

Figure A-1 shows a example of response spectra for several power plants.[45] The frequency range, of 1 to 10 *Hz,* is the subject of USGS earthquake hazard studies, as discussed above.

Figure A-1. Spectral Acceleration (*g*) vs. Frequency (Hz)

Curves are response spectral values (5% damping) at an annual exceedance frequency of10^{-5}



Source: NRC Generic Issue -99, *Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants*, Figure 1, August 2010.

Notes: For illustrative purposes only. Originally prepared to compare seismic hazard results for four early site permit submittals. Solid line represents submittals to 1989. Dashed Lines represent Electric Power Research Institute Seismicity Owners Group Study.

NUREG/CR-6926 references the American Society of Civil Engineers (ASCE) Standard 7-05 *Minimum Design Loads for Buildings and Other Structures* seismic hazard maps. The maximum considered earthquake (MCE) is based on spectral accelerations with 2%/50 yr probability (2% probability of being equaled or exceeded in any single year in 50 years or otherwise stated as a 2% annual exceedance probability). (To obtain the design earthquake spectral response accelerations (DS) used in structural design, the spectral accelerations are multiplied by 2/3.) At sites in seismically active regions in the Western United States (WUS), the corresponding DS hazard is approximately 10%/50 yr (return period of 475 yr). In the Central and Eastern United

[45] Frequency *Hz* (Hertz) refers to the number of cycles per second (which is inverse of the ground motion wave period ─ the time between two wave peaks). Thus, 0.2-*s* is the equivalent of 5 *Hz* (1/0.2-*s*), and 1-*s* is the equivalent of 1 Hz (1/1-*s*).

States (CEUS) this hazard is approximately 4%/50 yr (return period of approximately 1,200 yr), These are due to differences in the typical slopes of seismic hazard curves in the WUS and CEUS.

Appendix B. Terms

Boiling water reactor (BWR) directly generates steam inside the reactor vessel.

Deterministic Seismic Hazard Assessment (DSHA) focuses on a single earthquake event to determine the finite probability of occurring.

Double-ended guillotine break (DEGB) represents a break of the largest diameter pipe in the primary system that the emergency core cooling system (ECCS) must be sized to provide adequate makeup water to compensate for.

Light water reactor systems use ordinary water as a fuel moderator and coolant, and uranium fuel artificially enriched to 4.5%-5% fissile uranium-235. Includes BWR and PWR types.

Loss of Coolant Accident (LOCA) is the most severe operating condition for a reactor that can contribute to a reactor core meltdown.

Operating Basis Earthquake is the maximum vibratory ground motion that a reactor could continue operation without undue risk and safety of the public.

Pressurized water reactor (PWR) uses two major loops to convert the heat generated by the reactor core into steam outside of the reactor vessel.

Probabilistic Seismic Hazard Assessments (PSHA) attempt to quantify the probability of exceeding various ground-motion levels at a site given all possible earthquakes.

Safe Shutdown Earthquake (also design basis earthquake) is the maximum vibratory ground motion at which certain structures, systems, and components are designed to remain functional.

Seismic Core Damage Frequency is the probability of damage to the core resulting from a seismic initiating event.

Author Contact Information

Anthony Andrews
Specialist in Energy and Defense Policy
aandrews@crs.loc.gov, 7-6843

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Peter Folger, Specialist in Energy and Natural Resources Policy, Congressional Research Service