3.0 Description of NRC Licensed Reactor Facilities and the Decommissioning Process

This chapter provides information on both the operating nuclear power plants and those being decommissioned. First, a general description of the nuclear power plants and sites is provided in Section 3.1 to help the reader understand the types of reactor facilities that will be decommissioned, the location of the radioactive material in these facilities, and the structures, systems and components (SSCs) that will be referred to later in this document and that are important in the decommissioning process. Next, the methods that are commonly used during decommissioning are described in Section 3.2. Section 3.3 addresses the decommissioning experience of the currently decommissioning plant sites, their chosen method for decommissioning, and the activities that are being used to decommission the facilities.

There are currently 22 nuclear power reactors at 21 sites that are permanently shutdown: 19 of these reactors are in various stages of decommissioning, and 3 sites have finished decommissioning and no longer maintain a license. The decommissioning efforts at these 22 plants equates to over 200 equivalent years of experience decommissioning commercial power reactors since the 1988 *Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities*, NUREG-0586 (1988 GEIS; NRC 1988) was published. There are also currently 104 nuclear plants that have a license and are either operating or have not yet certified that they have permanently ceased power operations. Between 2006 and 2035, these 104 plants will either permanently cease operations or renew their licenses. Ultimately, they will all permanently cease operations and be decommissioned.

3.1 Plants, Sites, and Reactor Systems[a]

Between 1957 and 1996, the U.S. Nuclear Regulatory Commission (NRC) issued 126 operating licenses for commercial power reactor operation at 80 sites. The history of and experience with the 22 reactors that are being decommissioned currently or have completed decommissioning are addressed in Section 3.3. Because each of the remaining 104 operating plants will eventually enter the decommissioning process, their attributes and characteristics are included in this section to ensure that this Supplement is appropriate for future decommissioning plants.

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[a] Much of the information in this section was taken from NUREG-1437, *Generic Environmental Impact Statement for License Renewal of Nuclear Plants* (NRC 1996) and from NUREG-1628, *Staff Responses to Frequently Asked Questions Concerning Decommissioning of Nuclear Power Reactors* (NRC 2000a). This information has been supplemented and updated as appropriate to include all operating and currently decommissioning nuclear plants.

The material presented in this section is also provided as background information for the reader.

Nuclear power reactor facilities are located in 35 of the contiguous States, with none in Alaska or Hawaii. Thirty-nine sites contain two or three nuclear power reactors (units) per site. Of the 126 plants, 98 are located east of the Mississippi River with most of the nuclear capacity located in the northeast (New England States, New York, and Pennsylvania), the midwest (Illinois, Michigan, and Wisconsin) and the southeast (Virginia, North and South Carolina, Georgia, Florida, and Alabama).

Typically, nuclear power plants are sited in flat or rolling countryside, in wooded or agricultural areas away from urban areas. Most are located on or near rivers or lakes. Several plants are located in arid regions, and 19 plants are located along the seacoast on bays or inlets. More than 50 percent of the sites have 80-km (50-mile) population densities of less than 77 persons/km^2 (200 persons/mi^2) and over 80 percent have 80-km (50-mile) densities of less than 193 persons/km^2 (500 persons/mi^2). The most notable exception is the Indian Point Station, located within 80 km (50 mi) of New York City, which has a projected 1999 population density within 80 km (50 mi) of more than 770 persons/km^2 (2000 persons/mi^2). Indian Point has one permanently shutdown reactor and two operating reactors.

Site areas range from a minimum of 34 ha (84 ac) for the San Onofre Nuclear Generating Station, (a three unit site, with one permanently shutdown reactor) in California to 12,000 ha (30,000 ac) for the McGuire Nuclear Station in North Carolina (three operating units). Almost 60 percent of plant sites cover from 200 to 800 ha (500 to 2000 ac). Larger land-use areas are associated with plant cooling systems that include reservoirs, artificial lakes, and buffer areas.

Appendix F contains summary tables for both permanently shutdown and currently operating nuclear power facilities showing location, reactor type, thermal power, site area, cooling system and cooling water source, and licensing dates.

3.1.1 Types of Nuclear Power Reactor Facilities

In the United States, nearly all reactors used for commercial power generation have been conventional (thermal) light water reactors (LWRs) that use water as a moderator and coolant. The two types of LWRs are pressurized water reactors (PWRs) and boiling water reactors (BWRs). Of the 123 LWRs, 80 are PWRs and 43 are BWRs. The three plants that are not LWRs are Fermi, Unit 1, which is a permanently shutdown fast breeder reactor (FBR), and Peach Bottom, Unit 1, and Fort St. Vrain, which are permanently shutdown high-temperature gas-cooled reactors (HTGRs). The licensees for Fermi, Unit 1, and Peach Bottom, Unit 1, have elected to place both facilities in long-term storage. Fort St. Vrain has had its license terminated following completion of decommissioning activities.

Brief descriptions of these different types of reactors are given below as background.

3.1.1.1 Pressurized Water Reactors

In PWRs, water is heated to a high temperature under pressure inside the reactor. The water is then pumped in the primary circulation loop to the steam generator. Within the steam generator, water in the secondary circulation loop is converted to steam that drives the turbines. The turbines turn the generator to produce electricity. The steam leaving the turbines is condensed by water in the tertiary loop and returned to the steam generator. The tertiary loop water flows either to cooling towers, where it is cooled by evaporation or discharged to a body of water such as a river, lake, or other heat sink. The tertiary loop is open to the atmosphere, but the primary and secondary cooling loops are not (see Figure 3-1).

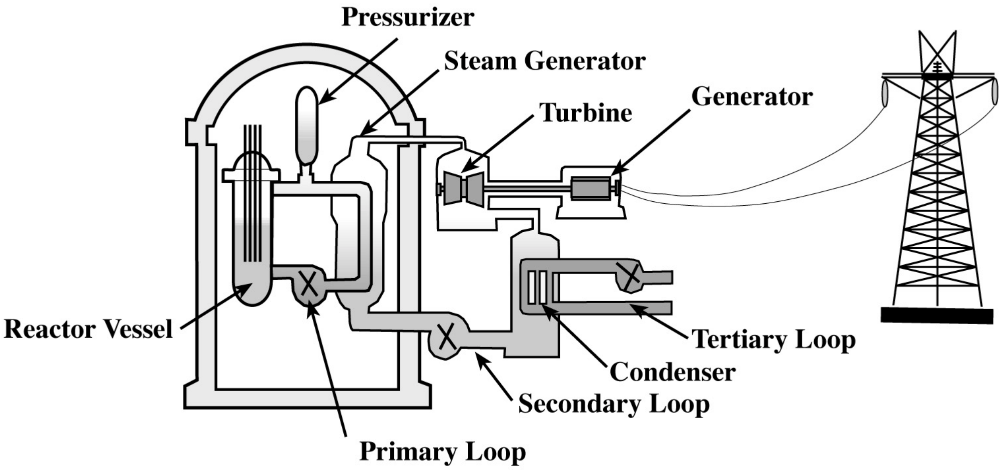


Figure 3-1. Pressurized Water Reactor

3.1.1.2 Boiling Water Reactors

The BWRs generate steam directly within the reactor vessel. The steam passes through moisture separators and steam dryers and then flows to the turbine. By generating steam directly in the reactor vessel, the power generation system contains only two heat transfer loops. The primary loop transports the steam from the reactor vessel directly to the turbine, which generates electricity. The secondary coolant loop removes excess heat from the primary loop in the condenser. From the condenser the primary condensate proceeds into the feedwater stage and the secondary coolant loop removes the excess heat to the environment (see Figure 3-2).

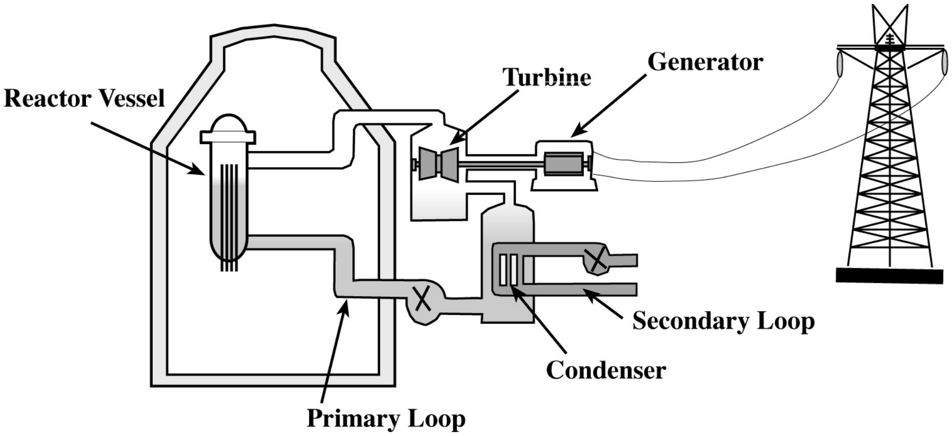


Figure 3-2. Boiling Water Reactor

3.1.1.3 Fast Breeder Reactors

In the FBR, such as Fermi, Unit 1, liquid sodium is used as the reactor coolant instead of water. The FBR also uses plutonium for fuel instead of the fissile isotope of uranium, as does an LWR. During the chain reaction, while some neutrons are fissioning plutonium atoms and releasing heat energy, others are captured by uranium atoms, which are then converted into more plutonium atoms. A fast breeder can produce 1.4 new plutonium atoms for every one fissioned–enough to refuel another reactor in 10 years. Fast breeders also generally have a higher power density in the core (thus, a smaller reactor) and better heat transfer characteristics, which improves power-plant efficiency. The Fermi, Unit 1, reactor also utilized a steam cycle to generate electricity, similar to a PWR. However, the Fermi, Unit 1, reactor had two sodium loops. Primary-loop liquid sodium was circulated through the reactor core, where it absorbed the heat generated by the reactor, and then through a heat exchanger, where its heat was transferred to the second (intermediate) sodium loop. The intermediate-loop liquid sodium was then circulated through a steam generator. The steam produced in the steam generators was then circulated to the turbine generators to produce electricity.(a)

(a) For more information, see http://pw1.netcom.com/~res95/energy/nuclear/breeder.htm (April 19, 2001).

At this time, there are no FBRs operating or under construction in the United States. Fermi, Unit 1, is currently in SAFSTOR. The environmental impacts described in this Supplement for FBRs are applicable to Fermi, Unit 1.

3.1.1.4 High-Temperature Gas-Cooled Reactors

Commercial HTGRs, operated in the United States at Peach Bottom, Unit 1, and Fort St. Vrain, use helium gas instead of water (as in LWRs) to transfer the heat from the reactor core to produce steam. In HTGRs, the entire primary coolant system, including the reactor, the steam generators, and the helium circulators, is housed within a prestressed concrete or steel reactor vessel. The helium circulators pump the pressurized coolant through the core, where it absorbs the heat from the fission process. The helium then enters the steam generators, which transfer the heat to the secondary system. The secondary system is a steam cycle similar to that found in any modern fossil-fuel facility. Superheated steam is produced in the steam generators and routed to the turbine generator, which generates the electricity (Fuller 1988).

At this time, there are no HTGRs operating or under construction in the United States. Decommissioning at Fort St. Vrain is complete and the license is terminated, and Peach Bottom, Unit 1, is currently in SAFSTOR. The environmental impacts described in this Supplement for HTGRs are applicable to Peach Bottom, Unit 1.

3.1.2 Types of Structures Located at a Nuclear Power Facility

As discussed in Chapter 1, the definition of decommissioning includes the reduction of residual radioactivity to a level that permits release of the property and termination of the license. As a result, the decontamination and/or dismantlement of those SSCs that are radioactive are by definition, included within the scope of this Supplement as part of decommissioning. If the structures must be decontaminated or parts of the structures removed to meet the requirements for the termination of the NRC license, those activities are also considered within scope as part of the decommissioning process. This includes removing nonradiological structures necessary to decontaminate another structure. Additionally, the impacts of dismantling all SSCs that were built or installed at the site to support power production are considered in this Supplement. This section discusses all the structures that will be referred to later in the document as background information for the reader.

Nuclear power plants generally contain similar facilities. They all contain a nuclear steam supply system, as described in Section 3.1.1 above. Additionally, there are a number of common SSCs necessary for plant operation. However, the layout of buildings and structures varies considerably among the sites. For example, control rooms may be located in the auxiliary building, in a separate control building, or in a radwaste and control building. Thus, the following list describes typical structures located on most sites.

- Containment or reactor building: The containment or reactor building in a PWR is a massive concrete or steel structure that houses the reactor vessel, reactor coolant piping and pumps, steam generators, pressurizer, pumps, and associated piping. The reactor building structure of a BWR generally includes a containment structure and a shield building. The containment is a massive concrete or steel structure that houses the reactor vessel, the reactor coolant piping and pumps, and the suppression pool. It is located inside a somewhat less substantive structure called the shield building. The shield building for a BWR also generally contains the spent fuel pool and the new fuel pool.

  The reactor building for both PWRs and BWRs is designed to withstand such disasters as hurricanes and earthquakes. The containment's ability to withstand such disasters and to contain the effects of accidents initiated by system failures are the principal protections against releasing radioactive material to the environment.

  The containment building for the FBR is a reinforced concrete structure that contains the upper end of the reactor vessel and the fuel-handling equipment.

  The HTGRs have two containment structures. Peach Bottom's inner containment structure is made of a steel pressure vessel and Fort St. Vrain's was made of prestressed concrete. This inner vessel houses the entire primary coolant system, the interconnecting ducts and plenums, the reactor core assembly, and the steam generator. The inner vessel is housed inside a second containment structure, which is designed to contain the entire primary coolant system helium under conditions postulated for the design-basis accident.

- Fuel building: For PWRs, the fuel building has a fuel pool that is used for the storage and servicing of spent fuel and the preparation of new fuel for insertion into the reactor. This building is connected to the reactor building by a transfer tube or channel that is used to move new fuel into the reactor and to move spent fuel out of the reactor for storage.

- Turbine building: The turbine building houses the turbine generators, condenser, feedwater heaters, condensate and feed water pumps, waste-heat rejection system, pumps, and equipment that supports those systems. Primary coolant is circulated through these systems in BWRs, thereby causing them to become slightly contaminated. However, primary coolant is not circulated through the turbine building systems in PWRs. The turbine building does not normally become contaminated during power generation at PWRs.

- Auxiliary buildings: Auxiliary buildings house such support systems as the ventilation system, the emergency core cooling system, the laundry facilities, water treatment system, and waste treatment system. The auxiliary building may also contain the emergency diesel generators and, in some PWRs, the fuel storage facility. Often, the facility's control room is also located in the auxiliary building.

- Diesel generator building: Often, there is a separate building for housing the emergency diesel generators if they are not located in the auxiliary building. The emergency diesel generators do not become contaminated or activated.

- Pumphouses: Various pumphouses may be present onsite for circulating water, standby service water, or makeup water. Pumphouses that carry clean water do not require radiological decommissioning.

- Cooling towers: Cooling towers are structures that are designed to remove excess heat from the condenser without dumping the heat directly into water bodies, such as lakes or rivers. There are two principal types of cooling towers: mechanical draft towers and natural draft towers. Most nuclear plants that have once-through cooling do not have cooling towers associated with them (see the descriptions in Section 3.1.3). However, five facilities with once-through cooling also have cooling towers.

- Radwaste facilities: If the radwaste facilities are not contained in the auxiliary building, they may be located in a separate solid radwaste building. An interim radwaste storage facility may also be used.

- Ventilation stack: Many older nuclear power plants, particularly BWRs, have ventilation stacks to discharge gaseous waste effluents and ventilation air. These stacks can be 90 m (300 ft) tall or more and contain monitoring systems to ensure that radioactive gaseous discharges are below fixed release limits. Radioactive gaseous effluents are treated and processed prior to discharge out the stack.

The following structures may also be part of the nuclear reactor facility but are not evaluated in this Supplement.

- Independent spent fuel storage installations (ISFSI): An ISFSI is designed and constructed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage. ISFSIs may be located at the site of a nuclear power plant or at another location. The most common design for an ISFSI, at this time, is a concrete pad with dry casks containing spent fuel bundles. ISFSIs are used by operating plants that require increased spent fuel storage capability because their spent fuel pools have reached capacity. Decommissioning facilities also use ISFSIs. The first dry-storage installation was licensed by the NRC in 1986. As of January 21, 2000, there were 14 nuclear power facilities licensed to use dry storage: Surry, Oconee, H.B. Robinson, Calvert Cliffs, Fort St.

Vrain, Palisades, Point Beach, Prairie Island, Davis-Besse, Susquehanna, Arkansas Nuclear One, North Anna, Trojan, and U.S. Department of Energy (DOE [TMI-2 fuel debris]).

An ISFSI can be constructed and operated and decommissioned either under the same license that is used for the operating or decommissioning facility called a general license under 10 CFR Part 50 or a specific license under 10 CFR Part 72 license. If a licensee chose to operate the ISFSI under a Part 50 license, it could, by way of a license-amendment request, change the ISFSI to a Part 72 license, thus allowing termination of the Part 50 license at the end of the decommissioning process. The NRC staff would also be required to conduct an environmental assessment of the licensee's proposal.

- <u>Switchyard</u>: A plant site also contains a large switchyard, where the electric voltage is stepped up and fed into the regional power distribution system. The switchyard is an integral part of the electric power transmission grid, and may remain on the site even after termination of the license.

- <u>Administrative, training, and security buildings</u>: Normally, the administrative, training, and security buildings are located outside the radiation protection zones, and no radiological hazards are present.

3.1.3 Description of Systems

After permanent cessation of operations and transfer of the fuel from the reactor vessel, licensees begin to shut down systems that are no longer operated in a decommissioning plant. However, specific systems will continue to be used during the different phases of the decommissioning process although in some cases in reduced roles. This section provides background information related to the systems, explains the differences between the systems' use during operations and during the decommissioning process, and explains how their continued operation could impact the environment during the decommissioning process. Lobner et al. (1990) provides more comprehensive descriptions of these systems in U.S. commercial LWRs.

- <u>Cooling and auxiliary water systems</u>: The predominant water use at an operating nuclear power plant is for removing excess heat generated in the reactor by the condenser cooling system. The quantity of water that is used for condenser cooling in an operating plant is a function of several factors, including the capacity rating of the plant and the increase in cooling water temperature from the discharge to the intake. The cooling water system for the reactor is not operated after the facility has permanently ceased power operations and the fuel has been removed from the reactor vessel. Therefore, water use is greatly reduced when operations cease. However, systems are not immediately drained upon cessation of operation and are frequently left in place for a period of time to provide shielding to the workers.

There are two major types of cooling systems for operating plants: once-through cooling and closed-cycle cooling.

In a once-through cooling system, circulating water for condenser cooling is obtained from an adjacent body of water, such as a lake or river, passed through the condenser tubes, and returned at a higher temperature to the adjacent body of water. Flow through the condenser for a 1000-MW plant during operations is typically 45 to 65 m^3/s (700,000 to 1,000,000 gpm) (NRC 1996). The waste heat is dissipated to the atmosphere mainly by evaporation from the water body and, to a much smaller extent, by conduction, convection, and thermal radiation loss.

In a closed-cycle system at an operating plant, the cooling water is recirculated through the condenser after the waste heat is removed by dissipation to the atmosphere, usually by circulating the water through large cooling towers constructed for that purpose. The average for makeup water withdrawals for a 1000-MW plant during operations is typically about 0.9 to 1.1 m^3/s (14,000 to 18,000 gpm). Recirculating cooling systems consist of either natural draft or mechanical draft cooling towers, cooling ponds, lakes, or canals. Because the predominant cooling mechanism associated with closed-cycle systems is evaporation, most of the water used for cooling is consumed and is not returned to the water source.

In addition to removing heat from the reactor of an operating facility, cooling water is also provided to the service water system and to the auxiliary water system. These systems account for 1 to 15 percent of the water needed for the condenser cooling. The auxiliary water systems include emergency core cooling systems, the containment spray and cooling system, the emergency feedwater system, the component cooling water system, and the spent fuel pool water systems. Most of these systems would not be needed following permanent cessation of operations. However, some, such as the systems for the spent fuel pool cooling, will be used after the plant has shut down.

- Waste systems (gaseous, liquid, solid, and nonradioactive): The gaseous waste management system in an operating nuclear facility collects fission products, mainly noble gases, that accumulate in the primary coolant. It is designed to reduce the radioactive material in gaseous waste before discharge to meet the dose design objectives in 10 CFR Part 50, Appendix I. During decommissioning, the gaseous waste management system is used during the decontamination and dismantlement of certain tanks or pipes. It is also used during dismantlement to assist in the control of radioactive dust or loose contamination. In addition, high-efficiency particulate air (HEPA) filters are used to remove radioactive material on a localized basis. For example, when removing concrete with a power hammer or drill in the containment building, a temporary plastic tent equipped with a HEPA filter prevents contaminated dust particles from entering the building. A second set of HEPA filters is located on the exhaust vent pathway for the building. The quantities of gaseous effluents released from operating plants and those in the decommissioning process are controlled by the administrative limits that are defined in the Offsite Dose Calculation Manual (ODCM), which is specific for each plant. The limits in the ODCM are designed to provide reasonable assurance that radioactive material discharged in gaseous effluents are not in excess of the limits specified in 10 CFR Part 20, Appendix B, thereby limiting the exposure of a member of the public in an unrestricted area.

The liquid radioactive waste system in operating nuclear power plants is used to collect and process liquid wastes collected from equipment leaks, valve and pump seal leaks, laundry wastes, personnel and equipment wastes, and steam generator blowdown (for PWRs), as well as building, laboratory, and floor drains. Each of these sources of liquid wastes receives varying degrees and types of treatment before storage, reuse, or discharge to the environment. During decommissioning, any radioactive liquids from operation of decommissioning activities in the facility will be processed and disposed of, thus necessitating the use of the liquid radioactive waste system. Some systems such as the laundry will likely still operate for a period of time, but others like the steam generator blowdown will not. Controls for limiting the release of radiological liquid effluents are described in the facility's ODCM. Controls are based on (1) concentrations of radioactive materials in liquid effluents and projected dose or (2) dose commitments to a member of the public. Concentrations of radioactive material that may be released in liquid effluents to unrestricted areas are limited to the concentration specified in 10 CFR Part 20, Appendix B, Table 2.

Solid low-level waste (LLW) from nuclear power plants is generated by removal of radionuclides from liquid waste streams, filtration of airborne gaseous emissions, and removal of contaminated material. The major source of solid LLW during decommissioning is the decommissioning process itself. Removal of contamination involves the use of protective clothing and cleaning rags. Dismantlement results in concrete or metal that has low levels of contamination or activation products. While the amount of liquid and gaseous radioactive waste generated is usually lower for decommissioning plants than for operating plants, the quantity of solid LLW being generated is significantly higher during decommissioning.

Solid waste is packaged in containers to meet the applicable requirements of 49 CFR Parts 171 through 177. Disposal and transportation are performed in accordance with the applicable requirements of 10 CFR Part 61 and 10 CFR Part 71, respectively.

Solid radioactive waste generated during either decommissioning or operations is usually shipped to a LLW processor or, in some cases, directly to a LLW disposal site. Volume reduction may occur both onsite and offsite. The most common onsite volume reduction techniques are high-pressure compacting in waste drums, dewatering and evaporating wet wastes, monitoring waste streams to segregate wastes, and sorting. Offsite waste management vendors compact wastes at ultra-high pressures, incinerate dry active waste, separate and incinerate oily and organic wastes, and asphalt-solidify resins and sludges before the waste is sent to the LLW site.

Nonradioactive wastes, including storm water system and sewage waste, are also generated during the decommissioning process. For example, use of hazardous oils or other chemicals in solvent cleaning and repair of equipment produces some nonradioactive wastes. Also, during decommissioning, additional quantities of nonradioactive waste (paint, asbestos) are generated or removed. Disposal of essentially all of the hazardous chemicals used at nuclear power plants is regulated by the Resource Conservation and Recovery Act (RCRA) of 1976 or by National Pollutant Discharge Elimination System (NPDES) permits, which are regulated by the U.S. Environmental Protection Agency and administered by the States to control the amount and types of pollutants that may be discharged from the plant.

Mixed waste is regulated under RCRA, the Atomic Energy Act, and NRC and is sent to a facility that is licensed to handle mixed waste.

- <u>Miscellaneous mechanical systems</u>: A variety of existing plant mechanical systems may continue to be used during plant decommissioning, including

    - the fire protection system

    - the heating, ventilation, and air conditioning (HVAC) system

    - the fuel-handling system

    - various cranes and hoists.

  The use of these systems generally does not have a direct impact on the environment. For example, the HVAC system that is used inside a contaminated area would be exhausted to the gaseous waste management system.

- <u>Instrumentation and control systems</u>: While most instrumentation and control systems in the plant can be deactivated after permanent shutdown and defueling of the reactor, a few may continue to be used to support decommissioning operations, including

  - the radiation monitoring system, which detects, measures, and records radiation levels during decommissioning operations and alerts plant staff of off-normal readings, and

  - the security system, which monitors the plant protected area to prevent uncontrolled access.

  In most cases, these systems are altered or reduced during the decommissioning process. The use of these systems during the decommissioning process does not impact the environment.

- Electrical systems: Numerous electrical systems may continue to be used during decommissioning operations. These include systems needed to provide uninterrupted power, lighting, and communication. In some cases, licensees have installed a new power distribution system, re-energizing only those loads that are necessary for continued use during decommissioning. In many facilities, the circuits that are being used are color-coded so that workers can easily identify the live circuits. Both of these practices are intended to prevent workers from cutting into a live wire during the decommissioning process.

- Spent fuel storage systems: Before beginning the decommissioning process, the licensee must certify to the NRC that it has permanently removed the fuel from the reactor vessel. The fuel is first moved into the spent fuel pool, which is a specially designed water-filled basin. Even after the nuclear reactor is shut down, the fuel continues to generate decay heat from the radioactive decay of fission products. The rate at which the decay heat is generated decreases the longer the reactor has been shut down. Therefore, the longer the time from last criticality, the less heat the spent fuel gives off. Storing the spent fuel in a pool of water provides an adequate heat sink for the removal of heat from the irradiated fuel. In addition, the fuel is located far enough under water that the radiation emanating from the fuel is shielded by the water, thus protecting workers from the radiation. After the fuel has cooled adequately, it can be stored in an ISFSI in air-cooled dry casks. Typically, transfer of spent fuel to an ISFSI occurs after the fuel has cooled for 5 years.

  After removal of the fuel to the spent fuel pool, it is common for the licensee to reduce the security area at the facility to a "nuclear island" that focuses primarily on the storage area for the spent fuel. This allows the spent fuel to be protected and the security system to cover only the storage location for the spent fuel.

  At this time, there are no facilities for permanent disposal of high-level radioactive wastes (HLW). The Nuclear Waste Policy Act of 1982 defined the goals and structure of a program for permanent, deep geologic repositories for high-level radioactive waste and unreprocessed spent fuel. Under this Act, the DOE is responsible for developing permanent disposal capacity for the spent fuel and other high-level nuclear wastes. At the present time, DOE, as directed by Congress, is investigating a site in Yucca Mountain, Nevada, for a possible disposal facility. A high-level waste repository would be built and operated by DOE and licensed by the NRC.

The Commission believes (10 CFR 51.23(a)) there is reasonable assurance that at least one mined geological repository will be available in the first quarter of the 21st Century and that, within 30 years beyond the licensed life of operation for any reactor, sufficient repository capacity will be available to dispose of the reactor's high-level waste and spent fuel generated up to that time.

Until a high-level waste repository is available or some interim central waste storage facility is approved and licensed, licensees generally store the fuel onsite either in dry storage (ISFSI) or in wet storage in a spent fuel pool. Licensees are prohibited from shipping spent fuel from one reactor spent fuel pool to another without NRC approval by license amendment.

The Commission has independently, in a separate proceeding (the Waste Confidence Proceeding), made a finding that there is

> reasonable assurance that, if necessary, spent fuel generated in any reactor can be stored safely and without significant environmental impacts for at least 30 years beyond the licensed life for operation (which may include the term of a revised license) of that reactor at its spent fuel storage basin, or at either onsite or offsite independent spent fuel storage installations (54 FR 39767).

The Commission has committed to review this finding at least every 10 years. In its most recent review, the Commission concluded that experience and developments since 1990 were not such that a comprehensive review of the Waste Confidence Decision was necessary at this time (64 FR 68005). Accordingly, the Commission reaffirmed its findings of insignificant environmental impacts cited above. This finding is codified in the Commission's regulations at 10 CFR 51.23(a). The staff relies on the Waste Confidence Rule, but for completeness has elected to include in this Supplement information related to the storage and maintenance of fuel in a spent fuel pool.

- <u>Transportation systems</u>: There are four broad classes of shipments to and from operating nuclear power plants: (1) routinely generated LLW transported from plants to disposal facilities, (2) routine LLW shipped to offsite facilities for volume reduction, (3) nuclear fuel shipments from fuel-fabrication facilities to plants for loading into reactors, and (4) spent fuel shipments to other nuclear power plants with available storage space (an infrequent occurrence that is usually limited to plants owned by the same utility).

The transportation of radioactive materials is regulated jointly at the Federal level by the U.S. Department of Transportation (DOT) and the NRC. The responsibilities of the two agencies are delineated in a Memorandum of Understanding (see 44 FR 38690). Most LLW is shipped in packages authorized by the DOT. Some packages for larger quantities of LLW require NRC certification. The LLW packages can be loaded onto trucks or trains for shipment to the LLW disposal site. In general, the areas regulated by the agencies are as follows:

- DOT – Regulates shippers and carriers of radioactive material and the conditions of transport, including routing, tiedowns, radiological controls, vehicle requirements, hazard communication, handling, storage, emergency response information, and employee training. DOT regulations are located in the Code of Federal Regulations, Title 49, "Transportation."

- NRC – Regulates users of radioactive material and the design, construction, use, and maintenance of shipping containers used for larger quantities of radioactive material and fissile material such as uranium. NRC regulations are located in 10 CFR Part 71, "Packaging and Transportation of Radioactive Material."

Title 10 CFR 71.47 states that under normal transportation conditions, each package of radioactive materials must be designed and prepared for shipment such that the radiation level does not exceed 2 mSv/h (200 mrem/h) at any point on the external surface of the package and 0.1 mSv/h (10 mrem/h) at any point 1 m (3.3 ft) from the packaging surface. This type of shipment is called a nonexclusive use shipment. If the package exceeds the limits specified for nonexclusive use shipments, it must be transported by exclusive use shipment only. The radiation limits for exclusive use packages are the following:

- At any point on the package surface: 2 mSv/h (200 mrem/h). For closed transport vehicle only: 10 mSv/h (1000 mrem/h)

- At 2 m (6.6 ft) from lateral surfaces of vehicle: 0.1 mSv/h (10 mrem/h)

- At all external surfaces of the vehicle: 2 mSv/h (200 mrem/h)

- In the occupied area of the vehicle: 0.02 mSv/h (2 mrem/h), with certain exceptions.

For more information regarding waste packaging and radioactive transportation regulations, see 10 CFR Part 71.

The frequency of waste shipments increases sharply during the decommissioning period. In some cases, such as the shipment of large components (e.g., steam generators, reactor vessels, or pressurizers), the waste packaging is unique compared to most shipments during operations. However, the licensee is still required to meet the regulations discussed above, unless the NRC approves an exemption after a thorough analysis of the licensee's proposal.

3.1.4 Formation and Location of Radioactive Contamination and Activation in an Operating Plant

During reactor operation, a large inventory of radioactive fission products builds up within the fuel. Virtually all of the fission products are contained within the fuel pellets. The fuel pellets are enclosed in hollow metal rods, which are hermetically sealed to prevent further release of fission products. Occasionally fuel rods develop small leaks allowing a small fraction of the fission products to contaminate the reactor coolant. The radioactive contamination in the reactor coolant is the source of gaseous, liquid, and solid radioactive wastes generated at LWRs during operation.

There are two sources of radioactive material: contamination and activation. Contaminated materials are unintentionally transported through the facility by workers, equipment, and, to some degree, air movement. Although many precautions are taken to prevent the movement of contaminated material in a nuclear facility and to clean up any contaminated materials that may be found, it is likely that contamination will occur in the reactor building, around the spent fuel pool, and around specific SSCs in the auxiliary building and other buildings and equipment in the area near the reactor. The areas known to contain contamination are labeled by the licensee, who routinely checks for contamination and removes as much as possible during operations. Radioactive contamination may be deposited from the air or dissolved in water and subsequently deposited onto material such as concrete. Radioactive contamination is generally located on or near the surface of materials such as metals, high-density concrete, or painted walls. It can travel farther into unpainted surfaces or lower-density concrete. Radioactive contamination can usually be removed from surface areas by washing, scrubbing, spraying, or, in extreme cases, by physically removing the outer layers of the surface material.

Activation products are also formed during reactor operation. Activation products are radioactive materials created when stable substances are bombarded by neutrons. Concrete and steel surrounding the core of the reactor are the most common types of activated products. Activation products cannot be removed by the processes used to remove contamination. Activation products are incorporated into the molecular structure of the material and cannot be wiped off or removed. The entire structure must be removed and treated as radioactive waste. Activated metal and concrete contain the single largest inventory of radionuclides with the exception of the spent fuel, in facilities that are being decommissioned. The radioactive decay of activation products is the main source of radiation exposure to plant personnel.

The spent fuel contains the largest amount of radioactive material at a permanently shutdown facility followed by the reactor vessel, internals, and bioshield. Systems containing smaller amounts of radioactive material include the steam generator, pressurizer, piping of the primary system and other systems, piping, as well as the radwaste systems. Minor contamination is found in the secondary systems and miscellaneous piping.

3.2 Decommissioning Options

This Supplement evaluates the environmental impacts of three decommissioning options or combinations of the options. These options, first identified in the 1988 Generic Environmental Impact Statement (GEIS) using the acronyms DECON, SAFSTOR, and ENTOMB, are defined as follows:

> DECON: The equipment, structures, and portions of the facility and site that contain radioactive contaminants are promptly removed or decontaminated to a level that permits termination of the license shortly after cessation of operations.
>
> SAFSTOR: The facility is placed in a safe, stable condition and maintained in that state (safe storage) until it is subsequently decontaminated and dismantled to levels that permit license termination. During SAFSTOR, a facility is left intact, but the fuel has been removed from the reactor vessel, and radioactive liquids have been drained from systems and components and then processed. Radioactive decay occurs during the SAFSTOR period, thus reducing the quantity of contaminated and radioactive material that must be disposed of during decontamination and dismantlement.
>
> ENTOMB: Radioactive SSCs are encased in a structurally long-lived substance, such as concrete. The entombed structure is appropriately maintained, and continued surveillance is carried out until the radioactivity decays to a level that permits termination of the license.

The choice of decommissioning option is left entirely to the licensee, provided that it can be performed according to the NRC's regulations. This choice is communicated to the NRC and the public in the post-shutdown decommissioning activities report. In addition, the licensee may choose to combine the DECON and SAFSTOR options. For example, after power operations cease at a facility, a licensee could use a short storage period for planning purposes, followed by removal of large components (such as the steam generators, pressurizer, and reactor vessel internals), place the facility in storage for 30 years, and eventually finish the decontamination and dismantlement process.

Although the selection of the decommissioning option is up to the licensee, the NRC requires the licensee to re-evaluate its selection if the option (1) could not be completed as described,

(2) could not be completed within 60 years of the permanent cessation of plant operations, (3) included activities that would endanger the health and safety of the public by being outside of the NRC's health and safety regulations, or (4) would result in a significant impact to the environment.

To date, most utilities have used DECON or SAFSTOR to decommission reactors. Several sites have performed some incremental decontamination and dismantlement during the storage period of SAFSTOR, a combination of SAFSTOR and DECON. A site using DECON may have a short period of time (1 to 4 years) when the facility is in SAFSTOR. Several licensees continue to conduct limited decommissioning activities during a SAFSTOR period as personnel, money, or other factors become available. This process of occasionally conducting active decontamination and dismantlement is referred to as incremental DECON. No utilities have used the ENTOMB option for a commercial nuclear power reactor.

The following sections provide a general overview of each decommissioning option.

3.2.1 DECON

The DECON decommissioning option involves removing or decontaminating equipment, structures, and portions of the facility and site that contain radioactive contaminants to a level that permits termination of the license, as defined in Regulatory Guide 1.184 (NRC 2000a).

There are several advantages to using the DECON option of decommissioning. One is that the facility license is quickly terminated so that the facility and site become available for other purposes. By beginning the decontamination and dismantlement process soon after permanent cessation of operation, the available work force can be maintained and is highly knowledgeable about the facility. The availability of facilities willing to accept LLW may also be a factor in the licensee's decision to pursue the DECON option. Currently, the estimated cost of decommissioning a site using DECON is less than SAFSTOR due primarily to price escalation in the disposal of LLW. Because most activities that occur during DECON also occur during SAFSTOR, the price for decommissioning at a later date is greater because of the cost of storage and inflation (NRC 2000c). DECON also eliminates the need for long-term security, maintenance, and surveillance of the facility, which is required for the other decommissioning options.

The major disadvantages of DECON are the higher worker dose and significant initial expenditures. Also, compared to SAFSTOR, DECON requires a larger potential commitment of disposal site space (NRC 2000c).

The general activities that may occur during DECON are listed below (NRC 2000d):

- draining (and potentially flushing) of some contaminated systems and removal of resins from ion exchangers
- setup activities such as establishing monitoring stations or designing and fabricating special shielding and contamination-control envelopes to facilitate decommissioning activities
- reduction of site-security area (setup of new security monitoring stations)
- modification of the control room or establishing an alternate control room
- site surveys
- decontamination of radioactive components, including use of chemical decontamination techniques
- removal of reactor vessel and internals
- removal of other large components, including major radioactive components
- removal of the balance of the primary system (charging system, boron control system, etc.)
- general activities related to removing other significant radioactive components
- decontamination and/or dismantlement of structures or buildings
- temporary onsite storage of components
- shipment and processing of LLW, including compaction or incineration of the waste
- removal of the spent fuel and greater than Class C (GTCC) waste to an ISFSI
- removal of hazardous radioactive (mixed) wastes
- changes in management and staffing.

3.2.2 SAFSTOR

The SAFSTOR decommissioning option involves placing the facility in a safe, stable condition and maintaining that state for a period of time, followed by subsequent decontamination and dismantlement to levels that permit license termination. During the storage period of

SAFSTOR, the facility is left intact. The fuel has been removed from the reactor vessel and radioactive liquids have been drained from systems and components and processed. Radioactive decay occurs during the storage period, reducing the quantity of contaminated and radioactive material that must be disposed of during decontamination and dismantlement.

There are several advantages to using the SAFSTOR option of decommissioning. A substantial reduction in radioactive material as a result of radioactive decay during the storage period reduces worker and public doses below those of the DECON alternative. Since there is potentially less radioactive waste, less waste-disposal space is required. Moreover, the costs immediately following permanent cessation of operations are lower than costs during the first years of DECON because of reduced amounts of activity and a smaller work force (NRC 2000c).

However, because of the time gap between cessation of operations and decommissioning activities, SAFSTOR can result in a shortage of personnel familiar with the facility at the time of dismantlement and decontamination. During the prolonged period of storage, the plant requires continued maintenance, security, and surveillance. Also, uncertainties regarding the availability and cost of LLW sites in the future could mean higher costs for decontamination and dismantlement (NRC 2000c).

Activities that typically occur during the preparation and storage stages of the SAFSTOR process are described below (NRC 2000d).

<u>During preparation</u>:

- draining (and potential flushing) of some systems and removal of resins from ion exchangers

- spent fuel pool cooling systems reconfiguration

- decontamination of highly contaminated and high dose areas as necessary

- performance of a radiological assessment as a baseline before storage

- removal of LLW that is ready to be shipped

- shipment and processing or storage of the fuel and GTCC waste

- de-energizing or deactivating systems and equipment

- reconfiguration of ventilation systems, fire protection systems, and spent fuel pool cooling system for use during storage

- establishment of inspection and monitoring plans for use during storage

- maintenance of any systems critical to final dismantlement during storage

- changes in management and staffing.

During storage:

- performance of preventative and corrective maintenance on plant systems that will be operating and/or functional during storage

- maintenance to preserve structural integrity

- maintenance of security systems

- maintenance of radiation effluent and environmental monitoring programs

- processing of any radwaste generated (usually small amounts).

Following the storage period, the facility is decontaminated and dismantled to radiological levels that allow termination of the license. Activities during this period of time will be the same activities that occur for DECON.

3.2.3 ENTOMB

The ENTOMB decommissioning method was defined in the Supplementary Information to the 1988 Decommissioning Rule (53 FR 24018) as the option in which radioactive contaminants are encased in a structurally long-lived material, such as concrete. The entombed structure is appropriately maintained and surveillance is continued until the radioactivity decays to a level permitting unrestricted release of the property (NRC 1988).

Currently, 10 CFR 50.82 (a)(3) requires that decommissioning be completed within 60 years of permanent cessation of operations, and completion of decommissioning beyond 60 years be approved by the NRC only when necessary to protect public health and safety. The factors that could be considered by the Commission in evaluating an option that provides for the completion of decommissioning beyond 60 years of permanent cessation of operation include unavailability of waste disposal capacity and site-specific factors affecting the licensee's capability to carry out decommissioning, including the presence of other nuclear facilities at the site.

The current regulations, pertaining to the decommissioning of nuclear reactors promulgated in 1988, are also structured to favor decommissioning options that result in unrestricted release of the site. As noted in the supplementary information for the June 27, 1988, final rule, the ENTOMB option was not specifically precluded because it was recognized that it might be an allowable option for protecting public health and safety.

The 1997 Rule for Radiological Criteria for License Termination (64 FR 39058) established criteria (10 CFR Part 20, Subpart E) that allow for both restricted and unrestricted release of property. Under a restricted release, the dose to the average member of the critical group must not exceed 0.25 mSv/yr (25 mrem/yr) total effective dose equivalent (TEDE) and must be as low as reasonably achievable (ALARA) with the restrictions in place. If the restrictions were no longer in effect, the dose due to residual radioactivity could not exceed 1 mSv/yr (100 mrem/yr) (or 5 mSv/yr [500 rem/yr], if additional conditions are met) TEDE and must be ALARA. These caps were chosen to provide a safety net in the highly unlikely event that the restrictions failed.

In the Staff Requirements Memorandum on the ENTOMB option, dated July 20, 2000 (NRC 200b), the Commission directed that

> [T]he staff closely coordinate this rulemaking effort for this rulemaking with the ongoing efforts to update the generic environmental impact statement for the decommissioning of power reactors. The staff should include the entombment option in the GEIS recognizing that not all entombment proposals can be forecast but that the GEIS would provide a bounding analysis. The staff should also address the issue of entombing Greater Than Class C waste for this category of waste.

On September 18, 2001, the Commission approved the staff's rulemaking plan (see Section 2.2.2) for potential development of a rule to allow entombment as a decommissioning option for power reactors. On October 16, 2001, the Commission issued an advance notice of proposed rulemaking (ANPR) on Entombment Options for Power Reactors (66 FR 32551) to invite early input from interested stakeholders on issues related to entombment of power reactors. The ANPR identifies a number of rulemaking options related to entombment. Based on comments received from stakeholders the staff may propose changes to the regulations. Any rulemaking effort on the part of the NRC staff will require an environmental assessment (10 CFR 51.21).

The assessment of impacts associated with the ENTOMB option presented in this GEIS is independent of a prospective rulemaking before the Commission. The staff is making the assumption that environmental issues arising from any rulemaking effort will be addressed in the rulemaking and its supporting environmental documentation. These issues may include

(1) the long-term onsite retention of radioactive materials, including those that may be classified as GTCC, (2) issues related to long-term NRC oversight and monitoring requirements, (3) durability of institutional controls and site-engineered barriers, and (4) site-specific requirements.

The purpose of the entombment process is to isolate the entombed radioactive waste so that the reactor facility can be released and the license terminated. Therefore, prior to entombment, (1) an accurate characterization of the radioactive materials that are to remain is needed, and (2) the adequacy of the entombment configuration to isolate the entombed radioactive waste must be determined. Because of the requirement in the regulation to complete decommissioning within 60 years, no licensee has proposed the use of ENTOMB as the preferred decommissioning option for any of the nuclear power reactors currently undergoing decommissioning. The staff can envision a large number of entombment scenarios arranged along a continuum, differing primarily on the amount of decontamination and dismantlement done prior to the actual entombment.

The staff evaluated the impacts associated with the entombment options by developing two scenarios that have been designated ENTOMB1 and ENTOMB2. These two scenarios were developed specifically to envelope a wide range of potential options by describing two possible extreme cases of entombment. ENTOMB1 assumes significant decontamination and dismantlement and removal of all contamination and activation involving long-lived radioactive isotopes prior to entombment. ENTOMB2 assumes significantly less decontamination and dismantlement, significantly more engineered barriers, and the retention onsite of long-lived radioactive isotopes. Both options assume that the spent fuel would be removed from the facility and either transported to a permanent HLW repository or placed in an onsite interim spent fuel storage installation (ISFSI).

ENTOMB1 is envisioned by the staff to begin the decommissioning process in a manner similar to the DECON option. The reactor would be defueled and the fuel initially placed into the spent fuel pool for some period prior to disposal at a licensed HLW repository or placed in an onsite ISFSI. Any decommissioning activity would be preceded by an accurate radiological characterization of SSCs throughout the facility. Active decommissioning would begin with draining and decontamination of SSCs throughout the facility with the goal of isolating and fixing contamination. SSCs would either be decontaminated or removed and either shipped to a LLW burial site or placed inside the reactor containment building. Offsite disposal of resins and considerable amounts of contaminated material would occur. There would likely be a chemical decontamination of the primary system. The reactor pressure vessel (RPV) and reactor internals would be removed, either intact or after sectioning, and disposed of offsite. Any other SSCs that have long-lived activation products would be removed. Interim dry storage of the vessel, vessel internals, and any other SSCs containing long-lived activation products could occur onsite until a final disposal site for this waste (predominately GTCC waste) is identified. Steam generators and the pressurizer, depending on whether or not the components are contaminated with long-lived radioisotopes, would either be removed and disposed of offsite or retained inside the reactor containment. The spent fuel pool would be drained and decontaminated. The reactor building or containment would then be filled with SSCs contaminated with relatively short-lived isotopes from the balance of the facility. Material would be placed in the building in a manner that would minimize the spread of any contamination (i.e., dry, contamination fixed, isolated). Engineered barriers would be put in place to deny access and eliminate the possibility of the release of any contamination to the environment. The reactor building or containment would be sealed and made weather tight.

The license termination monitoring program would be submitted and the site would be characterized. A partial site release would be completed for almost all of the site and the balance of the plant. The staff makes no assumptions as to when the license would be terminated and whether it would be terminated under the restricted or unrestricted provisions of 10 CFR Part 20, Subpart E. These decisions would likely be addressed as part of the staff's rulemaking effort related to entombment explained above. The staff does assume that there would a monitoring program period as long as 20 to 30 years to demonstrate that there was isolation of the contamination and adequate permanence of the structure.

The general activities that would occur during ENTOMB1 are listed below:

- planning and preparation activities

- draining (and potentially flushing) of contaminated systems and removal of resins from ion exchangers

- reduction of site-security area (optional)

- deactivation of support systems

- decontamination of radioactive components, including use of chemical decontamination techniques

- removal of the reactor vessel and internals

- removal of other large components, including major radioactive components

- removal of fuel from the spent fuel pool to an ISFSI

- dismantlement of remaining radioactively contaminated structures and placement of the dismantled structures in the reactor building

- installation of engineered barriers and other controls to prevent inadvertent intrusion and dispersion of contamination outside of the entombed structure

- filling of the void spaces in the previous reactor building structure with grout (concrete).

ENTOMB2 is also envisioned by the staff to begin the decommissioning process in a manner similar to the DECON option. The reactor would be defueled and the fuel initially placed into the spent fuel pool for some period prior to disposal at a licensed HLW repository or placed in an onsite ISFSI. Any decommissioning activity would be preceded by an accurate radiological characterization of SSCs throughout the facility. Active decommissioning would begin with the draining and decontamination of SSCs throughout the facility with the goal of isolating and fixing contamination. The spent fuel pool would be drained and decontaminated. SSCs would either be decontaminated or removed and either shipped to a LLW burial site or placed inside the reactor containment building (PWR) or the reactor building (BWR). Disposal offsite of resins would occur. The primary system would be drained the RPV filled with contaminated material, all penetrations sealed, the RPV head reinstalled, and the reactor vessel filled with low-density concrete. Reactor internals would remain in place. Emphasis would be placed on draining and drying all systems and components and fixing contamination to prevent movement either by air or liquid means. The steam generators and pressurizer would be laid up dry and remain in place. The reactor building or containment would then be filled with contaminated SSCs from the balance of the facility. Material would be placed in the building in a manner that would minimize the spread of any contamination (i.e. dry, contamination fixed, isolated).

Engineered barriers would be put in place to deny access and eliminate the possibility of the release of any contamination to the environment. The ceiling of the containment or reactor building, in the case of BWRs, would be lowered to near the refueling floor and to the top of the pressurizer for PWRs. The cavity of the remaining structure would be filled with a low-density concrete grout. The resulting structure would be sealed and made weather tight and covered with a engineered cap designed to deny access, and prevent the intrusion of water or the release of radioactive contamination to the environment.

The license termination monitoring program would be submitted and the site would be characterized. A partial site release would be completed for almost all of the site and the balance of the plant. The license would be likely terminated under the restricted release provisions of 10 CFR Part 20, Subpart E, after a site-monitoring program that demonstrates the isolation of the contamination and the permanence of the structure. Monitoring could be as long as 100 years.

The general activities that would occur during ENTOMB2 are listed below:

- planning and preparation activities

- draining (and potentially flushing) of contaminated systems and removal of resins from ion exchangers
- deactivation of support systems
- removal of fuel from the spent fuel pool to an ISFSI
- dismantlement of all radioactively contaminated structures (other than the reactor building) and placement of the dismantled structures in the reactor building
- lowering of the ceiling of the reactor building to near the refueling floor (in BWRs) or near the top of the pressurizer (in PWRs)
- installation of engineered barriers and other controls to prevent inadvertent intrusion and dispersion of contamination outside of the entombed structure
- filling of the cavity of the reactor building structure with low-density grout (concrete)
- placement of an engineered cap over the entombed structure to further isolate the structure from the environment.

The advantages of both ENTOMB options are reduced public exposure to radiation due to significantly less transportation of radioactive waste to an LLW disposal site and corresponding reduced cost of LLW disposal. An additional advantage of ENTOMB2 is related to the significant reduction in the amount of work activity, and thus a significant reduction in occupational exposures, as compared to the DECON or SAFSTOR decommissioning options.

3.3 Summary of Plants That Have Permanently Ceased Operations

Twenty-two of the commercial nuclear reactors licensed by the NRC have permanently shut down and have had their licenses terminated or are currently being decommissioned. This section presents the significant characteristics of these plants, the decommissioning options being used by each plant, and each plant's decommissioning activities.

3.3.1 Plant Sites

An overview of the shutdown plants can be found in Table 3-1, which includes 22 units shut down between 1963 and 1997. Table 3-2 summarizes important characteristics of the shutdown plants. The thermal power capabilities of the reactors ranged from 23 to 3411 MW(t). The reactors operated from just a few days (Shoreham) to 33 years (Big Rock Point). Since 1987, an average of one plant per year has been shut down.

Table 3-1. Summary of Shutdown Plant Information

Types and Number of Shutdown Reactors	
BWR	8
PWR	11
HTGR	2
FBR	1
Decommissioning Option	
SAFSTOR	14
DECON	7
Accident cleanup followed by storage	1
Fuel Location	
Fuel onsite in pool	13
No fuel onsite[(a)]	8
Fuel onsite in ISFSI	1
Plan to move fuel to an ISFSI between 2000 and 2005	9

(a) Includes Three Mile Island, Unit 2, which has approximately 900 kg of fuel remaining onsite due to the accident.

Three of the 22 plants (Fort St. Vrain, Shoreham, and Pathfinder) have completed decommissioning and have had their 10 CFR Part 50 licenses terminated. Two of these three (Fort St. Vrain and Shoreham) used the DECON process for decommissioning. One facility, Shoreham, operated less than three full power days before being shut down and decommissioned so there was relatively little contamination. Another facility, Pathfinder, was placed in SAFSTOR and subsequently decommissioned. Eleven of the plants shut down prematurely. Three Mile Island, Unit 2, ceased power operations as a result of a severe accident. Three Mile Island, Unit 2, has been placed in a monitored storage mode until Unit 1 permanently ceases operation, at which time both units are to be decommissioned.

Ten of the permanently shutdown plants were part of the U.S. Atomic Energy Commission's (AEC’s) Demonstrations Program, including Big Rock Point; Dresden, Unit 1; Fermi, Unit 1; GE-VBWR; Humboldt Bay, Unit 3; Indian Point, Unit 1; La Crosse; Pathfinder; Peach Bottom, Unit 1; and, Saxton. These plants were prototype designs that were jointly funded by the AEC and commercial utilities. One of the plants, Pathfinder, has completed decommissioning and had its license terminated.

The most recent of the Demonstration Program reactors to shut down was Big Rock Point, which operated for 33 years and permanently shut down in 1997.

Table 3-2. Permanently Shutdown Plants

Nuclear Plant	Reactor Type	Thermal Power	Shutdown Date [(a)]	Decommissioning Option[(b)]	Location	Fuel Status and License Termination Date
Plants Currently in Decommissioning Process						
Big Rock Point	BWR	240 MW	08/30/97	DECON	Michigan	Fuel in pool
Dresden, Unit 1	BWR	700 MW	10/31/78	SAFSTOR	Illinois	Fuel in pool
Fermi, Unit 1	FBR	200 MW	09/22/72	SAFSTOR	Michigan	No fuel onsite
GE-VBWR	BWR	50 MW	12/09/63	SAFSTOR	California	No fuel onsite
Haddam Neck	PWR	1825 MW	07/22/96	DECON	Connecticut	Fuel in pool
Humboldt Bay, Unit 3	BWR	200 MW	07/02/76	SAFSTOR[(c)]	California	Fuel in pool
Indian Point, Unit 1	PWR	615 MW	10/31/74	SAFSTOR	New York	Fuel in pool
La Crosse	BWR	165 MW	04/30/87	SAFSTOR	Wisconsin	Fuel in pool
Maine Yankee	PWR	2700 MW	12/06/96	DECON	Maine	Fuel in pool
Millstone, Unit 1	BWR	2011 MW	11/04/95	SAFSTOR	Connecticut	Fuel in pool
Peach Bottom, Unit 1	HTGR	115 MW	10/31/74	SAFSTOR	Pennsylvania	No fuel onsite
Rancho Seco	PWR	2772 MW	06/07/89	SAFSTOR[(c)]	California	Fuel in pool/Partial DECON proposed in 1997
San Onofre, Unit 1	PWR	1347 MW	11/30/92	SAFSTOR[(c)]	California	Fuel in pool
Saxton	PWR	28 MW	05/01/72	SAFSTOR[(c)]	Pennsylvania	No fuel onsite/Currently in DECON
Three Mile Island, Unit 2	PWR	2772 MW	03/28/79	Accident cleanup followed by storage	Pennsylvania	Approx 900 kg fuel onsite/ Post-defueling monitored storage
Trojan	PWR	3411 MW	11/09/92	DECON	Oregon	Fuel in pool
Yankee Rowe	PWR	600 MW	10/01/91	DECON	Massachusetts	Fuel in pool
Zion, Unit 1	PWR	3250 MW	02/21/97	SAFSTOR	Illinois	Fuel in pool
Zion, Unit 2	PWR	3250 MW	09/19/96	SAFSTOR	Illinois	Fuel in pool
Terminated Licenses						
Fort St. Vrain	HTGR	842 MW	08/18/89	DECON	Colorado	Fuel ISFSI/License terminated in 1997
Pathfinder	BWR	190 MW	09/16/67	SAFSTOR	South Dakota	No fuel onsite/License terminated in 1992
Shoreham	BWR	2436 MW	06/28/89	DECON	New York	No fuel onsite/License terminated in 1995

(a) The shutdown date corresponds to the date of the last criticality.
(b) The option shown in the table for each plant is the option that has been officially provided to NRC. Plants in DECON may have had a short (1 to 4 yr) SAFSTOR period. Likewise, plants in SAFSTOR may have performed some DECON activities or may have transitioned from the storage phase into the decontamination and dismantlement phase of SAFSTOR.
(c) These plants have recently performed or are currently performing the decontamination and dismantlement phase of SAFSTOR.

Eight of the decommissioned or decommissioning plants are located in the northeast (or mid-Atlantic states), six in the west, six in the midwest, and one in the east. The majority of the shutdown plants (13) are situated on freshwater or impoundments, five others are in coastal or estuarine environments, and three others are on the Great Lakes.

3.3.2 Description of Decommissioning Options Selected

Seven decommissioned units are located on multi-unit sites in which the remaining units continue to operate and one multi-unit site shutdown both units permanently. All eight of these licensees chose SAFSTOR as the decommissioning option. In most cases, SAFSTOR was chosen so that all units on a site could be decommissioned simultaneously. For various reasons, however, most shutdown units have done some decontamination and dismantlement.

The reasons cited by licensees for choosing DECON have included the availability of LLW capacity, availability of staff familiar with the plant, available funding, the licensee's intent to use the land for other purposes, influence by State or local government to complete decommissioning, or a combination of other reasons.

A number of the plants have combined the DECON and SAFSTOR process by either entering shorter SAFSTOR periods or by doing an incremental DECON, allowing the plant to use resources and "decommission as they go." Sites have combined the options, usually to achieve economic advantages. For example, one site decided to shorten the SAFSTOR period and begin incremental dismantlement out of concern over future availability of a waste site and future costs of disposal. One site that prematurely shut down had a short SAFSTOR period to allow short-lived radioactive materials to decay and to conduct more detailed planning. Safety is another reason for combining the two options. Because of seismic safety concerns, one site undertook a major dismantling project to remove a 76-m (250-ft) concrete vent stack after it had been in SAFSTOR for 10 years.

The licensee determines the physical condition of the site after the decommissioning process. Some licensees intend to restore the site to “greenfield” status at the end of decommissioning, while others may install a non-nuclear facility. The NRC’s regulatory authority is only over that portion of the facility that is contaminated. Some licensees will leave structures standing at the time of license termination, and others will not. While undergoing the decommissioning process, some licensees have opted for partial site release to decrease the size of the site area.

3.3.3 Decommissioning Process

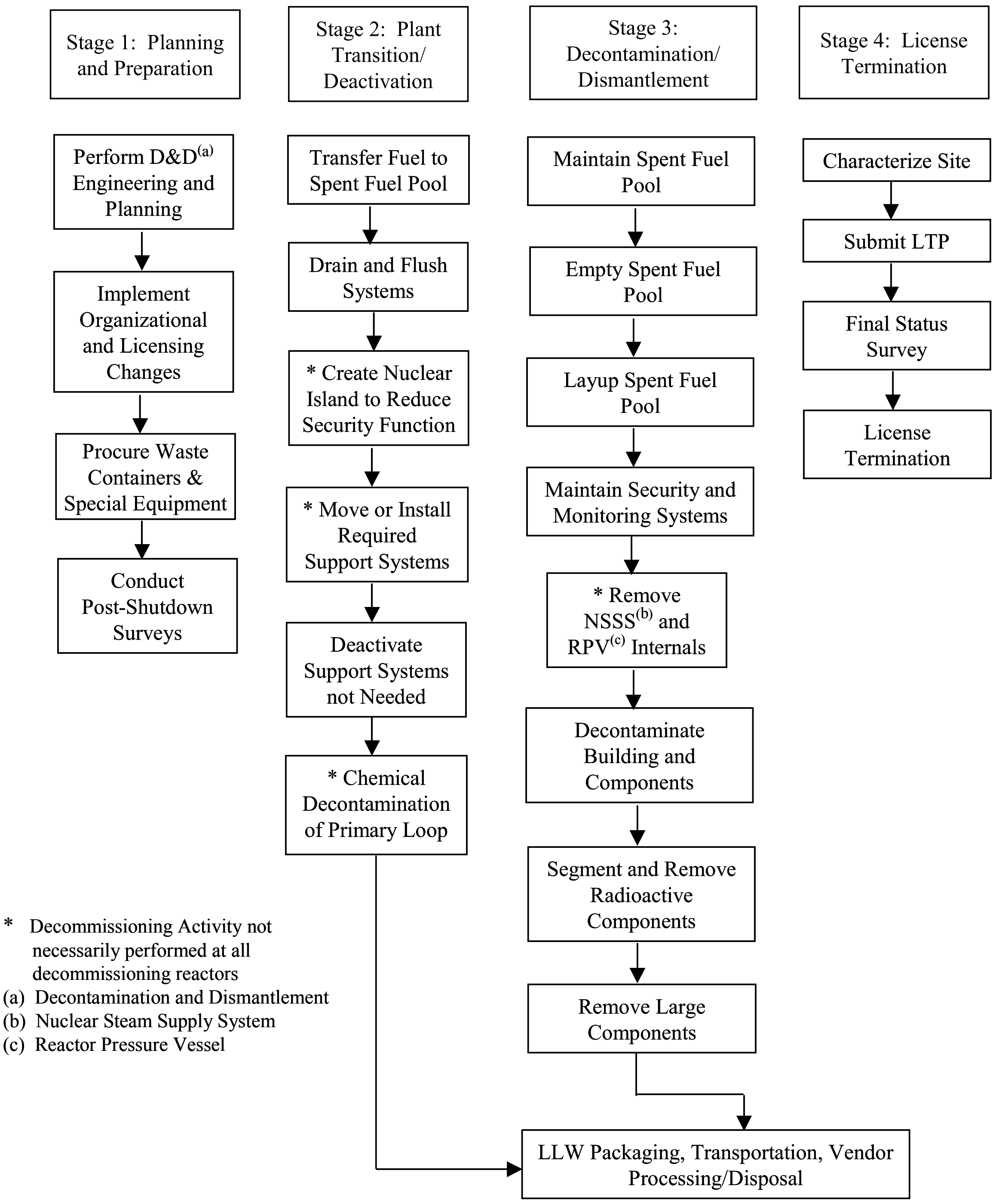
The processes of decommissioning a power reactor facility for the SAFSTOR and DECON options can be divided into four stages, as shown in Figure 3-3. Figure 3-4 identifies the comparable stages that could be postulated for the two ENTOMB options. The order of each step and the duration of each stage vary, depending on plant-specific characteristics, such as location, operating history, reactor vendor, and licensee. The staff considered the differences in timing and choice of activities in evaluating the environmental impacts of decommissioning based on the experiences of currently decommissioning facilities.

Stage 1 in Figures 3-3 and 3-4 includes the licensee's initial preparations to shut down the plant and begin decommissioning. This stage is primarily administrative. Stage 1 typically lasts 1½ to 2½ years, regardless of the decommissioning option chosen. The main activities during the planning and preparation stage are determining the decommissioning option, making changes to the organization structure (layoffs, hiring experienced decommissioning contractors, etc.), and initiating licensing-basis changes.

The planning and preparation activities of Stage 1 vary, depending on when the licensee decides to cease operation. If the end of service is planned, the licensee may make plans for the decommissioning process and may even submit the PSDAR in advance of shutdown. This allows the plant to start major decommissioning activities immediately following the certification of permanent shutdown and the removal of the fuel (see Chapter 2, “Background Information Related to Decommissioning Regulations,” for a discussion of major decommissioning activities). If the end of service is unplanned, the licensee will probably not be ready to start decommissioning activities immediately following the certification of permanent shutdown and removal of fuel. Therefore, the order and duration of the activities in Stage 1 might vary compared to a planned shutdown. For most plants, the organizational changes will include a reduction in the number of staff as well as implementation of an employee-retention program

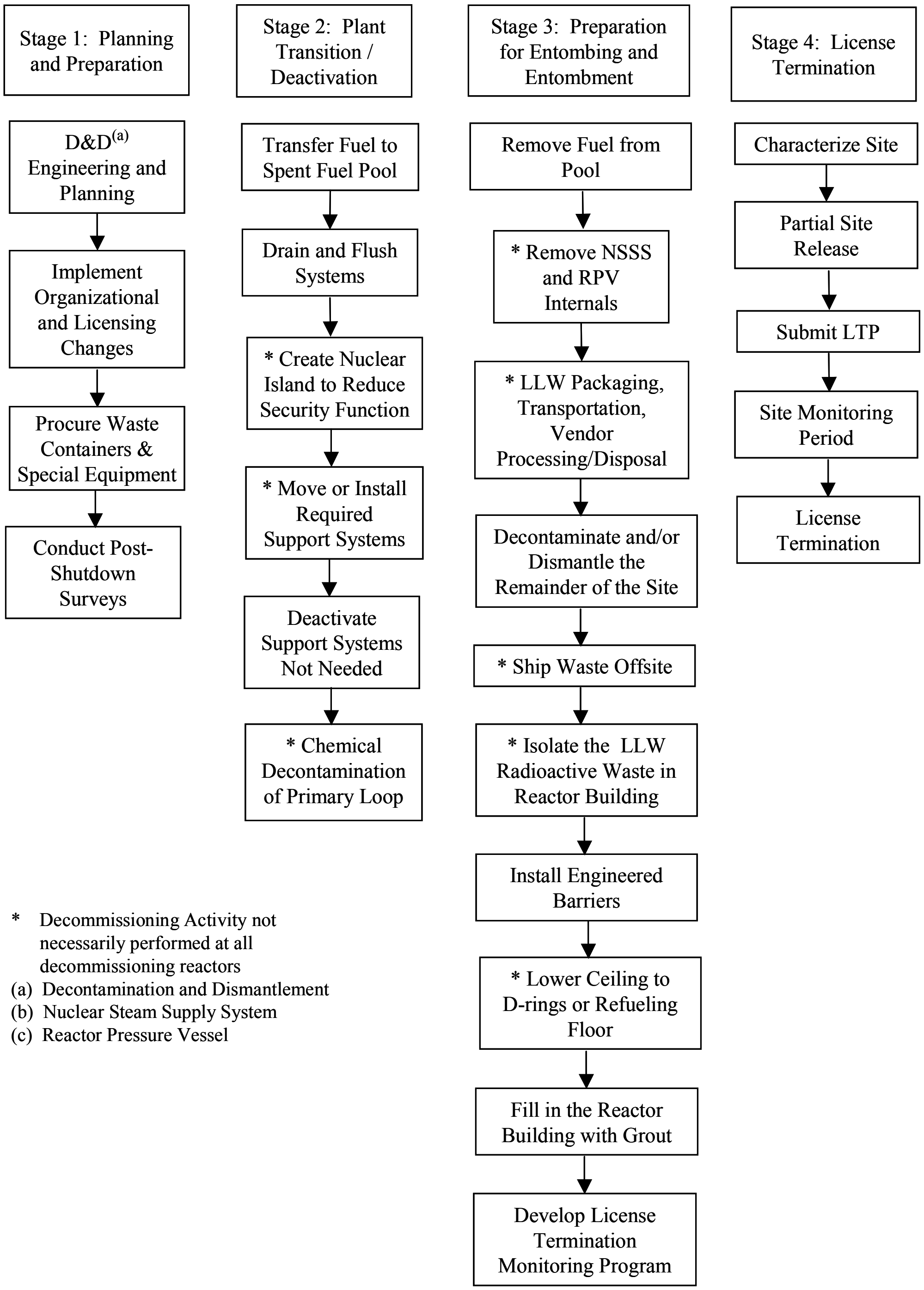
Stage 1 in Figures 3-3 and 3-4 includes the licensee's initial preparations to shut down the plant and begin decommissioning. This stage is primarily administrative. Stage 1 typically lasts 1½ to 2½ years, regardless of the decommissioning option chosen. The main activities during the planning and preparation stage are determining the decommissioning option, making changes to the organization structure (layoffs, hiring experienced decommissioning contractors, etc.), and initiating licensing-basis changes.

The planning and preparation activities of Stage 1 vary, depending on when the licensee decides to cease operation. If the end of service is planned, the licensee may make plans for the decommissioning process and may even submit the PSDAR in advance of shutdown. This allows the plant to start major decommissioning activities immediately following the certification of permanent shutdown and the removal of the fuel (see Chapter 2, “Background Information Related to Decommissioning Regulations,” for a discussion of major decommissioning activities). If the end of service is unplanned, the licensee will probably not be ready to start decommissioning activities immediately following the certification of permanent shutdown and removal of fuel. Therefore, the order and duration of the activities in Stage 1 might vary compared to a planned shutdown. For most plants, the organizational changes will include a reduction in the number of staff as well as implementation of an employee-retention program to encourage the needed staff to stay on. However, one site actually had to increase staffing levels at the time of the permanent cessation of operation to start the DECON process. Initial plant characterization will be made during the planning activities and will continue throughout the decommissioning process. Because these activities are mostly planning, administrative,



* Decommissioning Activity not necessarily performed at all decommissioning reactors
(a) Decontamination and Dismantlement
(b) Nuclear Steam Supply System
(c) Reactor Pressure Vessel

Figure 3-3. Reactor Decommissioning Process - DECON or SAFSTOR



* Decommissioning Activity not necessarily performed at all decommissioning reactors
(a) Decontamination and Dismantlement
(b) Nuclear Steam Supply System
(c) Reactor Pressure Vessel

Figure 3-4. Reactor Decommissioning Process - ENTOMB and organizational in nature, there is little potential for onsite or offsite impacts from these activities and only small amounts of decommissioning related LLW generated.

Stage 2 in Figures 3-3 and 3-4 involves the transition of the plant from reactor operation to decommissioning. Stage 2 will last from about ½ to 1½ years for plants in SAFSTOR, DECON, and ENTOMB. All plants will have to transfer fuel out of the reactor and into the spent fuel pool. Isolation and stabilization of all unnecessary SSCs are also conducted during this stage.

Licensing-basis changes will continue during this stage, and the licensee may request an exemption from offsite emergency preparedness requirements.

For DECON and SAFSTOR, there are a number of activities during Stage 2 that the plant can either choose not to perform or can perform at a later date. Chemical decontamination of the primary system and creation of a nuclear island are the two main activities that several decommissioning sites have undertaken. Chemical decontamination is optional for ENTOMB1 and would not likely occur for ENTOMB2. Support systems no longer necessary to reactor operation may also be removed for all four options. Likewise, additional support systems needed for decommissioning activities may be installed at this stage for DECON, SAFSTOR, and ENTOMB1. Changes to electrical systems are common during Stage 2.

Chemical decontamination of the primary system has been performed at several facilities, resulting in a reduction of total person-rem during decommissioning activities. One facility evaluated conducted a system decontamination, aiming at significant reduced dose to workers and reduced cost, by reducing both the amount and level of contamination from disposal of contaminated piping. This chemical decontamination was performed following the removal of the steam generators, pressurizer, and reactor coolant pump motors, as well as most of the auxiliary piping. At a second facility evaluated, a chemical decontamination was considered necessary to keep doses within previously issued environmental assessments. The chemical decontamination was performed early in the decommissioning process to allow dismantling to proceed unimpeded. Other plants, both operating and permanently shutdown, have also performed chemical decontamination.

Some plants have also created nuclear islands, which are used to reduce the scope of the required safeguards and security systems to the storage facilities only. Focusing security on the physical protection of the fuel can be a cost savings. Creating a nuclear island may involve installing an electrical power supply at the spent fuel pool, installing or modifying chemistry controls, designing and constructing a new heat removal system, and moving or installing new security-related equipment. For plants going into SAFSTOR, creation of a nuclear island is primarily a cost savings, but for plants in active decontamination and dismantlement, work activities may be done more conveniently when workers are not constrained by security requirements. ENTOMB2 would not benefit from the “nuclear island” concept.

Environmental impacts may vary at each site, depending on the activities and the timing of the activities performed. Examples of impacts include activities such as chemical decontamination, which result in the use of small quantities of water and produce LLW as well as some liquid effluents that would not be released unless they are below the limits allowed by the regulations in 10 CFR Part 20. Smaller amounts of waste will likely be generated during the creation of a nuclear island or the rewiring of a facility.

Stage 3 in Figure 3-3 involves decontamination and dismantlement of the plant for DECON, SAFSTOR, and ENTOMB1. For ENTOMB2, Stage 3 involves dismantlement of all radioactively contaminated SSCs external to the reactor building and placement of these SSCs in the reactor building, followed by lowering the ceiling to the D-rings (PWRs) or refueling floor (BWRs). For both ENTOMB options, it includes installation of grout and engineered barriers and development of the license termination monitoring program. For those sites that have a SAFSTOR period, Stage 3 includes the storage time. The decontamination and dismantlement activities performed for SAFSTOR can occur before, after, or during the storage period. For the SAFSTOR period, Stage 3 can be from just a few years to about 54 years. For a site going straight through the DECON option, the time for Stage 3 would be expected to take between 3½ and 10 years. For either ENTOMB option Stage 3 would be expected to take 2 to 4 years.

The greatest variability in the decommissioning process is seen in Stage 3 and is related to dismantlement. Every plant that has completed decommissioning or has started dismantlement has performed the activities in different ways and at different times during the decommissioning process. Two examples of large-component removal are at Rancho Seco and Trojan. Rancho Seco has started its dismantlement on the secondary side, removing the moisture separators, diesel generators, steam piping, and related components. Dismantlement of the equipment in the auxiliary building was also initiated. Plans for large-component removal are still in process. The primary issues related to decisions on large-component removal are how to transport the components. Because there are no convenient waterways for transport, the large components from Rancho Seco will have to be shipped by both road and rail, which will require segmentation or cutting up the larger components. Trojan took a different approach to dismantlement, based on the ability to ship by barge and the availability of disposal at Hanford. Trojan removed its four steam generators and pressurizer, pumped grout into them, and shipped them by barge for burial at Hanford. Following that activity, the reactor vessel and internals were removed whole, filled with grout, welded closed, and shipped. For Trojan, removing and shipping these large components as whole units saved millions of dollars and significantly reduced dose to workers.

Stage 4 of decommissioning is license termination. Activities for this stage, which are similar for all options, include final site characterization, final radiation survey submission of final license termination plan, and final site survey. The ENTOMB options would include both a partial site release and a site monitoring program.

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