summary and the response to comments in Appendices A and B of this report. However, due to the length of these two appendices, the staff decided to publish these two appendices and the appendix containing the transcripts and comment letters in a second volume. In addition to the scoping meetings, meetings were held with EPA and CEQ between February and November 2000 to obtain input on the scope of the environmental review.

Site visits were conducted by the NRC staff and its contractor at six nuclear reactor facilities that are in various stages of decommissioning. The site visits were conducted to obtain information and to familiarize the NRC team with the current types of activities conducted and the resulting impacts during decommissioning. In addition to the site visits, the Nuclear Energy Institute arranged access to additional site-specific decommissioning data. In addition to the six sites visited, data was received for three other nuclear power reactor facilities.

Information used in this report was also obtained from docketed material, such as post-shutdown decommissioning activity reports (PSDARs), effluent release reports, license termination plans (LTPs), and decommissioning funding plans.

1.3 Scope of This Supplement

Except for decommissioning planning activities, this Supplement considers only activities that occur following certification that fuel has been removed from the reactor. Figure 1-1 illustrates the decommissioning process. Licensee decommissioning activities are listed in the top part of the timeline. Regulatory activities are summarized by the lower part of the timeline. This section discusses licensee decommissioning activities that are within scope and also explains why some activities and impacts are not in scope for this Supplement. Table 1-1 briefly lists decommissioning activities that are within and outside the scope of this Supplement. Additional discussion of the out-of-scope activities is provided in Appendix D.

Impacts related to the decision to permanently cease operations are outside the scope of this Supplement. This includes impacts that result directly and immediately from the act of permanently ceasing operations, regardless of when or why the decision was made. For example, when a reactor ceases operation, the flow of warmer water into the canal, lake, or river that receives the plant's thermal discharges is stopped, and this may impact the organisms in the vicinity of the thermal outfall. However, this impact is not within the scope of this Supplement because it is essentially a restoration of the existing conditions.

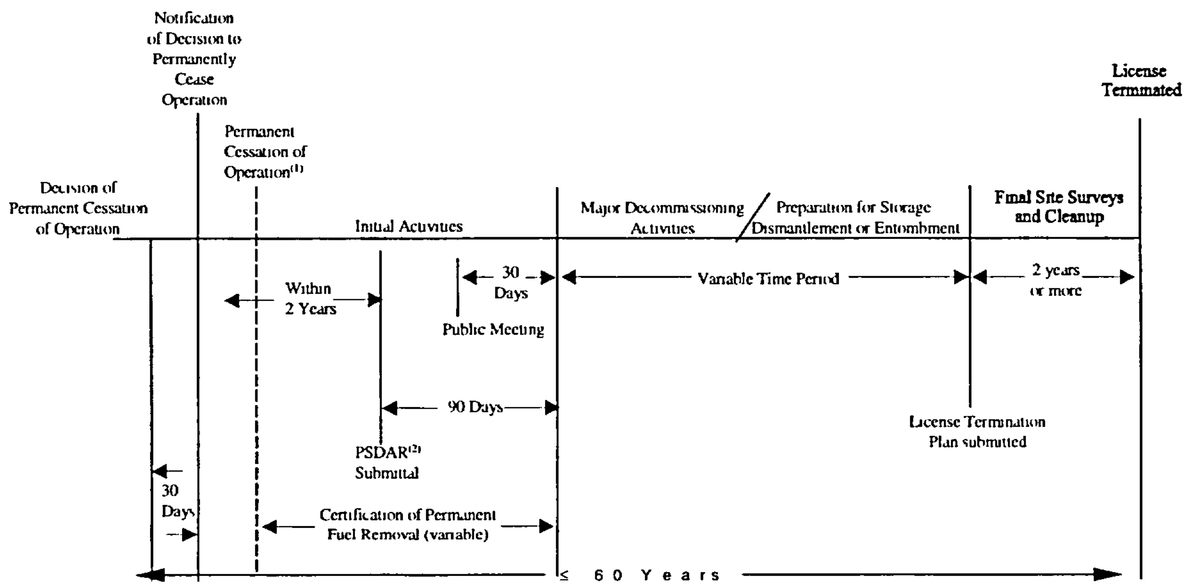

Figure 1-1.
Decommissioning Timeline

(1) The cessation of operations may occur before, concurrent with, or following the certification to permanently cease operations.
(2) The PSDAR may be submitted before permanent cessation of operations.

The licensee may declare or certify the date for permanent cessation of operations prior to the end of the license term and while still operating. In such cases, the decommissioning planning activities prior to shutdown and activities and impacts that occur following the actual shutdown of the facility are within the scope of this Supplement. In some circumstances, the licensee may not operate the facility for a period of many years without certifying that they have permanently ceased power operations. In these cases, the activities occurring before the certification is completed would be considered part of the operational phase of the facility and would be within the scope of the site-specific environmental impact statement (EIS) that covers reactor operations but are outside the scope of this Supplement.

The NRC definition for *decommission* in 10 CFR 50.2 is "to remove a facility or site safely from service and reduce residual radioactivity to a level that permits (1) Release of the property for unrestricted use and termination of the license; or (2) Release of the property under restricted conditions and termination of the license." This Supplement is not limited only to activities directly related to the removal of radioactive material from facilities or that must be performed to facilitate removal of contaminated SSCs. The staff has included activities and impacts related to removing uncontaminated SSCs that were required for reactor operation, such as the intake structure or cooling towers. Including uncontaminated SSCs in this Supplement is consistent with an expectation under NEPA that all impacts associated with an activity and that public concerns about the scope of the review be considered.

Various activities that are performed in conjunction with decommissioning are not considered within the scope of this Supplement, but are reviewed and regulated by the NRC under other licenses. These activities include

- independent spent fuel storage installation (ISFSI) construction, maintenance, and decommissioning – An ISFSI can be 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 under a specific license under 10 CFR Part 72. If a licensee chose to operate the ISFSI under a Part 50 license, it could choose to continue to maintain their Part 50 license, or seek a site -specific 10 CFR Part 72 license for the ISFSI, thus allowing termination of the Part 50 license and the end of the reactor decommissioning process. The NRC staff would also be required to conduct an environmental assessment of the licensee's request for a site-specific 10 CFR Part 72 license.

- spent fuel storage and maintenance – 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 that 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 has elected to include in this Supplement information related to the storage and maintenance of fuel in a spent fuel pool for completeness.

Table 1-1. Activities and Impacts Within or Outside the Scope of This Supplement

In Scope
• Activities performed to remove the facility from service from the time that the licensee certifies that the facility has permanently ceased operations
• Activities (and the resulting impacts) performed in support of radiological decommissioning, including decontamination and dismantlement of radioactive structures and any activities required to support the decontamination and dismantlement process
• Activities performed in support of dismantlement of nonradiological structures, systems, and components (SSCs) required for the operation of the reactor, such as diesel generator buildings and cooling towers
• Activities performed up to license termination and their resulting impacts as provided in the definition of decommissioning Nonradiological impacts occurring after license termination from activities conducted during decommissioning
• Activities related to release of the facility
• Human health impacts from radiological and nonradiological decommissioning activities
• Activities related to preparing the facility for entombment
Out of Scope[a]
• Activities and the resulting impacts (other than planning activities) that are performed before permanent cessation of operation is certified
• Radiological impacts following license termination
• Activities (and the resulting impacts) performed to dismantle structures on the site that are not radiologically contaminated and were not required for operation of the reactor (e.g., training building and administration building)
• Activities performed to support installation of alternate energy-generating facilities during or following the decommissioning process
• Site restoration activities performed during or after the decommissioning process
• Activities (and their impacts) performed after license termination, such as - any additional non-NRC required monitoring to evaluate radiological impacts - site restoration - continued use of site for power production or other activities
• Activities performed at facilities that are separately licensed or regulated - independent spent fuel storage installation (ISFSI) construction, maintenance, or decommissioning - interim storage of Greater-than-Class-C Waste - spent fuel storage,[b] maintenance, and disposal on or away from a reactor location - low-level waste (LLW) disposal at a licensed LLW site or treatment at compactor facilities
• Activities to install engineered barriers and institutional controls for restricted release
• Public perceptions and psychological impacts
• Activities at facilities that have been permanently shut down by a major accident
• Issues related to the ENTOMB option after the facility begins the entombment period

(a) A detailed discussion of the reasons for determining that activities are out of scope can be found in Appendix D.

(b) As discussed in the text, the staff relies on the Waste Confidence Decision Review (54 FR 39767 and 64 FR 68005) but has chosen to include information related to the storage and maintenance of fuel in a spent fuel pool for completeness in this Supplement.

- spent fuel transport and disposal away from the reactor location – Transportation of spent fuel and other high-level nuclear wastes is governed by regulations in 10 CFR Part 71, "Packaging and Transportation of Radioactive Material." Disposal of spent fuel and high-level wastes are governed by the Nuclear Waste Policy Act (NWPA) of 1982, as amended, which defined the goals and structure of a program for permanent, deep geologic repositories for the disposal of high-level radioactive waste and nonreprocessed spent fuel. Under this Act, the U.S. Department of Energy (DOE) is responsible for developing permanent disposal capacity for spent fuel and other high-level nuclear wastes. Title 10 CFR Part 60 contains rules governing the licensing to receive and possess source, special nuclear, and by-product material at a geological repository operations area that is sited, constructed, or operated in accordance with the NWPA. However, the Commission issued the final rule to supercede the generic criteria in 10 CFR Part 60 for disposal at a geological repository with specific criteria in 10 CFR Part 63, issued on November 2, 2001 (66 FR 55732).

- LLW disposal at a licensed LLW site or treatment of LLW at compactor facilities – Regulations related to LLW disposal are in 10 CFR Part 61 and 10 CFR Part 20, Subpart K. A final GEIS supporting the regulations in 10 CFR Part 61, "Final Generic Environmental Impact Statement for 10 CFR Part 61" was published as NUREG-0945 (NRC 1982).

A further description of these activities and the basis for not including them in the scope of this supplement is in Appendix D.

The decommissioning process continues until the licensee requests termination of the license and demonstrates that radioactive material has been removed to levels that permit termination of the NRC license. Once the NRC determines that the decommissioning is completed, the license is terminated. At that point, the NRC no longer has regulatory authority over the site, and the owner of the site is no longer subject to NRC regulations. As a result, activities performed after license termination and the resulting impacts are outside the scope of this Supplement. These activities may include any non-NRC required monitoring, site restoration (grading, planting of vegetation, etc.), continued dismantlement or continued use of the site for activities such as power production using natural gas, oil, or coal.

Any potential radiological impacts following license termination that are related to activities performed during decommissioning are not considered in this Supplement. Such impacts are covered by the Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities, NUREG-1496 (NRC 1997).

Any potential nonradiological impacts resulting from decommissioning and occurring after termination of the license are considered within the scope of this Supplement. Onsite disposal has been proposed by the industry as a method to dispose of slightly radiologically contaminated building rubble provided that the waste is buried onsite below grade, for example, in existing underground portions of the dismantled plant in such a manner as to meet the site release criteria of 10 CFR Part 20, Subpart E. This concept has been referred to as "Rubblization" (the disposal onsite of slightly contaminated material in a manner to meet the 10 CFR Part 20 release criteria).[a] On February 14, 2000, the staff informed the Commission of licensee interest in this method and the staff's intent to address Rubblization in this Supplement (NRC 2000). The staff has determined that the long-term radiological aspects of Rubblization, or onsite disposal of slightly contaminated material, would require a site-specific analysis and would be addressed at the time the LTP is submitted. The nonradiological impacts, occurring both during the decommissioning period (e.g., noise, dust, land disturbance), and the long-term impacts occurring after the decommissioning activities are completed (e.g., concrete leaching into the groundwater) can be evaluated generically and are included in the evaluation of each of the applicable environmental issues in Chapter 4 of this document.

Public perceptions and psychological impacts related to the risk of a radiological accident during decommissioning are not addressed in the 1988 GEIS and are not addressed in this Supplement. The U.S. Supreme Court stated in *Metropolitan Edison Co. v. People Against Nuclear Energy*, 460 U.S. 766, at 774-775, that such psychological effects or impacts raised policy questions that fell outside of NEPA. This court case involved an organization of residents living in the area of Three Mile Island, People Against Nuclear Energy (PANE), that claimed the NRC should consider, as part of an EIS, the severe psychological stress caused to its members by the restart of Three Mile Island, Unit 1, after the accident at Three Mile Island, Unit 2. However, in *Metropolitan Edison Co., et al. v. People Against Nuclear Energy* (1983), the Supreme Court read NEPA to require

> a reasonably close causal relationship between a change in the physical environment and the effect at issue .... a <u>risk</u> of an accident is not an effect on the physical environment .... We believe that the element of risk lengthens the causal chain beyond the reach of NEPA.

(a) The term "rubblization" is frequently used to describe the crushing of structural material (e g., concrete) to facilitate disposal. The material may be concrete that is uncontaminated or contaminated with radiological material. The staff used the term Rubblization to describe the process of onsite disposal of slightly contaminated material in a manner to meet the site release criteria. For this report, in order to avoid confusion, the staff chose to use the term "demolition" instead of rubblization as the verb to describe the process of crushing structural material to allow for easy burial or disposal.

The decommissioning activities following shutdown of a facility after a major accident resulting in significant contamination of the site are outside the scope of this Supplement. For most types of accidents, decommissioning would be treated on a site-specific basis and, therefore, cannot be considered in a generic sense.

1.4 Categories for Environmental Impacts and Extent of Issues

In the analysis of potential issues in decommissioning activities, two areas in particular were found to benefit from categorization: (a) ranking the significance and severity of potential environmental impacts for proposed decommissioning activities and (b) sorting potential issues as either generic or site-specific.

1.4.1 Levels of Significance of Environmental Impacts

For decommissioning, the staff is using a standard of significance derived from the CEQ terminology for "significantly" (40 CFR 1508.27, which considers "context" and "intensity"). The NRC has defined three significance levels: SMALL, MODERATE, and LARGE.

SMALL – Environmental impacts are not detectable or are so minor that they will neither destabilize nor noticeably alter any important attribute of the resource. For the purposes of assessing radiological impacts in this Supplement, the NRC has concluded that those impacts that do not exceed permissible levels in the Commission's regulations are considered small.

MODERATE – Environmental impacts are sufficient to alter noticeably, but not to destabilize, important attributes of the resource.

LARGE – Environmental impacts are clearly noticeable and are sufficient to destabilize important attributes of the resource.

The discussion of each environmental issue in this Supplement includes an explanation of how the significance level was determined. In determining the significance level, the NRC staff assumed that ongoing mitigation measures would continue (including those mitigation measures implemented during plant construction and/or operation) during decommissioning, as appropriate. Benefits of additional mitigation measures during or after decommissioning are not considered in determining significance levels.

1.4.2 Regulatory Distinction of Generic and Site-Specific Approaches

In addition to determining the significance of environmental impacts, this Supplement includes a determination of whether the analysis of the environmental issue could be applied to all plants, and whether additional mitigation measures would be warranted. An environmental issue may be assigned to one of two categories (generic or site-specific) described below.

- Generic – For each environmental issue, the analysis reported in this Supplement shows the following:

  (1) Environmental impacts associated with the issue have been determined to apply either to all plants, or for some issues to plants having a specific size, specific location, or having a specific type of cooling system or other site characteristics, and

  (2) A single significance level (i.e., SMALL, MODERATE, or LARGE) has been assigned to the impacts, and

  (3) Mitigation of adverse impacts associated with the issue has been considered in the analysis, and it has been determined that additional plant-specific mitigation measures are not likely to be sufficiently beneficial to warrant implementation.

- Site-specific – For each environmental issue that was determined to be site-specific, the analysis reported in this Supplement has shown that one or more of the generic criteria was not met. Therefore, additional plant-specific review is required.

1.5 Uses of This Supplement

This Supplement can be used by the public to understand the decommissioning process, the activities performed during decommissioning, and the potential environmental impacts resulting from these activities. The Supplement does not (1) establish or revise regulations, (2) impose requirements, (3) provide relief from requirements, or (4) provide guidance on the decommissioning process.

This Supplement identifies activities that can be bounded by a generic evaluation. It also identifies the decommissioning activities and associated environmental issues that will likely require site-specific analysis before performing a decommissioning activity.

Licensees can rely on the information in this Supplement as a basis for meeting the requirements in 10 CFR 50.82(a)(6)(ii). This requirement states that the licensee must not perform any decommissioning activity that causes any significant environmental impact not previously reviewed. Prior to conducting a decommissioning activity, the licensee must make a determination that the resulting environmental impacts fall within the bounds of this Supplement or of another EIS related to its facility. When finalized, licensees are expected to reflect the environmental impacts described in this Supplement rather than those in the 1988 GEIS. For any decommissioning activity that does not meet these conditions, the regulations prohibit the licensee from undertaking the activity until it performs a site-specific analysis of the activity. Depending on the results of the site-specific evaluation, the staff may determine that it is appropriate to consult with another agency about the potential impacts. Such agencies could include the U.S. Fish and Wildlife Service or a State Historic Preservation Office. If the activity would result in an impact that is outside the bounds of the GEIS or other environmental assessments, the licensee would be required to submit a license-amendment request. The NRC staff periodically inspects the licensee's procedures and documentation to ensure that a proper environmental review is part of the screening criteria used for proposed changes to the facility.

In addition to the NRC staff's review of the licensee's procedures and documentation, there are two points during the decommissioning process when the licensee performs an evaluation of environmental impacts. The first evaluation occurs when the licensee must submit a PSDAR to the NRC (within two years following permanent cessation of operation). The PSDAR must include a discussion that provides the reasons for concluding that the environmental impacts associated with the licensee's planned site-specific decommissioning activities will be bounded by an appropriate previously issued environmental assessments, including this Supplement. If the licensee identifies environmental impacts that are not bounded by a previous NRC environmental assessment, the licensee must address the impacts in a request for a license amendment regarding the activities. The licensee must also submit a supplement to its environmental report (ER) that describes and evaluates the additional impacts. The NRC will review the supplement to the ER in conjunction with its review of the license-amendment request.

The second evaluation is near the end of decommissioning at the time when the licensee submits an application for license termination. In accordance with 10 CFR 50.82(a)(9), a licensee must submit its LTP at least 2 years before the anticipated termination date of the license. The LTP must be a supplement to the Final Safety Analysis Report or its equivalent for the facility and is submitted as a license amendment. The NRC requires an environmental review as part of the review of the license-amendment request. Thus, the LTP must include a supplement to the ER that describes any new information or significant environmental change associated with the licensee's proposed termination activities. The NRC staff will also rely upon this supplement as a basis for determining if anticipated decommissioning impacts require an additional review.

1.6 Development of This Supplement

The requirements in 10 CFR Part 51 were followed for the development of this Supplement. This included conducting scoping meetings and obtaining public comments (see Appendix N). From these meetings and meetings with other appropriate government agencies, the staff defined the scope of this Supplement (see Sections 1.2 and 1.3). During the scoping process, the staff developed an evaluation process for determining the environmental impacts from decommissioning. Section 4.2 provides additional discussion of the process and Appendix E provides a detailed description of the analysis used to identify the environmental impacts from decommissioning. The evaluation process involved determining the specific activities that occur during decommissioning and obtaining data from site visits and from an information request to decommissioning plants that was related to the impact of these activities at currently decommissioning facilities. The data obtained from the decommissioning sites were analyzed and then evaluated against a list of variables that defined the parameters for plants that are currently operating but which will one day be decommissioned. This evaluation resulted in a range of impacts for each environmental issue that may be used for comparison by licensees that are or will be decommissioning their facilities.

1.7 Parts of This Supplement

Chapter 2 provides background, describing the basis for the current regulations and summarizing the regulations. Chapter 3 describes the types of plants covered by this Supplement, which includes permanently shutdown reactor facilities as well as operating facilities that will eventually cease power operations. Chapter 3 also describes the location and types of buildings on the sites, the systems that may still be active after permanent shutdown, and changes in effluents after permanent shutdown. Chapter 4 describes activities conducted during the decommissioning process and impacts that could arise from these activities. The analysis of the impacts is based on variables such as the option of decommissioning, location of plant, type of plant, and timing of the activity. Chapter 5 discusses the "No Action" alternative to decommissioning, which is the abandonment of the facility after the cessation of operations. Chapter 6 contains the summary of findings and conclusions.

1.8 References

10 CFR 20. Code of Federal Regulations, Title 10, *Energy*, Part 20, "Standards for protection against radiation."

10 CFR 50. Code of Federal Regulations, Title 10, *Energy*, Part 50, "Domestic licensing of production and utilization facilities."

10 CFR 51. Code of Federal Regulations, Title 10, *Energy*, Part 51, "Environmental protection regulations for domestic licensing and related regulatory functions."

10 CFR 60. Code of Federal Regulations, Title 10, *Energy*, Part 60, "Disposal of high-level radioactive wastes in geologic repositories."

10 CFR 61. Code of Federal Regulations, Title 10, *Energy*, Part 61, "Licensing requirements for land disposal of radioactive waste.

10 CFR 63. Code of Federal Regulations, Title 10, *Energy*, Part 63, "Disposal of high-level radioactive wastes in a geologic repository at Yucca Mountain, Nevada."

10 CFR 71. Code of Federal Regulations, Title 10, *Energy*, Part 71, "Packaging and transportation of radioactive material."

10 CFR 72. Code of Federal Regulations, Title 10, *Energy*, Part 72, "Licensing requirements for the independent storage of spent nuclear fuel, high-level radioactive waste, and reactor-related greater-than-Class-C waste."

40 CFR 1508. Code of Federal Regulations, Title 40, *Protection of the Environment*, Part 1508, "Terminology and Index."

54 FR 39767. "10 CFR Part 51 Waste Confidence Decision Review." *Federal Register*. September 28, 1989.

64 FR 68005. "Waste Confidence Decision Review." *Federal Register*. December 6, 1999.

66 FR 55732. "Disposal of High-Level Radioactive Wastes in a Proposed Geologic Repository at Yucca Mountain, Nevada." *Federal Register*. November 2, 2001.

Metropolitan Edison Co., et al v. People Against Nuclear Energy, 460 U.S. 766, at 774-775. 1983.

National Environmental Policy Act (NEPA) of 1969, as amended, 42 USC 4321 et seq.

Nuclear Waste Policy Act of 1982, as amended, 42 USC 10.101 et seq.

U.S. Nuclear Regulatory Commission (NRC). 1982. *Final Generic Environmental Impact Statement for 10 CFR Part 61*. NUREG-0945, NRC, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1988. *Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities.* NUREG-0586, NRC, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1997. *Final Generic Environmental Impact Statement in Support of Rulemaking on Radiological Criteria for License Termination of NRC-Licensed Nuclear Facilities.* NUREG-1496, Vol. 1, NRC, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 2000. "SECY-00-0041 Use of Rubblized Concrete Dismantlement to Address 10 CFR Part 20, Subpart E, Radiological Criteria for License Termination." NRC, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 2001. Letter from U.S. NRC to Distribution: "Subject: Issuance of a scoping summary report of comments received related to the intent to develop a Supplement to NUREG-0586." Dated April 17, 2001.

2.0 Background Information Related to Decommissioning Regulations

This section provides background information that will assist the reader in understanding the requirements for decommissioning and license termination. The basis for the current decommissioning regulations and a summary of the current regulations are provided below. This chapter and Chapter 3, "Description of NRC Licensed Reactor Facilities and the Decommissioning Process," will give the reader a basic understanding of the overall reactor decommissioning process and environmental impact assessments used during the process.

2.1 Basis for Current Regulations

In the mid-1990s, the Commission initiated an effort to significantly change the regulations for decommissioning power reactor facilities. The new regulations were intended to make the decommissioning process more current, efficient, and uniform. On July 29, 1996, a final rule revising 10 CFR 50.82, "Decommissioning of Nuclear Power Reactors," was published in the Federal Register (61 FR 39278). This rule redefined the decommissioning process and modified the regulations written in 1988, which had required submittal of a detailed decommissioning plan before the start of decommissioning.

The regulations were revised based on experience gained from reactor decommissionings that had occurred during the 1980s and early 1990s. Review of the activities that occur during decommissioning showed that they are similar to the activities that occur during the construction, operation, maintenance, and refueling outages of a power reactor (e.g., decontamination, steam generator replacement, and pipe removal). However, the magnitude of some activities during decommissioning (e.g., removal of piping) is considerably greater than during operations. Activities associated with the decommissioning of facilities had resulted in impacts consistent with or less than those evaluated in the 1988 *Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities* (GEIS), NUREG-0586 (NRC 1988). Based on the above reasons, the Commission determined that review and approval by the U.S. Nuclear Regulatory Commission (NRC) staff of a detailed decommissioning plan was not necessary.

2.2 Summary of Current Regulations

2.2.1 Regulations for Decommissioning Activities

The current regulations (10 CFR 50.82) specify the regulatory actions that both the NRC and the licensee must take to decommission a nuclear power facility. Once the licensee decides to permanently cease operations, it must submit, within 30 days, a written certification to the NRC.

The notification must contain the date on which the power-generating operations ceased or will cease. The licensee must permanently remove all fuel from the reactor and submit a written certification to the NRC confirming the completion of fuel removal. Once this certification has been submitted, the licensee is no longer permitted to operate the reactor, or to put fuel back into the reactor vessel. After certification that the fuel is removed, the annual license fee to the NRC is reduced as well as the licensee's obligation to adhere to certain requirements that are needed only during reactor operations.

In addition to the certifications, the licensee must submit a post-shutdown decommissioning activities report (PSDAR) to the NRC and any affected States no later than 2 years after the date of permanent cessation of operations. Section 10 CFR 50.82 requires that the PSDAR include

- a description of the licensee's planned major decommissioning activities
- a schedule for completing these activities
- an estimate of the expected decommissioning costs
- a discussion that provides the reasons for concluding that the environmental impacts associated with site-specific decommissioning activities will be bounded by an appropriate previously issued environmental impact statement (EIS).

After receiving a PSDAR, the NRC publishes a notice of receipt in the Federal Register, makes the PSDAR available for public review and comment, and holds a public meeting in the vicinity of the facility to discuss the licensee's plans. The NRC will examine the PSDAR to determine if the required information is included and will inform the licensee in writing if there are deficiencies that must be addressed before the licensee initiates any major decommissioning activities. The regulations require a 90-day waiting period after submittal of the PSDAR before the licensee may commence major decommissioning activities.

The purpose of the PSDAR is to provide the NRC and the public with a general overview of the licensee's proposed decommissioning activities. The PSDAR serves to inform the NRC staff of the licensee's expected activities and schedule, which facilitates planning for inspections and decisions regarding NRC oversight activities. The PSDAR is also a mechanism for informing the public of the proposed decommissioning activities before those activities are conducted.

Prior to submission of the PSDAR, the licensee can conduct a variety of activities at the site including activities to ensure the safe shutdown of the facility. Systems can be drained, components removed, and certain structures demolished. However, the licensee is prohibited from undertaking any major decommissioning activity as defined in 10 CFR 50.2.

Once the PSDAR has been submitted and the 90-day period has been completed, the licensee may begin major decommissioning activities, which may include the following:

- permanent removal of major radioactive components, such as the reactor vessel, steam generators, or other components that are comparably radioactive
- permanent changes to the containment structure
- dismantling of components containing Greater-than-Class-C (GTCC) Waste.[a]

In accordance with 10 CFR 50.82(a)(6)(ii), licensees shall not perform any decommissioning activities "that result in significant environmental impacts not previously reviewed." If any decommissioning activity does not meet this requirement, the licensee must submit a license-amendment request before conducting the activity. The licensee also must submit a supplement to its environmental report (ER) that relates to the additional impacts. The NRC will review the ER Supplement, and prepare an environmental assessment (EA) or EIS, and amendment to the license in conjunction with its review.

The licensee can choose (1) to immediately decontaminate and dismantle the facility (DECON), or (2) to place the facility in long-term storage (SAFSTOR) followed by subsequent decontamination and dismantlement, or (3) to perform some incremental decontamination and dismantlement activities before or during the storage period of SAFSTOR. Under the current regulations, unless the licensee receives permission to the contrary, the site must be decommissioned within 60 years. Chapter 3 describes in more detail the decommissioning

(a) The NRC has adopted a waste classification system for low-level radioactive waste based on its potential hazards, and has specified disposal and waste form requirements for each of the general classes of waste: A, B, and C. The classifications are based on the key radionuclides present in the waste and their half-lives. Tables defining these three classes are contained in 10 CFR 61.55. In general, requirements for waste form, stability, and disposal methods become more stringent when going from Class A to Class C. GTCC waste exceeds the concentration limits in 10 CFR 61.55 and is generally unsuitable for near-surface disposal as low-level waste (LLW), even though it is legally defined as LLW. The NRC's regulations in 10 CFR 61.55(a)(2)(iv) require that this type of waste must be disposed of in a geologic repository unless approved for an alternative disposal method on a case-specific basis by the NRC. 10 CFR Part 72 allows for interim storage of GTCC from a commercial power reactor.

options available to the licensee. In this Supplement, the staff also evaluates another option called ENTOMB, which encases the radioactive contaminants in a structurally long-lived material.

2.2.2 Regulations for License Termination

In order to terminate the license and allow release of the site, the licensee must submit a license termination plan (LTP). In accordance with 10 CFR 50.82(a)(9), an application for license termination must be accompanied or preceded by an LTP, which is subject to NRC review and approval. The licensee must submit the LTP at least 2 years before the date of license termination. The LTP approval process is by license amendment. By regulation, the LTP must include the following:

- a site characterization
- identification of remaining dismantlement activities
- plans for site remediation
- detailed plans for the final survey of residual contamination
- a description of the end-use of the site (if restricted use is proposed)
- an updated site-specific estimate of remaining decommissioning costs
- a supplement to the ER.

The licensee must submit the LTP as a supplement to its Final Safety Analysis Report or as an equivalent document, thus formalizing the steps necessary to revise the document.

After receiving the LTP, the NRC will place a notice of receipt of the plan in the Federal Register and will make the plan available to the public for comment. The NRC will schedule a public meeting near the facility to discuss the plan's contents and the staff's process for reviewing the submittal. The NRC will also offer an opportunity for a public hearing on the license-amendment request associated with the LTP. At this stage, a site-specific EA is required. Depending on the circumstances, the EA evaluation can result in the development of a full EIS. If the LTP demonstrates that the remainder of decommissioning activities will be performed in accordance with NRC regulations, are not detrimental to the health and safety of the public, and will not have a significant adverse effect on the quality of the environment, the

Commission will approve the plan by a license amendment (subject to whatever conditions and limitations the Commission deems appropriate and necessary).

After the approval of the LTP, the NRC will continue its inspection of the site. These inspections will include validation of commitments made in the LTP. Inspections may also include confirmatory surveys to verify that areas of the site have been decontaminated to the limits established in the LTP.

On July 21, 1997, the NRC published (also in the Federal Register) a final rule entitled, "Radiological Criteria for License Termination" (64 FR 39058) prescribing specific radiological criteria for license termination. At the end of the LTP process, if the NRC determines that the remaining dismantlement has been performed in accordance with the approved LTP, and if the final radiation survey and associated documentation demonstrate that the facility and site are suitable for release, then the Commission will terminate the license.

The radiological criteria for license termination are given in 10 CFR Part 20, Subpart E. There are two broad categories of uses for the facility after the license termination: unrestricted use and restricted use.

Unrestricted use means that there are no NRC-imposed restrictions on how the site may be used. State and local jurisdictions may, and have, imposed additional restrictions or requirements on licensees. The licensee is free to continue to dismantle any remaining buildings or structures and to use or sell the land for any type of application. The Commission has established a 0.25 mSv/yr (25 mrem/yr) total effective dose equivalent (TEDE) to an average member of the critical group[(a)] as an acceptable criterion for release of any site for unrestricted

(a) The "critical group" is that group of individuals reasonably expected to receive the highest exposure to residual radioactivity within the assumptions of a particular scenario. The average dose to a member of the critical group is represented by the average of the doses for all members of the critical group, which in turn is assumed to represent the most likely exposure situation. For example, when considering whether it is appropriate to "release" a building that has been decontaminated (allow people to work in the building without restrictions), the critical group would be the group of employees that would regularly work in the building. If radiation in the soil is the concern, then the scenario used to represent the maximally exposed individual is that of a resident farmer. The assumptions used for this scenario are prudently conservative and tend to overestimate the potential doses. The added "sensitivity" of certain members of the population, such as pregnant women, infants, children, and any others who may be at higher risk from radiation exposures, are accounted for in the analysis. However, the most sensitive member may not always be the member of the population that receives the highest dose. This is especially true if the most sensitive member (e.g., an infant) does not participate in activities that provide the greatest dose or if they do not eat specific foods that cause the greatest dose.

use. The licensee will be required to show that the site can meet this criterion before the license will be terminated for unrestricted use. In addition, the licensee will need to show that the amounts of residual radioactivity have been reduced to levels that are as low as reasonably achievable (ALARA).[a] For sites that have been determined to be acceptable for unrestricted use, there are no requirements for further measurement of radiation levels. It is not expected that these radiation levels would change (other than to be reduced over time through radioactive decay), and there would be no mechanism for further contamination or radiological releases.

Restricted use means that there are restrictions on the facility use after license termination. A site would be considered acceptable for license termination under restricted conditions if the licensee can demonstrate that further reductions in residual radioactivity necessary to meet the requirements for unrestricted use would result in net public or environmental harm, or were not being made because the residual levels were ALARA. In addition, the licensee must have made provisions for legally enforceable institutional controls (e.g., use restrictions placed in the deed for the property) that provide reasonable assurance that the radiological criteria set by the NRC (0.25 mSv/yr [25 mrem/yr] TEDE to an average member of the critical group) will not be exceeded. The licensee must also have provided sufficient financial assurance to an amenable independent third party to assume and carry out responsibilities for any necessary control and maintenance of the site. There are also regulations relating to the documentation of how the advice of individuals and institutions in the community who may be affected by decommissioning has been sought and incorporated in the LTP if the license is to be terminated under restricted conditions.

Residual radioactivity at the site must be reduced so that if the institutional controls were no longer in effect, there would be reasonable assurance that the TEDE from residual radioactivity distinguishable from background to the average member of the critical group would be ALARA and would not exceed either 1 mSv/yr (100 mrem/yr) or 5 mSv/yr (500 mrem/yr). In the latter case, the licensee must (1) demonstrate that further reductions in residual radioactivity necessary to comply with the 1 mSv/yr (100 mrem/yr) value are not technically achievable, would be prohibitively expensive, or would result in net public or environmental harm, (2) make provisions for durable institutional controls, and (3) provide sufficient financial assurance to enable a responsible government entity or independent third party to carry out periodic checks of the facility no less frequently than every 5 years to ensure that the institutional controls remain in place.

(a) The ALARA concept means that all doses are to be reduced below required levels to the lowest reasonably achievable level considering economic and societal factors. Determination of levels that are ALARA must consider any detriments, such as deaths from transportation accidents, that are expected to potentially result from disposal of radioactive waste.

Alternate release criteria may be used in specific cases. The use of alternate criteria to terminate a license requires the approval of the Commission after consideration of the NRC staff's recommendations that address comments provided by the U.S. Environmental Protection Agency and any public comments submitted pursuant to 10 CFR 20.1405. These alternate criteria are expected to be used only in very rare cases.

To date, the three NRC-licensed facilities (Shoreham, Fort St. Vrain, and Pathfinder) that have completed the decommissioning process have had their licenses terminated, allowing unrestricted use of the sites. License termination plans have been submitted for three other facilities. The LTPs describe plans for unrestricted use of the sites following license termination. No nuclear power licensees have indicated that they plan for restricted use of the site after license termination.

A proposed rule was issued on September 4, 2001 (66 FR 46230) for partial site release prior to license termination. Partial site release means release of part of a nuclear power reactor facility or site for unrestricted use prior to NRC approval of the LTP. The NRC proposes to add a new section to 10 CFR Part 50, separate from the existing rules for decommissioning and radiological criteria for license termination, that identifies the requirements and criteria necessary for partial site release. The proposed rule includes associated amendments to 10 CFR Part 2 and 10 CFR Part 20. The purpose of this rulemaking is to ensure that any remaining residual radioactive material from licensed activities on a portion the site released for unrestricted use will meet the radiological criteria for license termination.

Licensees will be required to submit information necessary to demonstrate the following:

- The release of radiologically impacted property complies with the radiological criteria for unrestricted use in 10 CFR 20.1402 (0.25 mSv/yr [25 mrem/yr] to the average member of the critical group and ALARA).

- The licensee will continue to comply with all other applicable regulatory requirements that may be affected by the release of property and changes to the site boundary. This would include, for example, requirements in 10 CFR Parts 20, 50, 72, and 100.

- Records of property-line changes and the radiological conditions of partial site releases are being maintained to ensure that the dose from residual material associated with these releases can be accounted for at the time of any subsequent partial releases and at the time of license termination.

The proposed rule provides additional flexibility to licensees who are releasing property that has never been radiologically impacted. While an amendment of the Part 50 operating license is required to release radiologically impacted property, the proposed rule offers the opportunity for a letter submittal for partial releases if the licensee can demonstrate that there is no reasonable potential for residual radioactivity from license activities.

2.3 References

10 CFR 2. Code of Federal Regulations, Title 10, *Energy*, Part 2, "Rules of practice for domestic licensing proceedings and issuance of orders."

10 CFR 20. Code of Federal Regulations, Title 10, *Energy*, Part 20, "Standards for protection against radiation."

10 CFR 50. Code of Federal Regulations, Title 10, *Energy*, Part 50, "Domestic licensing of production and utilization facilities."

10 CFR 61. Code of Federal Regulations, Title 10, *Energy*, Part 61, "Licensing requirements for land disposal of radioactive waste."

10 CFR 72. Code of Federal Regulations, Title 10, *Energy*, Part 72, "Licensing requirements for the independent storage of spent nuclear fuel high-level radioactive waste and reactor-related greater-than-Class-C waste."

10 CFR 100. Code of Federal Regulations, Title 10, *Energy*, Part 100, "Reactor site criteria."

61 FR 39278. "Decommissioning of Nuclear Power Reactors. Final Rule." *Federal Register.* July 29, 1996.

64 FR 39058. "Radiological Criteria for License Termination. Final Rule." *Federal Register.* July 21, 1997.

66 FR 46230. "Releasing Part of a Power Reactor Site or Facility for Unrestricted Use Before the NRC Approves the License Termination Plan. Proposed Rule." *Federal Register.* September 4, 2001.

U.S. Nuclear Regulatory Commission (NRC). 1988. *Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities.* NUREG-0586, NRC, Washington, D.C.

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 shut down: 19 of these reactors are in various stages of decommissioning, and reactors at 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. The material presented in this section is also provided as background information for the reader.

(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.

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 9700 ha (24,000 ac) for the Turkey Point Plant in Florida (two 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). Fermi, Unit 1, is currently performing the decontamination and dismantlement phase of SAFSTOR (see Section 3.2). Peach Bottom, Unit 1, is 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).

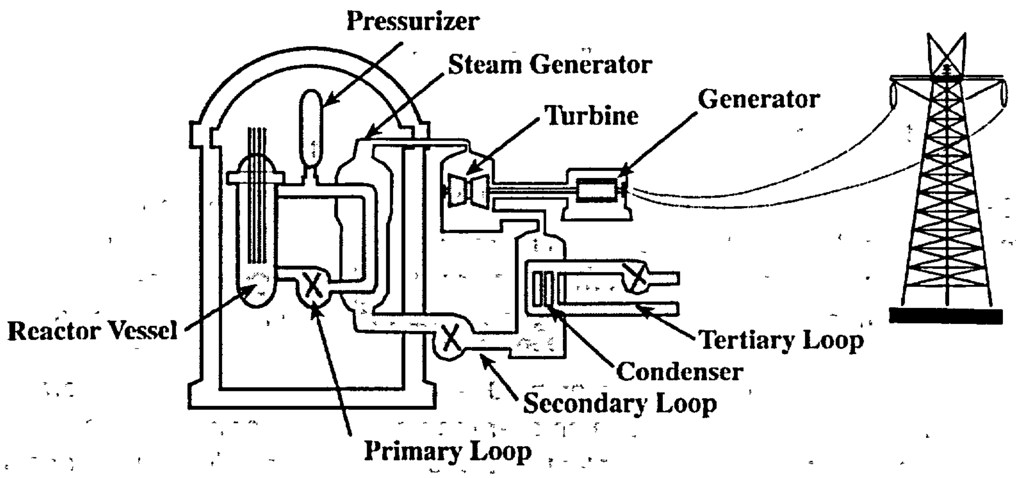

Figure 3-1. Pressurized Water Reactor

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).

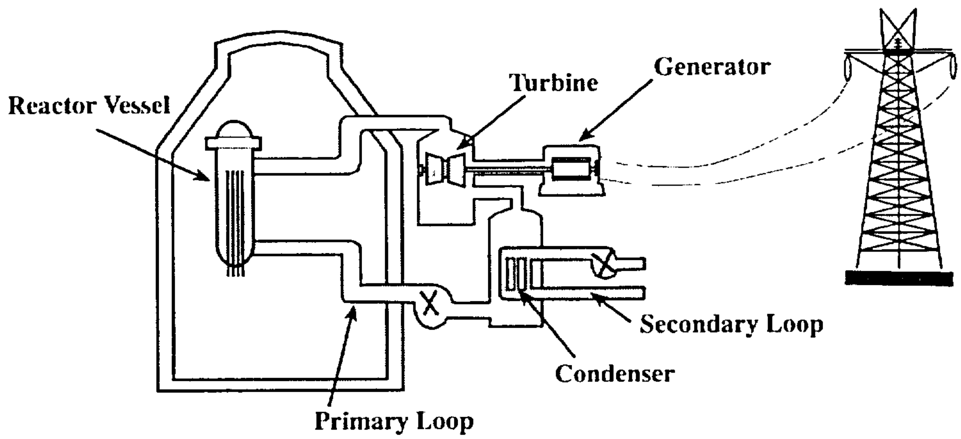

Figure 3-2. Boiling Water Reactor

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 Fermi, Unit 1, FBR used the fissile isotope of uranium as fuel. 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. Depending on design, 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.

At this time, there are no commercial 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 steel-domed structure that contains the upper end of the reactor vessel and the fuel-handling equipment. Below ground there is considerable concrete shielding.

  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 feedwater 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. Primary coolant is not circulated through the turbine building systems in PWRs. However, it is not unusual for portions of the turbine building to become mildly 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 August 21, 2002, there were 23 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, Dresden, Hatch, McGuire, Oyster Creek, Peach Bottom, Yankee Rowe, Fitzpatrick, Rancho Seco, Maine Yankee, and U.S. Department of Energy (DOE [TMI-2 fuel debris]) at Idaho National Engineering and Environmental Laboratory.

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, seek a site-specific 10 CFR Part 72 license for the ISFSI, 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 request for a site-specific 10 CFR Part 72 license.

- Switchyard: 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.

- Administrative, training, and security buildings: 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. The systems described below are typical and may differ at specific facilities.

- Cooling and auxiliary water systems: 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) or similar document, 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 (EPA) and administered by EPA, or if authorized, 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.

- Miscellaneous mechanical systems: 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.

- Instrumentation and control systems: 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 HLW 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 HLW 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 HLW and spent fuel generated up to that time.

Until a HLW 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.

- Transportation systems: 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, trains, barges, or other ships 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. Most of the contamination in the reactor coolant system is from the activation of corrosion products and not from leaking fuel.

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 (or portions) that have been activated 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, both of structures as well as corrosion 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. The determination of SAFSTOR includes those activities necessary for the final decontamination and dismantlement of the facility. 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. The definition of SAFSTOR also includes the decontamination and dismantlement of the facility at the end of the storage period.

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 (PSDAR). 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 (excluding the onsite storage of spent fuel), 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.

<u>During storage</u>:

- 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 2000b), 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. NRC published an Advance Notice of Proposed Rulemaking (ANPR) on October 16, 2001 (66 FR 52551) seeking stakeholder input on three proposed regulatory options and whether entombment was a viable decommissioning alternative. The ANPR comment period closed on December 31, 2001. NRC received 19 comments from: six States; eight licensees; the Nuclear Energy Institute (NEI); the U.S. Environmental Protection Agency (EPA); the Conference of Radiation Control Program Directors' E-24 Committee on Decommissioning and Decontamination (CRCPD E-24 Committee); the Southeast Compact Commission (SCC); and a private individual.

Generally, the eight utilities and NEI stated that they would have entombment available as a decommissioning option; however, none unequivocally committed to using entombment for their decommissioning process. Some Agreement State commenters endorsed the 10 CFR Part 20 dose limits, with one State adding that a time limit to reach the dose rates should be considered. Although one State advocated extending the decommissioning period beyond 60 years, most were silent on the decommissioning regulations in 10 CFR Part 50. The staff notes that there was no consensus on a preferred option. NRC staff has considered the comments received and has prepared a paper transmitting the staff's recommendations to the Commission. As of the date of this publication the Commission has not acted on the staff's recommendations.

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 ISFSI. Licensees choosing ENTOMB will adapt the entombment option to fit their specific site requirements.

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 be 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, may 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. The resulting structure would be sealed and made weather tight and covered with an 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
- potentially 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 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.

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.

Eleven 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; Yankee Rowe; 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 3-1. 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.

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 shut down 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 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, 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

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 ISFSI
Fermi, Unit 1	FBR	200 MW	09/22/72	SAFSTOR[c]	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[d]
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 ISFSI/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[d]
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 in 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.
(d) Licensee is in process of transferring fuel to dry storage in onsite ISFSI.

auxiliary piping. At a second facility evaluated, a chemical decontamination was considered necessary to keep doses within previously issued EAs. 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 reduce the scope of the required safeguards and security systems to only the fuel storage facilities and isolate the spent fuel so decontamination and dismantlement can proceed on the balance of the facility without the potential for affecting the spent fuel. 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 concrete 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.

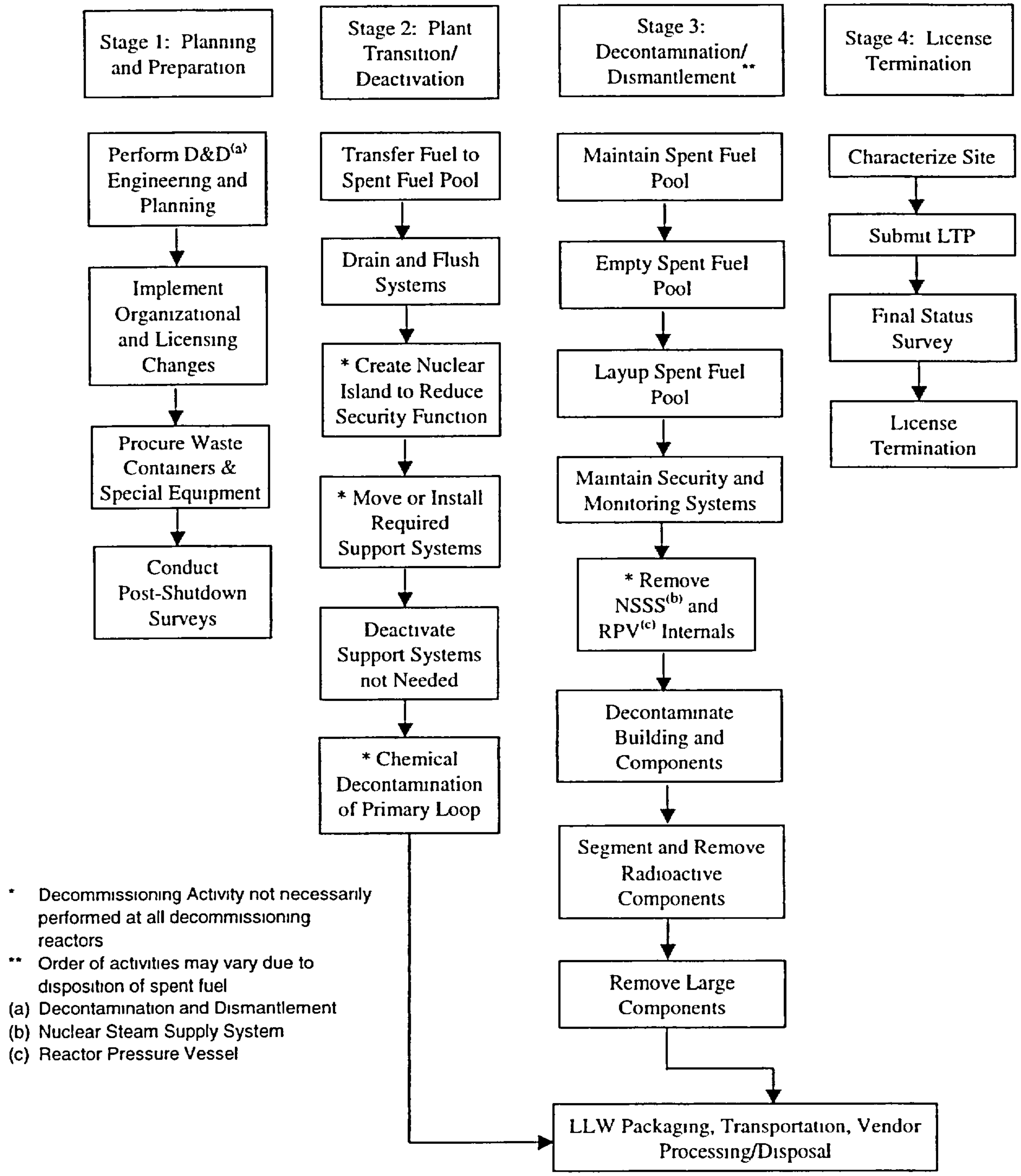

Figure 3-3. Reactor Decommissioning Process - DECON or SAFSTOR