REGULATORY ANALYSIS FOR JAPAN LESSONS-LEARNED TIER 3 ISSUE ON EXPEDITED TRANSFER OF SPENT FUEL

U.S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation

FOREWORD

On March 11, 2011, the Great Tōhoku earthquake and subsequent tsunami in Japan resulted in significant damage to the site of the Fukushima Dai-ichi nuclear power station. The spent fuel pools and the used fuel assemblies stored in the pools remained intact at the plant. Even so, the event led to questions about the safe storage of spent fuel. In a memorandum to the Commission entitled, "Updated Schedule and Plans for Japan Lessons Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," dated May 7, 2013 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13105A122), the staff outlined a plan for evaluating whether the U.S. Nuclear Regulatory Commission (NRC) should undertake a regulatory action to require the expedited transfer of spent fuel from pools to dry cask storage containers at U.S. nuclear power plants.

To determine if additional studies are needed to further assess potential regulatory action on expedited transfer, the staff has performed this regulatory analysis. The staff assessed the potential safety benefits by using the Commission's 1986 Safety Goal Policy Statement (Ref. 4). Then, to provide additional information to support the Commission's deliberations, the staff performed a cost-benefit analysis. The staff concluded that requiring the expedited transfer of spent fuel would provide only a minor or limited safety benefit (i.e., below the safety goal screening criteria), and that its expected implementation costs would not be warranted. The results of this analysis support the staff recommendation that the NRC conduct no further generic assessments on expedited transfer, and that this Tier 3 Japan lessons learned activity be closed. The NRC staff continues to believe, based on this analysis and previous studies that spent fuel pools provide adequate protection of public health and safety.

EXECUTIVE SUMMARY

The NRC evaluates within this analysis whether additional study of expedited transfer of spent fuel from spent fuel pools (SFPs) (i.e., expedited transfer) to dry cask storage might be warranted. This analysis was undertaken to support development of a technical basis for the program plan described in a memorandum to the Commission, "Updated Schedule and Plans for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," dated May 7, 2013 (Ref. 1). In the memorandum, the staff outlined a three-phase plan for evaluating if regulatory action should be pursued to require licensees to expedite transfer of spent fuel from SFPs to dry cask storage. The program plan calls for preparing this analysis under Phase 1 to help determine if additional study is warranted. If the results of Phase 1 indicate that additional study is warranted, Phases 2 and 3 of the program plan would be conducted to refine assumptions used in the analyses to determine whether any regulatory action is warranted. The Phase 1 screening analysis is documented in this regulatory analysis, and considers the results of the SFP study (SFPS) (Ref. 2), along with previous studies. For this analysis, the NRC evaluated the merits of additional research by comparing the status quo to a scenario in which expedited transfer would be required.

The SFPS provides consequence estimates of a hypothetical SFP accident initiated by a low likelihood seismic event at a reference plant for both a fully loaded (high-density) and minimally loaded (low-density) SFP. The SFPS contributed to the resolution of this Tier 3 issue by providing a measure of the change in potential consequences resulting from a change in spent fuel storage density for a reference plant. The staff completed a regulatory analysis in Appendix D of the SFPS, which indicates that expediting movement of spent fuel for the reference plant would provide only a minor or limited safety benefit, and that this benefit would be outweighed by the expected implementation costs. The staff's analysis herein expands the regulatory analysis in the SFPS by covering SFP designs used in the operating and decommissioned reactors in the United States.

To determine if additional studies are needed to further assess potential regulatory action on expedited transfer, the staff conducted a two-part analysis of expedited transfer. The staff first assessed the potential safety benefits by using the Commission's 1986 Safety Goal Policy Statement (Ref. 4). Although the regulatory analysis guidelines would normally allow the staff to stop the evaluation upon finding that the proposed action does not provide a sufficient safety enhancement to meet the threshold of the safety goal screening, the staff proceeded to perform a cost-benefit analysis to provide additional information for the Commission's consideration.

Whereas the SFPS addressed the consequences of a selected event at a reference plant, this analysis is expanded to consider a variety of possible initiating events and to determine whether expedited spent fuel transfer may be warranted at SFPs across the U.S. fleet of nuclear power plants and independent wet spent fuel storage facilities. The staff accounted for the differences in the SFPs by categorizing them into several groups with similar properties. The categorization process is further described in Section 4.1.1 of this regulatory analysis. The staff used conservative values for parameters in the base case analysis to ensure that effects of design, operational and other site variations among the licensed reactor fleet were encompassed. The base case was supplemented with low and high sensitivity calculations to address uncertainties in the analysis.

To the extent practicable, the staff used conservative estimates and assumptions to bound the variations in SFP parameters across the fleet for this analysis. This analysis determines

whether regulatory action may be appropriate, or whether additional generic studies are needed. In accordance with Phases 2 and 3 of the program plan, if the Commission directs additional studies, then the staff would refine the conservative assumptions used in this regulatory analysis to increase realism, and consider additional factors such as the risks associated with the transfer of spent fuel assemblies to casks, and storage of the casks in the associated storage facilities. These risks were not included in this study so as to bias the results in favor of taking regulatory action. The staff's judgment is that these refinements would likely reduce the benefit associated with expedited transfer, resulting in a more negative cost-benefit assessment.

The staff used the U.S. Geological Survey (USGS) 2008 model to evaluate seismic hazards at central and eastern U.S. (CEUS) nuclear power plant sites in this analysis. Although the USGS model considers sites in the western United States (including Columbia, Diablo Canyon, Palo Verde, and San Onofre), the staff has not performed the necessary analyses for these sites to include them in this analysis. Considering the robust designs of SFPs, especially in more seismically active areas in the western United States, the staff concludes that public health and safety are adequately protected. Upon completion of the Near-Term Task Force Recommendation 2.1 seismic reevaluation, the staff will confirm that the seismic risk for SFPs is consistent with the risk assumed in this analysis.

This analysis and the supporting references, in general, do not include events caused by sabotage. For nuclear power plants, security requirements are established to provide high assurance of adequate protection from radiological sabotage of the nuclear power plant reactor and SFP. The NRC continually monitors threat conditions and, as was done after the September 11, 2001 attacks, makes adjustments, as appropriate in the governing security requirements and in actions to oversee their effective implementation. Based on the staff's view that security issues are effectively addressed in the existing regulatory program, they are not part of this analysis.

In this analysis, the risks associated with a severe SFP accident at the plants studied are compared to the Safety Goal Policy Statement (Ref. 4) to determine if requiring the expedited transfer of spent fuel to dry cask storage would provide more than a minor safety benefit. Despite the large releases for some low probability accident progressions analyzed, the projected consequences indicate that there are no offsite early fatalities from acute radiation effects. The analysis also shows that the risk of an individual dying from cancer from the radioactive release is less than 0.76% of the Commission's Quantitative Health Objective of two in one million ($2x10^{-6}$) per year. The risks are similar between different spent fuel loading and mitigation scenarios because of modeled offsite protective actions that include evacuation, sheltering, relocation, and decontamination. Additionally, these individual risks are dominated by long-term exposures to very lightly contaminated areas for which doses are small enough for the areas to be considered habitable.

In addition, the staff conducted a cost-benefit analysis, which finds that the added costs involved with expedited transfer of spent fuel to dry cask storage to achieve the low-density SFP storage alternative are not warranted in light of the benefits from such expedited transfer. The combination of high estimates for important parameters assumed in some of the sensitivity cases presented in this analysis result in large economic consequences, such that, the calculated benefits from expedited transfer of spent fuel to dry cask storage for those cases outweigh the associated costs. However, even in these cases, there is only a limited safety benefit when using the QHOs and the expected implementation costs would not be warranted. In addition, in the staff's judgment, the various assumptions made in the analysis of the "base

case” result in an overall cost-benefit assessment that is appropriately conservative for a generic regulatory decision and justify using the “base case” as the primary basis for the staff’s recommendation. Based on the generic assessment and the other considerations detailed in this analysis, the staff finds that additional studies are not needed to reasonably conclude that the expedited transfer of spent fuel to dry cask storage would provide only a minor or limited safety benefit (i.e., below the safety goal screening criteria), and that its expected implementation costs would not be warranted.

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ABBREVIATIONS AND ACRONYMS

ac	alternating current
ADAMS	Agencywide Documents Access and Management System
BLS	Bureau of Labor Statistics
BWR	boiling-water reactor
CEUS	central and eastern United States
CFR	*Code of Federal Regulations*
CoC	certificate of compliance
CPI-U	consumer price index—all urban consumer inflator
Cs	cesium
DOE	U.S. Department of Energy
DSC	dry storage cask systems
EPA	U.S. Environmental Protection Agency
EPRI	Electric Power Research Institute
FR	*Federal Register*
FTE	full-time equivalent
GMPE	ground motion prediction equation
GWd	gigawatt-day
ISFSI	independent spent fuel storage installation
LCF	latent cancer fatality
LNT	linear no-threshold
LOOP	loss of offsite power
MACCS2	MELCOR Accident Consequence Code System, Version 2
MELCOR	(not an acronym)
MTU	metric ton heavy metal or metric ton uranium
MW_t	megawatt thermal
NGO	non-government organizations
NPV	net present value
NRC	Nuclear Regulatory Commission
NTTF	Near-Term Task Force
OCP	operating cycle phase
OMB	Office of Management and Budget
ORIGEN	(not an acronym)
PAG	protective action guides
PGA	peak ground acceleration
PRM	petition for rulemaking
PSHA	probabilistic seismic hazard assessment
PWR	pressurized water reactor
RA	regulatory analysis
SCALE	(not an acronym)

SFP	spent fuel pool
SOARCA	State-of-the-Art Reactor Consequence Analyses
SRM	staff requirements memorandum
USGS	U.S. Geological Survey
VSL	value of a statistical life

1. INTRODUCTION

The NRC evaluates within this regulatory analysis whether additional study of expedited transfer of spent fuel from spent fuel pools (SFPs) (i.e., expedited transfer) to dry cask storage might be warranted. The NRC evaluated the merits of additional research by comparing the status quo to one in which expedited transfer would be required. The staff assessed the potential safety benefits of requiring expedited transfer by using the Commission's 1986 Safety Goal Policy Statement (Ref. 4). Then, to provide additional information to support the Commission's deliberations, the staff performed a cost-benefit analysis of requiring expedited transfer. This work was conducted in accordance with the program plan described in a memorandum to the Commission, "Updated Schedule and Plans for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," dated May 7, 2013 (Ref. 1).

In conducting the analyses described herein, the staff considered the results of the Spent Fuel Pool Study (SFPS) (Ref. 2) along with previous studies and operating experience. The SFPS analyzed the risks and consequences of postulated spent fuel pool accidents for a reference plant (a General Electric (GE) Type 4 boiling-water reactor (BWR) with a Mark I containment). Since seismic events dominate SFP damage risk, seismic events were modeled. The other risk contributors, such as equipment failures and human errors, were derived from previous studies and were factored into the analysis. Mechanistic modeling was applied to develop the source term for the SFP accident since it differs from that associated with severe core damage accidents. The consequences of a SFP accident, which results in the loss of cooling or the loss of pool water inventory and a radiological release, are dominated by the long-lived isotopes, such as cesium. The results of the SFPS showed that the overall level of safety with respect to spent fuel storage in a SFP currently achieved at the reference plant is high and that the level of risk at the reference plant is very low. The staff therefore found that adequate protection is assured. Additionally, the SFPS included a regulatory assessment that considered various initiating events and concluded that the incremental safety benefit associated with expedited transfer of spent fuel at the reference plant was minor, far from the threshold that the NRC uses to inform its decisionmaking, and was also not warranted in light of the added costs involved with expediting the movement of spent fuel from the pool to achieve low-density fuel pool storage. The regulatory analysis is included in Appendix D of the SFPS. The results of the SFPS are consistent with earlier research conducted over the last several decades, as summarized in NUREG 1353, "Regulatory Analysis for the Resolution of Generic Issue 82, Beyond Design Basis Accidents in Spent Fuel Pools," dated April 1989; in NUREG/CR 6451, "A Safety and Regulatory Assessment of Generic BWR and PWR [pressurized-water reactor] Permanently Shutdown Nuclear Power Plants," dated April 1997, and in NUREG 1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," dated February 2001.

The SFPS was an important input to this analysis but is not the sole technical study or basis for the following analysis and related findings. The SFPS addressed the consequences of a selected seismic event that could result in the loss of SFP integrity at a reference plant. The staff's analysis herein expands the regulatory analysis in the SFPS by covering SFP designs used in the operating and decommissioned reactors in the United States (as used throughout this document, the operating reactor fleet includes the recently licensed but not yet operating AP1000 plants).

This Tier 3 analysis assesses whether the proposed expedited spent fuel transfer alternative would have more than a minor safety benefit, and in doing so the staff uses the quantitative

health objectives (QHOs). The QHOs are used as a surrogate for the safety goal as outlined in the Commission's Safety Goal Policy Statement (Ref. 4). A further discussion of the basis and background for using the QHOs in assessing SFP accidents is included in Section 3 of this regulatory analysis. The staff relied on information from past studies, the recently completed SFPS, and operating experience in conducting this analysis.

To determine if additional studies are needed to further assess whether expedited transfer should be required, the staff conducted a two-part analysis. The staff first assessed the potential safety benefits of requiring expedited transfer using the Commission's 1986 Safety Goal Policy Statement to conduct a safety goal screening evaluation. Although the agency's guidance would normally allow the staff to stop the evaluation upon determining that the proposed action does not provide a sufficient safety enhancement to meet the threshold of the safety goal screening, the staff proceeded to perform a cost benefit analysis (summarized below) to provide the Commission additional information.

In addition to safety benefits, the staff's cost-benefit analysis considers wider societal measures, such as averted offsite property damage. The staff developed estimates of benefits and costs, which are quantified, when possible, together to conclude whether requiring expedited transfer would be cost-beneficial[1].

Within this cost-benefit analysis, the staff developed a base case that generally used conservative assumptions for key parameters such as conditional probabilities of pool failures and zirconium fires to increase the calculated net benefits of the expedited transfer of spent fuel alternative for each SFP grouping and to generally bound the parameters that vary among spent fuel pools. The benefits calculated for these base case evaluations provide only a minor or limited safety benefit that is far from the threshold that the NRC uses to inform its regulatory decisionmaking. In addition, the benefits calculated for the base case evaluations are less than the estimated costs for expedited transfer of spent fuel. There are some plants that for a particular parameter are not bounded by the base case. However, the amount of conservatism used in the other parameters overwhelm the slight non-conservatism in the particular outlying parameter. Therefore, the overall results of the base case is conservative for all plants. This analysis approach greatly simplifies the analysis and precludes the need to model each plant in detail. To provide additional information for the Commission's consideration, the staff also analyzed additional cases where the key input parameters are varied to provide a low to high estimate of the calculated benefits. In addition, to identify the specific effect of certain parameters, the staff performed sensitivity studies where only one parameter was varied from a low to high value. Sensitivity studies were conducted on key factors such as the dollars per person-rem conversion factor and consideration of consequences beyond 50 miles (80 kilometers) to measure each attribute's effect upon the overall result. The sensitivity of the dollars per person-rem conversion factor is important because it provides the Commission with additional information to inform regulatory decisionmaking. The cost-benefit analysis used key insights from operating experience and the recent SFPS, such as the plant damage state for seismic events, probability of a release for specific pool damage states, and the expected amount and type of radioactive material released.

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[1] Cost-beneficial means that the benefits of the proposed action are equal to, or exceed, the costs of the proposed action.

1.1 Statement of the Problem

The federal government's decision to stop work on a deep geologic repository at Yucca Mountain, and the events in Japan following the March 2011 earthquake, have rekindled public and industry interest in understanding the consequences from postulated accidents associated with high-density SFP storage, and the relative benefits of low-density SFP storage. In response to these events, as discussed in a memorandum to the Commission, "Updated Schedule and Plans for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel" (Ref. 1), the staff determined that it should confirm whether high-density SFP configurations continue to provide adequate protection and assess whether any safety benefits (or detriments) would occur in requiring the expedited transfer of spent fuel to dry cask storage.

U.S. nuclear power plants store spent fuel in pools for varying periods of time using a high-density configuration. Various risk studies (such as NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants," February 2001 (Ref. 5)) have shown that storage of spent fuel in a high-density configuration in SFPs is safe, and that the risk of accidental release of a significant amount of radioactive material to the environment is low. These studies used simplified and sometimes bounding assumptions and models to characterize the likelihood and consequences of beyond-design-basis SFP accidents.[2] As part of the NRC's post-9/11 security assessments, SFP modeling using detailed thermal-hydraulic and severe accident progression models integrated into the MELCOR code were developed and applied to assess the realistic heatup of spent fuel under various pool draining conditions. Moreover, in conjunction with these post-September 11 security assessments, the NRC in 2009 issued 10 CFR 50.54(hh)(2) (Ref. 6) as a final rule, which requires reactor licensees to develop and implement strategies intended, in part, to maintain or restore SFP cooling capabilities in the event of explosions or fires caused by beyond-design-basis events.

The NRC had previously restated its views on the safety of spent fuel stored in high-density configurations in a response to Petition for Rulemaking (PRM)-51-10 (Ref. 7) and PRM-51-12 (Ref. 8) (73 FR 46204, August 8, 2008), and in revising NUREG-1437, "Generic Environmental Impact Statement for License Renewal of Nuclear Plants, Draft Report for Comment" (Ref. 9). However, the NRC's position relies, in part, on the findings of the aforementioned security assessments, which are not publicly available.

1.2 Overview of the Safety Goal Screening Evaluation

As part of the NRC staff's regulatory analysis, the risks associated with a severe SFP accident at the plants studied are compared to the Commission's 1986 Safety Goal Policy Statement (Ref. 4) to determine if requiring the expedited transfer of spent fuel to dry cask storage would provide more than a minor safety benefit. Despite the large releases for some low probability accident progressions analyzed, the projected consequences indicate there are no offsite early fatalities from acute radiation effects. The analysis also shows that the risk of an individual dying from cancer from the radioactive release is less than 0.76% of the Commission's QHO of two in one million ($2x10^{-6}$) per year. The risks are similar between different spent fuel loading and mitigation scenarios because of modeled offsite protective actions that include evacuation, sheltering, relocation, and decontamination. Additionally, these individual risks are dominated

[2] An overview of previous studies is provided in section 10.2 to the SFPS (Ref. 2).

by long-term exposures to very lightly contaminated areas for which doses are small enough for the areas to be considered habitable. The QHOs are used as a surrogate for the safety goal as outlined in the Commission's 1986 policy statement. Section 3 below discusses the safety goal screening evaluation in more detail.

1.3 Overview of the Cost-Benefit Analysis

This analysis uses information contained in the SFPS for its structural analysis and related damage characterization, its accident progression analysis, and its offsite consequences analysis. These results are supplemented with results from previous studies and conservative assumptions in the cost-benefit analysis to broaden the assessment to generically address the SFP risk at multiple facilities.

This analysis calculates the potential benefit per reactor year resulting from expedited fuel transfer by comparing the safety of high-density fuel pool storage relative to low-density fuel pool storage and related alternatives. The comparison uses the initiating frequency and consequences from the SFPS as an indicator of any changes in the NRC's understanding of safe storage of spent fuel following a beyond-design-basis seismic event. The staff also used calculated results from previous SFP studies (i.e., NUREG-1353 and NUREG-1738) to extend the applicability of this evaluation to include other initiators, which could challenge SFP cooling or integrity and incorporated inputs representing the range of U.S. SFP characteristics to extend the analysis applicability to SFPs within other U.S. reactor designs.

Within this cost-benefit analysis, the staff developed a base case that generally used conservative assumptions for key parameters such as conditional probabilities of pool failures and zirconium fires to increase the calculated net benefits of the expedited transfer of spent fuel alternative for each SFP grouping and to generally bound the parameters that vary among spent fuel pools. The benefits calculated for these base case evaluations provide only a minor or limited safety benefit that is far from the threshold that the NRC uses to inform its regulatory decisionmaking. In addition, the benefits calculated for the base case evaluations are less than the estimated costs for expedited transfer of spent fuel. There are some plants that for a particular parameter are not bounded by the base case. However, the amount of conservatism used in the other parameters overwhelms the slight non-conservatism in the particular outlying parameter. Therefore, the overall results of the base case are conservative for all plants. This analysis approach greatly simplifies the analysis and precludes the need to model each plant in detail. To provide additional information for the Commission's consideration, the staff also analyzed additional cases where the key input parameters are varied to provide a low to high estimate of the calculated benefits. In addition, to identify the specific effect of certain parameters, the staff performed sensitivity studies where only one parameter was varied from a low to high value. Section 4 below discusses the staff's cost-benefit analysis in more detail.

2. ANALYSIS OF IDENTIFIED ALTERNATIVE

The U.S. Nuclear Regulatory Commission (NRC) considered the regulatory baseline and one alternative to change this baseline as discussed below. The baseline is used to estimate the incremental costs of the alternative.

2.1 Regulatory Baseline—Maintain the Existing Spent Fuel Storage Requirements

The baseline would be maintained if the Commission decides not to require the expedited transfer of spent fuel from pools to dry cask storage, but to continue with the NRC's existing licensing requirements for spent fuel storage. Spent fuel must now be moved into dry cask storage only as necessary to accommodate fuel assemblies being removed from the core during refueling operations. Fuel storage in the spent fuel pool (SFP) is managed to maintain sufficient empty space in the pool for removal of one full core of reactor fuel in case of emergencies (referred to as full core discharge) or other operational contingencies. The NRC also assumes in this analysis that all applicable requirements and guidance to date have been implemented, there are no unevaluated degraded or nonconforming conditions, and no implementation is assumed for related generic issues or other staff requirements or guidance that is unresolved or still under review.

The baseline condition is the storage of spent fuel in high-density racks[3] in the SFP, a relatively full SFP, and compliance with all current regulatory requirements. The regulatory requirements include design features intended to prevent a substantial loss in water inventory under accident conditions and those requirements for emergency abnormal conditions associated with the following[4]:

- Title 10 of the *Code of Federal Regulations* (10 CFR) 50.54(hh)(2) (Ref. 6) with respect to spent fuel configuration and SFP preventive and mitigative capabilities

For the purpose of evaluating the potential benefits of expedited transfer of spent fuel to dry cask storage, this analysis used a conservative approach by crediting successful mitigation for the low-density SFP alternative and assumed no successful mitigation for the high-density SFP storage regulatory baseline. Furthermore, because SFPs have limited available storage, even after licensees expanded their storage capacity using high-density storage racks, the current practice of transferring spent fuel to dry storage in accordance with 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste," (Ref. 12) is assumed to continue.[5]

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[3] Most nuclear power plant SFPs were originally designed for temporary storage of spent fuel. Starting in the 1980s, most pools were "re-racked" to use hardware that stores the assemblies in a more closely spaced arrangement, thus allowing the storage of more assemblies in a high-density configuration.

[4] The following regulatory requirements apply to operating power reactors considered in this analysis.

[5] Maintenance of the existing SFP storage requirements would not limit the Commission's authority to add new requirements or update regulatory guidelines, as necessary. These actions and activities are a part of the regulatory baseline. However, these activities would be pursued as separate regulatory actions to resolve particular technical issues. In the baseline case, the NRC would take no

The NRC has required through orders that licensees enhance their ability to respond to beyond-design-basis events. The additional capabilities to do so were not quantitatively considered in this analysis. The orders include:

- Order EA-12-049 (Ref. 10) that requires licensees to develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and SFP cooling capabilities following a beyond-design-basis external event

- Order EA-12-051 (Ref. 11) that requires licensees to install reliable means of remotely monitoring wide-range SFP levels to support effective prioritization of event mitigation and recovery actions in the occurrence of a beyond-design-basis external event

2.2 Expedited Transfer Alternative—Low-Density Spent Fuel Pool Storage

This proposed alternative would require older spent fuel assemblies[6] to be expeditiously moved from SFP storage to dry cask storage beginning in year 2014, to achieve and maintain a low-density loading of spent fuel in the existing high-density racks as a preventive measure. Because of the low-density SFP loading, this alternative has less long-lived radionuclide inventory in the SFP, a lower overall heat load in the pool, and a slight increase in the initial water inventory that displaces the removed spent fuel assemblies.

Because of the uncertainty over the availability of a spent fuel repository, many plants have plans to establish onsite storage capacity (in-pool capacity and dry storage) sufficient to store all of the spent fuel discharged over the operating life of the plant until repository capacity becomes available. As of early 2013, all but 5 of the 65 U.S. sites with operating nuclear power reactors had either built or were seeking licenses to build dry storage facilities (Ref. 19).

Recently, some non-government organizations (NGOs) concerned about the hazards of nuclear power indicated preference for onsite dry storage instead of reprocessing or central storage. Those NGOs have also called for spent fuel to be placed in onsite dry casks after, at most, five years of cooling in spent-fuel pools.

There are cost and risk impacts associated with the transfer of spent fuel from the SFP to cask storage and during long-term cask storage.[7] These cost and risk impacts reduce the overall net benefit of this alternative in relation to the regulatory baseline. However, the added risks of handling and moving casks were conservatively not included in this analysis to maximize the delta benefit of the expedited transfer alternative.

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action to require facilities to expedite the movement of spent fuel to achieve low-density loading in the SFP.

[6] Older spent fuel assemblies are those that have been placed in the SFP to cool for at least five years after discharge from the reactor core.

[7] EPRI report TR-1021049 (Ref. 17) assesses the cost and risk impacts from a worker dose perspective associated with transfer of spent nuclear fuel from SFPs to dry storage after five years of cooling. The report concludes that expedited fuel movement would result in an increase cost to the U.S. nuclear industry of $3.6 billion, with the increase primarily related to the additional capital costs for new casks and construction costs for the dry storage facilities.

3. SAFETY GOAL SCREENING EVALUATION

The Commission has directed that NRC's regulatory actions affecting nuclear power plants be evaluated for conformity with NRC's Policy Statement on Safety Goals for the Operations of Nuclear Power Plants (Ref. 4). The Safety Goal Policy Statement sets out two qualitative safety goals and two quantitative objectives. Both the goals and objectives apply only to the risks to the public from the accidental or routine release of radioactive materials from nuclear power plants.

The two qualitative safety goals are as follows:

(1) Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health.

(2) Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The following quantitative health objectives are to be used in determining achievement of the above safety goals:

(1) The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed 1/10 of 1 percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

(2) The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed 1/10 of 1 percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

An important part of the implementation of the policy statement is its incorporation into the NRC's processes for evaluating possible changes in regulations or other requirements imposed on licensees. Within the NRC's Regulatory Analysis Guidelines, the safety goal screening evaluation is designed to answer when a regulatory requirement should not be imposed generically on nuclear power plants because the residual risk is already acceptably low. This evaluation is intended to eliminate some proposed requirements from further consideration independently of whether they could be cost-beneficial. Note that performing a safety goal screening evaluation requires judgment by the NRC staff and Commission as to whether the evaluation provides an unreasonable finding on whether a proposed action provides more than a marginal safety improvement.

The Safety Goals for the Operation of Nuclear Power Plants: Policy Statement defines the early fatality area calculation as that within 1.6 kilometers (1 mile) from the site boundary. The prompt fatality QHO represents a $5x10^{-7}$ per year objective for an average individual within 1 mile ("Safety Goals for Nuclear Power Plant Operation," NUREG-0880, Rev. 1, issued May 1983.) (Ref. 14)

The second quantitative objective of the policy relates to ensuring that the cancer fatality risks from nuclear power plant operations remain a small fraction of the overall cancer risks from all

causes. The cancer fatality QHO represents a $2x10^{-6}$ per year objective for an average individual within 16 kilometers (10 miles) (NUREG-0880). The staff assessed the criteria based on recent data (http://www.cancer.org/research/cancerfactsfigures/index), and found that the total fatality rate from cancer in the United States is 580,350 per 315,747,500 persons (http://www.census.gov/popclock/) or a risk of $1.84x10^{-3}$ per year. 1/10 of 1 percent of this value results in a safety goal of $1.84x10^{-6}$ per year (i.e., little changed from the value in NUREG-0880).

Using the bounding frequency of damage to the spent fuel of $3.46x10^{-5}$ per year[8], which considers all initiators that could challenge SFP cooling or integrity, and the estimates from the SFPS for conditional individual latent cancer fatality risk within a ten-mile radius of $4.4x10^{-4}$ yields a conservative high estimate of individual latent cancer fatality risk of $1.52x10^{-8}$ cancer fatalities per year. This calculated value of $1.52x10^{-8}$ individual latent cancer fatality risk per reactor-year associated with a SFP accident is less than one percent of the $1.84x10^{-6}$ per year societal risk goal value based on the calculation area specified in the Safety Goal Policy Statement.[9] The factors leading to this low likelihood, as discussed above, are summarized in Figure 1.

Comparing the results of this analysis to the NRC Safety Goal Policy Statement involves important limitations.

(1) First, the safety goal is intended to encompass all accident scenarios on a nuclear power plant site, including those involving reactors and spent fuel. This analysis does not examine reactor scenarios that would need to be considered, although the analysis does consider the most important contributors to SFP risk. As a result, comparison of the calculated individual latent cancer fatality (LCF) risk to the NRC Safety Goal Policy Statement is incomplete. However, it is intended to show that SFP risk is less than one percent of the individual LCF risk that corresponds to the overall or total safety goal for latent cancer fatalities for a nuclear power plant site. It is unlikely that the additional reactor accident scenarios would contribute significantly to overall risks and introduce significant challenges to the Commission's Safety Goal Policy Statement.

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[8] See Table 43 in Appendix C for frequencies of all groups. The value of the highest frequency of group 4 is 3.46×10^{-5} per year is and is greater than the frequency of any of the other groups.

[9] The safety goals and related QHOs were developed to assess aggregate risks and to be used for making decisions on rulemakings or other major agency actions. It is necessary to keep this in mind when using the QHOs to evaluate specific issues or plant specific concerns. In this case, the risks associated with high-density loadings in spent fuel pools contribute only a small fraction of the overall societal risk goal and so the staff concludes that the issue would not result in additional risks that would cause the cumulative risk of nuclear power to exceed the established safety goals.

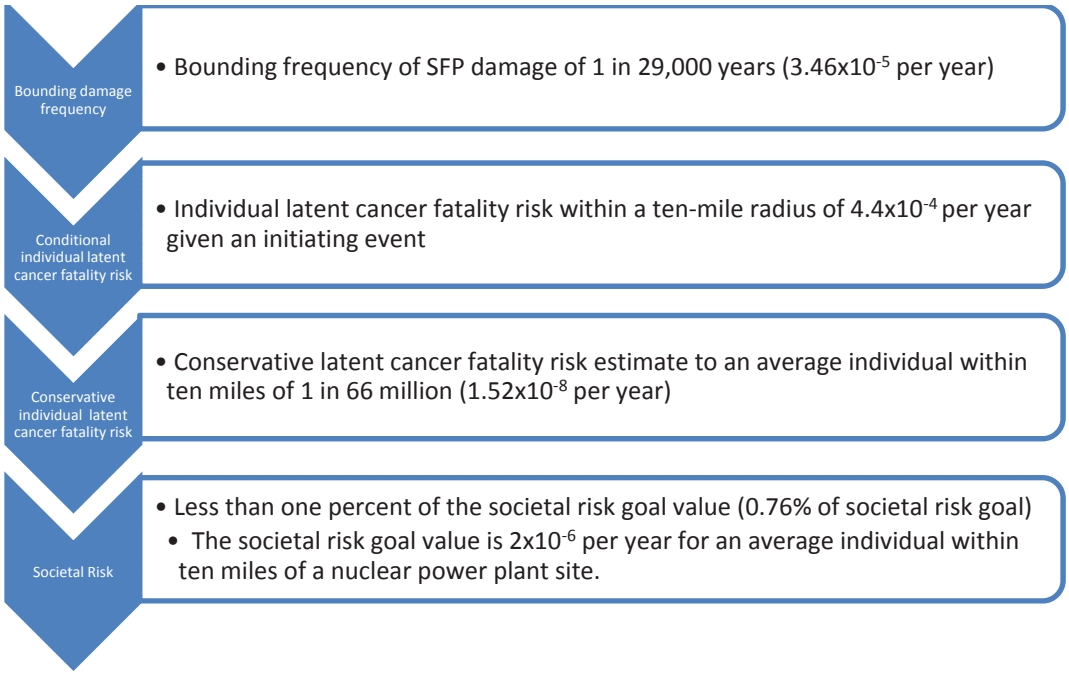

Figure 1 Factors Used in Evaluating Societal Risk

(2) The QHOs effectively establish expectations related to the frequency of severe accidents associated with nuclear reactors and the potential for release of radioactive materials from an operating reactor core. Previous NRC evaluations of SFPs, including NUREG-1353 and NUREG-1738, compared the estimated risks from SFP accidents to the QHOs as part of the rationale for determining appropriate regulatory actions. Some considerations in comparing SFP risks to the QHOs are that the potential consequences of a SFP accident can exceed those of reactor accidents in terms of the amount of long-lived radioactive material released, the land area affected, and the economic consequences. The safety goal relates the risks to an individual from nuclear power in comparison to other risks that an individual faces. The staff uses the safety goal in regulatory decisionmaking processes as a measure of health consequences to determine if a potential action provides a substantial safety improvement. Although a SFP accident might affect larger areas and more people than a reactor accident, protective actions such as relocation of the public and decontamination of affected areas would result in the risks to individuals beyond ten miles to be similar to individuals located closer to the plant. For this reason, the staff uses the existing QHOs for determining whether the substantial safety enhancement threshold is met.

(3) A possible issue with use of the existing guidance and QHOs for SFP accidents relates to the inclusion of emergency planning (i.e., evacuation, sheltering, and relocation of populations) within the analyses. Given that the same measures would be taken for releases following accidents involving high-density or low-density spent fuel pools, the difference in risks to individuals does not increase as much as might be expected from the large differences in the amount of radioactive material released and populations affected. So while the risk of individuals, either close to or far from the plant, remains below the QHOs, the total or cumulative radiation dose to the population might be higher for a SFP accident than for a reactor accident. This would be in large part due to low doses to larger populations associated with the potentially expanded land areas affected

by a SFP accident. The discussions of larger affected populations and areas regarding SFP accidents, as compared to reactor accidents, leads to questions about the use of QHOs as a screening metric as well as questions about underlying Commission policies on estimating the health effects of ionizing radiation (i.e., linear no-threshold model).

The significant difference between the calculated consequences of a SFP accident and a reactor accident has led some stakeholders to propose alternate performance measures to help in the decisionmaking process. Such measures could include a revised consideration of economic consequences, collective dose to populations, or other estimates that reflect the large consequences and reduce the influence of the low event frequencies and implementation of protective actions in assessing the overall societal risks associated with SFP accidents. However, the Commission has previously directed that these performance measures should be consistent with the overall safety goals the Commission policy established and should not be so conservative that it creates a de facto new policy.[10] In addition, the Commission stated in the staff requirements memorandum for SECY-12-0110, "Consideration of Economic Consequences within the U.S. Nuclear Regulatory Commission's Regulatory Framework," that developing guidance for other regulatory applications should be limited and should be resourced as a lower priority than applying State-of-the-Art Reactor Consequence Analyses (SOARCA) insights and improving guidance and analysis tools.

The development of surrogate measures for SFPs could be useful if the conditional probability of a significant SFP accident is very high for particular event scenarios (a so-called cliff-edge effect). Although the staff has used various conservative assumptions in this assessment in order to estimate the potential benefits of reducing the density of spent fuel stored in pools, the expected ability of pools to retain their integrity and the availability of mitigation capabilities leads the staff to conclude that exceeding design basis values associated with SFPs are unlikely to result in such a cliff-edge effect and that the frequency of damage to stored fuel is appropriately low to satisfy overall societal risk goals. Therefore, the staff has not identified this as an area for which it needs to develop new methodologies, guidance, or criteria. In the SRM for SECY-12-0110, the Commission directed the staff to proceed with improvements to the guidance for estimating offsite economic costs. The staff is continuing its efforts and planning related to the SRM and is scheduled to provide the Commission with a paper in December 2013. Factors considered likely to change as a result of the staff's activities (e.g., dollars per person-rem conversion factor) have been addressed in this evaluation through the presentation of additional cases and sensitivity studies.

The staff has concluded that the continued operation of nuclear power plants with high-density loadings in their SFPs does not challenge the NRC's safety goals or related QHOs. Therefore, in the staff's judgment, a regulatory action to require reducing the inventory of spent fuel in the pools would provide no more than a minor safety improvement.

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[10] Commission Guidance on Implementation of the NRC's Safety Goal Policy," memorandum from the Secretary of the Commission to the EDO, dated November 6, 1987.

4. COST-BENEFIT ANALYSIS

To support Commission's deliberations, the staff conducted a cost-benefit analysis using current policies and guidance. Recently the staff completed the SFPS, producing updated consequence estimates which were used in this analysis. The SFPS provides consequence estimates of a hypothetical SFP accident initiated by a low likelihood seismic event at a reference plant for both a fully loaded (high-density) and minimally loaded (low-density) SFP. Appendix D of the SFPS evaluates whether the benefits would be cost-justified and substantial enough at the reference plant to require a change from high- to low-density storage configurations in the SFP.

To determine whether further study of expedited spent fuel transfer may be appropriate, the staff herein conducts a more expansive analysis using insights from the SFPS and previous studies. This generic analysis addresses the different types of SFPs at U.S. nuclear power plants. The process the staff used to conduct the generic analysis is described in the following sections and referenced appendices.

4.1 Spent Fuel Pool Characteristics and Operation Strategies

4.1.1 Spent Fuel Pool Groupings

Based on the variation in SFP configurations, rack designs, and SFP capacities provided in detail in Appendix A, the following groupings were created for use in this analysis.

Table 1 Average Reactor Operation Expectancy by Grouping

SFP Group No.	Description	No. of reactor units	No. of spent fuel pools	Average Year when the Reactor Operating License Expires
1	BWR Mark I and Mark II with nonshared SFPs	31	31	2037
2	PWR and Mark III with nonshared SFPs	49	49	2040
3	AP1000 SFPs	4	4	2078
4	Reactor units with shared SFPs	20	10	2038
5	SFPs located below grade[1]	(included in Group 2 numbers)		
6	Decommissioned plants with spent fuel stored in pool[2,3]	7	6	N/A
7	Decommissioned plants with fuel stored in an ISFSI using dry casks	21	N/A	N/A

1. Group 5 is a special set of currently operating PWRs where damage to the pool structure would not result in a rapid loss of water inventory.
2. The Zion 1 and 2 decommissioned reactor units share a single SFP.
3. The GE-Hitachi Morris wet ISFSI site is included in Group 6.

This cost-benefit analysis focuses on the first four groups identified in Table 1. Group 5 SFPs are excluded from the analysis because they are a special set of SFPs that are less susceptible to the formation of small or medium leaks due to the absence of open space around the pool liner and concrete structure. The spent fuel in Group 6 SFPs are no longer receiving discharged fuel following reactor decommissioning and several plants had extended plant

outages before announcing cessation of plant operation. The spent fuel in Group 7 is already in dry cask storage.

4.1.2 Operation Strategies

The operation strategies include the interim storage operations to expand onsite storage and cask loading strategies; these strategies are provided in detail in Appendix B.

4.2 Estimation and Evaluation of Costs and Benefits

This section discusses how the costs and benefits of the proposed alternative are evaluated and presented relative to the baseline. Ideally, all costs and benefits are converted into monetary values. The total of costs and benefits are then algebraically summed to determine whether the difference between the costs and benefits is a positive benefit. However, in some cases the assignment of monetary values to benefits is not provided because meaningful quantification is not possible.

4.2.1 Identification of Affected Attributes

This section identifies the factors within the public and private sectors that the expedited transfer are expected to affect. These factors are classified as attributes using the list of potential attributes provided by the NRC in Chapter 5 of its Regulatory Analysis Technical Evaluation Handbook (NUREG/BR-0184) (Ref. 15). The basis for selecting each attribute is presented below.

Affected attributes are the following:

- Public Health (Accident). This attribute measures expected changes in radiation exposure to the public caused by changes in accident frequencies or accident consequences associated with the proposed action (i.e., delta risk). The expected changes in radiation exposure are measured over a 50-mile (80-kilometer) radius from the plant site. The dose to the public is from reoccupation of the land and other activities following a severe accident. In addition, the dose to the public includes the occupational dose to workers for cleanup and decontamination of the contaminated land offsite.

- Occupational Health (Accident). This attribute measures occupational health effects, both immediate and long-term, associated with site workers because of changes in accident frequency or accident consequence. The short-term occupational exposure related to the accident occurs at the time of the accident and during the immediate management of the emergency and during decontamination and decommissioning of the onsite property. The radiological occupational exposure resulting from cleanup and refurbishment or decommissioning activities of the damaged facility to occupational workers are found within the long-term occupational exposure.

- Occupational Health (Routine). This attribute accounts for radiological exposures to workers during normal facility operations (i.e., nonaccident situations). These occupational exposures occur during dry storage cask (DSC) loading and handling activities; ISFSI operations, maintenance, and surveillance activities; and preparing to ship the spent fuel offsite.

This attribute represents an estimate of health effects incurred during normal facility operations so accident probabilities are not relevant. As is true of other types of exposures, a net decrease in worker exposures is taken as a positive benefit; a net increase in worker exposures is taken as a negative benefit.

- Offsite Property. This attribute measures the expected total monetary effects on offsite property resulting from the proposed action. Changes to offsite property can take various forms, both direct, (e.g., land, food, and water) and indirect (e.g., tourism). This attribute is typically the product of the change in accident frequency and the property consequences from the occurrence of an accident.

  The offsite property costs are any property consequences resulting from any radiological release from the occurrence of an accident. Normal operational releases and those releases before severe accident are outside the scope of this cost-benefit analysis.

- Onsite Property. This attribute measures the expected monetary effects on onsite property, including replacement power costs, decontamination, and refurbishment costs, from the proposed action. There are two forms of onsite property costs that are evaluated. The first type is the cleanup and decontamination costs for the damaged unit. The second type is the cost to replace the energy from the damaged or shutdown units.

- Industry Implementation. This attribute accounts for the projected net economic effect on the affected licensees to implement the mandated changes. Costs include procedural and administrative activities. Additional costs above the regulatory baseline are considered negative and cost savings are considered positive.

- Industry Operation. This attribute accounts for the projected net economic effect caused by routine and recurring activities required by the proposed alternative on all affected licensees.

- NRC Implementation. This attribute accounts for the projected net economic effect on the NRC to place the proposed alternative into operation. NRC implementation costs and benefits incurred in addition to those expected under the regulatory baseline are included. Additional rulemaking, policy statements, new or expedited revision of guidance documents, and inspection procedures are examples of such costs.

- NRC Operation. This attribute accounts for the projected net economic effect on the NRC after the proposed action is implemented. Additional inspections, evaluations, or enforcement activities are examples of such costs.

Attributes that are not expected to be affected under any of the alternatives include the following: public health (routine), other government, general public or antitrust considerations, safeguards and security considerations, regulatory efficiency, improvements in knowledge, and environmental considerations addressing section 102(2) of the National Environmental Policy Act of 1979.

4.2.2 **Methodology for Evaluation of Benefits and Costs**

This section describes the process used to evaluate benefits and costs associated with the proposed alternatives. The benefits (values) include desirable changes in affected attributes (e.g., monetary savings and improved security and safety). The costs (impacts or burdens) include undesirable changes in affected attributes (e.g., increased monetary costs and decreased security and safety).

The cost-benefit analysis methodology is specified by various guidance documents. The two documents that govern the NRC's voluntary regulatory analysis process are NUREG/BR-0058, Revision 4, "Regulatory Analysis (RA) Guidelines of the U.S. Nuclear Regulatory Commission," dated September 2004 (RA Guidelines) (Ref. 3), and NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook," dated January 1997 (RA Handbook) (Ref. 15). The analysis identifies all attributes impacted by the proposed alternative and analyzes them either quantitatively or qualitatively.

For the quantified cost-benefit analysis, the NRC staff develops expected values for each cost and benefit. The expected value is the product of the probability of the cost or benefit occurring and the consequences that would occur assuming the event happens. For each alternative, the staff first determines the probabilities and consequences for each cost and benefit, including the year the consequence is incurred. The NRC staff then discounts the consequences in future years to the current year of the regulatory action for purposes of evaluating benefits and costs (i.e., providing a net present value). Finally, the NRC staff sums the costs and the benefits for each alternative and compares them.

After performing a quantitative regulatory analysis, the NRC staff adds attributes that could only be qualified.[11] Based on the qualification of each attribute, uncertainties, sensitivities, and the quantified costs and benefits, the staff provides a recommendation for each alternative. If the benefits, both quantified and qualified, are greater than the quantified and qualified costs, then the staff recommends the alternative be implemented. If the benefits, both quantified and qualified, are less than the quantified and qualified costs, then the staff recommends the alternative not be implemented.[12]

There are a number of tables presented throughout this analysis. Generally, the tables include the SFP group[13], the case, the dose averted, the dose conversion factor, and the benefits/costs/cost offsets provided based on the net present value (NPV)[14]. There are two formats that the case information is presented in the tables. In one format, the information is

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[11] See the NRC's Regulatory Analysis Technical Evaluation Handbook, Section 4.3, "Estimation and Evaluation of Values and Impacts" (Ref. 15).

[12] See the NRC's Regulatory Analysis Technical Evaluation Handbook, Section 4.5, "Decision Rationale" (Ref. 15). Nonquantifiable attributes can only be factored into the decision in a judgmental way; the experience of the decisionmaker will strongly influence the weight that they are given. Qualitative attributes may be significant factors in regulatory decisions and should be considered, if appropriate.

[13] Information on the SFP groups is found in Section 4.1.1 and Appendix E.

[14] Information on net present value is found in Appendix C, Section C.1.3 and Appendix D, Section D.1.

presented as low estimate, base case, and high estimate. In the other format, the base case evaluations are presented as Expedited Transfer Alternative–Low-density storage for each SFP group.

The dose averted and the dose conversion factors are only provided in tables that relate to health benefits. The dose averted is the amount of probability-weighted dose (i.e., risk) that is prevented due to the alternative based on a linear no threshold dose response model per year (i.e., the delta risk per year between the regulatory baseline and the alternative). The dose conversion factor (dollar per person-rem) is used to monetize the averted dose to allow comparison to other attributes.[15] The product of the dose averted and the dose conversion factor provides the monetized benefit per year.

The last row of the tables in this analysis provides the total benefit or cost offset for the attribute in 2012 dollars and is provided based on the NPV. The benefits and cost offsets are calculated by using the benefit/cost offset per year and applying it to the average remaining life of the affected entities. The way to apply the information to the average life is by discounting each year in the future by the discount rate. The formula for calculating NPV is

$$NPV = {}^{FV}/_{(1+r)^t}$$

where FV is future value, r is the discount rate, and t is the number of years from the base year to the year the benefit/cost offset is incurred. For example, $100 in year 2013 (FV) would be worth $97 in 2012 dollars (NPV) at a 3 percent discount rate. To determine the total benefit/cost/cost offset for an attribute, each year of the attribute is summed into a total that is provided within the table.

4.2.3 **Assumptions**

This section provides an overview of the assumptions used by the staff in this analysis to estimate the costs and benefits associated with expedited transfer. This section describes:

- Assumptions associated with economic modeling, the definition of representative plants, projection of future spent fuel discharges, and requirements for dry storage. This includes assumptions regarding fuel burnup, decay heat, and cesium-137 source term, as well as wet and dry storage technology capacity and heat load capability.

- Assumptions associated with SFP accident modeling and evaluation. This includes assumptions regarding the probability of initiating events challenging SFP integrity and spent fuel cooling, radiological release source term, atmospheric modeling and meteorology, post-accident radiological doses, population demographics and surrounding area economic data, long-term habitability criteria, and emergency response modeling.

- Assumptions associated with time periods required to load dry storage cask systems (DSCs) and occupational dose received during cask loading operations.

[15] Additional information on dollar per person-rem is found in Appendix C, Section C.2.5 and Appendix D, Section D.2.

- Assumptions regarding the costs of construction and operation of an at-reactor ISFSI, cost increases associated with expedited transfer, cost increases associated with the need for a short-term increase in DSC fabrication capacity, costs to load additional DSCs, and the need to increase shielding capability of DSCs to store spent fuel with shorter cooling times.

Assumptions used are documented throughout this report. For reader convenience, major assumptions are listed in Table 2.

Table 2 Major Assumptions

Topical Area	Major Assumption	Comment
Overall Approach	The fleet of U.S. reactor SFPs were classified in the following groups: 1. BWRs with elevated pools 2. PWRs and BWRs with dedicated pools near grade 3. New AP1000 reactors 4. PWRs that share a single pool 5. PWRs with pools that cannot rapidly drain 6. Decommissioning reactors For the first four groups, representative characteristics of the spent fuel and SFP loading conditions that were conservative with respect to the majority of SFPs within each group were selected. The remaining two groups were not evaluated due to the much lower potential for runaway zirconium oxidation.	The configuration of the plant is considered in determining potential bounding conditions regarding the potential drainage paths from the pools and the potential for natural circulation air cooling. The inventory of fuel, reactor thermal power, and fuel burn-up at reactors within each group are considered in determining the representative inventory of radioactive material present in the pool. Plant characteristics and accident progression for BWRs with elevated pools were drawn from the SFPS. Remaining plant characteristics and accident progression assumptions are drawn from NUREG-1353 and NUREG-1738.
Regulatory Baseline Condition	High-density loading configuration with one full core reserve capacity during which mitigation capability is assumed to be ineffective.	This loading configuration approximates the maximum fuel inventory normally maintained in the SFP. The assumption of ineffective mitigation maximizes the potential release frequency.
Alternative Condition	Low-density loading configuration with fuel decayed more than five years removed from the SFP and mitigation 95% effective.	This loading configuration approximates the minimum fuel inventory for an operating reactor SFP. The assumption of 95% effective mitigation minimizes the frequency of potential releases.
Seismic Hazard Characterization	Seismic hazard models – this analysis used the USGS 2008 model instead of the model currently under development in an ongoing regulatory program. While the USGS (2008) hazard model is not sufficiently detailed for regulatory decisions, it is appropriate to use for this analysis	A new probabilistic seismic hazard model is currently being developed and will consist of two parts: (1) a seismic source zone characterization and (2) a ground motion prediction equation (GMPE) model. Although part (1) is now complete (Ref. 16), it was not

Topical Area	Major Assumption	Comment
	because it was the most recent and readily available hazard model for the central and eastern U.S. plant sites. Hazards for the western sites will be evaluated when the updated model is complete.	available at the start of this analysis. In addition, the GMPE update is still in progress. Furthermore, the NRC is currently developing an independent probabilistic seismic hazard assessment (PSHA) computer code to incorporate part (1) and part (2) when complete.
Earthquake Frequency	Earthquake frequencies are based on hazard curves developed from 2008 USGS data for two bins having peak ground accelerations of 0.7g and 1.2g, respectively. Large earthquakes with frequencies on the order of a few occurrences every 100,000 years to once every 1,000,000 years have the potential to damage the SFP structure.	The USGS data provides a consistent method of quantifying earthquake frequency east of the Rockies. The low and base cases use the seismic hazard estimate for the SFPS reference plant, which results in higher earthquake frequency estimates than the USGS model for most plants. The high case uses the USGS model results for the site within each group with the highest earthquake frequency.
Cask Drop Frequency	A cask drop frequency of $2x10^{-7}$ per year is used for each SFP.	This value is drawn from an evaluation in NUREG-1738 and represents the potential for cask drops during routine transfer activities to maintain assumed SFP storage inventory. Additional cask movements associated with achieving low-density SFP storage are conservatively not evaluated.
AC Power Fragility	AC power is conservatively assumed to fail during earthquake and cask drop initiators to reflect loss of installed forced cooling and coolant makeup systems.	This assumption results in loss of forced cooling and other minor coolant leaks progressing to uncover the stored fuel unless mitigation is effectively deployed.
Liner Fragility	The values conservatively selected for the base case are: • 0.7g PGA earthquake - 10% for BWRs with elevated pools (SFPS) and 5% for all other groups • 1.2g PGA earthquake - 100% for BWRs with elevated pools and 50% for all other groups • Cask drop event - 100%	Liner Fragility represents the conditional probability of leakage from the SFP at locations that uncover the stored fuel, given an earthquake or cask drop occurs. The high case uses 100% for all initiators.
Other Initiating Event Frequencies	Loss of forced cooling and loss of coolant inventory events are conservatively represented by a total initiating event frequency of $2.37x10^{-7}$ per year.	Individual initiating events affecting loss of forced cooling, loss of AC power, loss of coolant inventory, and seal failures were drawn from NUREG-1738 and NUREG-1353.
Unavailability of	The conservative values selected for	Unavailability of natural circulation

Topical Area	Major Assumption	Comment
Natural Circulation Air Cooling – Partial Drain Conditions	the base case are: • 8% – 0.7g earthquake for BWRs with elevated pools (SFPS) • 100% – 0.7g earthquake for all other groups • 100% for the 1.2g earthquake • 100% for the cask drop event • 100% for all other initiators	air cooling reflects various conditions that could lead to inadequate heat removal and progression to runaway zirconium cladding oxidation. Conditions bounded by this result include: • fuel with high decay heat • recently discharged fuel in a contiguous pattern rather than distributed pattern • partial drain conditions with racks that block air cooling The high case uses 100% for all initiators.
Mitigation	Effective deployment of mitigation is conservatively assumed to reduce the frequency of release for low-density storage cases by a factor of 19.	Conservative assumption to maximize difference in release frequency between low-density and high-density storage configurations.
Release Frequency Determination	The release frequencies are calculated as the product of the frequency fuel becomes uncovered and the unavailability of air cooling. The frequency fuel becomes uncovered is the product of the initiating event frequency, ac power fragility, and liner fragility for the seismic and cask drop initiators. For all other initiators, the initiating event frequency is the frequency fuel becomes uncovered. For low-density storage configurations, the release frequency is reduced by a factor of 19 to reflect mitigation.	The earthquake and cask drop initiators dominate the events potentially leading to inadequate cooling of the fuel because these events are most likely to cause a leak from the pool at or below the elevation of the stored fuel. Other initiators are conservatively assumed to progress such that the coolant inventory does not adequately cool the stored fuel because of uncertainties in the accident progression.
Cs-137 Release fraction	The SFP Group 1 high-density loading release fractions are: • 3% for the low estimate • 40% for the base case • 90% for the high estimate	The SFPS (Table 27) shows that for the high-density scenarios involving a leak without mitigation measures, the maximum release is approximately 40%, which was used for the base case. A 90% release fraction is used for the high estimate to account for SFP variations within the group and uncertainties in the accident progression.
	The SFP Groups 2, 3 and 4 high-density loading release fractions used are: • 10% for the low estimate • 75% for the base case • 90% for the high estimate	These release fractions are consistent with the range of release fractions used in previous SFP studies.

Topical Area	Major Assumption	Comment
	The SFP Group 1, 2, 3, and 4 low-density loading release fractions are: • 0.5% for the low estimate • 3% for the base case • 5% for the high estimate	The SFPS (Table 28) shows that for the low-density scenarios involving a leak without mitigation measures, the maximum release is approximately 3%, which was used for the base case. A 5% release fraction is used for the high estimate to account for SFP variations within the group and uncertainties in the accident progression. The release fractions are the same for all groups because only the most recently discharged fuel is expected to be involved.
Radionuclide Source Term	A source term calculated by the MELCOR code based on the cesium release fraction.	The MELCOR code models the fuel damage state, radionuclide release, and holdup of aerosols.
Atmospheric Modeling and Meteorology	The atmospheric transport and dispersion model used in this analysis is based on the MACCS2 model developed using weather data for the Peach Bottom site, which is described in Section 7.1.2 of the SFPS.	A straight-line Gaussian plume segment dispersion model is used for the atmospheric transport.
Population and Economic Data	Representative site demographics are selected to represent the 90th percentile, the mean, the median, and the 20th percentiles. For each representative site, the site population and economic data is established for use in the consequence analysis.	Representative sites for the 90th percentile, the mean, the median, and the 20th percentile are Peach Bottom, Surry, Palisades, and Point Beach, respectively. To identify the specific effect of these values, the staff performed sensitivity studies where only one parameter was varied from a low to high value. Section 4 discusses this sensitivity study in more detail.
Emergency Response Model	The site-specific emergency response model from the SFPS is used to model evacuation timing and speed within the emergency planning zone.	The conditional individual risk measures near the site are expected to be relatively insensitive to site-specific characteristics (i.e., emergency response measures). This is because the predicted releases allow time for effective protective actions to limit exposures to the public.
Long-Term Habitability Criteria	The long-term phase is modeled for 50 years to calculate the consequences of exposure to the average person assuming habitation is limited to areas where annual dose	The selected habitability criteria affect the values of offsite property damage used in this analysis. Certain metrics such as offsite property damage, the number of

Topical Area	Major Assumption	Comment
	is within the criteria. The base case uses habitability criteria of 2 rem in the first year and 500 mrem each year thereafter. The high case uses a criterion of 2 rem annually.	displaced individuals (either temporarily or permanently) and the extents to which such actions may be needed are inversely proportional to changes in collective dose resulting from changes in habitability criteria.
Accident Occupational Exposure	Occupational exposures related to accident mitigation and recovery are estimated based on actual worker doses collected for the Fukushima Dai-ichi site.	The assumed accident period extends for one year and involves a work force of 3,700 people.
Health Consequences	The Linear No Threshold (LNT) dose-response model is used as the base for reporting results. The dose truncation methodology, introduced in the SOARCA analyses documented in NUREG-1935, is provided as a sensitivity analysis.	For large populations exposed to low annual doses, which is the case for some of the SFP accident scenarios, the health effects to populations in habitable zones dominate the health effects when the LNT model is used.
Implementation Cost Approach and Timing of Cask Loading	For the regulatory baseline, the plant is expected to load the required number of dry storage casks each refueling cycle to retain sufficient space in the SFP to discharge one full core of fuel. For the low-density storage alternative in Groups 1, 2, and 4, the plant is assumed to transfer all fuel that has greater than 5 years decay within a 5 year period and then continue loading dry storage casks each refueling cycle as necessary to maintain a full core reserve. For the low-density storage alternative in Group 3, the plant is expected to begin loading dry storage casks once the pool reaches the allowed capacity in a low-density (1x4) configuration.	Group dry storage cask loading is based on a representative plant selected within each group. The total number of dry storage casks necessary for the low-density storage alternative is higher than for the regulatory baseline because fuel assemblies that have decayed for shorter periods have higher decay heat levels, and the higher decay heat per assembly reduces the allowed capacity below its nominal capacity.
Occupational Dose	For the low-density storage alternative, each cask loaded in addition to the number required by the regulatory baseline is estimated to result in an incremental 400 person-mrem dose.	This radiation dose is consistent with the exposure value used in EPRI TR-1021049 (Ref. 17) and in EPRI TR-1018058 (Ref. 18), which analyzed worker impacts associated with loading spent fuel for transport to the proposed Yucca Mountain repository.
Incremental Upfront Cost of	Each additional dry storage cask is expected to require engineering,	Each of these cost components are further described in

Topical Area	Major Assumption	Comment
ISFSI Capacity	design and construction costs of $657,700 in 2012 dollars.	EPRI TR-1021048, "Industry Spent Fuel Storage Handbook."
Incremental Cost of Additional Cask purchase and Loading	The base cost for purchase and loading of a dry storage cask is assumed to be $1,300,000. When only 5-year decayed, high-burnup fuel is available for loading, additional shielding; engineering, licensing, and operational expenses are assumed to increase the cost to $1,466,400 per cask.	These cost estimates are based on the DSC unit costs that EPRI used for a generic interim storage facility and documented in EPRI TR-1025206.
Incremental Annual ISFSI Operating Costs	The majority of reactor sites in Groups 1, 2, and 4, have operational ISFSIs, and the incremental operating cost for increased capacity is considered negligible for these groups. For Group 3, maintenance of low-density storage is expected to require early operation at an incremental cost of $1.1 million per year.	EPRI reports a wide variability in published estimates of annual ISFSI operating costs that range from $212,000 to $2 million per year in 2012 dollars and reported their estimate of $1.1 million per year for an ISFSI at an operating nuclear power plant site.

4.2.4 Sensitivity Analysis

Table 3 provides a list of sensitivity studies performed to estimate the effect upon the results of variations in input parameters. The output from the sensitivity studies is used to determine the importance of the evaluated parameters. The table below provides the parameter evaluated in the left column, what the parameter value is for the base case for the staff's recommendation and sensitivities that the staff performed as additional information for the Commission, and whether it was determined to be a key parameter[16]. Additional detail describing these sensitivity studies is contained in Section 4.3 of this analysis and in Appendix D.

Table 3 Sensitivity Study Parameters

Parameters	Methodology		Key Parameter
	Base Case	Sensitivity	
Present value calculations	7% net present value	2% and 3% net present value	Yes
Dollar per person-rem conversion factor	$2,000	$4,000	Yes
Replacement energy costs (annual) (Constant 2012 dollars)	$2.3 million	Range: $729,000 to $57.3 million Average: $10.1 million Median: $6.7 million	No
Calculated consequences from site	50 miles	Beyond 50 miles	Yes

[16] A key parameter is a variable that can significantly affect calculation results.

Parameters	Methodology		Key Parameter
	Base Case	Sensitivity	
Uniform fuel pattern during outage	1x4 arrangement	Uniformly arranged for a short period	No
Population density	Surry	Range: Point Beach to Peach Bottom Median: Palisades	No
Habitability criteria	2 rem in the first year and 500 mrem each year thereafter	500 mrem per year and 2 rem per year	Yes
Seismic initiator frequency[1]	Bin 3: 1.65×10^{-5} Bin 4: 4.90×10^{-6}	Bin 3: 2.24×10^{-5} – 5.64×10^{-5} Bin 4: 7.09×10^{-6} – 2.00×10^{-5}	Yes

1 As discussed in section 3.2 of the SFPS, damage to the SFP and other relevant structures, systems, and components is not credible for events in Bins 1 and 2. These bins are further discussed in Appendix C, Section C 2.2.

4.3 Evaluation of Alternative—Low-Density Spent Fuel Pool Storage

This section discusses the costs and benefits of the evaluated alternative (i.e., expedited transfer) relative to the baseline or current practices. As described in the previous section, costs and benefits are provided for the various attributes addressed within a regulatory analysis and for a range of assumptions for various parameters (i.e., low estimate, base case, and high estimate). Information is also provided regarding the sensitivity of the cost/benefit assessments to several key factors. A qualitative discussion is provided for those issues not easily represented in monetary values.

4.3.1 Public Health (Accident)

This attribute measures expected changes in radiation exposure to the public caused by change in accident frequencies or accident consequences associated with the proposed action. The expected changes in radiation exposure are predicted over a 50-mile radius from the plant site. The calculated radiation dose to the public is primarily from reoccupation of the land and other activities following the SFP accident. In addition, the calculated radiation dose to the public includes the occupational dose to workers for cleanup and decontamination of contaminated land not onsite. The incremental radiation doses are calculated by subtracting the values for the alternative from those of the regulatory baseline. The difference (delta) is the averted dose benefit of this alternative in units of person-rem. The quantitative results for public health that could affect SFP risk are provided for each SFP grouping. These values are based on MACCS2 analyses and probabilistic considerations described in further detail in Appendix C of this analysis. The assumptions with regard to the base case seismic event frequencies are discussed in Appendix section C.2.2 and with regard to release frequencies are found in Appendix section C.2.3 of this cost-benefit analysis.

As Table 4 shows, the base case of the delta benefit for averted public health (accident) radiation exposure from a SFP accident resulting in spent fuel damage is approximately 1,740 person-rem for the Group 1 SFP and varies for each grouping. This dose represents the reduction of public health risk that results from a policy decision to transfer spent fuel from the SFP to dry storage in order to achieve low-density spent fuel loading in the pool. For a single

BWR Mark I or Mark II reactor with a non-shared SFP (Group 1), the averted delta dose exposure is approximately 69.6 person-rem per year over a remaining licensed commercial operation of the reactor of 24-years (until year 2037). The value assumes a U.S. reactor site average population density of approximately 300 people per square mile within a 50-mile radius from the site. The calculated dose is the difference between an uncontrolled release of radionuclides from a full high-density SFP with no credit for successful mitigation to a full low-density SFP with credit for successful mitigation. The averted doses reflects the calculated health benefits that result if adherence to the EPA intermediate phase protective action guides that allow a dose of 2 rem in the first year and 500 mrem each year thereafter are used.

To provide the Commission with additional information to inform its regulatory decisionmaking, an evaluation of the sensitivity of the results to a change in the dollar per person-rem conversion value from $2,000 to $4,000 per person-rem averted was performed and the results are also provided in Table 4.

Table 4 Summary of Public Health (Accident) for Expedited Transfer Alternative– Low-density Spent Fuel Pool Storage (Base case with $2,000 and $4,000 per person-rem)

SFP Group	Case	Dose conversion factor ($/person-rem)	Dose (averted person-rem per pool	Benefits (2012 million dollars)		
				2% NPV	3% NPV	7% NPV
1	Alternative 2 - Low-density storage	$2,000	1,740	$2.72	$2.42	$1.62
		$4,000		$5.43	$4.85	$3.24
2	Alternative 2 - Low-density storage	$2,000	1,630	$2.45	$2.15	$1.38
		$4,000		$4.90	$4.30	$2.75
3	Alternative 2 - Low-density storage	$2,000	3,020	$3.14	$2.37	$0.99
		$4,000		$6.28	$4.75	$1.98
4	Alternative 2 - Low-density storage	$2,000	1,690	$2.62	$2.33	$1.54
		$4,000		$5.25	$4.66	$3.08

4.3.1.1 Population Demographic Sensitivity

Population densities and distributions characteristics for SFP sites are examined to provide perspective on how important changes to these site demographic characteristics are for this cost-benefit analysis. The base case and the three additional site population densities and distributions near SFP locations and the results are discussed in Appendix C Section C.2.12.

4.3.1.2 Habitability Criteria Sensitivity

A long-term cleanup policy for recovery after a severe nuclear power plant accident does not currently exist. The actual decisions regarding how land would be recovered and populations relocated after an accident would be made by a number of local, State, and Federal jurisdictions and would most likely be based on a long-term cleanup strategy, which is currently being developed by the NRC, the U.S. Environmental Protection Agency (EPA), and other Federal agencies. Furthermore, a cleanup standard may not have an explicit dose level for cleanup. Instead, the cleanup strategy may give local jurisdictions the ability to develop localized cleanup goals after an accident, to allow for a number of factors that include sociopolitical, technical, and economic considerations.

For habitability, most States adhere to EPA intermediate phase protective action guides that allow a dose of 2 rem in the first year and 500 mrem each year thereafter. This habitability criterion was used in previous SFP studies, which used 4 rem in 5 years to represent these protective action guideline levels (e.g., 2 rem in year one, followed by 0.5 rem each successive

year). Further discussion of this approach is provided in Appendix section C.2.13 of this analysis.

4.3.1.3 Seismic Initiator Frequency Assumptions Sensitivity

Although the SFPS reference plant hazard exceedance frequencies curves discussed in Appendix section C.2.2 of this cost-benefit analysis falls close to the upper end of each group in terms of hazard estimates, there are some central and eastern United States (CEUS) sites that exceed those estimates. To analyze the seismic risk hazard for these CEUS sites in each SFP group, a high estimate using the largest site hazard exceedance frequency curve in the group is used to in this sensitivity study. The seismic frequencies are provided in Table 37 in Appendix section C.2.2. Other bounding seismic assumptions include the loss of all ac power for all SFP initiators, a conservative liner fragility value is discussed in Appendix section C.2.3 even though a detailed analysis may be able to justify a value of factor of 2 or more lower, and assuming a bounding value of 1.0 for the conditional probability of failure to successfully mitigate the high-density storage spent fuel accident. These conservative (bounding) assumptions were used in order to calculate a high value estimate for the seismic initiating frequency sensitivity analysis in order to analyze the effect on the public health (accident) attribute. Further discussion of this approach is provided in Appendix section C.2.4 of this analysis.

4.3.1.4 Sensitivity to a Uniform Fuel Pattern during an Outage

The base case of this cost-benefit analysis assumes that each licensee has prearranged the SFP such that discharged assemblies can be placed directly into a 1x4 arrangement for the discharges of the last two outages. However, those requirements do allow for the fuel to be stored in a less favorable configuration for some time following discharge if other considerations prevent prearrangement. To capture the effects of nonbeneficial arrangement of discharged fuel, this cost-benefit analysis evaluates the situation in which the discharged spent fuel is uniformly arranged during the outage to evaluate the effect of this aspect on public health (accident) attribute. For the offsite consequence analysis, the sequences with recently discharged fuel in a uniform configuration were binned in a similar manner to the low-density and high-density (1x4) loading scenarios. Because licensees are required to move their recently discharged fuel to a more favorable configuration after a certain amount of time, this sensitivity assumes that the high-density uniform case becomes identical to the high-density (1x4) case by the end of operating cycle phase 2 (OCP 2) or within 25 days.[17] Further discussion of this approach is provided in Appendix section C.2.15 of this analysis.

4.3.2 Occupational Health (Accident)

Occupational health measures both short-term and long-term health effects associated with site workers as a result of changes in accident frequency or accident mitigation. Within the regulatory baseline, the short-term occupational exposure related to the accident occurs at the time of the accident and during the immediate management of the emergency and during decontamination and decommissioning of the onsite property. The radiological occupational exposure resulting from cleanup and refurbishment or decommissioning activities of the damaged facility to occupational workers are estimated within the long-term occupational

[17] To analyze this scenario the plant operating cycle is divided into numerous small periods of time or operating cycle phases (OCPs). The definitions for the modeled operating cycle phases is provided in Table 16 of the SFPS.

exposure. The quantitative results for occupational health (accident) considering the contribution of all initiators that could affect SFP risk is provided in Table 5 and is based on the release frequencies discussed in Appendix section C.2.1and the occupational health (accident) assumptions found in Appendix section C.2.9. The high estimate also incorporates the seismic initiator frequency assumptions described in Section 4.3.1.3.

Table 5 Summary of Occupational Health (Accident) Benefits for Low-density Spent Fuel Pool Storage (Base case with $2,000 and $4,000 per person-rem and with Low and High Estimates)

SFP Group	Occupational Health (Accident) Case	Dose conversion factor ($/person-rem)	Dose averted per pool (person-rem)	Benefits (2012 dollars)		
				2% NPV	3% NPV	7% NPV
1	Low Estimate	$2,000	0.60	$942	$840	$562
		$4,000		$1,884	$1,730	$1,203
	Base Case	$2,000	5.49	$8,579	$7,652	$5,121
		$4,000		$17,159	$15,763	$10,959
	High Estimate	$2,000	67	$105,037	$93,684	$62,697
		$4,000		$210,075	$192,988	$134,171
2	Low Estimate	$2,000	0.34	$500	$400	$300
		$4,000		$1,000	$900	$600
	Base Case	$2,000	4.36	$6,600	$5,800	$3,700
		$4,000		$13,100	$11,500	$7,400
	High Estimate	$2,000	25	$37,300	$32,700	$21,000
		$4,000		$74,600	$65,500	$41,900
3	Low Estimate	$2,000	0.71	$700	$600	$200
		$4,000		$1,500	$1,100	$500
	Base Case	$2,000	9.16	$9,500	$7,200	$3,000
		$4,000		$19,100	$14,400	$6,000
	High Estimate	$2,000	52	$54,200	$41,000	$17,100
		$4,000		$108,400	$82,000	$34,200
4	Low Estimate	$2,000	0.30	$500	$400	$300
		$4,000		$900	$800	$600
	Base Case	$2,000	3.91	$6,000	$5,400	$3,600
		$4,000		$12,100	$10,700	$7,100
	High Estimate	$2,000	22	$34,300	$30,500	$20,200
		$4,000		$68,700	$61,000	$40,400

As Table 5 shows, the total delta benefit for short- and long-term occupational health (accident) range between 3.91 and 9.16 person-rem averted per SFP for the base case. The estimated total benefit of the occupational health (accident) attribute for low-density SFP storage relative to the regulatory baseline, using the $2,000 per person-rem averted conversion factor, net present value ranges are insignificant for the base case and do not warrant further sensitivity analysis. The high estimate includes the conservative inputs and assumptions for the seismic initiator frequency sensitivity analysis discussed in Section 4.3.1.3 of this cost-benefit analysis.

4.3.3 Occupational Health (Routine)

Occupational health (routine) accounts for radiological exposures to workers during normal facility operations (i.e., non-accident situations). These occupational exposures occur during DSC loading and handling activities, ISFSI operations, and maintenance and surveillance activities. The assumptions in relation to the exposures for occupational health (routine) are found in Section 4.3.3 of this cost-benefit analysis.

Table 6 Summary of Occupational Health (Routine) Costs for Low-Density Spent Fuel Pool Storage (Base Case with $2,000 and $4,000 per Person-rem)

SFP Group	No. of DSCs required through end of operation		Delta Dose (p-rem)	Dose conversion factor ($/p-	Costs (2012 dollars)		
	High-density storage (Alternative 1)	Low-density storage (Alternative 2)			2% NPV	3% NPV	7% NPV
1	107	119	6.84	$2,000	$25,400	$27,800	$28,200
				$4,000	$50,800	$55,600	$56,300
2	75	90	8.55	$2,000	$27,200	$29,100	$28,900
				$4,000	$54,500	$58,300	$57,700
3	77	87	5.70	$2,000	$14,500	$12,900	$6,400
				$4,000	$29,000	$25,800	$12,800
4	130	141	6.27	$2,000	$22,700	$24,700	$24,800
				$4,000	$45,400	$49,400	$49,700

As Table 6 shows, the delta benefit for occupational health (routine) is an increase of between 5.70 and 8.55 person-rem in worker exposure resulting from DSC loading and handling activities; ISFSI operations; and maintenance and surveillance activities depending on the SFP grouping. The estimated cost to the occupational health (routine) for low-density spent fuel storage relative to the regulatory baseline for all SFP groups and calculated in accordance with the current regulatory framework, ranges from $14,500 to $27,200 (2 percent net present value), $12,900 to $29,100 (3 percent net present value), and $6,400 to $28,900 (7 percent net present value) using the $2,000 per person-rem averted conversion factor. These ranges are insignificant for this analysis and do not warrant further sensitivity analysis.

4.3.4 Offsite Property

The offsite property attribute measures the expected total monetary effects on offsite property resulting from the proposed action. Changes to offsite property can take various forms, both direct, (e.g., land, food, and water) and indirect (e.g., tourism). This attribute is the product of the change in accident frequency and the property consequences from the occurrence of a SFP accident.

For the regulatory baseline, the offsite property costs are any property consequences resulting from any radiological release from the occurrence of an accident. Plant releases not related to the severe accident analyzed are outside the scope of this cost-benefit analysis.

The cost offsets for the analyzed SFP accident are quantified relative to the regulatory baseline based on the MACCS2 calculation results and probabilistic considerations. The results for the consequences from a low-density spent pool accident are compared to those from the regulatory baseline SFP accident. The calculation is the difference between the calculated consequences resulting from a low-density and a high-density SFP accident. The results are provided in Table 7. The assumptions with regard to the base case seismic event frequencies are discussed in Appendix section C.2.2 and with regard to release frequencies are found in Appendix section C.2.3 of this cost-benefit analysis.

Table 7 Summary of Offsite Property Cost Offsets for Expedited Transfer Alternative– Low-Density Spent Fuel Pool Storage within 50 Miles (Base Case)

SFP Group	Case	Offsite Property Cost Offsets (2012 million dollars)		
		2% NPV	3% NPV	7% NPV
1	Alternative 2 - Low-density storage	$8.96	$7.99	$5.35
2	Alternative 2 - Low-density storage	$9.03	$7.93	$5.08
3	Alternative 2 - Low-density storage	$11.45	$8.66	$3.61
4	Alternative 2 - Low-density storage	$9.81	$8.71	$5.76

As Table 7 shows, the estimate of offsite property damage from a SFP accident resulting in spent fuel damage, ranges from $8.96 million (2 percent net present value) to $5.35 million (7 percent net present value) for Group 1 SFPs and varies for each grouping. This value assumes a U.S. reactor site average population density of approximately 300 people per square mile within a 50-mile radius from the site and is representative of the associated property values found near the Surry power plant site. This base case uses the EPA intermediate phase PAG level of 2 rem in the first year and 500 mrem annually to evaluate post-accident collective dose and offsite property costs as discussed in Appendix section C.2.13 of this cost-benefit analysis.

4.3.4.1 Population Demographic Sensitivity

Certain metrics such as property use, the number of displaced individuals (either temporarily or permanently), and the extent to which such actions may be needed are affected by the population size and the amount of economic activity in the vicinity of the postulated accident.

This examination provides a perspective on how important changes to these site demographic variables are for this cost-benefit analysis. The base case and the three additional site population densities, distributions, and economic characteristics near SFP locations are discussed in Appendix section C.2.12. It provides a basis for understanding the nature and the extent of the relationship between population densities, distributions characteristics, and property values near SFP sites.

4.3.4.2 Offsite Property Consequences beyond 50 Miles Sensitivity

Because a SFP accident under certain scenarios and environmental conditions could result in impacts to offsite property located beyond 50 miles from the postulated accident site, this case evaluates the sensitivity of offsite property cost offsets for damages occurring beyond 50 miles from the site, using the base case assumptions and the intermediate EPA PAG criterion. This is discussed in Appendix section C.2.12.

4.3.4.3 Offsite Property Costs Sensitivity to Habitability Criteria

As discussed in Section 4.3.1.2, a long-term cleanup policy for recovery after a severe nuclear power plant accident does not currently exist. The actual decisions regarding how land would

be recovered and populations relocated after an accident would be made by a number of local, State, and Federal jurisdictions and would most likely be based on a long-term cleanup strategy, which is currently being developed by the NRC, EPA, and other Federal agencies. Furthermore, a cleanup standard may not have an explicit dose level for cleanup. Instead, the cleanup strategy may give local jurisdictions the ability to develop localized cleanup goals after an accident, to allow for a number of factors that include sociopolitical, technical, and economic considerations. Given the uncertainties in which long-term habitability criterion would be used, Appendix section C.2.13 discusses this sensitivity analysis and analyze the effect on the costs for offsite property damage.

4.3.4.4 Offsite Property Cost Offset Sensitivity to Seismic Initiator Frequency Assumptions

Although the SFPS reference plant hazard exceedance frequencies curves discussed in Appendix section C.2.1 of this analysis fall close to the upper end of each SFP group in terms of hazard estimates, there are some CEUS sites that exceed those estimates. To analyze the seismic risk hazard for these CEUS sites, a high estimate using the bounding plant hazard exceedance frequency curve is used to produce the high estimate seismic bins and initiating event frequencies. This sensitivity analysis is discussed in Appendix section C.2.4 of this analysis.

4.3.4.5 Offsite Property Cost Offset Sensitivity to a Uniform Fuel Pattern during an Outage

As discussed in Section 4.3.1.4, the base case assumes that the licensee has prearranged the SFP such that discharged assemblies can be placed directly into a 1x4 arrangement for the discharges of the last two outages. This approach is consistent with Section 9.3 of the SFPS (Ref. 2). However, fuel is allowed to be stored in a less favorable configuration for some time following discharge if other considerations prevent prearrangement. To capture the effects of non-beneficial arrangement of discharged fuel, this cost-benefit analysis evaluates the situation in which the discharged spent fuel is uniformly arranged during the outage to evaluate the effect of this aspect on offsite property attribute.

For the offsite consequence analysis, the sequences with recently discharged fuel in a uniform configuration were binned in a similar manner to the low-density and high-density (1x4) loading scenarios. Because licensees are required to move their recently discharged fuel to a more favorable configuration after a certain amount of time, this sensitivity assumes that the high-density uniform case becomes identical to the high-density (1x4) case during operating cycle phase 3 (OCP3.). While the uniform case has different release categories, the situations that lead to release are largely the same as the low-density and high-density (1x4) base cases.

Table 65 in Appendix C provides a comparison of the effect on the offsite property cost offsets if a plant operator initially places discharged spent fuel in a uniform pattern and achieves the 1x4 pattern by the end of OCP2 (i.e., within 25 days) versus placing the fuel directly into the 1x4 pattern.

4.3.5 Onsite Property

This attribute measures the expected monetary effects on onsite property, including replacement power costs, decontamination, and refurbishment costs, from the proposed action.

There are two forms of onsite property costs that each alternative must disposition. The first type of onsite property costs are the cleanup and decontamination costs for the unit. The second type of onsite property costs is the cost to replace the energy from the damaged or shutdown unit(s). The cost offsets for low-density SFP storage are quantified relative to the regulatory baseline based on the probabilistic considerations provided in the SFPS (Ref. 2) and the onsite property estimates described in Appendix C.2.7.

Because many nuclear power plants have more than one reactor unit co-located on a plant site, it is assumed that a severe SFP accident that occurs at one unit would result in the cleanup and/or decommissioning costs and the loss of power generation for the affected unit. The postulated SFP accident might also result in the temporarily loss of power generation from the co-located unit. In modeling the replacement energy costs based on this scenario, it is assumed for the high estimate that replacement energy would be purchased for two units.

Based on these modeling assumptions, the onsite property results are provided in Table 8.

Table 8 Summary of Onsite Property Cost Offsets for Low-density Spent Fuel Pool Storage

Group	Case	Onsite Property Cost Offsets (2012 dollars)								
		Low Estimate			Base Case			High Estimate		
		2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
1	Onsite Property - Replacement Energy	$90	$80	$50	$9,620	$8,450	$5,270	$34,680	$30,440	$19,000
	Onsite Property - Cleanup, Decontamination, Repair, & Refurbishment	$5,900	$5,200	$3,100	$57,900	$50,200	$30,200	$173,600	$150,500	$90,500
	Group 1 Total	$5,990	$5,280	$3,150	$67,520	$58,650	$35,470	$208,280	$180,940	$109,500
2	Onsite Property - Replacement Energy	$50	$40	$30	$7,500	$6,480	$3,850	$27,010	$23,340	$13,880
	Onsite Property - Cleanup, Decontamination, Repair, & Refurbishment	$3,200	$2,800	$1,600	$44,300	$37,800	$21,700	$132,800	$113,400	$65,200
	Group 2 Total	$3,250	$2,840	$1,630	$51,800	$44,280	$25,550	$159,810	$136,740	$79,080
3	Onsite Property - Replacement Energy	$80	$60	$20	$11,510	$8,530	$3,250	$41,490	$30,740	$11,700
	Onsite Property - Cleanup, Decontamination, Repair, & Refurbishment	$4,700	$3,500	$1,300	$64,400	$47,300	$17,700	$193,100	$142,000	$53,200
	Group 3 Total	$4,780	$3,560	$1,320	$75,910	$55,830	$20,950	$234,590	$172,740	$64,900
4	Onsite Property - Replacement Energy	$50	$40	$20	$6,820	$5,960	$3,670	$23,710	$20,810	$12,990
	Onsite Property - Cleanup, Decontamination, Repair, & Refurbishment	$3,000	$2,600	$1,500	$40,800	$35,200	$20,900	$122,300	$105,700	$62,800
	Group 4 Total	$3,050	$2,640	$1,520	$47,620	$41,160	$24,570	$146,010	$126,510	$75,790

As Table 8 shows, based on these calculations, the delta cost offset for the frequency-weighted onsite property base case estimate ranges from $47,620 to $75,910 per pool (2 percent net present value) to $41,160 to $55,830 per pool (3 percent net present value), and to $20,950 to

$35,470 per pool (7 percent net present value). Low and high estimates are also provided in Table 8.

4.3.6 Industry Implementation

Industry implementation accounts for the projected net economic effect on the affected licensees to implement the mandated changes. Costs evaluated for dry storage include upfront and incremental dry storage cask (DSC) capital and loading costs. Additional costs above the regulatory baseline are considered negative and cost savings are considered positive. The quantitative results for industry implementation are given in terms of expected costs if a policy decision is made to accelerate the transfer of spent fuel stored in SFPs to dry storage. These expected costs are not frequency weighted. Assumptions used for developing the industry implementation cost model are discussed in Appendix sections C.1.7, C.4.3, and C.4.4.

4.3.6.1 Industry Implementation Cost Summary

Table 9 provides a summary of the industry implementation costs for each SFP group and provides the number of additional DSCs that are needed to store the hotter spent fuel.

Table 9 Industry Implementation Costs for Low-Density Spent Fuel Pool Storage for a Single Spent Fuel Pool

SFP Group	No. of additional DSCs needed	Implementation Costs (2012 million dollars		
		2% NPV	3% NPV	7% NPV
1	12	$52.6	$55.2	$52.3
2	15	$51.4	$53.8	$51.3
3	10	$42.4	$35.8	$16.7
4	11	$48.8	$50.4	$46.4

Table 9 shows, the incremental costs associated with DSC upfront costs and the earlier purchasing and loading of DSCs on a periodic basis. The estimated industry implementation costs for low-density spent fuel storage relative to the regulatory baseline and calculated in accordance with the current regulatory framework, ranges from $42.4 to $52.6 million (2 percent net present value), $35.8 to $55.2 million (3 percent net present value), and $16.7 to $52.3 million (7 percent net present value).

4.3.6.2 Implementation Costs to Install Open Frame Low-Density Racks in an Existing Spent Fuel Pool

The re-racking of a SFP with open frame low-density racks is a preventive risk reduction alternative, which is intended to reduce radiological material available and promote air cooling to prevent the onset of self-sustaining clad oxidation in the event of loss of SFP water inventory. As stated in the alternative, older spent fuel assemblies are expeditiously moved from SFP storage to dry cask storage beginning in year 2014 to achieve low-density spent fuel storage and provide an opportunity to re-rack the SFP. Re-racking a SFP involves replacing the existing high-density storage rack modules with new open frame low-density racks and is estimated to take approximately 2.5 years based on a hypothetical SFP re-racking schedule to install high-density racks provided in EPRI TR-1021048 (Ref. 19). The EPRI estimated schedule is provided in Figure 2.

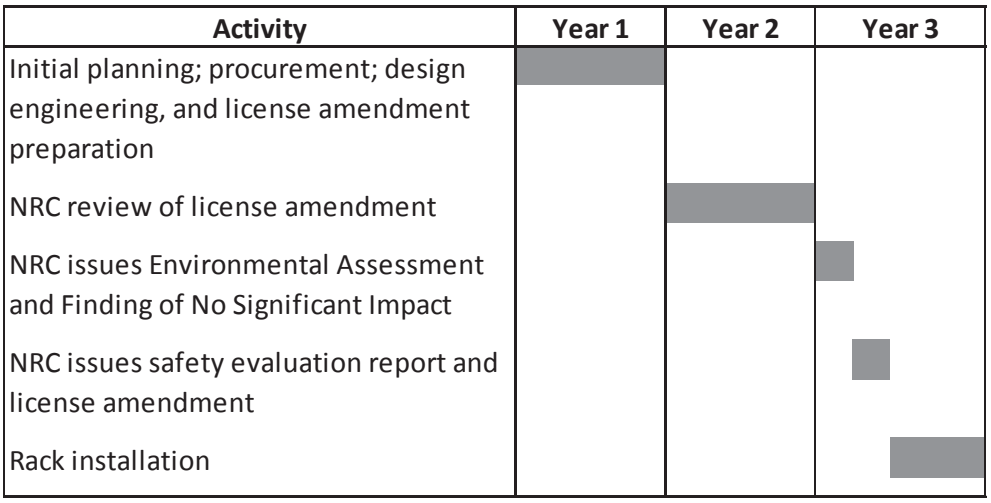

Figure 2 Estimated schedule for spent fuel pool re-racking project

The licensee would need to perform comprehensive safety analyses for the SFP re-rack project. These analyses will generally evaluate SFP criticality analysis; mechanical and structural design; seismic design; radiation protection provisions during rack removal and installation; changes to plant technical specifications; heavy loads analyses for the SFP during rack removal and installation; and SFP thermal-hydraulic; decay heat analyses; and radiological consequences of beyond-design-basis events. In addition to these design and engineering costs, other cost components include preparation of a license amendment and changes to the plant's technical specifications; specification and procurement of low-density replacement racks; rack manufacture, rack installation, and handling and disposal of the old high-density storage racks. One licensee estimated (Ref. 20) the cost for a single unit SFP re-rack project to be $7.5 million in 1979 which is equivalent to $23.7 million[18] in 2012 dollars.

This cost element was not included in this alternative because it would add substantial cost and is inefficient in terms of regulatory benefit given that much of the benefit is achieved by storing less fuel in the existing high-density racks for less cost. Based on insights from the SFPS, the staff believes that within the first few months after the fuel came out of the reactor, the decay heat in the freshly unloaded spent fuel is high enough to cause a zirconium fire even in the presence of convective cooling. Therefore, reracking the SFP to install open frame racks even with channel boxes removed to allow potential crossflow, would not necessarily prevent a radiological release during this time.

4.3.7 Industry Operation

Industry operation accounts for the projected net economic effect caused by routine and recurring activities required by the proposed alternative. Annual operating costs for an ISFSI during reactor operation include the costs associated with NRC inspections; security; radiation monitoring; ISFSI operational monitoring; technical specification and regulatory compliance,

[18] This cost was converted from the licensee's cost estimate of $7.5 million in 1979 dollars using the consumer price index cost inflator. The licensee's cost estimate includes the following: design, materials, fabrication; removal and disposal of old racks; transportation and installation of new racks; project management, licensing, quality assurance; contingency allowance; and allowances for funds used during construction.

including implementation of new certificate of compliance (CoC) amendments; personnel cost and code maintenance associated with fuel selection for dry storage; personnel costs for spent fuel management and fabrication surveillance activities; electric power usage for lighting and security systems; road maintenance to the ISFSI site; and miscellaneous expenses associated with ISFSI maintenance. NRC license fees for dry storage are included as part of the 10 CFR 50, "Domestic Licensing of Production and Utilization Facilities," operating license fees. As discussed in Appendix section C.4.4, incremental costs associated with annual ISFSI operating costs are insignificant for this analysis.

Industry operation also includes annual operating costs following reactor shutdown for decommissioning, which includes the costs associated with transporting spent fuel offsite. These costs were beyond the scope of the evaluation of expedited transfer of spent fuel to dry cask storage and are not included in this analysis.

The ability of a nuclear power plant operator to transfer spent fuel to dry storage during power operation is dependent upon what other activities are scheduled in the fuel handling area, plant-specific limitations on use of cask lifting crane or movement restrictions of heavy loads, or resource limitations if fuel handling equipment or personnel are shared between multiple reactor units. Furthermore, there could be operational impacts associated with large DSC loading campaigns as depicted in Figure 16 through Figure 19. These unintended consequences could include additional management support or attention to dry storage operations for longer periods, potential impacts on plant outage schedules or maintenance schedules because of increased staffing needs to support cask loading operations, and additional dry cask storage vendor oversight.

4.3.8 NRC Implementation

These costs, if calculated, would further reduce the calculated net benefit for this analysis.

4.3.9 NRC Operation

These costs, if calculated, would further reduce the calculated net benefit for this analysis.

4.3.10 Other Considerations

The other considerations are provided in relation to the regulatory baseline.

4.3.10.1 Seismic Hazard Model Uncertainties

There remain significant uncertainties in estimating the frequency of events for natural phenomena, which are postulated to challenge SFP cooling or integrity. This cost-benefit analysis uses the existing USGS 2008 model to evaluate seismic hazards at CEUS nuclear power plants. A new probabilistic seismic hazard model is currently being developed and will consist of two parts: (1) a seismic source zone characterization and (2) a ground motion prediction equation (GMPE) model. Although part (1) is now complete (Ref. 16), the GMPE update is still in progress. Furthermore, the NRC is currently developing an independent probabilistic seismic hazard assessment (PSHA) computer code to incorporate part (1) and part (2) when complete. While the USGS (2008) hazard model is not sufficiently detailed for regulatory decisions, it is used for this cost-benefit analysis because it is the most recent and readily available hazard model and was used in the SFPS.

4.3.10.2 Other Modeling Uncertainties

There are also significant uncertainties in the calculation of event consequences in terms of the dispersion and disposition of radioactive material into the site environs. This is due in part to significant uncertainties regarding the degree to which topographical features and other phenomena are modeled at distances away from the evaluated site. Estimating economic consequences also includes large uncertainties, as it is difficult to model the impact of disruptions to many different aspects of local economies and the loss of infrastructure on the general U.S. economy. An example of this is the supply chain disruptions that followed the 2011 Tohoku earthquake and subsequent tsunami on Japan or the 2004 Indian Ocean earthquake and tsunami on Thailand.

4.3.10.3 Cask Handling Risk

The NRC recognizes that there are costs and risks associated with the handling and movement of spent fuel casks. These cost and risk impacts, if included in this analysis, would further reduce the overall net benefit in relation to the regulatory baseline. These effects (e.g., the added risks of handling and moving casks) were conservatively ignored in order to calculate the maximum potential benefit by only comparing the safety of high-density fuel pool storage relative to low-density fuel pool storage and its implementation costs without consideration of cask movement risk.

4.3.10.4 Additional Repackaging Costs and Risk

Considering the uncertainty associated with the final disposal of spent fuel, there could be a potential impact of expedited transfer on the Department of Energy's (DOEs) cask standardization program and acceptance for final disposal. Should expedited transfer be required, it is expected that utilities would employ large capacity storage casks to minimize costs and handling. None of the proposed DOE repository designs were planned to accommodate the direct emplacement of large casks. Thus, the use of large canisters for storage may prove incompatible with a future repository design. There could be additional costs and risk associated with repackaging the spent fuel into canisters that are compatible with final disposal requirements. The staff is currently engaged in a significant effort with DOE and industry to address technical issues related to long term aging issues, such as canister and fuel cladding degradation. This ongoing DOE research effort could provide valuable insights with a direct impact on the potential costs and benefits of expedited spent fuel transfer to dry cask storage. These additional repackaging costs and risk were conservatively ignored to calculate the minimum implementation costs for the low-density fuel pool storage alternative.

4.3.10.5 Mitigating Strategies

The release of fission products to the environment from events that may cause the loss of SFP cooling or integrity, such as seismic events, missiles, heavy load drops, loss of cooling or make-up, inadvertent drainage or siphoning and pneumatic seal failures, are estimated to be range between 7.39×10^{-7} to 3.46×10^{-5} per year without successful mitigation. Operator diagnosis and recovery are important factors considered in the development of the event frequencies for these events and portions of this evaluation are premised on licensees having taken appropriate actions to understand the potential consequences of SFP accident events

and develop appropriate procedures and mitigating strategies to respond and mitigate the consequences.

The SFPS (Ref. 2) evaluated the potential benefits of mitigation measures required under 10 CFR 50.54(hh)(2) (Ref. 6), which were implemented following the September 11, 2001 attacks. These mitigation measures are intended to maintain SFP cooling in the event of a loss of large areas of the plant caused by explosions or fire. Neither the SFPS nor previous SFP studies considers the post-Fukushima improvements required by NRC and being implemented by the plants. These improvements are intended to increase the likelihood of restoring or maintaining power and mitigation capability during severe accidents.

The new SFP level instrumentation required under Order EA-12-051 and the mitigation strategies now required under Order EA-12-049 significantly enhance the likelihood of successful mitigation beyond that considered in this cost-benefit analysis because of the following features:

- Portable equipment with redundant sets (e.g., N+1) that is sufficient to supply all functions, simultaneously for the entire site, including equipment for the SFP. This portable equipment provides reasonable protection from seismic events, which are a dominant contributor to SFP risk.

- The mission time for this equipment is indefinite, versus the 12-hour mission time for the 50.54(hh)(2) equipment.[19]

- The new EA-12-049 mitigating strategies (Ref. 10) are capable of being deployed in all modes, which means that the new strategies can address SFP cooling issues that could occur in any operating cycle phase.

- The new SFP level instrumentation required under Order EA-12-051 (Ref. 11), ensures a reliable indication of the water level in the SFP for identification of the following pool water level conditions:
    - a level that is adequate to support operation of the normal fuel pool cooling system
    - a level that is adequate to provide substantial radiation shielding for a person standing on the SFP operating deck
    - a level where fuel remains covered and actions to implement makeup water addition should no longer be deferred

- The method of filling the SFP is via a connection to the normal SFP makeup system located away from the SFP floor, reducing the impacts on human performance because of potentially adverse environmental conditions (e.g., high temperature, humidity, and radiation) following an event.

[19] This section of the regulations deals with the development and implementation of guidance and strategies intended to maintain or restore core cooling, containment, and spent fuel pool cooling capabilities under the circumstances associated with loss of large areas of the plant resulting from explosions or fire.

This additional equipment, strategies, and features provided by Orders EA-12-049 and EA-12-051, provide additional accident mitigation capability and would further enhance the likelihood of successful mitigation, thereby further reducing the value for the conditional probability of release used in this cost-benefit analysis.

4.3.10.6 Cost Uncertainties

It is difficult to determine costs that could be incurred 50 to 100 years in the future. Changes in technology, regulation, or public policy could all have a profound effect on the actual cost. The purpose of including costs is to try to discern the benefit for the expedited transfer alternative. Of course, this analysis is based on best estimates of current spent fuel strategies and cost. If the U.S. government were to take possession of the spent fuel in order to provide storage at a non-operating plant site for extended periods, the costs could be heavily discounted, and the differences between storage alternatives in this analysis might be reduced.

4.3.10.7 Inadvertent Criticality

Design requirements and related safety analyses ensure fuel stored in the SFP will remain safely subcritical under conditions considered as part of the design basis, but rare conditions beyond the design basis may challenge some measures used to control reactivity. To maintain adequate margin to criticality in U.S. SFPs, the safety analyses credit the geometric configuration of the fuel and a combination of other measures that may include fixed neutron poison material (e.g., Boraflex) and limits on fuel reactivity. In addition, the presence of soluble boron in the coolant of PWR SFPs may be credited, but the stored fuel must remain subcritical assuming unborated water is present (10 CFR 50.68). Since these measures may be challenged by a beyond design-basis event, the NRC staff cannot rule out the potential for an inadvertent criticality event. However, the NRC staff judges that the potential consequences of a zirconium fire in the SFP and an associated hydrogen deflagration considered in this analysis would not be significantly affected by an inadvertent criticality event. The NRC staff bases this judgment on the following considerations:

- Fuel assembly geometric configuration would be maintained while water covers the fuel. Commercial reactor fuel assemblies are robust components designed to withstand the effects of design basis events, including safe shutdown earthquakes, while producing power in an operating reactor. The operating environment of a SFP is considerably less demanding than that of an operating reactor. The fuel racks are also designed to withstand design basis events, and the presence of water around the racks tends to dampen the effects of seismic events on these structures. While the earthquakes considered in this analysis are beyond what the fuel was designed to withstand, the NRC staff judges that fuel cladding and the fuel rack structure would not experience sufficient damage during a seismic event of these magnitudes to cause significant changes in the geometric configuration of the fuel.

- Potential criticality is limited by moderator availability and pool configuration. Many U.S. SFPs rely on the presence of neutron absorbing materials that are part of the storage rack structure to meet sub-criticality requirements under normal and credible abnormal events. The performance of these materials following a large beyond design basis seismic event has not been fully analyzed. It is possible that the environmental conditions after the beyond design basis seismic event could cause degradation of these materials. However, the presence of a moderator is necessary for an inadvertent

criticality event to occur, and an adequate moderator would only be present during the drain down/boil off phase or during recovery actions. While neither of these scenarios has been analyzed, the sustainable power of the inadvertent criticality event would be limited to a level significantly below the operating reactor, since the SFP is an open system and significant heat generation would create steam voids that provide inadequate moderation. Therefore, the additional fission product inventory in the fuel would not be significant. In addition, the required moderator for criticality limits the effect of any inadvertent criticality event because the water would provide shielding and reduce the fraction of radioactive material that would be released.

- Consequences of an inadvertent criticality event would be insignificant relative to consequences of a zirconium fire: Fuel assemblies that experienced zirconium cladding ignition could have sufficient cladding damage where further agitation, such as seismic aftershocks, would relocate fuel fragments in a non-uniform configuration. In this scenario, a large majority of the radioactive source term material would have already been released during the zirconium fire. The release from a subsequent inadvertent criticality event would be primarily a hazard to onsite workers with little offsite impact. The staff expects that any sustained inadvertent criticality event would be orders of magnitude lower than the power generated in the reactor with a corresponding lower production of short half-lived releasable material, making the inadvertent criticality event an insignificant contributor to the consequences of the zirconium fire. Therefore, the NRC staff judges that the consequences of a potential inadvertent criticality event following a zirconium fire fuel need not be considered. Furthermore, if a SFP criticality event did occur and generated short-lived radionuclides that are associated with offsite early fatalities, the emergency response as modeled effectively prevents any early fatality risk. This occurs in part because the modeled accident progression results in releases that are long compared with the time needed for relocation.

4.4 Presentation of Results

This section presents the analytical results, including discussion of supplemental considerations, uncertainties in estimates, and results of sensitivity analyses on the overall benefits.

4.4.1 Cost-Benefit Analysis

4.4.1.1 Summary Table

Table 10 provides the quantified and qualified costs and benefits for low-density SFP storage for each spent fuel group. For the quantitative analysis, the low estimate, base case, and high estimate results within 80 kilometers (50 miles) are reported.

The calculated benefits for requiring low-density SFP storage (Alternative) for the low estimate and base case are less than industry costs to achieve a low-density spent fuel loading pattern for each SFP group. As might be expected for estimates that include a compounding of the most conservative assumptions, all of the SFP group high estimate cases result in calculated benefits that are greater than the estimated costs.

Similar to the seismic event analyzed for the SFPS, no offsite early fatalities are calculated to occur. This results from the following two reasons:

(1) In comparison to reactors, SFPs have a larger proportion of longer-lived radionuclides, which are less likely to cause the significant doses required for acute health effects.

(2) Despite the large releases for certain predicted SFP accident progressions, the release from the most recently discharged fuel (which contains the shorter-lived radionuclides) is predicted to be insufficiently fast and insufficiently large to reach the acute thresholds associated with offsite early fatalities. When doses do exceed minimum levels for early fatalities, emergency response, as treated in the SFPS, effectively prevents any early fatality risk, at least in part because the modeled accident progression results in releases that are long compared with the time needed for relocation.

In addition, the predicted long-term exposure of the population, which could result in latent cancer fatality risk, is also low for the following reasons:

(1) The individual latent individual latent cancer fatality risk within 0 to 10 miles is predicted to be on the order of 2.4×10^{-10} to 1.5×10^{-8} per year, based on the linear no threshold (LNT) dose response model.

(2) The risk within 10 miles of the analyzed accident is dominated by low dose received at a low dose rate. Using truncation levels that do not quantify the effects of doses below 620 mrem/year (i.e., those arising from representative background radiation including average annual medical exposures) reduces the estimated individual LCF risk by up to a few orders of magnitude for the accident as modeled.

(3) Average individual latent cancer fatality risk is low but decreases slowly as a function of distance from the plant. Additionally, the predicted individual risks of latent cancer fatalities are dominated by long-term exposures to very lightly contaminated areas for which doses are small enough to be considered habitable.

Table 10 Summary of Totals for Alternatives

Net Monetary Savings (or Costs) – Total Present Value	Sensitivity Studies	Qualitative Benefits and (Costs)
Regulatory Baseline – Maintain the Existing Spent Fuel Storage Requirements		
$0	None	None.
Expedited Transfer Alternative – Low-density Spent Fuel Pool Storage		
Group 1 – BWR Mark I and Mark II with non-shared SFPs		
Group 1 Industry (Costs): *Base case* ($52 million) using a 7% discount rate **NRC (Costs):** Not calculated **Benefits:** *Base case* $7 million using a 7% discount rate **Group 1 Net Benefit = Benefits + (Costs)** **Base case: $7M + ($52M) = ($45M)** **Conclusion: Not cost beneficial**	**Group 1 Sensitivity Studies** **Industry (Costs) Sensitivity Studies** ($53 million) using a 2% discount rate ($55 million) using a 3% discount rate **Benefit Sensitivity Studies** *Low estimate* $0.2 million using a 2% discount rate $0.2 million using a 3% discount rate $0.1 million using a 7% discount rate *High estimate* $123 million using a 2% discount rate $109 million using a 3% discount rate $73 million using a 7% discount rate **Net Benefit Sensitivity Studies** *Low estimate* ($52.8M) using a 2% discount rate ($54.8M) using a 3% discount rate ($51.9M) using a 7% discount rate *High estimate* $70 million using a 2% discount rate $54 million using a 3% discount rate $21 million using a 7% discount rate	**Qualitative Benefits and (Costs)** **Qualitative (Costs):** Cost Uncertainties (Repackaging Costs) **Qualitative Benefits:** Modeling Uncertainties. (Cask Handling Risk) Mitigating Strategies
Group 2 – PWR and BWR Mark III with non-shared SFPs		
Group 2 Industry (Costs): *Base case* ($51 million) using a 7% discount rate **NRC (Costs):** Not calculated **Benefits:** *Base case* $6.4 million using a 7% discount rate **Group 2 Net Benefit = Benefits + (Costs)** **Base case: $6.4M + ($51M) = ($45M)** **Conclusion: Not cost beneficial**	**Group 2 Sensitivity Studies** **Industry (Costs) Sensitivity Studies** ($51 million) using a 2% discount rate ($54 million) using a 3% discount rate **Benefit Sensitivity Studies** *Low estimate* $0.3 million using a 2% discount rate $0.3 million using a 3% discount rate $0.2 million using a 7% discount rate *High estimate* $137 million using a 2% discount rate $121 million using a 3% discount rate $77 million using a 7% discount rate	**Qualitative Benefits and (Costs)** **Qualitative (Costs):** Cost Uncertainties (Repackaging Costs) **Qualitative Benefits:** Modeling Uncertainties. (Cask Handling Risk) Mitigating Strategies

Net Monetary Savings (or Costs) – Total Present Value	Sensitivity Studies	Qualitative Benefits and (Costs)
	Net Benefit Sensitivity Studies *Low estimate* ($50.7M) using a 2% discount rate ($53.7M) using a 3% discount rate ($50.8M) using a 7% discount rate *High estimate* $86 million using a 2% discount rate $67 million using a 3% discount rate $26 million using a 7% discount rate	
Group 3 – New reactor SFPs		
Group 3 Industry (Costs): *Base case* ($17 million) using a 7% discount rate **NRC (Costs):** Not calculated **Benefits:** *Base case* $4.6 million using a 7% discount rate **Group 3 Net Benefit = Benefits + (Costs)** **Base case: $4.6M + ($17M) = ($12M)** **Conclusion: Not cost beneficial**	**Group 3 Sensitivity Studies** **Industry (Costs) Sensitivity Studies** ($42 million) using a 2% discount rate ($36 million) using a 3% discount rate **Benefit Sensitivity Studies** *Low estimate* $0.3 million using a 2% discount rate $0.3 million using a 3% discount rate $0.1 million using a 7% discount rate *High estimate* $108 million using a 2% discount rate $81 million using a 3% discount rate $34 million using a 7% discount rate **Net Benefit Sensitivity Studies** *Low estimate* ($41.7M) using a 2% discount rate ($35.7M) using a 3% discount rate ($16.9M) using a 7% discount rate *High estimate* $66 million using a 2% discount rate $45 million using a 3% discount rate $17 million using a 7% discount rate	**Qualitative Benefits and (Costs)** **Qualitative (Costs):** Cost Uncertainties (Repackaging Costs) **Qualitative Benefits:** Modeling Uncertainties. (Cask Handling Risk) Mitigating Strategies
Group 4 – Reactor units with shard SFPs		
Group 4 Industry (Costs): *Base case* ($46 million) using a 7% discount rate **NRC (Costs):** Not calculated **Benefits:** *Base case* $7.3 million using a 7% discount rate **Group 4 Net Benefit = Benefits + (Costs)**	**Group 4 Sensitivity Studies** **Industry (Costs) Sensitivity Studies** ($49 million) using a 2% discount rate ($50 million) using a 3% discount rate **Benefit Sensitivity Studies** *Low estimate* $0.3 million using a 2% discount rate $0.3 million using a 3% discount rate $0.2 million using a 7% discount rate *High estimate* $205 million using a 2% discount rate	**Qualitative Benefits and (Costs)** **Qualitative (Costs):** Cost Uncertainties (Repackaging Costs) **Qualitative Benefits:** Modeling Uncertainties. (Cask Handling Risk) Mitigating Strategies

Net Monetary Savings (or Costs) – Total Present Value	Sensitivity Studies	Qualitative Benefits and (Costs)
Base case: $7.3M + ($46M) = ($39M) **Conclusion: Not cost beneficial**	$182 million using a 3% discount rate $120 million using a 7% discount rate **Net Benefit Sensitivity Studies** *Low estimate* ($48.7M) using a 2% discount rate ($49.7M) using a 3% discount rate ($48.8M) using a 7% discount rate *High estimate* $156 million using a 2% discount rate $132 million using a 3% discount rate $74 million using a 7% discount rate	

4.4.1.2 Implementation and Operation Costs–Low- Density Spent Fuel Pool Storage Alternative

4.4.1.2.1 Spent Fuel Pool Group 1 – BWR Mark I and Mark II reactors with non-shared spent fuel pool

Table 11 Summary of Total Implementation and Operation Costs for Low-Density Spent Fuel Pool Storage—Spent Fuel Pool Group 1

Attribute	Costs per SFP (2012 dollars in millions)		
	2% NPV	3% NPV	7% NPV
Occupational Health (Routine)	$0.03	$0.03	$0.03
Industry Implementation	$52.61	$55.17	$52.28
Industry Operation	nc	nc	nc
NRC Implementation	nc	nc	nc
NRC Operation	nc	nc	nc
Total per pool	$52.64	$55.20	$52.31
Total for 31 pools	$1,632	$1,711	$1,622

nc = not calculated

The low-density SFP storage alternative for BWR Mark I and Mark II reactors with a non-shared SFP total implementation and operation costs is the summation of those costs for the industry and the NRC. As shown in Table 11, the total estimated costs for a single Group 1 SFP to achieve and maintain a low-density SFP loading ranges from $52.64 million (2 percent net present value), to $55.20 million (3 percent net present value), and to $52.31 million (7 percent net present value). The total cost for all 31 SFPs in this group is approximately $1.6 billion. These costs are dominated by the capital costs for the DSCs and the loading costs for the storage systems to achieve low-density storage in the SFP than that required for the regulatory baseline.

4.4.1.2.2 Spent Fuel Pool Group 2 – PWR and BWR Mark III reactors with non-shared spent fuel pool

Table 12 Summary of Total Implementation and Operation Costs for Low-Density Spent Fuel Pool Storage—Spent Fuel Pool Group 2

Attribute	Costs per SFP (2012 dollars in millions)		
	2% NPV	3% NPV	7% NPV
Occupational Health (Routine)	$0.03	$0.03	$0.03
Industry Implementation	$51.37	$53.80	$51.33
Industry Operation	nc	nc	nc
NRC Implementation	nc	nc	nc
NRC Operation	nc	nc	nc
Total per pool	$51.40	$53.83	$51.36
Total for 49 pools	$2,519	$2,638	$2,517

nc = not calculated

The low-density SFP storage alternative for PWR and BWR Mark III reactors with a non-shared SFP total implementation and operation costs is the summation of those costs for the industry and the NRC. As shown in Table 12, the total estimated costs for a single Group 2 SFP to achieve and maintain a low-density SFP loading ranges from $51.40 million (2 percent net present value), to $53.83 million (3 percent net present value), and to $51.36 million (7 percent net present value). The total cost for all 49 SFPs in this group range is approximately $2.56 billion. These costs are dominated by the capital costs for the DSCs and the loading costs for the storage systems to achieve low-density storage in the SFP than that required for the regulatory baseline.

4.4.1.2.3 Spent Fuel Pool Group 3 – New power reactors with non-shared spent fuel pool

Table 13 Summary of Total Implementation and Operation Costs for Low-Density Spent Fuel Pool Storage—Spent Fuel Pool Group 3

Attribute	Costs per SFP (2012 dollars in millions)		
	2% NPV	3% NPV	7% NPV
Occupational Health (Routine)	$0.01	$0.01	$0.01
Industry Implementation	$42.41	$35.75	$16.74
Industry Operation	nc	nc	nc
NRC Implementation	nc	nc	nc
NRC Operation	nc	nc	nc
Total per pool	$42.42	$35.76	$16.75
Total for four pools	$169.7	$143.1	$67.0

nc = not calculated

The low-density SFP storage alternative for new reactors with a non-shared SFP total implementation and operation costs is the summation of those costs for the industry and the NRC. As shown in Table 13, the total estimated costs for a single Group 3 SFP to achieve and maintain a low-density SFP loading ranges from $42.42 million (2 percent net present value), to

$35.76 million (3 percent net present value), and to $16.75 million (7 percent net present value). The total cost for all four SFPs in this group range between $67 and $170 million. These costs are dominated by the capital costs for the DSCs, the loading costs for the storage systems to achieve low-density storage in the SFP, and the additional ISFSI annual operation and maintenance costs required for establishing and storing spent fuel at the ISFSI 15 years earlier than that required for the regulatory baseline.

4.4.1.2.4 Spent Fuel Pool Group 4—Reactor units with a shared spent fuel pool

Table 14 Summary of Total Implementation and Operation Costs for Low-Density Spent Fuel Pool Storage—Spent Fuel Pool Group 4

Attribute	Costs per SFP (2012 dollars in millions)		
	2% NPV	3% NPV	7% NPV
Occupational Health (Routine)	$0.02	$0.02	$0.03
Industry Implementation	$48.78	$50.41	$46.39
Industry Operation	nc	nc	nc
NRC Implementation	nc	nc	nc
NRC Operation	nc	nc	nc
Total per pool	$48.80	$50.43	$46.41
Total for 10 pools	$488.0	$504.3	$464.1

nc = not calculated

The low-density SFP storage alternative for reactor units with a shared SFP total implementation and operation costs is the summation of those costs for the industry and the NRC. As shown in Table 15, the total estimated costs for a single Group 4 shared SFP to achieve and maintain a low-density SFP loading ranges from $48.80 million (2 percent net present value), to $50.43 million (3 percent net present value), and to $46.41 million (7 percent net present value). The total cost for all 10 SFPs in this group range between $511 and $555 million. These costs are dominated by the capital costs for the DSCs, and the loading costs for the storage systems to achieve low-density storage in the SFP than that required for the regulatory baseline.

4.4.1.3 Total Benefits and Cost Offsets

4.4.1.3.1 Spent Fuel Pool Group 1 – BWR Mark I and Mark II reactors with non-shared spent fuel pool

Table 15 Summary of Total Benefits and Cost Offsets for Low-Density Spent Fuel Pool Storage—Spent Fuel Pool Group 1

Attribute	Benefits and Cost Offsets (2012 dollars in millions)		
	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$0.05 - $35.6	$0.04 – $31.7	$0.03 – $21.2
Occupational Health (Accident)	<$0.01 – $0.1	<$0.01 – $0.09	<$0.01 – $0.06
Offsite Property	$0.17 – $85.7	$0.15 – $76.4	$0.10 – $51.1
Onsite Property	<$0.01 – $1.1	<$0.01 – $0.99	<$0.01 – $0.60
Total per pool	$0.24 – $123	$0.21 – $109	$0.15 – $73.0
Total for 31 pools	$7.4 – $3,800	$6.5 – $3,380	$4.7 – $2,260

The SFP Group 1 total benefits are shown in the Table 15. These benefits include the public health (accident) and occupational health (accident) benefits summed with the cost offsets. The cost offsets consists of the sum of the offsite property and onsite property attributes relative to the regulatory baseline. The offsite property cost offset is the largest contributor to the benefits, of which the majority of those costs occur during the long-term phase.

4.4.1.3.2 Spent Fuel Pool Group 2 – PWR and BWR Mark III reactors with non-shared spent fuel pool

Table 16 Summary of Total Benefits and Cost Offsets for Low-Density Spent Fuel Pool Storage—Spent Fuel Pool Group 2

Attribute	Benefits and Cost Offsets (2012 dollars in millions)		
	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$0.06 – $38.7	$0.05 – $34.0	$0.03 – $21.8
Occupational Health (Accident)	<$0.01 – $0.11	<$0.01 – $0.96	<$0.01 – $0.06
Offsite Property	$0.27 – $97.5	$0.24 – $85.6	$0.15 – $54.8
Onsite Property	<$0.01 – $1.2	<$0.01 – $1.0	<$0.01 – $0.59
Total per pool	$0.35 – $138	$0.31 – $122	$0.20 – $77.3
Total for 49 pools	$17 – $6,760	$15 – $5,9800	$10 – $3,790

The SFP Group 2 total benefits are shown in the Table 16. These benefits include the public health (accident) and occupational health (accident) benefits summed with the cost offsets. The cost offsets consists of the sum of the offsite property and onsite property attributes relative to the regulatory baseline. The offsite property cost offset is the largest contributor to the benefits, of which the majority of those costs occur during the long-term phase.

4.4.1.3.3 Spent Fuel Pool Group 3 – AP1000 power reactors with non-shared spent fuel pool

Table 17 Summary of Total Benefits and Cost Offsets for Low-Density Spent Fuel Pool Storage—Spent Fuel Pool Group 3

Attribute	Benefits and Cost Offsets (2012 dollars in millions)		
	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$0.06 – $31.9	$0.05 – $24.1	$0.02 – $10.1
Occupational Health (Accident)	<$0.01 – $0.97	<$0.01 – $0.07	<$0.01 – $0.03
Offsite Property	$0.26 – $74.5	$0.20 – $56.3	$0.08 – $23.5
Onsite Property	<$0.01 – $1.1	<$0.01 – $0.78	<$0.01 – $0.29
Total per pool	$0.34 – $108	$0.27 – $81.3	$0.12 – $33.9
Total for 4 pools	$1.4 – $430	$1.1– $330	$0.5 – $140

The SFP Group 3 total benefits are shown in the Table 17. These benefits include the public health (accident) and occupational health (accident) benefits summed with the cost offsets. The cost offsets consists of the sum of the offsite property and onsite property attributes relative to the regulatory baseline. The offsite property cost offset is the largest contributor to the benefits, of which the majority of those costs occur during the long-term phase.

4.4.1.3.4 Spent Fuel Pool Group 4 – Reactor units with a shared spent fuel pool

Table 18 Summary of Total Benefits and Cost Offsets for Low-Density Spent Fuel Pool Storage—Spent Fuel Pool Group 4

Attribute	Benefits and Cost Offsets (2012 dollars in millions)		
	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$0.06 – $52.1	$0.05 – $46.3	$0.03 – $30.6
Occupational Health (Accident)	<$0.01 – $0.13	<$0.01 – $0.11	<$0.01 – $0.07
Offsite Property	$0.27 – $151.2	$0.24 – $134.3	$0.16 – $88.9
Onsite Property	<$0.01 – $1.3	<$0.01 – $1.2	<$0.01 – $0.70
Total per pool	$0.35 – $205	$0.31 – $182	$0.21 – $120
Total for 10 pools	$3.5 – $2,050	$3.1 – $1,820	$2.1 – $1, 200

The SFP Group 4 total benefits are shown in the Table 18. These benefits include the public health (accident) and occupational health (accident) benefits summed with the cost offsets. The cost offsets consists of the sum of the offsite property and onsite property attributes relative to the regulatory baseline. The offsite property cost offset is the largest contributor to the benefits, of which the majority of those costs occur during the long-term phase.

4.4.1.4 Sensitivity Analysis

This section summarizes the results of the sensitivity analyses that were performed as an additional consideration in performing safety goal screening for the evaluated alternatives. In this section, a low and high estimate is provided that combines the range of expected SFP attributes with conservative assumptions to model the range of pool accidents postulated. These high and low estimates are expected to over and under estimate the consequences from SFP accidents for any individual SFPs assigned to the group.

4.4.1.4.1 Dollar per Person-rem Conversion Factor

The NRC is currently revising the dollar per person-rem averted conversion factor based on recent information regarding the value of a statistical life. However, until the NRC completes the update to NUREG-1530 (Ref. 21) and publishes the appropriate guidance documents, the NRC performs sensitivity analysis to estimate the impact on the calculated results when more current value of a statistical life (VSL) and cancer risk factors are used. The NRC used the EPAs VSL as an interim value in the sensitivity analysis as described in Appendix section D.2. The effect of using the higher dollar per person-rem conversion factor on the calculated results is provided below. As previously discussed, the consequences calculated for the high and low estimate are expected to over and under estimate respectively the consequences if compared to plant-specific SFP analyses within this SFP grouping.

4.4.1.4.1.1 Spent Fuel Pool Group 1—BWR Mark I and Mark II reactors with non-shared spent fuel pool

Table 19 Dollar Per Person-Rem Sensitivity Analysis of Net Benefits for Low-Density Spent Fuel Pool Storage within 50 miles—Group 1 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$96,000	$85,600	$57,200	$5,433,200	$4,845,800	$3,243,000	$71,176,000	$63,482,400	$42,485,000
Occupational Health (Accident)	$1,884	$1,680	$1,124	$17,158	$15,304	$10,242	$210,074	$187,367	$125,394
Offsite Property	$165,692	$147,782	$98,902	$8,959,243	$7,990,830	$5,347,787	$85,673,027	$76,412,549	$51,138,370
Onsite Property	$5,990	$5,280	$3,150	$67,520	$58,650	$35,470	$1,139,040	$989,660	$598,900
Total Benefits	***$269,600***	***$240,300***	***$160,400***	***$14,477,100***	***$12,910,600***	***$8,636,500***	***$158,198,100***	***$141,072,000***	***$94,347,700***
Occupational Health (Routine)	-$50,800	-$55,600	-$56,400	-$50,800	-$55,600	-$56,400	-$50,800	-$55,600	-$56,400
Industry Implementation	-$52,610,000	-$55,170,000	-$52,280,000	-$52,610,000	-$55,170,000	-$52,280,000	-$52,610,000	-$55,170,000	-$52,280,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$52,660,800***	***-$55,225,600***	***-$52,336,400***	***-$52,660,800***	***-$55,225,600***	***-$52,336,400***	***-$52,660,800***	***-$55,225,600***	***-$52,336,400***
Net Benefit	**-$52,391,000**	**-$54,985,000**	**-$52,176,000**	**-$38,184,000**	**-$42,315,000**	**-$43,700,000**	**$105,537,000**	**$85,846,000**	**$42,011,000**

nc = not calculated

As shown in Table 19, the dollar per person-rem sensitivity analysis does not achieve a positive net benefit for either the low estimate or base case when using a person-rem conversion factor twice as large as the conversion factor in NUREG-1530. When all the high estimates are combined, a positive net benefit is achieved. As Table 4 shows, the base case of the delta benefit for averted public health (accident) radiation exposure from a SFP accident resulting in spent fuel damage is approximately 1,740 person-rem for the Group 1 SFP. This dose represents the reduction of public health risk that results from a policy decision to transfer spent fuel from the SFP to dry storage in order to achieve low-density spent fuel loading in the pool. For a single BWR Mark I or Mark II reactor with a non-shared SFP (Group 1), the averted delta dose exposure is approximately 70 person-rem per year over a remaining licensed commercial operation of the reactor of 24 years (until year 2037). The value is based on a U.S. reactor site average population density of approximately 300 people per square mile within a 50-mile radius from the site. The calculated dose is the difference between an uncontrolled release of radionuclides from a full high-density SFP with no credit for successful mitigation to a full low-density SFP with credit for successful mitigation. The doses reflects the calculated health benefits that result if adherence to the EPA intermediate phase protective action guides that allow a dose of 2 rem in the first year and 500 mrem each year thereafter are used.

4.4.1.4.1.2 Spent Fuel Pool Group 2—PWR and BWR Mark III reactors with non-shared spent fuel pool

The effect of using the higher dollar per person-rem conversion factor on the calculated results is provided below. As previously discussed, the consequences calculated for the high and low estimate are expected to over and under estimate respectively the consequences if compared to plant-specific SFP analyses within this SFP grouping.

Table 20 Dollar Per Person-Rem Sensitivity Analysis of Net Benefits for Low-Density Spent Fuel Pool Storage within 50 miles—Group 2 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$118,000	$103,600	$66,400	$4,896,800	$4,301,400	$2,752,200	$77,482,600	$68,062,000	$43,549,200
Occupational Health (Accident)	$1,000	$800	$600	$13,200	$11,600	$7,400	$218,800	$192,200	$123,000
Offsite Property	$272,584	$239,442	$153,207	$9,031,983	$7,933,837	$5,076,442	$97,457,843	$85,608,518	$54,776,349
Onsite Property	$3,250	$2,840	$1,630	$51,800	$44,280	$25,550	$1,190,370	$1,018,500	$589,050
Total Benefits	***$394,800***	***$346,700***	***$221,800***	***$13,993,800***	***$12,291,100***	***$7,861,600***	***$176,349,600***	***$154,881,200***	***$99,037,600***
Occupational Health (Routine)	-$54,400	-$58,200	-$57,800	-$54,400	-$58,200	-$57,800	-$54,400	-$58,200	-$57,800
Industry Implementation	-$51,370,000	-$53,800,000	-$51,330,000	-$51,370,000	-$53,800,000	-$51,330,000	-$51,370,000	-$53,800,000	-$51,330,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$51,424,400***	***-$53,858,200***	***-$51,387,800***	***-$51,424,400***	***-$53,858,200***	***-$51,387,800***	***-$51,424,400***	***-$53,858,200***	***-$51,387,800***
Net Benefit	**-$51,030,000**	**-$53,512,000**	**-$51,166,000**	**-$37,431,000**	**-$41,567,000**	**-$43,526,000**	**$124,925,000**	**$101,023,000**	**$47,650,000**

nc = not calculated

As shown in Table 20, the dollar per person-rem sensitivity analysis does not achieve a positive net benefit when using a person-rem conversion factor twice as large as the conversion factor in NUREG-1530 for either the low estimate or base cases. When all the high estimates are combined, a positive net benefit is achieved.

4.4.1.4.1.3 Spent Fuel Pool Group 3—New power reactors with non-shared spent fuel pool

Table 21 Dollar Per Person-Rem Sensitivity Analysis of Net Benefits for Low-Density Spent Fuel Pool Storage within 50 miles—Group 3 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$129,600	$98,000	$41,000	$6,279,200	$4,748,800	$1,981,600	$63,827,600	$48,271,400	$20,143,000
Occupational Health (Accident)	$1,400	$1,200	$400	$19,000	$14,400	$6,000	$193,400	$146,200	$61,000
Offsite Property	$264,273	$199,864	$83,400	$11,451,619	$8,660,606	$3,613,942	$74,506,474	$56,347,594	$23,513,013
Onsite Property	$4,780	$3,560	$1,320	$75,910	$55,830	$20,950	$1,062,030	$781,900	$293,960
Total Benefits	***$400,100***	***$302,600***	***$126,100***	***$17,825,700***	***$13,479,600***	***$5,622,500***	***$139,589,500***	***$105,547,100***	***$44,011,000***
Occupational Health (Routine)	-$29,000	-$25,800	-$12,800	-$29,000	-$25,800	-$12,800	-$29,000	-$25,800	-$12,800
Industry Implementation	-$42,410,000	-$35,750,000	-$16,740,000	-$42,410,000	-$35,750,000	-$16,740,000	-$42,410,000	-$35,750,000	-$16,740,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$42,439,000***	***-$35,775,800***	***-$16,752,800***	***-$42,439,000***	***-$35,775,800***	***-$16,752,800***	***-$42,439,000***	***-$35,775,800***	***-$16,752,800***
Net Benefit	**-$42,039,000**	**-$35,473,000**	**-$16,627,000**	**-$24,613,000**	**-$22,296,000**	**-$11,130,000**	**$97,151,000**	**$69,771,000**	**$27,258,000**

nc = not calculated

As shown in Table 21, the dollar per person-rem sensitivity analysis does not achieve a positive net benefit when using a person-rem conversion factor twice as large as the conversion factor in NUREG-1530 for either the low estimate or base cases presented. The high estimates show a positive net benefit of between $27 and $97 million. This SFP group differs significantly from the other SFP groups analyzed in that these pools have not yet been constructed so that there is not a significant front ended DSC procurement cost difference between the two alternatives. However, in comparison to the base case, the high estimate includes additional conservative assumptions regarding seismic fragilities, release fractions, SFP inventories, long-term habitability criteria, and site population densities that are overly conservative for the four units with combined licenses.

4.4.1.4.1.4 Spent Fuel Pool Group 4—Reactor units with a shared spent fuel pool

Table 22 Dollar Per Person-Rem Sensitivity Analysis of Net Benefits for Low-Density Spent Fuel Pool Storage within 50 miles—Group 4 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$114,400	$101,600	$67,200	$5,246,400	$4,661,400	$3,083,600	$104,286,600	$92,655,000	$61,292,600
Occupational Health (Accident)	$1,000	$800	$600	$12,000	$10,800	$7,200	$250,000	$222,200	$147,000
Offsite Property	$271,158	$240,914	$159,368	$9,805,063	$8,711,458	$5,762,750	$151,185,571	$134,323,136	$88,856,614
Onsite Property	$3,050	$2,640	$1,520	$47,620	$41,160	$24,570	$1,349,250	$1,168,370	$700,210
Total Benefits	***$389,600***	***$346,000***	***$228,700***	***$15,111,100***	***$13,424,800***	***$8,878,100***	***$257,071,400***	***$228,368,700***	***$150,996,400***
Occupational Health (Routine)	$45,400	$49,400	$49,600	$45,400	$49,400	$49,600	$45,400	$49,400	$49,600
Industry Implementation	$48,780,000	$50,410,000	$46,390,000	$48,780,000	$50,410,000	$46,390,000	$48,780,000	$50,410,000	$46,390,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***$48,825,400***	***$50,459,400***	***$46,439,600***	***$48,825,400***	***$50,459,400***	***$46,439,600***	***$48,825,400***	***$50,459,400***	***$46,439,600***
Net Benefit	**-$48,436,000**	**-$50,113,000**	**-$46,211,000**	**-$33,714,000**	**-$37,035,000**	**-$37,562,000**	**$208,246,000**	**$177,909,000**	**$104,557,000**

nc = not calculated

As shown in Table 22, the dollar per person-rem sensitivity analysis does not achieve a positive net benefit when using a person-rem conversion factor twice as large as the conversion factor in NUREG-1530 for either the low estimate or base case presented. The high estimate shows a positive net benefit of between $105 and $208 million.

4.4.1.4.2 Consequences Extending Beyond 50 Miles

The Regulatory Analysis Handbook states that in the case of nuclear power plants, changes in public health and safety from radiation exposure and offsite property impacts should be examined over a 50-mile distance from the plant site, although alternative distances from the plant may be used for sensitivity analyses. For this cost-benefit analysis, supplemental information (e.g., analyses and results) based on MACCS2 calculated results, is performed which extends the analysis to consider consequences beyond 50 miles for each SFP group.

4.4.1.4.2.1 Spent Fuel Pool Group 1 – BWR Mark I and Mark II reactors with non-shared spent fuel pool

Table 23 Consequences Extending Beyond 50 Miles Sensitivity Analysis of Net Benefits for Low-Density Spent Fuel Pool Storage—Group 1 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$503,300	$448,900	$300,400	$22,835,700	$20,367,300	$13,630,700	$305,431,900	$272,417,500	$182,312,800
Occupational Health (Accident)	$942	$840	$562	$8,579	$7,652	$5,121	$105,037	$93,684	$62,697
Offsite Property	$573,290	$511,323	$342,198	$16,358,429	$14,590,231	$9,764,373	$323,691,221	$288,703,133	$193,211,821
Onsite Property	$5,990	$5,280	$3,150	$67,520	$58,650	$35,470	$1,139,040	$989,660	$598,900
Total Benefits	***$1,083,500***	***$966,300***	***$646,300***	***$39,270,200***	***$35,023,800***	***$23,435,700***	***$630,367,200***	***$562,204,000***	***$376,186,200***
Occupational Health (Routine)	-$25,400	-$27,800	-$28,200	-$25,400	-$27,800	-$28,200	-$25,400	-$27,800	-$28,200
Industry Implementation	-$52,610,000	-$55,170,000	-$52,280,000	-$52,610,000	-$55,170,000	-$52,280,000	-$52,610,000	-$55,170,000	-$52,280,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$52,635,400***	***-$55,197,800***	***-$52,308,200***	***-$52,635,400***	***-$55,197,800***	***-$52,308,200***	***-$52,635,400***	***-$55,197,800***	***-$52,308,200***
Net Benefit	**-$51,552,000**	**-$54,232,000**	**-$51,662,000**	**-$13,365,000**	**-$20,174,000**	**-$28,873,000**	**$577,732,000**	**$507,006,000**	**$323,878,000**

nc = not calculated

As shown in Table 23, calculated net benefits for requiring low-density SFP storage when considering consequences beyond 80 kilometers (50 miles) does not achieve a positive net benefit for either the low estimate or base cases presented. The high estimates show a positive

net benefit of between $324 and $578 million. In comparison to the base case, the high estimate includes additional conservative assumptions regarding seismic fragilities, release fractions, SFP inventories, long-term habitability criteria, and site population densities that when taken together result in a net beneficial result.

4.4.1.4.2.2 Spent Fuel Pool Group 2—PWR and BWR Mark III reactors with nonshared spent fuel pool

Table 24 Consequences Extending Beyond 50 Miles Sensitivity Analysis of Net Benefits for Low-Density Spent Fuel Pool Storage—Group 2 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$860,600	$755,900	$483,700	$20,609,300	$18,103,500	$11,583,500	$350,842,800	$308,185,900	$197,191,800
Occupational Health (Accident)	$500	$400	$300	$6,600	$5,800	$3,700	$109,400	$96,100	$61,500
Offsite Property	$1,860,702	$1,634,470	$1,045,811	$28,788,238	$25,288,046	$16,180,479	$402,559,059	$353,614,274	$226,259,013
Onsite Property	$3,250	$2,840	$1,630	$51,800	$44,280	$25,550	$201,170	$173,800	$103,350
Total Benefits	***$2,725,100***	***$2,393,600***	***$1,531,400***	***$49,455,900***	***$43,441,600***	***$27,793,200***	***$753,712,400***	***$662,070,100***	***$423,615,700***
Occupational Health (Routine)	-$27,200	-$29,100	-$28,900	-$27,200	-$29,100	-$28,900	-$27,200	-$29,100	-$28,900
Industry Implementation	-$51,370,000	-$53,800,000	-$51,330,000	-$51,370,000	-$53,800,000	-$51,330,000	-$51,370,000	-$53,800,000	-$51,330,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$51,397,200***	***-$53,829,100***	***-$51,358,900***	***-$51,397,200***	***-$53,829,100***	***-$51,358,900***	***-$51,397,200***	***-$53,829,100***	***-$51,358,900***
Net Benefit	**-$48,672,000**	**-$51,436,000**	**-$49,828,000**	**-$1,941,000**	**-$10,388,000**	**-$23,566,000**	**$702,315,000**	**$608,241,000**	**$372,257,000**

nc = not calculated

As shown in Table 24, calculated net benefits for requiring low-density SFP storage when considering consequences beyond 80 kilometers (50 miles) does not achieve a positive net benefit for either the low estimate or base cases presented. The high estimates show a positive net benefit of between $372 and $702 million. In comparison to the base case, the high estimate includes additional conservative assumptions regarding seismic fragilities, release fractions, SFP inventories, long-term habitability criteria, and site population densities that when taken together result in a net beneficial result.

4.4.1.4.2.3 Spent Fuel Pool Group 3 – New power reactors with non-shared spent fuel pool

Table 25 Consequences Extending Beyond 50 Miles Sensitivity Analysis of Net Benefits for Low-Density Spent Fuel Pool Storage—Group 3 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$844,600	$638,700	$266,500	$23,666,800	$17,898,700	$7,468,900	$263,568,800	$199,331,200	$83,178,000
Occupational Health (Accident)	$700	$600	$200	$9,500	$7,200	$3,000	$96,700	$73,100	$30,500
Offsite Property	$1,546,992	$1,169,956	$488,205	$27,166,671	$20,545,551	$8,573,353	$262,776,843	$198,732,300	$82,928,034
Onsite Property	$4,780	$3,560	$1,320	$75,910	$55,830	$20,950	$1,062,030	$781,900	$293,960
Total Benefits	***$2,397,100***	***$1,812,800***	***$756,200***	***$50,918,900***	***$38,507,300***	***$16,066,200***	***$527,504,400***	***$398,918,500***	***$166,430,500***
Occupational Health (Routine)	-$14,500	-$12,900	-$6,400	-$14,500	-$12,900	-$6,400	-$14,500	-$12,900	-$6,400
Industry Implementation	-$42,410,000	-$35,750,000	-$16,740,000	-$42,410,000	-$35,750,000	-$16,740,000	-$42,410,000	-$35,750,000	-$16,740,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$42,424,500***	***-$35,762,900***	***-$16,746,400***	***-$42,424,500***	***-$35,762,900***	***-$16,746,400***	***-$42,424,500***	***-$35,762,900***	***-$16,746,400***
Net Benefit	**-$40,027,000**	**-$33,950,000**	**-$15,990,000**	**$8,494,000**	**$2,744,000**	**-$680,000**	**$485,080,000**	**$363,156,000**	**$149,684,000**

nc = not calculated

As shown in Table 25, the dollar per person-rem sensitivity analysis does not achieve a positive net benefit when considering consequences beyond 80 kilometers (50 miles) for four of the nine cases presented. Two cases, the 2-percent and 3-percent discounted base cases and the high estimates show a positive net benefit range of between $2.7 and $485 million.

4.4.1.4.2.4 Spent Fuel Pool Group 4—Reactor units with a shared spent fuel pool

Table 26 Consequences Extending Beyond 50 Miles Sensitivity Analysis of Net Benefits for Low-Density Spent Fuel Pool Storage—Group 4 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$853,200	$758,100	$501,500	$24,572,200	$21,831,600	$14,441,900	$560,905,000	$498,344,700	$329,661,900
Occupational Health (Accident)	$500	$400	$300	$6,000	$5,400	$3,600	$125,000	$111,100	$73,500
Offsite Property	$1,898,771	$1,686,992	$1,115,969	$39,619,961	$35,200,961	$23,285,923	$779,796,081	$692,821,772	$458,311,191
Onsite Property	$3,050	$2,640	$1,520	$47,620	$41,160	$24,570	$1,349,250	$1,168,370	$700,210
Total Benefits	***$2,755,500***	***$2,448,100***	***$1,619,300***	***$64,245,800***	***$57,079,100***	***$37,756,000***	***$1,342,175,300***	***$1,192,445,900***	***$788,746,800***
Occupational Health (Routine)	-$22,700	-$24,700	-$24,800	-$22,700	-$24,700	-$24,800	-$22,700	-$24,700	-$24,800
Industry Implementation	-$48,780,000	-$50,410,000	-$46,390,000	-$48,780,000	-$50,410,000	-$46,390,000	-$48,780,000	-$50,410,000	-$46,390,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$48,802,700***	***-$50,434,700***	***-$46,414,800***	***-$48,802,700***	***-$50,434,700***	***-$46,414,800***	***-$48,802,700***	***-$50,434,700***	***-$46,414,800***
Net Benefit	**-$46,047,000**	**-$47,987,000**	**-$44,796,000**	**$15,443,000**	**$6,644,000**	**-$8,659,000**	**$1,293,373,000**	**$1,142,011,000**	**$742,332,000**

nc = not calculated

As shown in Table 26, the dollar per person-rem sensitivity analysis does not achieve a positive net benefit when considering consequences beyond 80 kilometers (50 miles) for four of the nine cases presented. Two cases, the 2-percent and 3-percent discounted base cases and the high estimates show a positive net benefit range of between $6.6 and $1,293 million.

4.4.1.4.3 Combined Effect of Consequences Extending Beyond 50 Miles and Dollar per Person-Rem Conversion Factor

This sensitivity analysis considers the combined effects of extending the analysis of consequences beyond 50 miles from the site and increasing the dollar per person-rem conversion value from $2,000 to $4,000 per person-rem averted. The combined effects of these two variables on the calculated net benefits are provided below.

4.4.1.4.3.1 Spent Fuel Pool Group 1 – BWR Mark I and Mark II reactors with non-shared spent fuel pool

Table 27 Combined Sensitivity Analysis that Analyzes Consequences beyond 50 Miles Using a Revised Dollar per Person-Rem Conversion Factor on the Net Benefits for Low-Density Spent Fuel Pool Storage—Group 1 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$1,006,600	$897,800	$600,800	$45,671,400	$40,734,600	$27,261,400	$610,863,800	$544,835,000	$364,625,600
Occupational Health (Accident)	$1,884	$1,680	$1,124	$17,158	$15,304	$10,242	$210,074	$187,367	$125,394
Offsite Property	$573,290	$511,323	$342,198	$16,358,429	$14,590,231	$9,764,373	$323,691,221	$288,703,133	$193,211,821
Onsite Property	$5,990	$5,280	$3,150	$67,520	$58,650	$35,470	$1,139,040	$989,660	$598,900
Total Benefits	***$1,587,800***	***$1,416,100***	***$947,300***	***$62,114,500***	***$55,398,800***	***$37,071,500***	***$935,904,100***	***$834,715,200***	***$558,561,700***
Occupational Health (Routine)	-$50,800	-$55,600	-$56,400	-$50,800	-$55,600	-$56,400	-$50,800	-$55,600	-$56,400
Industry Implementation	-$52,610,000	-$55,170,000	-$52,280,000	-$52,610,000	-$55,170,000	-$52,280,000	-$52,610,000	-$55,170,000	-$52,280,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$52,660,800***	***-$55,225,600***	***-$52,336,400***	***-$52,660,800***	***-$55,225,600***	***-$52,336,400***	***-$52,660,800***	***-$55,225,600***	***-$52,336,400***
Net Benefit	**-$51,073,000**	**-$53,810,000**	**-$51,389,000**	**$9,454,000**	**$173,000**	**-$15,265,000**	**$883,243,000**	**$779,490,000**	**$506,225,000**

nc = not calculated

As shown in Table 27, calculated net benefits for requiring low-density SFP storage when considering consequences beyond 50 miles combined with a revised dollar per person-rem

conversion factor does not achieve a positive net benefit for four of the nine cases presented. Two cases, the 2-percent and 3-percent discounted base cases and the high estimates show a positive net benefit range of between $173,000 and $883 million.

4.4.1.4.3.2 Spent Fuel Pool Group 2—PWR and BWR Mark III reactors with nonshared spent fuel pool

Table 28 Combined Sensitivity Analysis that Analyzes Consequences beyond 50 Miles Using a Revised Dollar per Person-Rem Conversion Factor on the Net Benefits for Low-Density Spent Fuel Pool Storage—Group 2 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$1,721,200	$1,511,800	$967,400	$41,218,600	$36,207,000	$23,167,000	$701,685,600	$616,371,800	$394,383,600
Occupational Health (Accident)	$1,000	$800	$600	$13,200	$11,600	$7,400	$218,800	$192,200	$123,000
Offsite Property	$1,860,702	$1,634,470	$1,045,811	$28,788,238	$25,288,046	$16,180,479	$402,559,059	$353,614,274	$226,259,013
Onsite Property	$3,250	$2,840	$1,630	$51,800	$44,280	$25,550	$201,170	$173,800	$103,350
Total Benefits	***$3,586,200***	***$3,149,900***	***$2,015,400***	***$70,071,800***	***$61,550,900***	***$39,380,400***	***$1,104,664,600***	***$970,352,100***	***$620,869,000***
Occupational Health (Routine)	-$54,400	-$58,200	-$57,800	-$54,400	-$58,200	-$57,800	-$54,400	-$58,200	-$57,800
Industry Implementation	-$51,370,000	-$53,800,000	-$51,330,000	-$51,370,000	-$53,800,000	-$51,330,000	-$51,370,000	-$53,800,000	-$51,330,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$51,424,400***	***-$53,858,200***	***-$51,387,800***	***-$51,424,400***	***-$53,858,200***	***-$51,387,800***	***-$51,424,400***	***-$53,858,200***	***-$51,387,800***
Net Benefit	**-$47,838,000**	**-$50,708,000**	**-$49,372,000**	**$18,647,000**	**$7,693,000**	**-$12,007,000**	**$1,053,240,000**	**$916,494,000**	**$569,481,000**

nc = not calculated

As shown in Table 28, calculated net benefits for requiring low-density SFP storage when considering consequences beyond 50-miles combined with a revised dollar per person-rem conversion factor does not achieve a positive net benefit for four of the nine cases presented. Two cases, the 2-percent and 3-percent discounted base cases and the high estimates show a positive net benefit range of between $7.7 and $1,053 million.

4.4.1.4.3.3 Spent Fuel Pool Group 3 – AP1000 power reactors with non-shared spent fuel pool

Table 29 Combined Sensitivity Analysis that Analyzes Consequences beyond 50 Miles Using a Revised Dollar per Person-Rem Conversion Factor on the Net Benefits for Low-Density Spent Fuel Pool Storage—Group 3 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$1,689,200	$1,277,400	$533,000	$47,333,600	$35,797,400	$14,937,800	$527,137,600	$398,662,400	$166,356,000
Occupational Health (Accident)	$1,400	$1,200	$400	$19,000	$14,400	$6,000	$193,400	$146,200	$61,000
Offsite Property	$1,546,992	$1,169,956	$488,205	$27,166,671	$20,545,551	$8,573,353	$262,776,843	$198,732,300	$82,928,034
Onsite Property	$4,780	$3,560	$1,320	$75,910	$55,830	$20,950	$1,062,030	$781,900	$293,960
Total Benefits	***$3,242,400***	***$2,452,100***	***$1,022,900***	***$74,595,200***	***$56,413,200***	***$23,538,100***	***$791,169,900***	***$598,322,800***	***$249,639,000***
Occupational Health (Routine)	-$29,000	-$25,800	-$12,800	-$29,000	-$25,800	-$12,800	-$29,000	-$25,800	-$12,800
Industry Implementation	-$42,410,000	-$35,750,000	-$16,740,000	-$42,410,000	-$35,750,000	-$16,740,000	-$42,410,000	-$35,750,000	-$16,740,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$42,439,000***	***-$35,775,800***	***-$16,752,800***	***-$42,439,000***	***-$35,775,800***	***-$16,752,800***	***-$42,439,000***	***-$35,775,800***	***-$16,752,800***
Net Benefit	**-$39,197,000**	**-$33,324,000**	**-$15,730,000**	**$32,156,000**	**$20,637,000**	**$6,785,000**	**$748,731,000**	**$562,547,000**	**$232,886,000**

nc = not calculated

As shown in Table 29, calculated net benefits for requiring low-density SFP storage when considering consequences beyond 50 miles combined with a revised dollar per person-rem conversion factor does not achieve a positive net benefit for the low estimate cases presented. The base cases and high estimates show a positive net benefit range of between $6.8 and $748 million.

4.4.1.4.3.4 Spent Fuel Pool Group 4 – Reactor units with a shared spent fuel pool

Table 30 Combined Sensitivity Analysis that Analyzes Consequences beyond 50 Miles Using a Revised Dollar per Person-Rem Conversion Factor on the Net Benefits for Low-Density Spent Fuel Pool Storage—Group 4 Spent Fuel Pool

Attribute	Low Estimate (2012 dollars)			Base Case (2012 dollars)			High Estimate (2012 dollars)		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
Public Health (Accident)	$1,706,400	$1,516,200	$1,003,000	$49,144,400	$43,663,200	$28,883,800	$1,121,810,000	$996,689,400	$659,323,800
Occupational Health (Accident)	$1,000	$800	$600	$12,000	$10,800	$7,200	$250,000	$222,200	$147,000
Offsite Property	$1,898,771	$1,686,992	$1,115,969	$39,619,961	$35,200,961	$23,285,923	$779,796,081	$692,821,772	$458,311,191
Onsite Property	$3,050	$2,640	$1,520	$47,620	$41,160	$24,570	$1,349,250	$1,168,370	$700,210
Total Benefits	***$3,609,200***	***$3,206,600***	***$2,121,100***	***$88,824,000***	***$78,916,100***	***$52,201,500***	***$1,903,205,300***	***$1,690,901,700***	***$1,118,482,200***
Occupational Health (Routine)	-$45,400	-$49,400	-$49,600	-$45,400	-$49,400	-$49,600	-$45,400	-$49,400	-$49,600
Industry Implementation	-$48,780,000	-$50,410,000	-$46,390,000	-$48,780,000	-$50,410,000	-$46,390,000	-$48,780,000	-$50,410,000	-$46,390,000
Industry Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Implementation	nc	nc	nc	nc	nc	nc	nc	nc	nc
NRC Operation	nc	nc	nc	nc	nc	nc	nc	nc	nc
Total Costs	***-$48,825,400***	***-$50,459,400***	***-$46,439,600***	***-$48,825,400***	***-$50,459,400***	***-$46,439,600***	***-$48,825,400***	***-$50,459,400***	***-$46,439,600***
Net Benefit	**-$45,216,000**	**-$47,253,000**	**-$44,319,000**	**$39,999,000**	**$28,457,000**	**$5,762,000**	**$1,854,380,000**	**$1,640,442,000**	**$1,072,043,000**

nc = not calculated

As shown in Table 30, calculated net benefits for requiring low-density SFP storage when considering consequences beyond 50-miles combined with a revised dollar per person-rem conversion factor does not achieve a positive net benefit for the low estimate cases presented. The base cases and high estimates show a positive net benefit range of between $5.8 and $1,854 million.

4.4.2 Disaggregation

In order to comply with the guidance provided in Section 4.3.2, "Criteria for the Treatment of Individual Requirements" of the Regulatory Analysis Guidelines (Ref. 3), the NRC conducted a screening review to ensure that the aggregate analysis does not mask the inclusion of individual requirements that are not cost-beneficial when considered individually and not necessary to meet the stated objectives. Consistent with the Regulatory Analysis Guidelines, the NRC evaluated, on a disaggregated basis, each new regulatory provision expected to result in incremental costs. Based on this screening review, the NRC did not identify any requirements needing further consideration. The NRC believes that each of these provisions described in Appendix section C.3 is necessary in the aggregate for the expedited transfer of spent fuel to DSCs. However, the NRC finds that requiring the accelerated transfer to DSCs would provide only limited safety benefits, far below the threshold that the NRC uses to inform it regulatory decisionmaking, and would not be cost-justified.

4.5 Decision Rationale

This section presents the decision rationale, including the basis for selection, and the decision criteria used.

Table 10 shows that the calculated benefits for requiring the low-density SFP storage alternative for the low estimate and base case are less than industry costs to achieve a low-density spent fuel loading pattern for each SFP group. As might be expected for estimates that include a compounding of the most conservative assumptions, all of the SFP group high estimate cases result in calculated benefits that are greater than the estimated costs.

Similar to the seismic event analyzed for the SFPS, no offsite early fatalities are calculated to occur. This results from the following two reasons:

(1) In comparison to reactors, SFPs have a larger proportion of longer-lived radionuclides, which are less likely to cause the significant doses required for acute health effects.

(2) Despite the large releases for certain predicted SFP accident progressions, the release from the most recently discharged fuel (which contains the shorter-lived radionuclides) is predicted to be insufficiently fast and insufficiently large to reach the acute thresholds associated with offsite early fatalities. When doses do exceed minimum levels for early fatalities, emergency response effectively prevents any early fatality risk, at least in part because the modeled accident progression results in releases that are long compared with the time needed for relocation.

In addition, the predicted long-term exposure of the population, which could result in latent cancer fatality risk, is also low for the following reasons:

(1) The individual latent individual latent cancer fatality risk within 0 to 10 miles is predicted to be on the order of 2.4×10^{-10} to 1.5×10^{-8} per year, based on the linear no threshold (LNT) dose response model.

(2) The risk within 10 miles of the analyzed accident is dominated by low dose received at a low dose rate. Using truncation levels that do not quantify the effects of doses below 620 mrem per year (i.e., those arising from representative background radiation including average annual medical exposures) reduces the estimated individual LCF risk by up to a few orders of magnitude for the accident as modeled.

(3) Average individual latent cancer fatality risk is low but decreases slowly as a function of distance from the plant. Additionally, the predicted individual risks of latent cancer fatalities are dominated by long-term exposures to very lightly contaminated areas for which doses are small enough to be considered habitable.

Sensitivity studies provided in Section 4.4.1.6 show that there are cases using conservative assumptions in each sensitivity study in which the low-density spent fuel storage alternative was cost-justified. However, after considering the analysis results, operating history, and limited safety benefits of possible plant changes, the staff finds that further study would be unlikely to support future actions requiring expedited transfer.

The NRC staff identified other considerations discussed in Section 4.3.10 that would further reduce the quantified benefits and make the proposed alternative less justifiable.

The outcome of this cost-benefit analysis indicates that undertaking additional study of the low-density SFP storage alternative is not justified. Except in those cases where action is needed to ensure adequate protection of public health and safety, the process used by the NRC when considering additional regulatory requirements is to assess the potential benefits from new regulations against a safety benefit threshold (e.g., the safety goal screening) and the costs of implementing new requirements. The potential benefits of a requirement to expedite the removal of spent fuel from storage pools could be to reduce the risk to the public from possible accidents involving SFPs. Assessments of risk and changes in risk from possible actions involve identifying what can go wrong, what are the consequences, and how likely is it to occur.

In the case of hypothetical accidents involving SFPs, the assessments have shown that impacts on public health and safety can be avoided but that the potential economic consequences can be very large. However, the assessments also show that the design and construction of SFPs, the characteristics of the spent fuel assemblies, and the availability of mitigating systems result in a very low likelihood that radioactive materials would be released because of an accident affecting a SFP. This evaluation of a low probability, high consequence event is similar to previous NRC risk assessments and related regulatory analyses for potential issues related to nuclear reactor and SFPs.

Based on the NRC's assessment of the costs and benefits, the agency has concluded that the risk of beyond-design-basis accidents in SFPs, while not negligible, is sufficiently low, far below the threshold NRC uses to inform its regulatory decisionmaking, and that the added costs involved with expediting the movement of spent fuel from the pool to achieve low-density fuel pool storage is not warranted.

5. CONCLUSION

To determine if additional studies are needed to further assess potential regulatory action on expedited transfer, the staff has conducted an analysis of expedited transfer of spent fuel to dry cask storage, in accordance with the agency's current policies and guidance.

The safety goal screening evaluation concludes that SFP accidents are a small contributor to the overall risks for public health and safety (less than one percent of the QHOs) and therefore any reductions in risk associated with expedited transfer of spent fuel would only have a marginal safety benefit. Due to the safety goal screening criterion not being satisfied, the staff finds that no further generic assessments are warranted. Although the regulatory analysis guidelines would normally stop the evaluation at this step because the risk is a small fraction of the safety goals, the staff proceeded to perform a cost-benefit analysis to provide additional information for the Commission's consideration.

The staff conducted a cost-benefit analysis, which finds that the added costs involved with expedited transfer of spent fuel to dry cask storage to achieve the low-density SFP storage alternative are not warranted in light of the marginal safety benefits from such an action. The combination of high estimates for important parameters assumed in some of the sensitivity cases presented in this analysis result in large economic consequences, such that, the calculated benefits from expedited transfer of spent fuel to dry cask storage for those cases outweigh the associated costs. However, even in these cases, there is only a marginal safety improvement in terms of public health and safety. In the staff's judgment, the assumptions made in this analysis were selected in a generally conservative manner such that the base case is the primary basis for the staff's recommendation. Based on the generic assessment and the other considerations detailed in this paper, the staff finds that additional studies are not needed to reasonably conclude that the expedited transfer of spent fuel to dry cask storage would provide only a marginal increase in the overall protection of public health and safety, and would not be warranted due to the expected implementation costs.

No further regulatory action is recommended for the resolution of this issue. The outcome of this cost-benefit analysis indicates that undertaking additional study of the low-density SFP storage alternative is not justified.

6. REFERENCES

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APPENDIX A: SPENT FUEL POOL CHARACTERISTICS

A.1 Spent Fuel Pool Configurations

The configuration of spent fuel storage pools is similar for most nuclear reactor and away-from-reactor storage facilities. The pools are rectangular in cross section and approximately 12 meters (40 feet) deep. Fuel assemblies are placed vertically in storage racks that maintain an adequate spacing to prevent criticality and to promote natural convective cooling in a water medium. The pools themselves are constructed of reinforced concrete with sufficient thickness to meet radiation shielding and structural requirements, and are lined with stainless steel plates of approximately 2.5-center (1/4-inch) thickness to ensure a leak-tight system.

A.1.1 Boiling-Water Reactors with Mark I and Mark II Containments

Boiling-water reactors (BWRs) with Mark I and Mark II containments are designed with the SFP located within the reactor building as shown in Figure 3. The bottom of the SFP is usually elevated approximately 15 meters (50 feet) above grade, which places the top of the pool at the level of the operating floor. The enclosing superstructure above the pool is typically a low-leakage steel, industrial-type building designed to house cranes that are used to move reactor components, spent fuel, and spent fuel casks. For a few reactor buildings, the enclosing superstructure is a reinforced concrete structure with strength similar to the lower portions of the reactor building, as depicted in Figure 3.

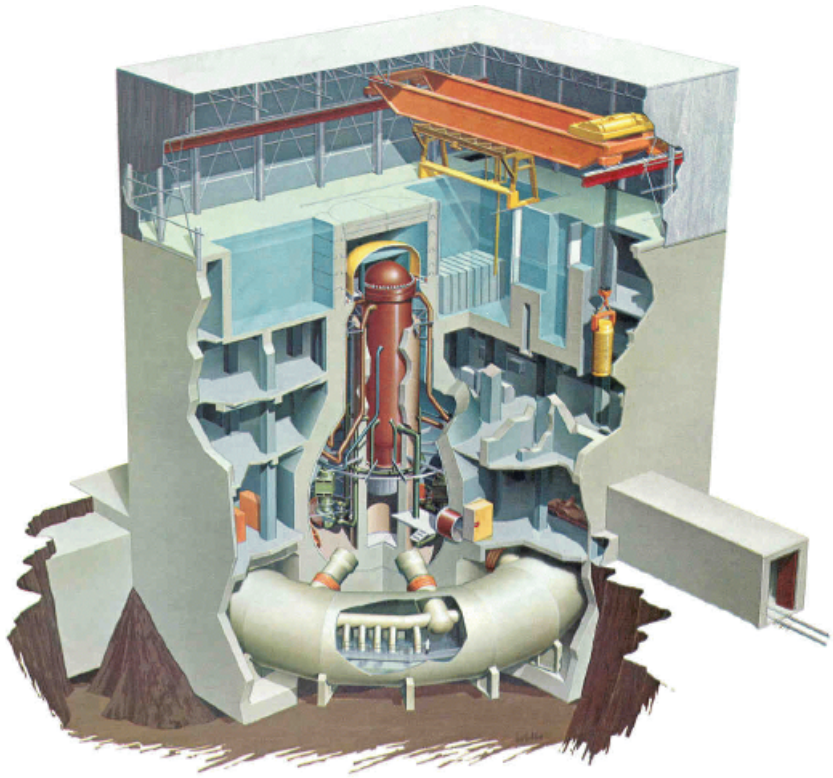

Figure 3 Schematic of a GE BWR Mark I Containment

Source: Reactor Concepts Manual: Boiling Water Reactor (BWR) Systems, p. 3–16 (Ref.A.1).

A.1.2 Pressurized-Water Reactors and Boiling-Water Reactors with Mark III Containments

Figure 4 shows the location of the SFP for the newer BWR Mark III design, which call for a ground-level storage pool to reduce seismic loads. The fuel building is located adjacent to the reactor building and is accessible for fuel servicing during plant operation. A lined fuel pool is used for the storage and servicing of spent fuel and the preparation of new fuel for insertion into the reactor. An area of the pool, separated by gates, is used for transfer of fuel to the reactor servicing pools located in the reactor building, and the receiving of spent fuel discharged from the reactor using a transfer tube. Another area of the fuel storage pool, also separated by gates, is used for the loading and decontamination of equipment and its containers for offsite shipping. Some of these SFPs are located below grade.

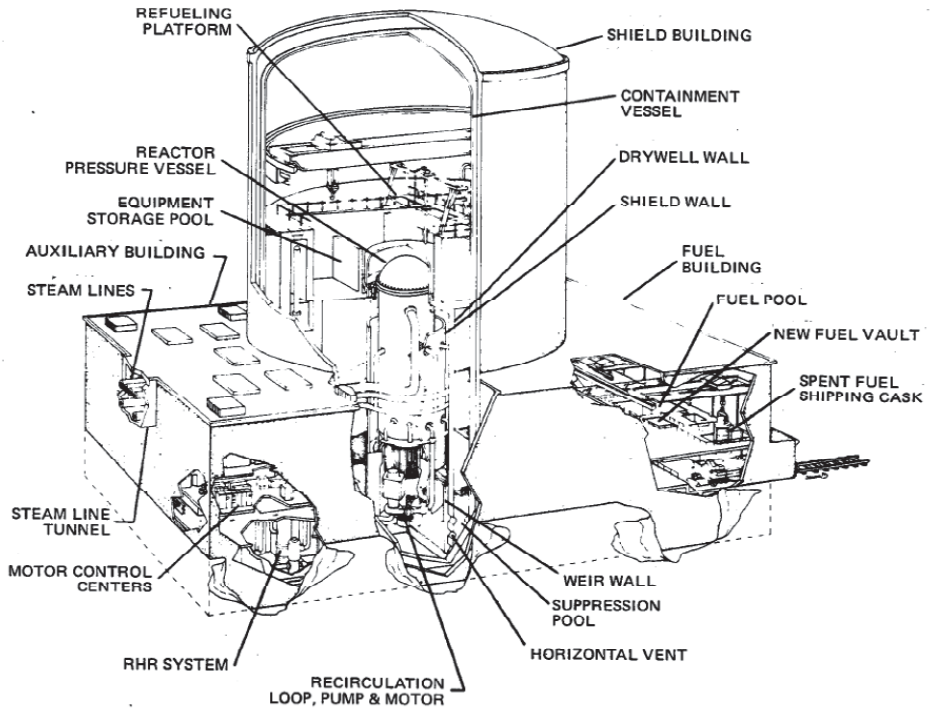

Figure 4 Schematic of a BWR Mark III reactor layout

Source: BWR/6 General Description of a Boiling Water Reactor, Figure 7-1 (Ref. A.2).

Pressurized-water reactor (PWR) designs have SFPs that are located close to grade level within the auxiliary building as shown in Figure 5. This design is typical of the fuel pool arrangement for PWRs.[20]

[20] The Shearon Harris spent fuel pools contain fuel from the Brunswick and Robinson reactors, but the BWR fuel is segregated from the PWR fuel and all transferred fuel has decayed for more than 10 years. The PWR pool reasonably represents this pool because the PWR fuel storage capacity is similar, the power and quantity of each representative refueling batch bounds the Harris conditions, and the stored BWR fuel is segregated such that it would not increase the severity of any potential release.

Nuclear power plant sites that contain two PWR reactors are usually arranged in a mirror image fashion, with the two SFPs (or a shared pool) located in a common area adjoining both reactor buildings or contained within the seismic Category I auxiliary building around or adjacent to the containment building. For single plant or two-plant arrangements, the building covering the SFP and crane structures is typically an ordinary steel industrial building.

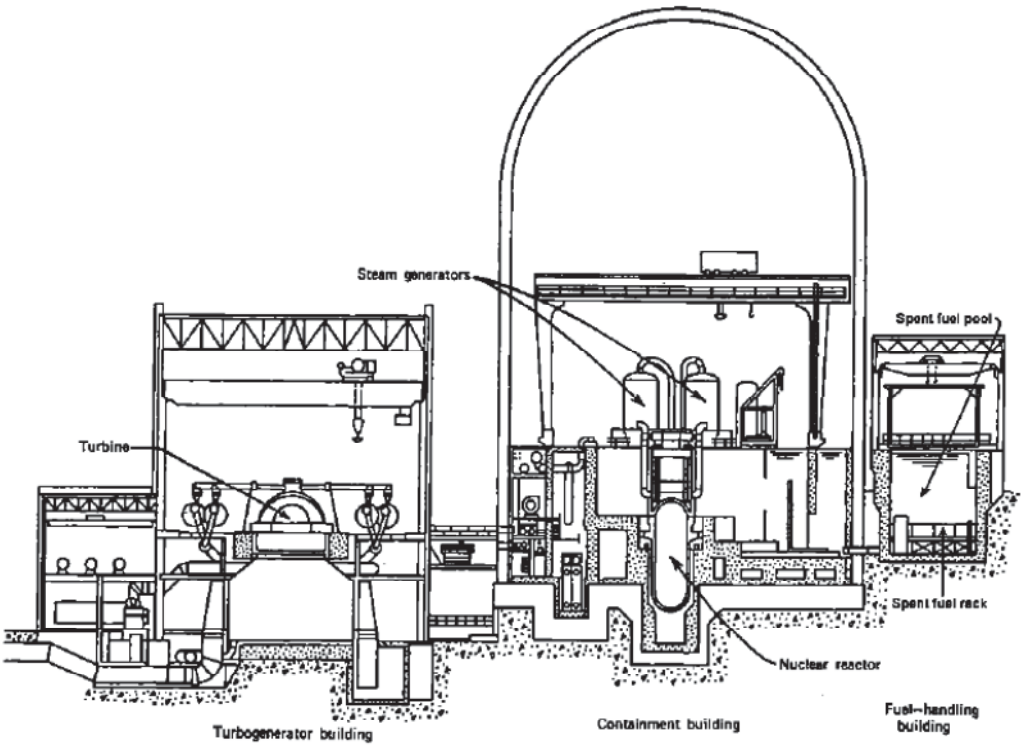

Figure 5 Schematic of a PWR layout

Source: Duderstadt and Hamilton, Figure 3-4 (Ref. A.3).

A.1.3 New Reactors

For the new reactors, the spent fuel storage facility is located within the seismic Category I auxiliary building fuel handling area. The walls of the SFP are an integral part of the seismic Category I auxiliary building structure as shown in Figure 6. The facility is protected from the effects of natural phenomena, such as earthquakes, wind and tornados, floods, and external missiles.

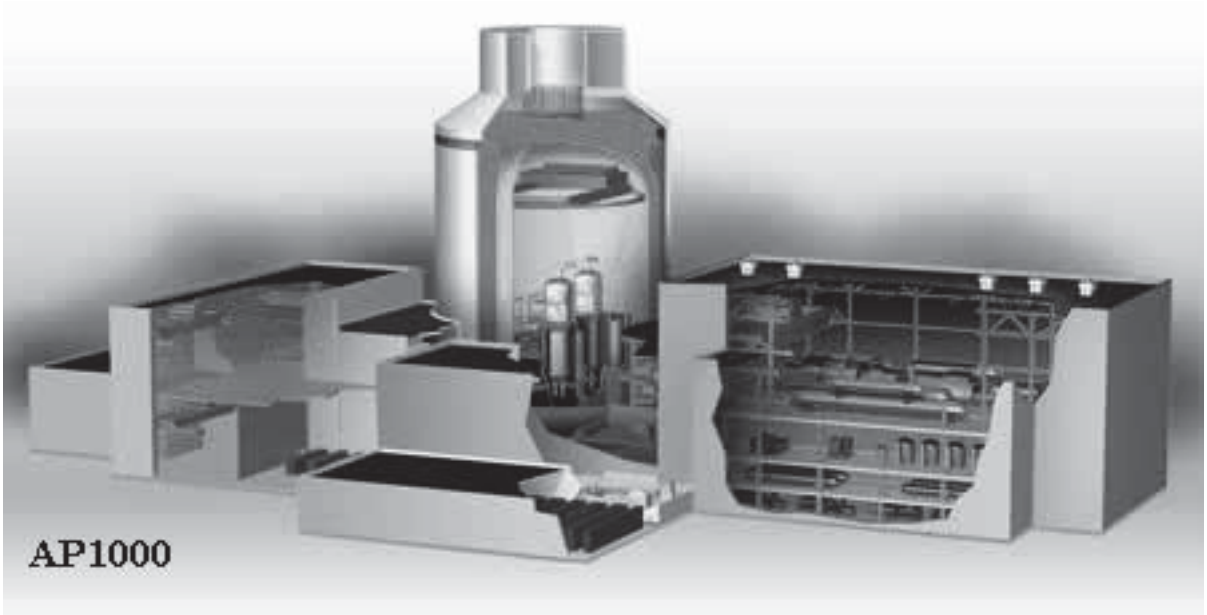

Figure 6 Schematic of an AP1000 reactor layout

Source: Nuclear Street (Ref. A.4).

A.1.4 Spent Fuel Pools at Non-Operating Plants

A SFP at non-operating plants is a special situation in which the reactor unit is no longer operating and spent fuel is stored in the unit's SFP for safe storage until it is placed in an ISFSI or shipped to a long-term Federal repository.

This grouping of pools was not evaluated due to its much lower potential for runaway zirconium oxidation. No further analysis is performed in this analysis for this grouping.

A.1.5 Decommissioned Plant Spent Fuel

A decommissioned plant spent fuel is a special situation in which the licensee requested a license for an independent spent fuel storage installation (ISFSI) to store the reactor unit's spent fuel. The spent fuel was relocated from wet storage in a SFP to dry storage containers at the ISFSI. The spent fuel will be held at the ISFSI until the U.S. Department of Energy is prepared to take possession of the spent fuel and transport it to a long-term repository.

This grouping also includes the GE–Hitachi Morris ISFSI, which is a wet pool storage design and is the only wet ''away from reactor'' ISFSI of its kind in the U.S. The major components of the Morris ISFSI include the stainless steel lined concrete storage basins, the pool structure, the spent fuel storage grid structure and fuel storage baskets that can store BWR spent fuel assemblies or PWR spent fuel assemblies, ancillary equipment necessary for the movement of spent nuclear fuel, e.g., cranes and basket grappling devices, and equipment necessary for the maintenance of the pool water quality and level (Ref. A.5). Because of the length of time that the discharged spent fuel stored at the Morris ISFSI has cooled, the licensee estimates that, based on evaporation rates, it will take approximately 140 days for the water level to expose the top of the stored fuel bundles (Ref. A.6). Furthermore, there is not sufficient energy in the stored fuel assemblies to ignite the fuel from either a partial or total loss of water.

Based on the characteristics of the spent fuel storage in this grouping, no further analysis is performed in this analysis for this grouping.

A.2 Spent Fuel Storage Options

The technologies available for spent fuel storage fall broadly into two categories—wet and dry—distinguished according to the cooling medium used. The wet option has historically been used for temporary storage in anticipation of the next step in the fuel cycle. More recently, a variety of dry storage options have been developed and applied in the U.S. and international markets.

A.2.1 Wet Storage

The majority of U.S. nuclear power plant spent fuel is stored in water pool storage (i.e., SFPs). SFPs have been used for storage of spent fuel as an established practice since the early days of nuclear power, due among other things, to the excellent properties of water for heat removal and shielding. The majority of reactor SFPs has been re-racked once, and some several times, to increase in-pool storage capacity. These pools are designed to the following principles as discussed in NUREG-0800, "Standard Review Plan," Section 9.1.2, "New and Spent Fuel Storage," (Ref. A.7):

- the capability to withstand and protect against natural phenomena (e.g., safe shutdown earthquake, design-basis tornado)
- the effectiveness of natural circulation of water through the spent fuel storage racks
- the ability to retain water and minimize leakage, which should be detectable, collectable, and quantifiable
- the configuration of the new fuel vault, the spent fuel storage pool, and their handling areas to preclude accidental falls of heavy objects on the new and spent fuel
- the ability to provide both radiological shielding for personnel by maintaining adequate water levels in the SFP
- the use of design features to maintain an adequate water inventory in the SFP under accident condition (e.g., weirs and gates, absence of unnecessary drains, and proper piping penetration levels)
- the use of appropriate monitoring systems to detect SFP water levels, pool temperature, building radiation levels, and to ensure an adequate degree of subcriticality

While there are many common features between SFPs, there are design differences.

A.2.1.1 Location

A.2.1.1.1 At-reactor pool located above grade

For boiling water reactor (BWR) Mark I and II designs, the SFP structures are located in the reactor building at an elevation several stories above grade.

A.2.1.1.2 At-reactor pool located near or below grade

The SFPs at pressurized water reactors (PWRs) and BWR Mark III operating reactors in the U.S. are located with the bottom of the pool at or below plant grade level. Because of the lower elevation, the seismic response is relatively low in comparison to the elevated pools in the BWR Mark I and Mark II plants. Some pools are located below grade, often in bedrock, such that even if a hole in the pool formed, it cannot rapidly drain this pool.

A.2.1.1.3 Away-from reactor or non-operating reactor pool

Away-from-reactor pools are used to provide interim spent fuel storage. Typically, they are divided into pools at the reactor site and pools away from the reactor site or offsite although this distinction is not important to this analysis. True away-from-reactor pools are independent of the reactor and all its services and can continue to operate after the reactor has been finally shut down and decommissioned. There are pools, however, that are located at reactors that are shut down but rely extensively on reactor services such as cooling water and water treatment, ventilation and electrical supplies. When reactors are shut down, special arrangements are usually taken because it could be impractical or uneconomic to continue to operate costly reactor-derived services if the spent fuel must remain in storage onsite for long periods. Dry storage facilities generally remove decay heat by passive cooling and have lower operating costs.

A.2.1.2 Functional Configuration

A.2.1.2.1 Dedicated pool

This is the simplest layout adopted for nuclear power plants in which a SFP supports a single nuclear power plant unit.

A.2.1.2.2 Shared pool

There are cases in which nuclear power plant units may be connected by water gates to share a SFP.

A.2.2 Dry Storage

Numerous companies supply dry storage technologies to U.S. commercial nuclear power plants, as shown in Table 70 located in Appendix F to this document. These dry storage cask systems[21] (DSCs) are certified by the NRC for storage of high burnup spent fuel (i.e., burnups greater than 45 GWd/MTU), using both regional and uniform loading of spent fuel in the packages. Although the dry storage design differs in design details, capacity, and loading steps, the scope of this analysis is limited to generic dry storage technologies, in order to develop a context for the cost-benefit analysis described in subsequent sections of this document.

A.3 Rack Designs

[21] The term dry storage cask system (DSC) includes dual-purpose canister based systems, dual-purpose casks, and storage-only dry storage casks and canister systems.

The design of storage racks and fuel element holder configurations varies considerably from facility to facility, both in general appearance and in details. In March 1979, the NRC issued NUREG/CR-0649, "Spent Fuel Heatup following Loss of Water during Storage" (Ref. A.8), which provided an analysis of spent fuel heatup following a hypothetical accident involving drainage of the storage pool. The report included analysis to assess the effect of decay time, fuel element design, storage rack design, packing density, room ventilation, drainage level, and other variables on the heatup characteristics of spent fuel stored in a SFP to predict the conditions under which clad failure would occur. The report concluded that the likelihood of clad failure caused by rupture or melting following a complete drainage is extremely dependent on the storage configuration and the spent fuel decay period. Furthermore, the minimum prerequisite decay time to preclude clad failures may vary from less than 10 days for some storage configurations to several years for others. The potential for reducing this critical decay time either by making reasonable design modifications or by providing effective emergency countermeasures was found to be significant. The NUREG/CR-0649 analysis assumed in most cases that a 41-centimeter (16-inch) open space is maintained between the baseplate and the bottom of the pool and between the sidewalls and the outermost basket or holder. The rack designs evaluated had center-to-center fuel element spacing that ranged from 21.6 centimeters (8.5 inches) to 53 centimeters (21 inches).

NUREG-1353, "Regulatory Analysis for the Resolution of Generic Issue 82, Beyond-Design-Basis Accidents in Spent Fuel Pools," which draws from the preceding report, concludes that if the decay heat level is high enough to heat the fuel rod cladding to about 900 degrees Celsius (C), the oxidation becomes self-sustaining, resulting in a Zircaloy cladding fire. NUREG-1353 used a conservative and bounding conditional probability of a Zircaloy cladding fire given a complete loss of water. The conservative and bounding values used were 1.0 for PWRs and 0.25 for BWRs in high-density configurations based on differences in assumed rack geometry.

NUREG/CR-6441, "Analysis of Spent Fuel Heatup following Loss of Water in a Spent Fuel Pool: A Users' Manual for the Computer Code SHARP" (Ref. A.9), was issued in 2002. This report included an analysis of spent fuel heatup, using representative design parameters and fuel loading assumptions. Sensitivity calculations were also performed in this NUREG to study the effect of fuel burnup, building ventilation rate, baseplate hole size, partial filling of the racks, and the amount of available space to the edge of the pool. The spent fuel heatup was found to be strongly affected by the total decay heat production in the pool, the availability of open spaces for airflow, and the building ventilation rate. SFP analyses performed by the NRC after this time do not use the SHARP computer code. Rather, the NRC uses the MELCOR computer code (owing to its mechanistic treatment of severe accident phenomena), with supporting analysis using the COBRA-SFS, FLOW3D, and Fluent codes, along with confirmatory experiments at Sandia National Laboratories.

The SFPS (draft) evaluated a BWR reference plant rack geometry with a cell pitch of 16 centimeters (6.3 inches); a closed rack design that inhibited or prevented cross-flow, while being relatively open at the top and bottom for axial flow; and a distance between the pool floor liner and the bottom of the rack baseplate of approximately 26 centimeters (10.2 inches).

A.4 REFERENCES

A.1 U.S. Nuclear Regulatory Commission (NRC) home page. "Reactor Concepts Manual: Boiling Water Reactor (BWR) Systems," [print graphic]. Retrieved from http://www.nrc.gov/reading-rm/basic-ref/teachers/03.pdf, accessed 10/30/2013.

A.2 GE Nuclear Energy, "BWR/6 General Description of a Boiling Water Reactor," [print graphic]. Retrieved from www4.ncsu.edu/~doster/NE405/Manuals/BWR6GeneralDescription.pdf, accessed July 15, 2013.

A.3 Duderstadt, J.J., and L.J. Hamilton, "Nuclear Reactor Analysis," John Wiley & Sons, New York, 1976.

A.4 Nuclear Street: Nuclear Powered Portal, "AP1000.jpg," [print graphic]. Retrieved from http://nuclearstreet.com/images/img/ap1000.jpg, accessed 7/31/2013.

A.5 U.S. Nuclear Regulatory Commission (NRC). "Risk Assessment of Operational Events Handbook," Volume 2, Revision 1.01, January 2008 (ADAMS Accession No. ML080300179).

A.6 U.S. Nuclear Regulatory Commission (NRC). "General Electric Company Notice of Issuance of an Environmental Assessment and Finding of No Significant Impact for License Renewal of the Morris Operation Independent Spent Fuel Storage Installation," 69 FR 71082, December 8, 2004.

A.7 U.S. Nuclear Regulatory Commission (NRC). NUREG-0800, "Standard Review Plan," Section 9.1.2, Revision 4, "New and Spent Fuel Storage," March 2007.

A.8 U.S. Nuclear Regulatory Commission (NRC). NUREG/CR–0649, "Spent Fuel Heat up Following Loss of Water during Storage," 1979.

A.9 U.S. Nuclear Regulatory Commission (NRC). NUREG/CR-6441, "Analysis of Spent Fuel Heatup following Loss of Water in a Spent Fuel Pool: A User's Manual for the Computer Code SHARP," March 2002 (ADAMS Accession No. ML021050336).

APPENDIX B: SPENT FUEL STORAGE STRATEGIES

B.1 Interim Storage Options to Expand Onsite Storage

The delay in the construction of the geologic repository mandated by Congress has caused nuclear power plants to store used fuel on site for longer than originally intended. The result is that many nuclear plants are running out of existing storage capacity. When a plant's used fuel pool nears its designed capacity, a company has two options:

- **Re-racking**. The first choice is to re-rack the used SFP, moving the fuel assemblies closer together. Eventually, even re-racked pools reach their capacity.

- **Dry Containers**. Many U.S. nuclear power plants are storing used spent fuel in large, rugged containers made of steel or steel-reinforced concrete. Depending on the design, a container can hold up to 37 PWR fuel assemblies or 87 BWR fuel assemblies. The containers have a 20-year license. After 20 years, with NRC approval, the license could be extended for up to 40 years.

Building a dry storage facility at a plant site requires an initial investment of approximately $10 million to $20 million. Once the facility is operational, it may cost $5 million to $7 million a year for the maintenance and security of the facility and for adding more containers as storage needs grow (Ref. B.1).

While re-racking is the most used method for expanding at-reactor spent fuel storage capacity over the past 40 years, utility experience with dry storage applications has grown significantly. In addition to the implementation and continued operation of dry storage at operating plant sites, numerous nuclear power plants that have permanently ceased operation have offloaded spent fuel from storage pools to at-reactor ISFSIs to facilitate decommissioning of the SFPs.

B.2 Cask Loading Strategies

Two cask loading strategies used to manage cask loading are 1) full core reserve (FCR) margin, and 2) SFP inventories. The first strategy is just-in-time cask loading, in which casks are loaded with a goal of maintaining FCR in the SFP. The second type of cask loading strategy employs larger loading campaigns with a goal of achieving additional space above that required for FCR in order to space cask loading campaigns further apart. When implementing this cask loading strategy, a plant might load 10 to 12 casks following every other refueling rather than five to six casks following every refueling outage.

The benefits of just-in-time cask loading are that:

- It minimizes near-term capital and operating expenditures since only enough casks to maintain FCR are loaded.

- Cask loading crews also do not have long periods of time between cask loading campaigns and may result in shorter learning curves for the next cask loading campaign.

The risks associated with a just-in-time loading strategy include:

- unexpected maintenance that requires offloading the reactor core at a time when the SFP has less than one FCR

- unexpected delays in delivery of storage casks caused by licensing issues or fabrication delays that might affect FCR capability

- increased outage times because of space limitations in the SFP

Benefits associated with larger loading campaigns include:

- There are fewer cask loading campaigns over the life of the plant (although the same number of casks would be loaded over the life of the plant) resulting in cost savings associated with mobilization/demobilization for cask loading, training, and dry runs.

- If a company owns multiple sites with operating ISFSIs and cask loading equipment is shared between sites, this results in fewer shipments of cask handling equipment between sites and possible cost savings.

- Larger loading campaigns would also provide more margin in SFPs over FCR, such that unexpected maintenance requiring off-loading of the reactor core can be accomplished and unexpected delays in delivery of storage casks are more likely to be accommodated.

- A negative benefit is that costs associated with large loading campaigns include increases in near-term capital and operating budgets because of purchasing and loading casks sooner than in a just-in-time loading scenario.

Risks associated with larger loading campaigns include:

- Longer cask loading cycles (months rather than weeks) to complete a loading campaign and possible impacts on plant maintenance activities or other SFP activities.

- Impacts on workers involved in cask loading operations. Shutdown nuclear operating plants have loaded between 15 and 60 casks in extended campaigns with reasonable schedules.

B.3 References

B.1 Nuclear Energy Institute, 2013. "Nuclear Waste Disposal," Retrieved from http://www.nei.org/resourcesandstats/documentlibrary/nuclearwastedisposal/factsheet/safelymanagingusednuclearfuel/, accessed 7/10/2013.

APPENDIX C: ANALYSIS MODEL INFORMATION

C.1 Economic Modeling and Representative Plant Assumptions

C.1.1 Compliance with Existing NRC Requirements

The regulatory baseline assumes full compliance with existing NRC requirements, including current regulations and relevant orders. This is consistent with NUREG/BR-0058, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission," Rev. 4 (Ref. C.1), which states that "in evaluating a new requirement…, the staff should assume that all existing NRC and Agreement State requirements have been implemented." For the purpose of evaluating the potential benefits of expedited transfer of spent fuel to dry cask storage, this analysis used a conservative approach by crediting successful application of post-9/11 and post-Fukushima mitigation capabilities for the low-density SFP alternative and assumed no successful mitigation for the high-density SFP storage regulatory baseline.

The data and assumptions used in analyzing the quantifiable impacts associated with each proposed alternative are discussed in this section. Information on attributes affected by the proposed regulatory framework alternatives is obtained from experienced NRC staff and other sources as referenced. The NRC considers the potential differences between the new requirements and the current requirements and incorporates the proposed incremental changes into this cost-benefit analysis.

C.1.2 Base Year

All monetized costs are expressed in 2012 dollars. Ongoing costs of operation related to the alternatives are assumed to begin in 2014 unless otherwise stated, and are modeled on an annual cost basis.

Estimates are made for one-time implementation costs. The NRC assumes that these costs will be incurred in the first year of the analysis unless otherwise noted.

Estimates are made for recurring annual operating expenses. The values for annual operating expenses are modeled as a constant expense for each year of the analysis horizon. An annuity calculation was performed to discount these annual expenses to 2012 dollar values.

C.1.3 Discount Rates

In accordance with guidance from the Office of Management and Budget (OMB) Circular No. A-4 (Ref. C.2) and NUREG/BR-0058, Revision 4 (Ref. C.1), present-worth calculations are used to determine how much society would need to invest today to ensure that the designated dollar amount is available in a given year in the future. By using present-worth, costs and benefits, regardless of when the cost or benefit is incurred in time, are valued to a reference year for comparison. The choice of a discount rate, and its associated conceptual basis, is a topic of ongoing discussion within the Federal government. Based on OMB Circular No. A-4, present-worth calculations are presented using 3 percent and 7 percent real discount rates. A 3 percent discount rate approximates the real rate of return on long-term government debt, which serves as a proxy for the real rate of return on savings to reflect reliance on a social rate of time preference discounting concept. A 7 percent rate approximates the marginal pretax real rate of return on an average investment in the private sector, and is the appropriate discount rate whenever the main effect of a regulation is to displace or alter the use of capital in the

private sector. A 7 percent rate is consistent with an opportunity cost of capital concept to reflect the time value of resources directed to meet regulatory requirements.

C.1.4 Cost/Benefit Inflators

The consequences for some attributes are estimated based on the values published in the NRC Regulatory Analysis Handbook. Within the NRC Regulatory Analysis Handbook, the information in relation to severe reactor accident consequences is provided in previous year dollars.
To evaluate the costs and benefits consistently, the consequences are inflated. The most common inflator is the Consumer Price Index for all urban consumers (CPI-U), developed by the U.S. Department of Labor, Bureau of Labor Statistics. Using the CPI-U, the previous year dollars were converted to the year 2012. The formula to determine the amount in 2012 dollars is

$$\frac{\mathrm{CPIU}_{2012}}{\mathrm{CPIU}_{\text{Base Year}}} * \mathrm{Consequence}_{\text{Base Year}} = \mathrm{Consequence}_{2012}$$

Values of CPI-U used in this cost-benefit analysis are summarized in Table 31.

Table 31 Consumer Price Index—All Urban Consumers Inflator

Base Year	CPI-U Inflator for Year 2012
2005	1.1756
2006	1.1389
2007	1.1073
2008	1.0664
2009	1.0702
2010	1.0529
2011	1.0207

Source: U.S. Department of Labor, Bureau of Labor Statistics, “Databases, Tables & Calculators by Subject: CPI Inflation Calculator (Ref. C.3).

C.1.5 Description of Representative Plants

Representative BWR Mark I and Mark II (Group 1)

The representative Group 1 plant is a single unit boiling-water reactor (BWR) Mark I or Mark II reactor with a rated capacity of approximately 3,500 megawatts thermal (MW_t) and a unit dedicated SFP. The representative BWR reactor began operating in the 1970s and will reach the end of its renewed operating license by year 2037. The NRC assumes the reactor core contains 764 assemblies and the SFP has a capacity of approximately 3,055 assemblies in a high-density 1x4 loading configuration. This number is based on a pool capacity of 3,819 assemblies, reduced by 764 assemblies to accommodate a full core offload capability using the existing high-density racking. In a low-density configuration, the SFP stores 852 assemblies in which the newly discharged spent fuel is arranged in a 1x4 configuration and the remaining fuel assemblies arranged in a checkerboard pattern. The unit operates on 24-month cycles, discharging approximately 284 assemblies per cycle. The representative BWR has already implemented dry storage.

Representative PWR or BWR Mark III (Group 2)

The representative Group 2 plant is a single unit pressurized-water reactor (PWR) with a rated capacity of approximately 3,400 MW_t and a unit dedicated SFP. The representative Group 2 reactor began operating in the 1970s and will reach the end of its extended operating license by year 2040. The NRC assumes the reactor core contains 193 assemblies and the SFP has a capacity of approximately 1,220 assemblies in a high-density 1x4 loading configuration. This number is based on a pool capacity of 1,414 assemblies, reduced by 193 assemblies to accommodate a full core offload capability using the existing high-density racking. In a low-density 1x4 with empties configuration, the SFP stores 312 assemblies. The unit operates on 18-month cycles, discharging approximately 78–84 assemblies per cycle. The representative PWR has already implemented dry storage.

Representative New Nuclear Plant (Group 3)

The representative new plant is an AP1000 PWR with a rated capacity of approximately 3,400 MW_t and a unit dedicated SFP. The representative Group 3 reactor begins operating in the year 2018 and will reach the end of its extended operating license by year 2078. The NRC assumes the reactor core contains 157 assemblies and the SFP has a capacity of approximately 1,000 assemblies in a high-density 1x4 loading configuration. This number is based on a pool capacity of 1,160 assemblies, reduced by 157 assemblies to accommodate a full core offload capability using the existing high-density racking. In a low-density 1x4 with empties configuration, the SFP stores 340 assemblies. The unit operates on either 18-month or 24-month cycles, discharging an estimated 69 assemblies per 18-month cycle or 77 assemblies per 24-month cycle (Ref. C.4, Section 9.1). The representative new nuclear plant is expected to begin dry storage in 2038 if high-density pool storage is allowed and will load a sufficient number of casks to maintain its full core offload capability.

Representative SFP Shared Between Units (Group 4)

This representative SFP is shared between two PWR units, each with a rated capacity of approximately 3,400 MW_t. The SFP, designed in two halves, is located outside the containment in the Auxiliary Building and provides underwater storage of spent fuel assemblies after their

removal from the reactor vessel of either reactor unit. The associated Group 4 reactor unit began operating in the 1970s and will reach the end of its extended operating license by year 2038. The NRC assumes each reactor core contains 193 assemblies and the SFP has a capacity of 1637 assemblies in a high-density 1x4 loading configuration. This number is based on a pool capacity of 1,830 assemblies, reduced by 193 assemblies to accommodate one unit's full core offload capability using the existing high-density racking. In a low-density 1x4 with empties configuration, the SFP stores 468 assemblies. The units operate on 24-month cycles, discharging approximately 78–84 assemblies per cycle on a 1-year staggered cycle. The representative shared SFP has already implemented dry storage.

C.1.6 Projected Number of Outages and Spent Fuel Assemblies

The spent fuel assembly inventory at a SFP is plant specific based on initial inventory, projected spent fuel discharged during each refueling outage, and operating cycle length. Additional spent fuel storage requirements are calculated using the SFP capacity and the cumulative spent fuel discharges. The cumulative number of fuel assemblies discharged is subtracted from the spent fuel pool capacity, assuming that each spent fuel pool retains space in the SFP to discharge one full core of fuel. During years in which no spent fuel is discharged at plants operating on 18-month or 24-month operating cycles, there would be no change in the SFP inventory. If there are more assemblies requiring storage than there is space in the SFP (including space to discharge one full core of fuel), these additional storage needs are assumed to be met using at-reactor dry storage rather than expansion of SFP capacity. The number of spent fuel assemblies required up to operating license expiration is calculated for each group based on the existing high-density SFP inventory, the number added from refueling outages, and the full reactor core inventory. These results are provided in Table 32.

Table 32 Number of Spent Fuel Assemblies Remaining through Operating License Expiration

Group No.	Category	Inventory	Number of Inventories	No. of spent fuel assemblies	Total
1	Current SFP inventory	3,055	1	3,055	
	refueling	284	12	3,408	
	reactor core	764	1	764	7,227
2	Current SFP inventory	1,220	1	1,220	
	refueling	78	18	1,404	
	reactor core	193	1	193	2,817
3a	Current SFP inventory	0	1	0	
	Refueling (18-month cycle)	69	40	2,760	
	reactor core	157	1	157	2,917
3b	Current SFP inventory	0	1	0	
	Refueling (24-month cycle)	77	30	2,310	
	reactor core	157	1	157	2,467
4	Current SFP inventory	1,637	1	1,637	
	refueling	78	24	1,872	
	reactor core	193	2	386	3,895

C.1.7 Dry Storage Capacity

Three companies supply most of the dry storage technologies to U.S. commercial nuclear power plants. These companies are Holtec International, Inc. (Holtec), NAC International, Inc. (NAC), and Transnuclear, Inc. (Transnuclear). The dry storage cask systems[22] (DSCs) for all three companies are certified by the NRC for storage of high burnup spent fuel (i.e., burnups greater than 45 GWd/MTU), using both regional and uniform loading of spent fuel in the packages. A summary of a representative sampling of dry storage canisters commercially available for spent fuel storage is provided in Table 33.

Table 33 Representative Sampling of Commercially Available BWR Spent Fuel Dry Storage Technology

Vendor Package	Fuel Type	Canister Type	Capacity (Assemblies)	Maximum Decay Heat Per Package[1] (kW)
Holtec HI-STORM 100	PWR	MPC-24	24	34
	PWR	MPC-32	32	34
Holtec HI-STORM FW	PWR	MPC-37	37	47
NAC UMS	PWR	24P	24	23
NAC MAGNASTOR	PWR	37P	37	35.5
Transnuclear NUHOMS	PWR	24PTH	24	40.8
	PWR	32PTH1	32	40.8
Transnuclear TN-40HT	PWR	Bolted	40	32
Holtec HI-STORM	BWR	MPC-68	68	34
Holtec HI-STORM FW	BWR	MPC-89	89	46.36
NAC MAGNASTOR	BWR	87B	87	33
Transnuclear NUHOMS	BWR	61BTH	61	31.2
Transnuclear TN-68	BWR	Bolted	68	30

The maximum decay heat per assembly for uniform loading is estimated by dividing the package decay heat by the number of assemblies. The maximum decay heat per assembly under regional loading schemes will generally be higher than the maximum decay heat per assembly assuming uniform loading for a smaller number of assemblies. Cask certificates of compliance provide the specific maximum assembly decay heat limits for each storage location in the basket.
Source: EPRI TR-1025206, p. 2-11 (Ref. C.5).

C.1.8 Discharged Spent Fuel Assemblies

The number of spent fuel assemblies in units of metric tons of uranium (MTU) that is discharged by a reactor unit during each refueling outage is estimated based on the unit's licensed thermal rating (megawatts thermal, MW_t, discharge burnup (BUP in MWd/MTU), capacity factor (CF in percent), and operating cycle length (CYL in years) as shown below.

$$MTU = \frac{MW_t \; x \; CYL \; x \; \frac{CF}{100} \; x \; \frac{365 \; days}{year}}{BUP}$$

Using the above formula, a 3,514 MW_t BWR reactor with a 24-month operating cycle operating at a 90 percent capacity factor and an average spent fuel assembly burnup of 45,000 MWd/MTU would discharge 51.3 MTU during each refueling cycle. The number of discharged

[22] The term dry storage cask system (DSC) includes dual-purpose canister based systems, dual-purpose casks, and storage-only dry storage casks and canister systems.

assemblies (ASSY) is estimated by dividing the MTU discharge value by the fuel assembly unit weight. Based on an average BWR fuel assembly unit weight of 0.18 MTU per assembly the equation yields approximately 285 assemblies.

C.1.9 Spent Fuel Assembly Decay Heat as a Function of Burnup and Cooling Time

As fuel assembly burnups increase, the decay heat of the fuel assembly (watts per assembly) and the Cesium-137 inventory in the spent fuel increase. Decay heat also can vary significantly with initial enrichment and assembly irradiation parameters. Spent fuel burnups have gradually increased since the 1990s with average BWR burnups about 43 GWd/MTU and range between 40 and 53 GWd/MTU and with average PWR burnups range between 40 and 55 GWd/MTU.

As shown in Figure 7, average decay heat for a 40 GWd/MTU PWR spent fuel assembly that has cooled for 5 years is approximately 1,100 watts per assembly based on approximately 2.3 kW/MTU times 0.45 MTU per assembly and a cesium-137 inventory of approximately 6.8×10^4 Ci per assembly. The average decay heat for a 55 GWd/MTU assembly that has cooled for five years is approximately 1,500 watts per assembly with a cesium-137 inventory of 9.6×10^4 Ci per assembly (Ref. C.5, p. 2-6). In comparison, a 40 GWd/MTU PWR spent fuel assembly that has cooled for 10 year has a decay heat of approximately 700 watts per assembly and a 55 GWd/MTU PWR spent fuel assembly has a decay heat of approximately 1,000 watts per assembly.

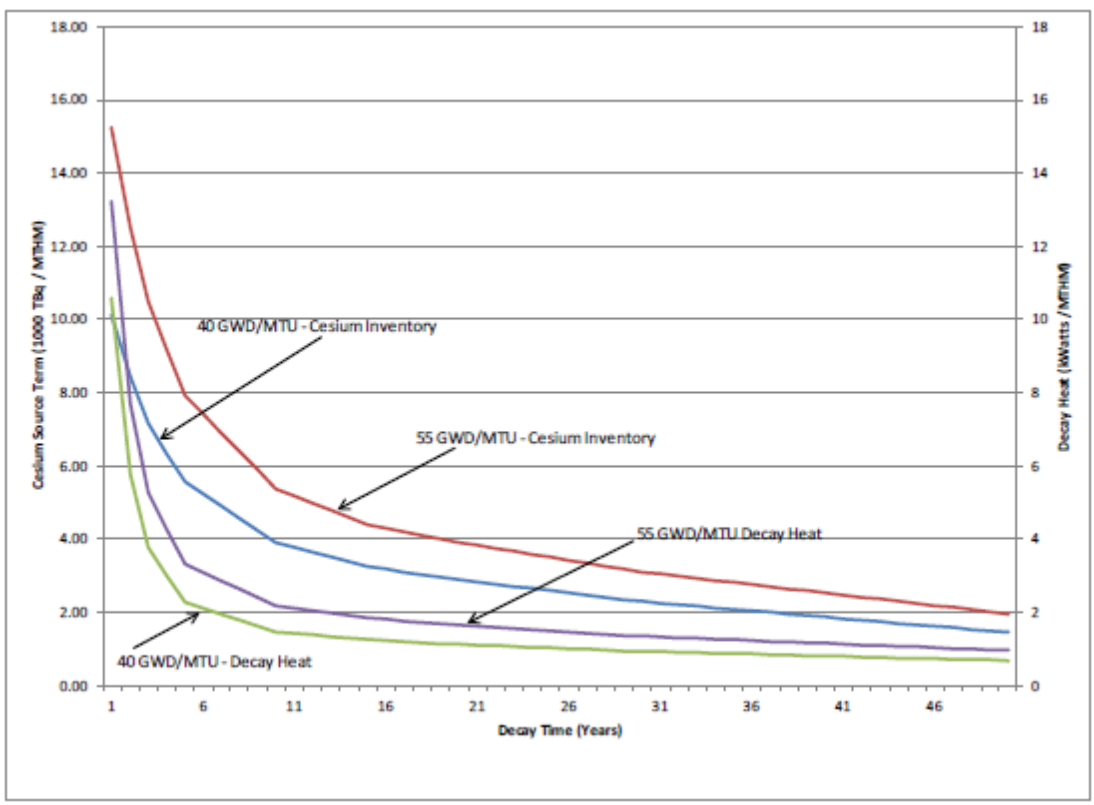

Figure 7 PWR spent fuel assembly decay heat and cesium inventory as a function of burnup and cooling time

Source: EPRI TR-1025206, p. 2-6 (Ref.C.5).

Average decay heat for a 40 GWd/MTU BWR spent fuel assembly that has cooled for five years is approximately 360 watts/assembly based on approximately 2.0 kW/MTU (from Figure 8) times 0.18 MTU per BWR assembly and a cesium-137 inventory of approximately 3.0×10^4 curies per assembly. The average decay heat for a 50 GWd/MTU assembly that has cooled for 5 years is approximately 520 watts per assembly with a cesium-137 inventory of 3.4×10^4 curies per assembly (Ref. C.5, p. 2-8). In comparison, a 40 GWd/MTU BWR spent fuel assembly that has cooled for 10 years has a decay heat of approximately 250 watts per assembly and a 50 GWd/MTU BWR spent fuel assembly has a decay heat of approximately 350 watts per assembly.

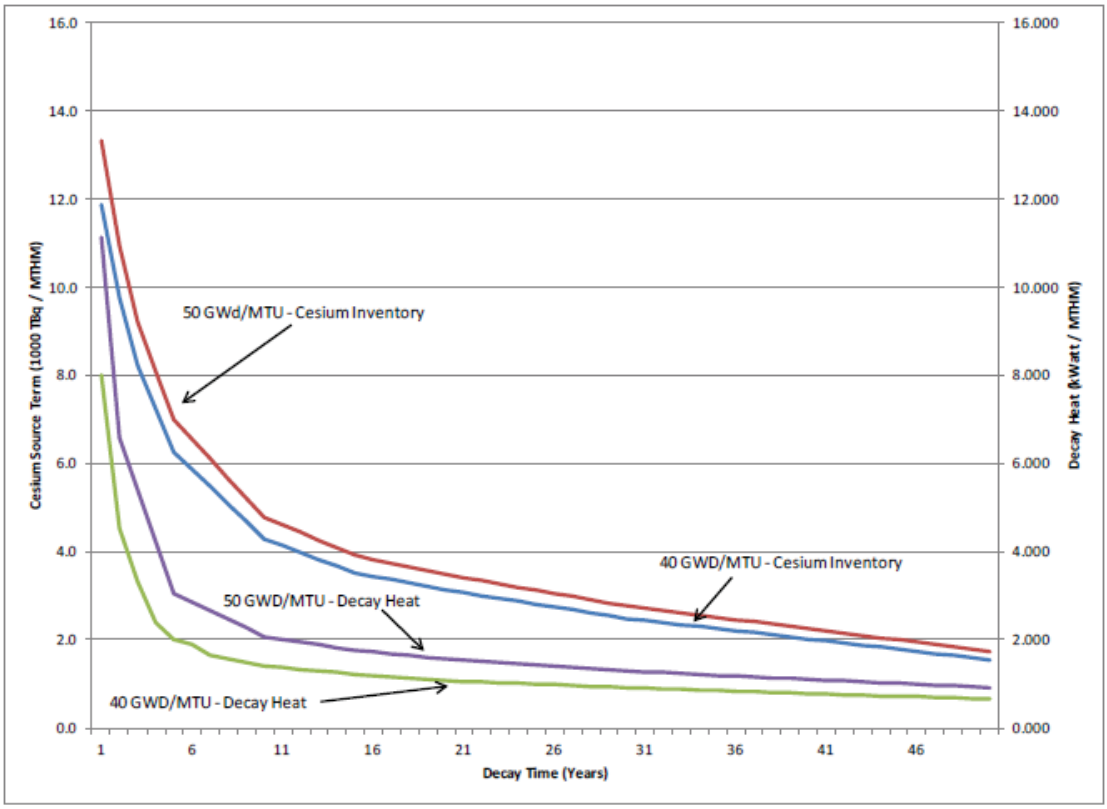

Figure 8 BWR spent fuel assembly decay heat and cesium inventory as a function of burnup and cooling time

Source: EPRI TR-1025206, p. 2-9 (Ref. C.5).

Based on an average PWR spent fuel assembly that emits 1,100 watts or an average BWR spent fuel assembly that emits 360 watts, Table 34 shows the number of spent fuel assemblies that could be stored assuming uniform fuel assembly burnup of 40 GWd/MTU and a five year decay time. Table cells that are not shaded identify those dry storage canisters that can be filled to capacity without exceeding the maximum decay heat per package rating, subject to restrictions on loading pattern. Shaded table cells identify those casks whose capacity loading is limited by the spent fuel assembly decay heat. For 55 GWd/MTU PWR assemblies that emit approximately 1,500 watts after they have cooled for five years or 50 GWD/MTU BWR assemblies that emit approximately 520 watts, fewer assemblies can be stored in the DSC than its design capacity due to decay heat limitations. The number of additional dry storage casks

that would be required for spent fuel cooled for five years depends on the vendor package selected and ranges between no additional canisters to almost twice as many additional canisters. Additional DSCs, which are required because of high heat load, are estimated in this cost-benefit analysis. For a sensitivity analysis, the maximum capacity based on decay heat limitations was also calculated if the spent fuel was allowed to cool for 10 years. As shown in Table 6, all of the lower heat rate fuel and most of the higher heat rate fuel could be loaded into casks without any decay heat limitations.

For this analysis, the Transnuclear TN-68 dry casks are selected as representative DSCs for the BWR spent fuel for Group 1. For Groups 2, 3, and 4, the Holtec Hi-Storm FW DSC is modeled as representative DSCs for the PWR spent fuel.

Table 34 Canister Storage Capacity Based on Decay Heat Limitations

Vendor Package	Fuel Type	Capacity (Assemblies)	Maximum Decay Heat Per Package (kW)	Maximum capacity based on decay heat			
				5 year cooling		10 year cooling	
				1100w (PWR) 360w (BWR) per assembly	1500w (PWR) 520w (BWR) per assembly	700w (PWR) 250w (BWR) per assembly	1000w (PWR) 350w (BWR) per assembly
Holtec HI-STORM 100	PWR	24	34	24.00	22.67	24.00	24.00
	PWR	32	34	30.91	22.67	32.00	32.00
Holtec HI-STORM FW	PWR	37	47	37.00	31.33	37.00	37.00
NAC UMS	PWR	24	23	20.91	15.33	24.00	23.00
NAC MAGNASTOR	PWR	37	35.5	32.27	23.67	37.00	35.50
Transnuclear NUHOMS	PWR	24	40.8	24.00	24.00	24.00	24.00
	PWR	32	40.8	32.00	27.20	32.00	32.00
Transnuclear TN-40HT	PWR	40	32	29.09	21.33	40.00	32.00
Holtec HI-STORM	BWR	68	34	68.00	65.38	68.00	68.00
Holtec HI-STORM FW	BWR	89	46.36	89.00	89.00	89.00	89.00
NAC MAGNASTOR	BWR	87	33	87.00	63.46	87.00	87.00
Transnuclear NUHOMS	BWR	61	31.2	61.00	60.00	61.00	61.00
Transnuclear TN-68	BWR	68	30	68.00	57.69	68.00	68.00

1. Shaded values identify where cask loading capacity is limited by the spent fuel decay heat.

The currently approved minimum cooling time for fuel stored in dry casks is seven years (10 years for some fuel types). Cask vendors would need to demonstrate, in an amendment request, that spent fuel that was cooled for a shorter period can be stored safely. The costs to prepare such an amendment request and for the NRC review are not included in this analysis. Furthermore, fuel selected must meet cask design specific fuel selection parameters that limit the maximum enrichment, maximum burnup, minimum cooling time, and maximum decay heat. The methodology used to estimate the capacity of the DSCs for spent fuel is subject to uncertainties resulting from decay heat and loading pattern restrictions. As a result, the actual DSC capacity may be higher or lower than those estimated.

C.1.10 Facility Life Cycle

Spent fuel storage involves a series of phases over the life cycle of the nuclear power plant for which it supports. The plant operational phases will have variable time requirements depending on the plant's refueling schedule, the capacity of the SFP, the term of the operating license, and the forecast schedule of removal of spent fuel from the SFP to the ISFSI.

At the expiration of a nuclear power plant's operating license, the full core is offloaded into the SFP. The licensee continues to store spent fuel in the pool following commercial operation[23] to allow the spent fuel to cool sufficiently before placing into dry storage.

C.1.11 Spent Fuel Pool Capacities

SFPs for all reactor types typically range from 9 to 18 meters (30 to 60 feet) in length and 6 to 12 meters (20 to 40 feet) in width, with a spent fuel capacity that ranges from 544 to 4,117 spent fuel assemblies for dedicated SFPs as shown in Table 72 in Appendix F. SFPs that are shared between units have capacities up to 4,628 fuel assemblies. This analysis assumes that plants with SFPs that are shared by multiple units reserve space for only one full core in the SFP.

For new reactors, spent fuel is stored in high-density racks which include integral neutron absorbing material to maintain the required degree of subcriticality. The SFP rack layout contains both Region 1 rack modules and Region 2 rack modules. The racks are designed to store fuel of the maximum design basis enrichment. Each rack in the SFP consists of an array of cells interconnected to each other at several elevations and to a thick base plate at the bottom elevation. These rack modules are free-standing, neither anchored to the pool floor nor braced to the pool wall. For the AP1000 reactors, the spent fuel storage racks include storage locations for 884 fuel assemblies and five defective fuel assemblies.

C.1.12 Spent Fuel Pool Cesium Inventory

The amount of cesium inventory in a SFP varies based on the number of spent fuel assemblies, the type of fuel stored, the discharge burnup, and the amount of time since the fuel was removed from the reactor core. The specific activity, $\frac{A}{M}$, in megacuries per metric tons of uranium (MCi/MTU) is relatively invariant and the assembly mass (in initial MTUs) is a reasonable scaling factor account for variations between different SFPs. This scaling factor is derived as follows assuming the two pools have similar distributions of burnup and cooling periods:

$$\frac{A_1}{M_1} \sim \frac{A_2}{M_2}$$

Where A_x is the absolute activity in megacuries (MCi) of SFP_x and M_x is the total amount of uranium in metric tons (MTU) stored in spent fuel pool x. The total amount of uranium, M_x, is estimated based on the number of spent fuel assemblies, N, and the average fuel assembly unit weight, m in MTU per assembly in the pool. A burnup scaling factor (BUP in MWd/MTU) can also be used in the above equation to yield:

$$\frac{A_1}{N_1\ x\ BUP_1\ x\ m_1} = \frac{A_2}{N_2\ x\ BUP_2\ x\ m_2}$$

Solving for the SFP absolute activity of the second pool yields:

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23 Decommissioning of the unit must be completed within 60 years of permanent cessation of operations under 10 CFR 50.82, "Termination of License." Completion of decommissioning beyond 60 years will be approved by the Commission only when necessary to protect public health and safety.

$$A_2 = \frac{A_1 \; x \; N_2 \; x \; BUP_2 \; x \; m_2}{N_1 \; x \; BUP_1 \; x \; m_1}$$

Using the above formula, a 3,514 MW_t BWR reactor with a SFP with absolute activity of 60 MCi from the storage of 3,055 BWR fuel assemblies with an average spent fuel assembly burnup of 45,000 MWd/MTU and with an average BWR fuel assembly unit weight of 0.18 MTU per assembly can be equated to a 3,400 MW_t PWR reactor with a SFP with an unknown absolute activity from the storage of 1,220 PWR fuel assemblies with an average spent fuel assembly burnup of 45,000 MWd/MTU and with an average PWR fuel assembly unit weight of 0.46 MTU per assembly as shown below.

$$A_{PWR} = \frac{59 \; x \; 1220 \; x \; 45000 \; x \; 0.46}{3055 \; x \; 45000 \; x \; 0.18} = \frac{1.49 \; x \; 10^9}{2.47 \; x \; 10^7} = 60.2 \; MCi$$

To test the accuracy of this estimate for high-density SFP scaling, the high-density Peach Bottom, Unit 3 SFP cesium inventories from 2001 and 2011 were used. The results showed that there is less than 1 percent error by using the scaling method described above.

Error is introduced when attempting to estimate a pool with a significantly different average cooling period for the spent fuel. To eliminate this source of error, the low-density loaded SFP inventory is estimated based on the low-density SFP characteristics evaluated in the SFPS and using the actual Cs-137 inventory of 22 MCi for all low-density SFPs and the formula above.

Table 72 located in Appendix F provides the estimated Cs-137 inventory for each SFP in a high-density loading configuration using the scaling factor discussed above. Cesium inventories used to analyze each SFP group are summarized in Table 35.

Table 35 Spent Fuel Pool Group Cesium Inventory

SFP Group	Pool Storage Case	Pool Cesium Inventory (MCi)		
		Sensitivity (Low Estimate)	Base Case	Sensitivity (High Estimate)
1	High-density	40.6	52.7	63.3
	Low-density	19.8	22.0	26.4
2	High-density	57.4	67.9	78.2
	Low-density	15.7	17.4	20.9
3	High-density	33.7	44.4	54.2
	Low-density	15.7	17.4	20.9
4	High-density	63.6	101.1	142.2
	Low-density	31.4	34.8	41.8

C.2 Spent fuel Pool Accident Modeling and Evaluation Assumptions

C.2.1 Seismic Hazard Model

This cost-benefit analysis uses the existing U.S. Geological Survey (USGS) 2008 model to evaluate seismic hazards at central and eastern United States (CEUS) nuclear power plants. A new probabilistic seismic hazard model is currently being developed and will consist of two parts: (1) a seismic source zone characterization and (2) a ground motion prediction equation (GMPE) model. Although part (1) is now complete (Ref. C.6), the GMPE update is still in

progress. Furthermore, the NRC is currently developing an independent probabilistic seismic hazard assessment (PSHA) computer code to incorporate part (1) and part (2) when complete. While the USGS (2008) hazard model is not sufficiently detailed for regulatory decisions, it is used for this analysis because it is the most recent and readily available hazard model and was used in the SFPS (Ref. C.7). Although the USGS 2008 model considers western U.S. sites (e.g., Columbia, Diablo Canyon, Palo Verde, and San Onofre), these sites are not addressed in Generic Issue 199 (Ref. C.8), which focused on the CEUS and, therefore, are not included in this analysis. Western sites will be considered on a site-specific basis in response to licensee requested information related to Recommendations 2.1 (Seismic Hazards Evaluations) and 2.3 (Seismic Walkdowns) of the Post-Fukushima Near-Term Task Force.

A comparison of the annual frequency of exceeding a given PGA for BWR Mark I and II sites (see Figure 9) shows that Peach Bottom (i.e., the reference plant) falls close to the upper end of the group located in the CEUS in terms of hazard estimates.

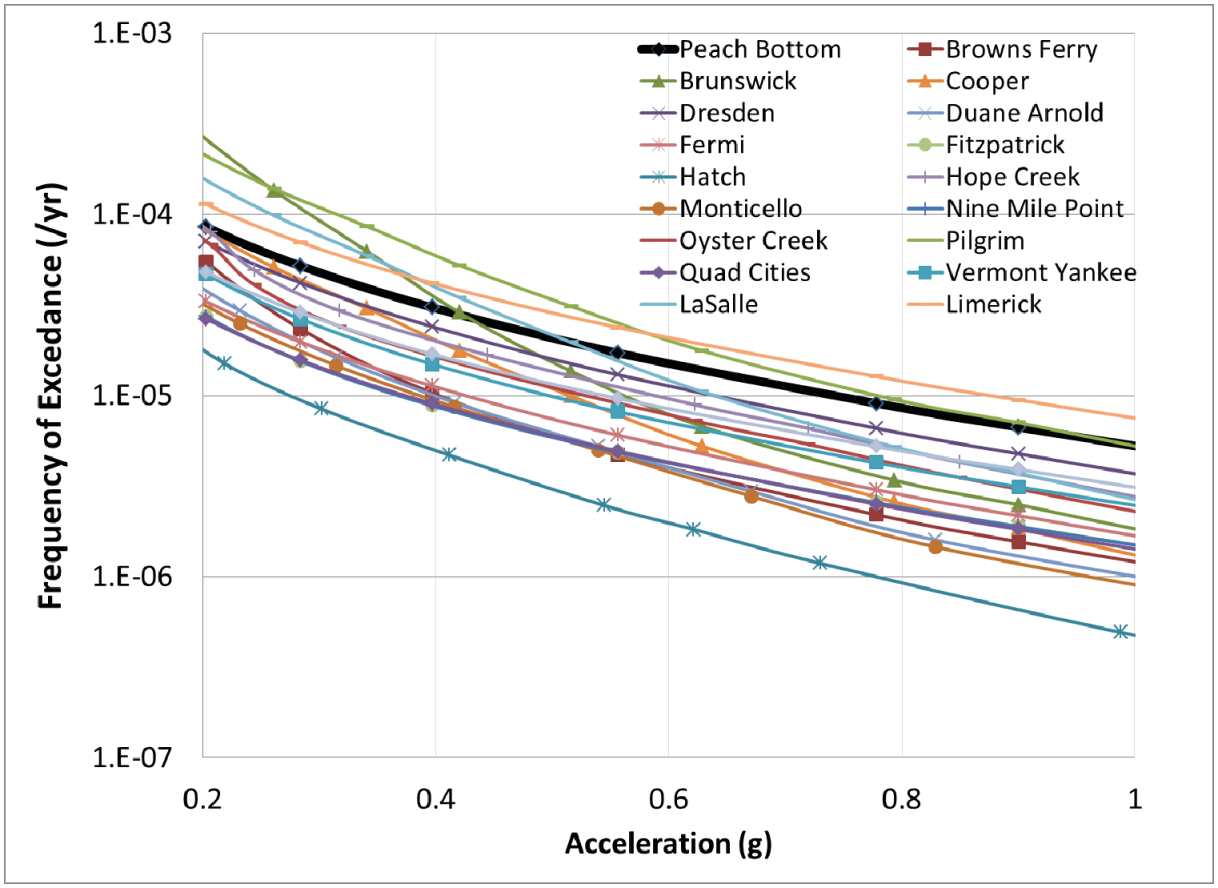

Figure 9 Comparison of annual PGA exceedance frequencies for U.S. BWR Mark I and Mark II reactors (USGS 2008 model)

A similar comparison of the annual frequency of exceeding a given PGA for PWR and BWR Mark III sites (Figure 10), for new reactors (Figure 11), and for reactors units with a shared SFP (Figure 12) shows that Peach Bottom falls close to the upper end of the group in terms of hazard estimates.

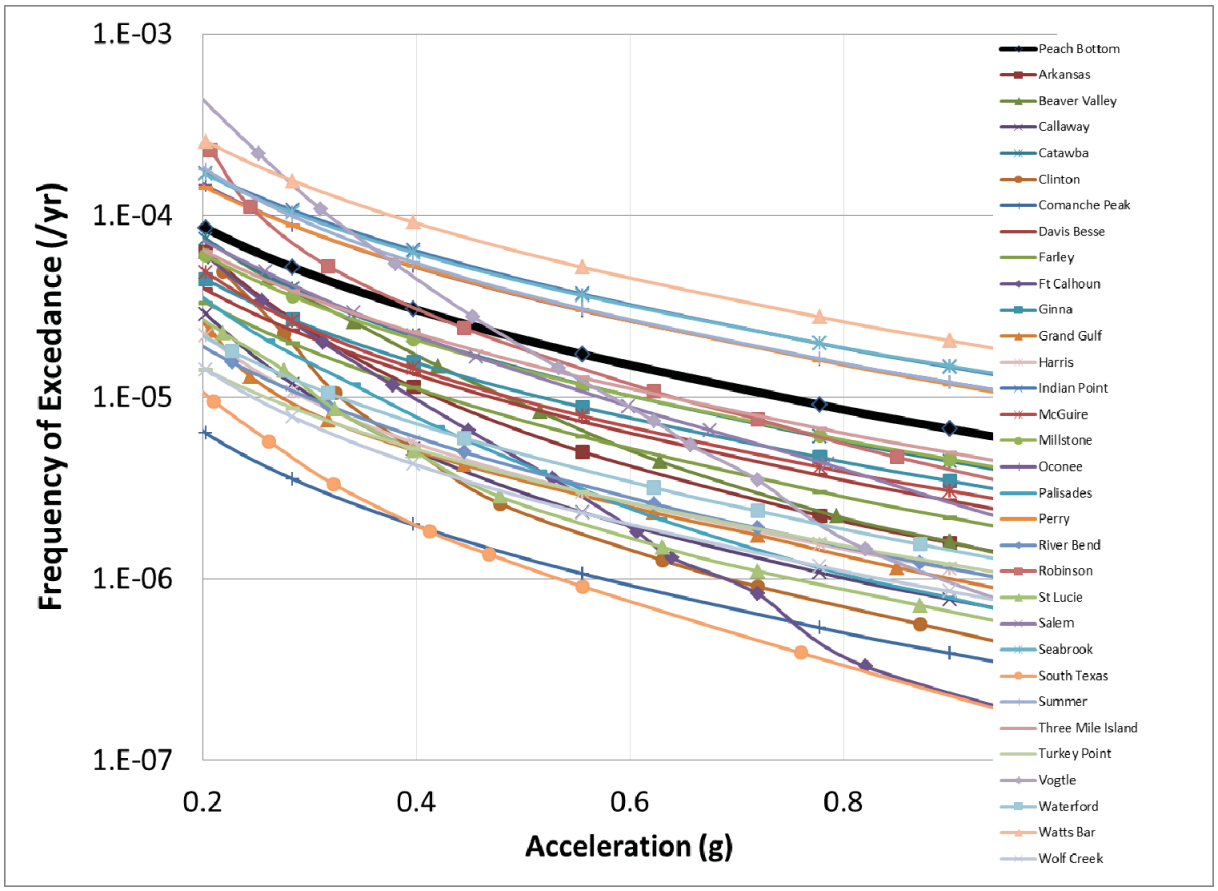

Figure 10 Comparison of annual PGA exceedance frequencies for U.S. PWR and BWR Mark III reactors (USGS 2008 model)

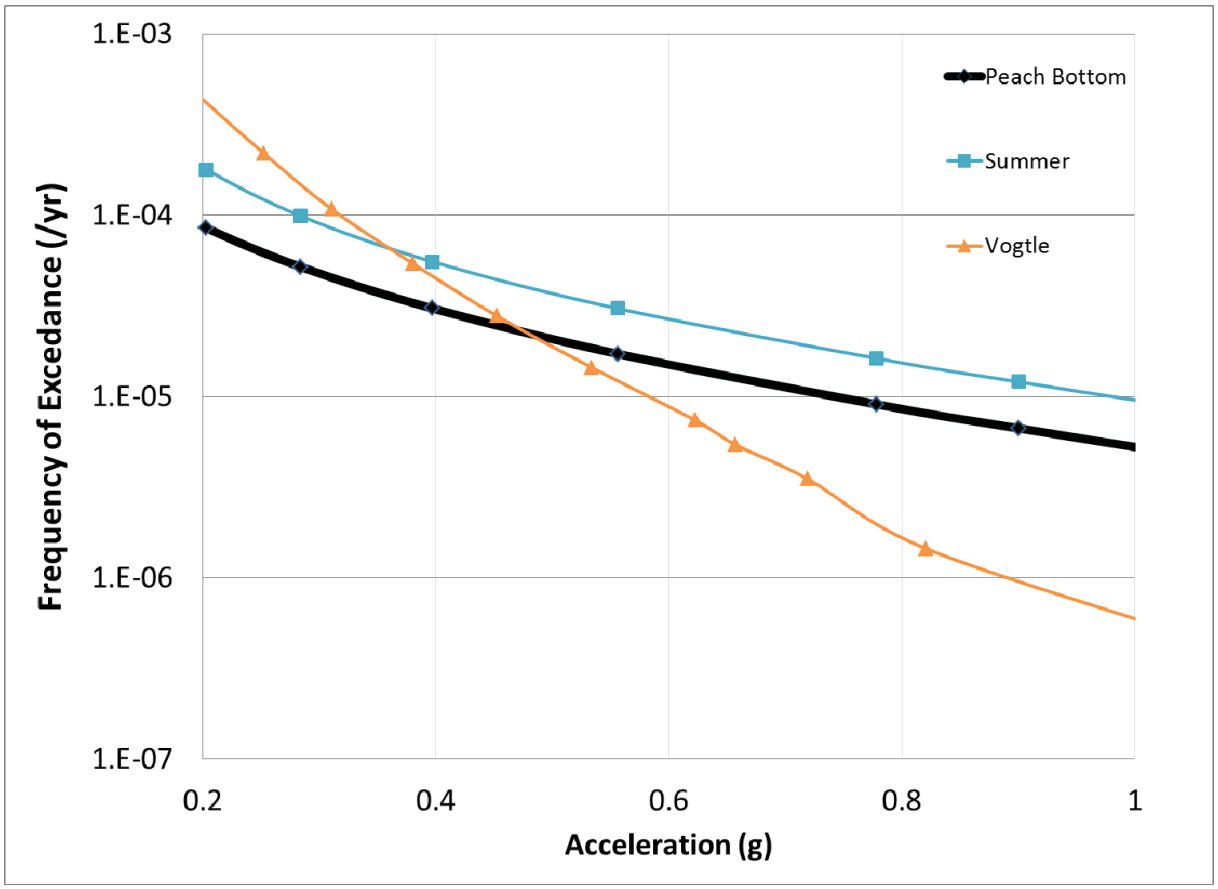

Figure 11 Comparison of annual PGA exceedance frequencies for new U.S. reactors (USGS 2008 model)

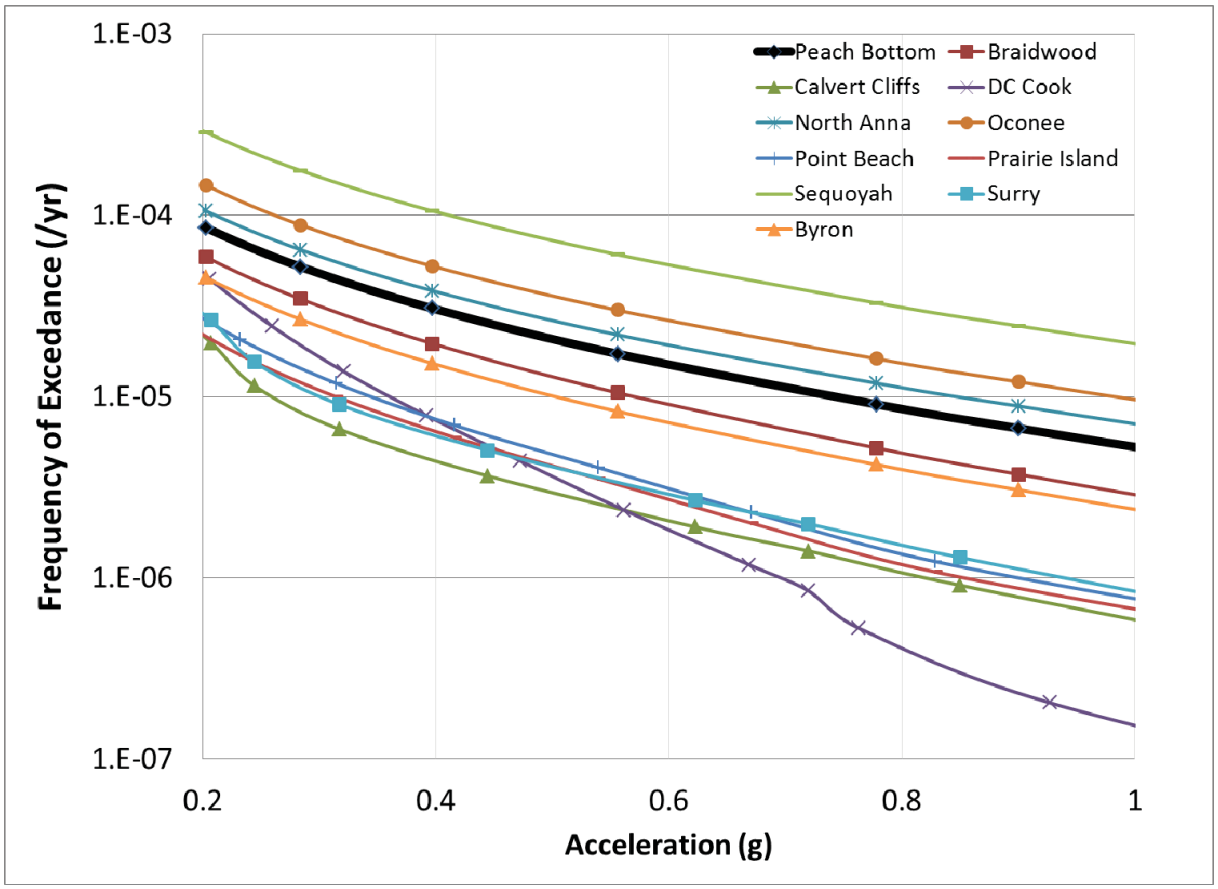

Figure 12 Comparison of annual PGA exceedance frequencies for U.S. reactors with a shared spent fuel pool (USGS 2008 model)

C.2.2 Characterization of Seismic Event Likelihood

As described in Section 3.2 of the SFPS (Ref. C.7), the hazard exceedance frequencies can be translated into initiating event frequencies by partitioning the PGA range into a number of discrete categories (bins) defined in terms of PGA intervals. These bins define a discrete number of seismic event scenarios with increasing intensity (PGA). Revision 1.01 of the NRC handbook entitled, "Risk Assessment of Operational Events, Volume 2—External Events," issued January 2008 (Ref. C.9), recommends the use of at least three bins unless plant-specific considerations require more bins. The SFPS used four bins.

Table 4 of the SFPS, reproduced in this analysis as Table 36, shows the resulting bins, along with the tabulated frequencies for various spectral and peak accelerations for Peach Bottom, the reference plant evaluated in that study. Note that for bin 4, the representative bin PGA has been set to 1.2g by convention, whereas for the other bins, it is the geometric mean of the interval endpoints.

Table 36 Seismic Bin Initiating Event Frequencies (Base Case)

Bin No.	Bin Range (g)	Bin PGA (g)	Approximate Initiating Event Frequency (USGS 2008 model) (/yr)
1	0.05 - 0.3	0.12	5.2×10^{-4}
2	0.3 - 0.5	0.4	2.7×10^{-5}
3	0.5 - 1.0	0.7	1.7×10^{-5}
4	> 1.0	1.2[1]	4.9×10^{-6}

1. Assumed based on PRA modeling convention.

Although the Peach Bottom hazard exceedance frequencies curves shown in Figures 7 through 10 fall close to the upper end of each group in terms of hazard estimates, there are some CEUS sites that exceed those estimates. For each SFP group, the site with the highest plant hazard exceedance frequency for peak ground accelerations greater than 0.6g was selected to produce the high estimate seismic bins and initiating event frequencies provided in Table 37.

Table 37 Seismic Bin Initiating Event Frequencies (High Estimate sensitivity)

SFP Group (Site Name)	Bin No.	Bin Range (g)	Bin PGA (g)	Approximate Initiating Event Frequency (USGS 2008 model) (/yr)
SFP Group 1 (Limerick)	1	0.05 - 0.3	0.12	6.8E-04
	2	0.3 - 0.5	0.4	3.6E-05
	3	0.5 - 1.0	0.7	2.2E-05
	4	> 1.0	1.2	7.1E-06
SFP Group 2 (Watts Bar)	1	0.05 - 0.3	0.12	1.7E-03
	2	0.3 - 0.5	0.4	8.1E-05
	3	0.5 - 1.0	0.7	4.9E-05
	4	> 1.0	1.2	1.5E-05
SFP Group 3 (Summer)	1	0.05 - 0.3	0.12	1.8E-03
	2	0.3 - 0.5	0.4	5.4E-05
	3	0.5 - 1.0	0.7	2.9E-05
	4	> 1.0	1.2	9.1E-06
SFP Group 4 (Sequoyah)	1	0.05 - 0.3	0.12	1.79E-03
	2	0.3 - 0.5	0.4	8.98E-05
	3	0.5 - 1.0	0.7	5.64E-05
	4	> 1.0	1.2	2.00E-05

The information above coupled with the review of previous studies (Ref. C.10) suggests that the base case frequency of a seismic event that could challenge the integrity of a SFP is on the order of 1.7×10^{-5} per year (i.e., approximately one event in 60,000 years) or less. Table 38 contrasts this frequency against other sources of information.

Table 38 Comparison of Seismic Frequencies from Various Sources

Source	Estimated initiating event frequency of a large seismic event	Notes
USGS 2008—Cost-benefit analysis base case	1.7×10^{-5} /year[2] (one event in 60,000 years)	Frequency of seismic bin 3 (0.5 to 1.0 g) of 4 bins
USGS 2008—Cost-benefit analysis high estimate sensitivity	5.6×10^{-5}/year[3] (one event in 18,000 years)	
NUREG-1738[1]	1.1×10^{-5}/year (one event in 90,000 years)	Frequency of seismic hazard between 0.51g to 1.02g

1. Initiating event frequency reported is based on the LLNL models (Ref. C.11).
2. This value is from Table 36 for Bin No. 3.
3. This value is the SFP group 4 Bin No. 3 value from Table 37 and is the greatest magnitude for any of the SFP groups.

C.2.3 Spent Fuel Pool Initiator Release Frequency

Section 1.5 of the SFPS (Ref. C.7) provides an overview of contributors to SFP risk. The majority of SFP risk emanates from a loss of water from a sizeable leak in the SFP or a boil off in which operator action to inject water into the pool for an extended period is precluded. The release frequency from the SFP can then be characterized as the frequency of the initiator causing fuel uncovery multiplied by the probability of a release given fuel uncovery for the specific initiating event. The total release frequency is the sum of the frequency of releases from cask drops, seismic events, and other initiators. This value is given by:

$$F_{release} = \sum_i F_{initiator_i} \, x \, P_{release_i}$$

Where $F_{initiator}$ includes:

F_{drop} = frequency of spent fuel uncovery from cask drops
$F_{seismic\text{-}bin\ 3}$ = frequency of spent fuel uncovery from seismic bin 3 event
$F_{seismic\text{-}bin\ 4}$ = frequency of spent fuel uncovery from seismic bin 4 event
F_{other} = frequency of spent fuel uncovery from sources other than cask drops and seismic
$P_{release}$ = probability of release given spent fuel uncovery for specific initiators

Source: Derived from SFPS, Section B.4 (Ref. C.7).

The SFPS provides a detailed analysis of the consequences, for a particular site and a calculation of $F_{seismic}$ for seismic bin 3, depicted as a hazard exceedance frequency range provided in Table 36.

The SFPS did not analyze initiators that contribute to SFP risk other than for seismic events defined by seismic bin no. 3. However past studies, such as NUREG-1353 (Ref. C.12) and NUREG-1738 (Ref. C.10), evaluated additional events that could contribute to risk and consequences from SFP accidents. Table 42 summarizes these initiating-event-class fuel uncovery frequencies. Uncovery frequencies taken from past studies depend on the assumptions stated in those studies. Additionally, seismic bin no. 4 is included by extrapolating the results of the SFPS. For seismic bin no. 3 and bin no. 4 events, the uncovery frequency is the product of the initiating event frequency, ac power fragility, and the liner fragility.

The SFPS (Ref. C.7) uses an alternating current (ac) power fragility value of 0.84 taken from NUREG-1150 (Ref. C.13) as a surrogate for the conditional probability of normal SFP cooling

and makeup not being available following a 0.7g earthquake. This simplifying assumption was made in light of the fact that the SFPS is not a probabilistic risk assessment but rather a consequence analysis with probabilistic considerations.

In reality, the availability of normal SFP cooling and makeup would be a combination of the ac power fragility, the fragility of the actual equipment and its support equipment, and operator actions to recover SFP cooling capabilities using additional mitigation equipment and strategies implemented in response to Order EA-12-049 (Ref. C.14). The modeling and consideration of these guidance and strategies to maintain or restore SFP cooling capabilities following a beyond-design-basis external event on a plant-specific basis may result in a value for SFP cooling and makeup failure conditional probability that may differ from the NUREG-1150. Because a documented ac power fragility analysis that covers U.S. SFPs is not readily available, a conservative bounding value of 1.0 is used in this analysis.

Section 4.1.5 of the SFPS (Ref. C.7) describes the results from the nonlinear finite element analysis to estimate the likelihood of leakage from concrete cracking and related SFP liner failure for the 0.7g earthquake. Figure 27 from the SFPS shows that the maximum membrane effective strain is about 3.7 percent. Based on this calculated liner strain for the 0.7g earthquake, a structural analysis of the pool estimates that the SFP in this study has a 90 percent probability of surviving the 0.7g earthquake with no liner leakage (or conversely, a 10 percent probability of damaging the liner such that leakage will occur). As a result, a liner fragility value of 0.1 is used in the SFPS for the seismic bin No. 3 initiating event. NUREG/CR-5176 (Ref. C.15) provides the fragility for the walls of a PWR located in the CEUS as having a 98 percent probability of surviving the 0.7g earthquake with no liner leakage (or conversely, a 2 percent probability of damaging the liner such that leakage will occur).

For the seismic bin 4 initiating event (i.e., 1.2g earthquake), a comparable structural analysis is not performed in the SFPS to determine the liner fragility value for the reference BWR Mark I plant. As a result, a bounding value of 1.00 for the seismic bin no. 4 earthquake is used in this analysis for Group 1 liner fragility high estimate, even though a detailed analysis may be able to justify a value a factor of 2 or more lower. NUREG/CR-5176 provides the fragility for the walls of a PWR located in the CEUS as having an 84 percent probability of surviving the 1.2g earthquake with no liner leakage (or conversely, a 16 percent probability of damaging the liner such that leakage will occur). As a result, a value of 0.16 is used for the seismic bin no. 4 earthquake low estimate in this analysis for Groups 2, 3, and 4 liner fragility. A summary of these liner fragility values is provided in Table 39.

Table 39 Liner Fragility Values as a Function of Spent Fuel Pool Group and Seismic Bin

SFP Group	Seismic Bin	Liner Fragility		
		Low Est.	Base Case	High Est.
1	Bin 3	10%	10%	100%
	Bin 4	50%	100%	100%
2, 3, & 4	Bin 3	2%	5%	25%
	Bin 4	16%	50%	100%

Past studies have reached generally similar conclusions about the relative contribution to risk from the seismic initiating events considered. Table 40 summarizes the impact of the above modeling assumptions when comparing the seismic initiating event fuel uncovery frequencies from previous SFP accident regulatory analyses.

Table 40 Frequency of Spent Fuel Pool Fuel Uncovery for Seismic Events

Reference	Reactor Type / SFP Grouping	Seismic Event Contribution to SFP Fuel Uncovery (per 10^6 reactor-years)	
		Base case	High estimate sensitivity
NUREG-1353 (Ref. C.12) (best estimate)	BWR[1]	6.7	N/A
	PWR	1.8	N/A
NUREG-1738[2] (Ref. C.10)	All	2.0	N/A
This analysis[3]	SFP Group 1	6.6	29
	SFP Group 2	3.3	27
	SFP Group 3	3.3	16
	SFP Group 4	3.3	34

1. The NUREG-1353 BWR seismic structural failure value was not multiplied by the stated conditional probability of having a zirconium fire of 0.25.
2. NUREG-1738 presented results for the two different seismic hazard models in wide use at the time (the Electric Power Research Institute and Lawrence Livermore National Labs models). The larger of the two values is listed above.
3. The base case initiating event frequency value is from Table 36. The high estimate sensitivity initiating event frequency value is from Table 37. The likelihood of fuel uncovery is a product of initiating event frequency, ac power fragility (1.0), and liner fragility (value depends on case being evaluated as displayed in Table 39). A value of 1.0 for ac power or pool liner failure mean represents a 100 percent likelihood of failure.

The SFPS evaluated a specific BWR Mark I reference site for a specific initiating event. When spent fuel in a pool becomes uncovered, it may still be coolable from natural circulation of air once the water level clears the baseplate of the racks, depending on the amount of decay heat during the operating cycle. In Section 12.1 of the SFPS, the fuel is estimated to be air coolable for all but roughly 10 percent of the operating cycle. Factors affecting this value include the amount of fuel in the pool, its configuration, geometry of the fuel racks, etc. A partial draindown event with channeled fuel or solid-walled high-density racks could impede airflow. In this case with no natural circulation of air through the racks, the fuel could only be cooled by steam generated by the fuel itself or through the application of water spray. For these mechanisms to be effective, a substantial fraction of the decay heat must be absorbed by the remaining water to generate adequate steam flow or adequate spray flow must be applied. Distributed fuel assemblies late in the operating cycle may lose a significant portion of the remaining decay heat to radiation heat transfer and limited convective heat transfer at temperatures below the runaway oxidation threshold, and therefore, the assemblies would not reach a self-sustaining oxidation condition.

The spent fuel is expected to retain an air coolable geometry following a seismic event that causes a moderate to large crack in the pool, and information provided in NUREG/CR-5176 (Ref. C.15), which concludes that there is high confidence that SFP racks are sufficiently robust to remain generally intact with their fuel channels open supports this assumption. Furthermore, prior studies conclude that severe earthquakes are not expected to result in catastrophic failure of SFP structural walls and floor or fuel racks. However, there is considerable variability in U.S. SFP size, capacity, rack type, and geometry as well as the amount and age of the fuel in the pool and its burnup. Because plant-specific analyses is not available to verify that U.S. SFPs

and racks retain their structural integrity and air-coolable geometry following a beyond-design basis seismic event for U.S. SFPs, a bounding approach was used to evaluate the sensitivity of assuming the spent fuel is not air-coolable following a seismic bin 3 or seismic bin 4 earthquake. For bin 3, this modeling represents the scenario in which the seismic event results in a partial draindown condition (i.e., liner tearing at the walls) with some water remaining at the bottom of the SFP. In the SFPS, the fuel is estimated to not be air coolable for 10 percent of the operating cycle following a Bin 3 seismic event based on the SFP configuration and other factors. This value was used for the base case of SFP Group 1. For stronger seismic events for SFP Group 1 , the other SFP Group base cases, and for all high estimates, a bounding value of 100 percent for the conditional probability of release was assumed as shown in Table 41.

Table 41 Fraction of Time Either Excessive Heat or a Partial Spent Fuel Pool Draindown Prevents Natural Circulation Cooling of the Spent Fuel

SFP Group	Seismic Bin	Inadequate Spent Fuel Cooling Fraction		
		Low Est.	Base Case	High Est.
1	Bin 3	10%	10%	100%
	Bin 4	30%	100%	
2, 3, & 4	Bin 3	10%	100%	100%
	Bin 4	30%		

For the postulated cask drop event, the spent fuel is expected to retain an air coolable geometry because a cask drop accident would most likely affect the fuel pool floor in the cask loading area. Typically overhead cranes used to move casks are designed to meet single failure proof criteria, and have interlocks and administrative controls that limit the motion of the crane over the SFP to the cask loading area, where no fuel is stored. Although improbable, crane failure is more likely to occur during hoisting operations when many components contribute to holding the cask than during translational motion when the hoist holding brakes are set. The hoisting activities occur over the cask loading area, and, in that location, the cask, if dropped, could have sufficient potential energy to damage the SFP floor. However, a structural analysis to evaluate all U.S. SFPs was not performed to verify that spent fuel and racks retain their structural integrity and air-coolable geometry following a cask drop event. Given the uncertainties and plant-specific variabilities involved, a bounding approach was used by assuming the spent fuel is not air-coolable following a cask drop accident. This was done by assigning a bounding value of 1.0 for the conditional probability of release for the cask drop unsuccessful mitigation event.

To calculate the total release frequency, the uncovery frequencies are multiplied by the conditional probability of release for each initiating event class. The conditional probability of release depends on the fraction of the operating cycle where the fuel is not air-coolable. As previously discussed in this section, given the uncertainties and plant-specific variability involved, a bounding approach was used. For SFP draindown events (e.g., seismic events and cask drops) the bounding approach used in this analysis assumes these events are not air-coolable. For the nonseismic and noncask drop events taken from previous studies, the nature of the events may lead to a situation similar to a partial draindown where the rack baseplate is not cleared and airflow is impeded. For these events, the spent fuel is not air-coolable and the conditional release probability is assumed to be 100 percent.

When mitigation is credited, the SFPS found that successful deployment of mitigation decreased the conditional probability by a factor of 19 for the seismic bin no. 3 event analyzed at the reference plant using mitigation measures required under 10 CFR50.54(hh)(2)

(Ref. C.16). The SFPS does not consider the post-Fukushima SFP instrumentation required under Order EA-12-051 (Ref. C.17) and severe accident mitigation equipment and mitigation strategies (Ref. C.18) required under Order EA-12-049 (Ref. C.14), which is being implemented by the plants and is intended to increase the likelihood of restoring or maintaining power and mitigation capability during severe accidents. In reality, the effectiveness of post-Fukushima improvements to severe accident mitigation measures will depend on a variety of factors, which the SFPS did not consider but are expected to increase the likelihood that deployment of mitigation measures is successful. Each plant has developed a plant-specific analysis and strategies for coping with the effects of the beyond-design-basis natural events that may challenge its SFP cooling and makeup capabilities. For the purposes of this analysis, it was estimated that mitigation if successfully deployed in time decreased the conditional probability by a factor of 19 for all initiating events as determined in the SFPS. This analysis used a conservative approach by crediting successful mitigation for the low-density SFP alternative and assumed no successful mitigation for the high-density SFP storage regulatory baseline.

Table 42 summarizes the non-seismic initiating event fuel uncovery frequencies, the conditional probability of release, and the total release frequency without mitigation.

Table 42 Release Frequencies for Spent Fuel Pool Initiators for Nonseismic Events

Initiating Event Class	Initiating Event Fuel Uncovery Frequency (per r-yr)	Conditional Probability of Release (Unsuccessful mitigation)	Release Frequency (Unsuccessful mitigation) (per r-yr)
Cask / heavy load drop	$2x10^{-7}$ (2)	8.2% - 100%	$1.64x10^{-8}$ – $2.00x10^{-7}$
LOOP – severe weather	$1x10^{-7}$ (2)	100%	$1.00x10^{-7}$
LOOP – other	$3x10^{-8}$ (2)	100%	$3.00x10^{-8}$
Internal fire	$2x10^{-8}$ (2)	100%	$2.00x10^{-8}$
Loss of pool cooling	$6x10^{-8}$ (1)	100%	$6.00x10^{-8}$
Loss of water inventory	$1x10^{-8}$ (2)	100%	$1.00x10^{-8}$
Inadvertent aircraft impacts	$6x10^{-9}$ (2)	100%	$6.00x10^{-9}$
Missiles – general	$1x10^{-8}$ (1)	100%	$1.00x10^{-8}$
Missiles - tornado	$1x10^{-9}$ (2)	100%	$1.00x10^{-9}$
Pneumatic seal failures	0 – $3x10^{-8}$ (1,4)	100%	0 – $3.00x10^{-8}$
Total			$2.53x10^{-7}$ – $4.37x10^{-7}$

1. Values from NUREG-1353 (Ref. C.12). These numbers are applicable to all reactors and were not adjusted by the stated conditional probability of having a zirconium fire of 0.25 for BWR reactors.
2. Values from NUREG-1738 (Ref. C.10).
3. The operating cycle phase is equal to 8.2% (e.g., 60/730) for 2-year refueling cycles and 11.0% (e.g., 60/547.5) for 18-month refueling cycles.
4. Although many plants use gates with mechanical seals that are kept under pressure by passive mechanical means (i.e., do not depend on air pressure, ac power, or dc power) to prevent leakage, there may be some plants that continue to use pneumatic seals. This analysis conservatively includes the pneumatic seal failures as an initiating event for U.S. PWR SFPs.

Table 43 provides the total release frequency by SFP group for all SFP event initiators.

Table 43 Total Release Frequency by Spent Fuel Pool Group

SFP Group	Seismic Bin	Bin Frequency (per year)	Liner Fragility	Fraction Not Air Coolable	Seismic Release Frequency (per year)	Non-Seismic Release Frequency (per year)	Total Release Frequency per Group (per year)
Low Estimate							
1	3	$1.65x10^{-5}$	10%	8%	$1.35x10^{-7}$	$2.53x10^{-7}$	$1.12x10^{-6}$
	4	$4.90x10^{-6}$	50%	30%	$7.35x10^{-7}$		
2,3,4	3	$1.65x10^{-5}$	2%	8%	$3.30x10^{-8}$	$2.83x10^{-7}$	$5.51x10^{-7}$
	4	$4.90x10^{-6}$	16%	30%	$2.35x10^{-7}$		
Base Case							
1	3	$1.65x10^{-5}$	10%	8%	$1.35x10^{-7}$	$4.37x10^{-7}$	$5.47x10^{-6}$
	4	$4.90x10^{-6}$	100%	100%	$4.90x10^{-6}$		
2,3,4	3	$1.65x10^{-5}$	5%	100%	$8.25x10^{-7}$	$4.67x10^{-7}$	$3.74x10^{-6}$
	4	$4.90x10^{-6}$	50%	100%	$2.45x10^{-6}$		
High Estimate							
1	3	$2.24x10^{-5}$	100%	100%	$2.24x10^{-5}$	$4.37x10^{-7}$	$2.99x10^{-5}$
	4	$7.09x10^{-6}$	100%	100%	$7.09x10^{-6}$		
2	3	$4.92x10^{-5}$	25%	100%	$1.23x10^{-5}$	$4.67x10^{-7}$	$2.79x10^{-5}$
	4	$1.51x10^{-5}$	100%	100%	$1.51x10^{-5}$		
3	3	$2.95x10^{-5}$	25%	100%	$7.38x10^{-6}$	$4.67x10^{-7}$	$1.69x10^{-5}$
	4	$9.10x10^{-6}$	100%	100%	$9.10x10^{-6}$		
4	3	$5.64x10^{-5}$	25%	100%	$1.41x10^{-5}$	$4.67x10^{-7}$	$3.46x10^{-5}$
	4	$2.00x10^{-5}$	100%	100%	$2.00x10^{-5}$		

C.2.1 Seismic Initiator Frequency Assumptions Sensitivity

As illustrated in Table 44, the combination of conservative seismic initiator modeling assumptions with the bounding seismic source zone characterization for any spent fuel pool located in the CEUS results in public health (accident) benefit values increasing by a factor between 4.5 and 9.3 times the averted public health (accident) dose calculated for the base case.

Table 44 Sensitivity of Public Health (Accident) Benefits within 50 Miles to Changes in Seismic Initiator Frequency Assumptions

SFP Group	Seismic Initiator Case	Dose (averted person-rem per pool)	Benefits (2012 million dollars)		
			2% NPV	3% NPV	7% NPV
1	Base Case	1,740	$2.72	$2.42	$1.62
	High Estimate	9,510	$14.86	$13.25	$8.87
2	Base Case	1,630	$2.45	$2.15	$1.38
	High Estimate	12,100	$18.23	$16.02	$10.25
3	Base Case	3,020	$3.14	$2.37	$0.99
	High Estimate	13,650	$14.21	$10.75	$4.49
4	Base Case	1,690	$2.62	$2.33	$1.54
	High Estimate	15,660	$24.23	$21.53	$14.24

Offsite Property Cost Offset Sensitivity to Seismic Initiator Frequency Assumptions

Although the SFPS reference plant hazard exceedance frequencies curves discussed in Appendix section C.2.1 of this analysis fall close to the upper end of each SFP group in terms of hazard estimates, there are some CEUS sites that exceed those estimates. To analyze the seismic risk hazard for these CEUS sites, a high estimate using the bounding plant hazard exceedance frequency curve is used to produce the high estimate seismic bins and initiating event frequencies. These seismic frequencies are provided in Table 37. Several other bounding assumptions are also made to arrive at the bounding SFP release frequency provided in Table 43 These include the loss of all ac power for all SFP initiators, a conservative liner fragility value (see Table 39) even though a realistic analysis may be able to justify a value that is lower by factor of 2 or more, and assuming a bounding value of 1.0 for the conditional probability for failure to successfully mitigate the high-density storage spent fuel accident. These conservative (bounding) assumptions were used to calculate the offsite property cost offset estimate sensitivity to the seismic initiating frequency assumptions provided in Table 45.

Table 45 Sensitivity of Offsite Property Cost Offset within 50 Miles to Changes in Seismic Initiator Frequency Assumptions

SFP Group	Seismic Initiator Case	Offsite Property Cost Offsets (2012 million dollars)		
		2% NPV	3% NPV	7% NPV
1	Base Case	7.65	6.83	4.57
	High Estimate	41.85	37.32	24.98
2	Base Case	11.50	10.10	6.46
	High Estimate	85.65	75.24	48.14
3	Base Case	12.07	9.13	3.81
	High Estimate	54.65	41.33	17.25
4	Base Case	14.35	12.75	8.44
	High Estimate	132.58	117.80	77.92

C.2.5 Duration of Onsite Spent Fuel Storage Risk

For this cost-benefit analysis, it is assumed that the each nuclear power plant operates through the term of its operating license and that the licensee continues to store spent fuel in the plant's SFP following commercial operation[24] to allow the spent fuel to cool sufficiently before placing into dry storage. Other than for operating reactors that have indicated they would not seek a license renewal, this analysis assumes that remaining operating reactors' operation expectancy will include a 20-year license renewal period, unless stated otherwise.[25] As a result, the average license will expire in 2039. Table 1 summarizes the average reactor operation expectancy by the identified SFP groupings.

[24] Decommissioning of the unit must be completed within 60 years of permanent cessation of operations under 10 CFR 50.82, "Termination of License." Completion of decommissioning beyond 60 years will be approved by the Commission only when necessary to protect public health and safety.

[25] Six U.S. nuclear power plant units have announced early retirements (with year of closure in parentheses) are Crystal River 3 (2013), Kewaunee (2013), San Onofre Units 2 and 3 (2013), Vermont Yankee (2014), and Oyster Creek (2019).

C.2.6 Dollar per Person-Rem Conversion Factor

Using the dollar value of the health detriment and a risk factor that establishes the nominal probability for stochastic health effects attributable to radiological exposure (fatal and nonfatal cancers and hereditary effects) provides a dollar per person-rem of $2,000, rounded to the nearest thousand, according to NUREG-1530, "Reassessment of NRC's Dollar per Person-Rem Conversion Factor Policy," dated December 1995 (Ref. C.19).

The NRC currently uses a value of statistical life (VSL)[26] of $3 million based on NUREG-1530, and a cancer risk factor of 7.0×10^{-4}, which is a reduction to the closest significant digit of a recommendation by the International Commission on Radiation Protection (ICRP) in Publication No. 60. Therefore, the dollar per person-rem is equal to $3 million times 7.0×10^{-4} rounded to the nearest thousand (because of uncertainties) or $2,000.

C.2.7 Onsite Property Decontamination, Repair, and Refurbishment Costs

SFP accident risks have significant contributions from onsite property monetary losses (e.g., repair and refurbishment) and plant decontamination. The risk dominant accident sequences involve the failure of the pool because of seismic or load drop events resulting in the loss of pool integrity. This scenario results in loss of SFP water inventory, Zircaloy cladding fire initiation with propagation through the spent fuel assemblies stored in the pool, and an uncontrolled radiological release from the reactor building. The NRC assumes that, based on the current regulatory framework, with insights from the Fukushima Dai-ichi accident, that onsite property would be radiologically affected in the following way. The consequences of a spent fuel fire are expected to be similar to the severe reactor accidents resulting in core damage and possible fuel melting as defined in NUREG/CR-5281, Section 3.2.4 (Ref. C.20). Based on this reference, the cleanup and decontamination costs are estimated to be approximately $165 million (1983 dollars) and the cost for permanent disposal of the damaged fuel is $26 million (1983 dollars). Using Table C.95 from the RA Handbook (Ref. C.21), the pool repair is expected to cost $72 million (1983 dollars). Adjusting these estimated costs using the CPI-U inflator formula and using a multiplier of three to model the high estimate and a divider of two to model the low estimate results in the values provided in Table 46.

Table 46 Onsite Property Decontamination, Repair, and Refurbishment Costs

	1983 dollars			2012 dollars		
Onsite Property Cost Element	Best Estimate	High Estimate	Low Estimate	Best Estimate	High Estimate	Low Estimate
Cleanup and decontamination	$165,000,000	$495,000,000	$82,500,000	$380,358,000	$1,141,074,000	$190,179,000
Repair Pool	$72,000,000	$216,000,000	$36,000,000	$165,974,000	$497,922,000	$82,987,000
Disposal of damaged fuel	$26,000,000	$78,000,000	$13,000,000	$59,935,000	$179,805,000	$29,968,000
Total	$263,000,000	$789,000,000	$131,500,000	$606,267,000	$1,818,801,000	$303,134,000

[26] The value of a statistical life (VSL) is the monetary value of a mortality risk reduction that would prevent one statistical (as opposed to an identified) death (Ref. C.22). The VSL is a key component in the calculation of the dollar per person-rem value, which is the product of the VSL multiplied by a risk coefficient.

C.2.8 Replacement Energy Costs

Replacement energy costs are the costs for replacing the energy from the nuclear power plant because of a plant shutdown to install required equipment or because of an accident.[27] The NRC assumes that replacement energy costs would be required until onsite decontamination and repair efforts are completed or the unit is retired. The NRC assumes that the cost per year of replacement energy would be about $2.3 million (2012 dollars).

The NRC assumes that licensees engage in power purchase agreements (PPA)[28] to economically purchase replacement power. A PPA is a legal contract between an electricity generator (licensee) and a power purchaser. The NRC assumes that a licensee will not be able to replace the power through other generation for 7 years and would have to buy power from the market. Although not all licensees may have PPAs, the licensee will still replace the lost energy any time that the nuclear power plant is not operating to meet its electrical power supply obligations. The NRC assumes that after 7 years, the onsite decontamination and repair efforts are completed or the unit is retired and other power sources will be developed to replace the unit's lost electrical generation capability. Therefore, the NRC assumes that the undiscounted cost of replacement energy would be $15.9 million.

C.2.9 Occupational Worker Exposure (Accident)

There are two types of occupational exposure related to accidents: short-term and long-term. The first occurs at the time of the accident and during the immediate management of the emergency. The second is a long-term exposure, presumably at significantly lower individual rates, associated with the cleanup and refurbishment or decommissioning of the damaged facility. The value gained in the avoidance of both types of exposure is conditioned on the change in frequency of the accident's occurrence.

The experiences at the Three Mile Island Unit 2 (TMI-2), the Chernobyl, and the Fukushima nuclear power plants illustrated that significant occupational exposures could result from performing activities outside the control room during a power reactor accident. At TMI-2, the average occupational exposure related to the incident was approximately 1.0 rem, with a collective dose of 1,000 person-rem occurring over a 4-month span, after which time occupational exposure approached pre-accident levels. For Chernobyl, the average dose for persons closest to the plant was 3.3 person-rem (Ref. C.21, p. 5.30), yielding a collective dose of 3,300 person-rem.

The accident at Fukushima involved release of both short-lived and long-lived radionuclides from the reactor cores within Units 1, 2, and 3, and no release from the fuel stored in the SFPs. Significant changes in the release of radioactivity occurred following changes in the status of the

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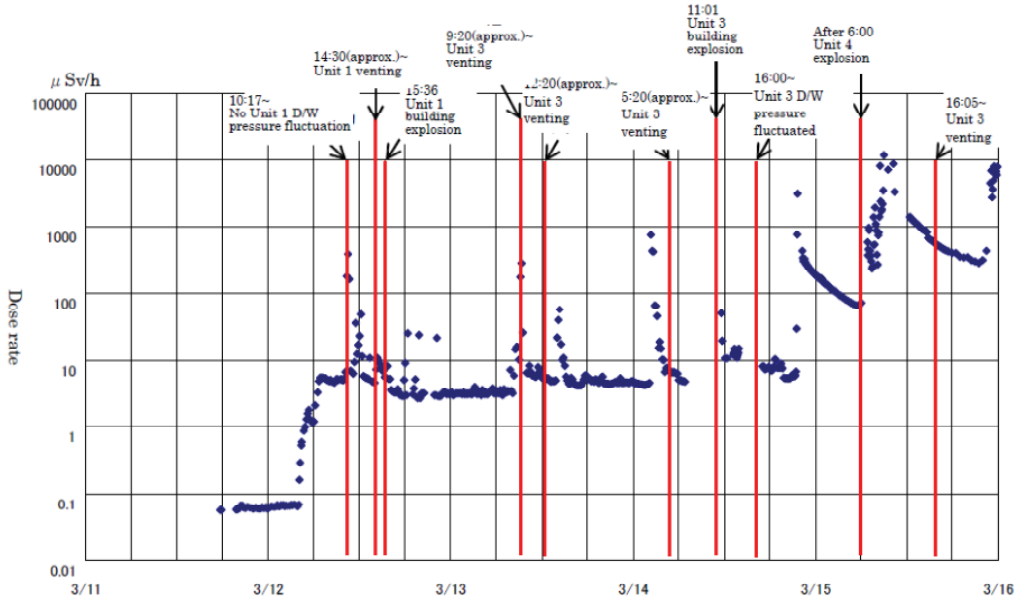
[27] The replacement energy cost is only the cost to buy the energy for production on the market. Therefore, the cost would be the cost of buying the cheapest energy. These estimates do not include transmission or distribution costs.

[28] A power purchase agreement is a contract between two parties, one who generates electricity for the purpose of sale (the seller) and one who is looking to purchase electricity (the buyer). The PPA defines all of the commercial terms for the sale of electricity between the two parties, including when the project will begin commercial operation, schedule for delivery of electricity, penalties for under delivery, payment terms, and termination.

core, primary containment, and secondary containment. After the Fukushima unit 1 building explosion on March 12, 2011, the unit 3 building explosion on March 14, and the unit 4 building explosion, which released radioactivity from Unit 3 because of a shared ventilation system, and the exposure of the unit 2 reactor fuel rods on March 15, radioactive materials were released into the environment and surrounding areas of the Fukushima Dai-ichi nuclear power plant. Measurement and evaluation of radiation exposure levels for workers engaged in emergency work at the Fukushima Dai-ichi NPS have been implemented continuously since the Tohoku earthquake.

As shown in Figure 13, the dose rate in the vicinity of the main gate at the Fukushima Dai-ichi site near the time of the Unit 4 explosion varied between 20 mrem and 1.0 rem per hour (between 200 and 10,000 μSv per hour).

Figure 13: Dose rate in vicinity of Fukushima Dai-ichi nuclear plant site main gate between March 11 and March 16, 2011



Source: Fukushima Nuclear Accident Analysis Report p. 371 (Ref. C.24).

On March 22 and 23, surveys of the airborne radioactivity and dose rates around the Fukushima Dai-ichi site were collected and documented. The dose rates are shown on Figure 14.

Figure 14: Fukushima Dai-ichi site dose rates between March 22 and March 23, 2011

Source: INPO 11-005, p 41 (Ref. C.23).

The distribution of total monthly exposure for workers engaged in radiation work at the Fukushima Dai-ichi nuclear plant site for the first 3 months following the March 2011 accident is provided in Table 47.

Table 47 Average Accident Occupational Exposure at Fukushima Dai-ichi Nuclear Power Plant from March to May 2011

Total Radiation Exposure (mSv)	Number of Plant Workers Exposed		
	March 2011[1]	April 2011[2]	May 2011[3]
≥ 250	6	0	0
200 - 249	2	0	0
150 - 199	14	0	0
100 - 149	77	0	0
50 - 99	309	3	0
20 - 49	859	81	19
10 - 19	1041	310	144
< 10	1434	3214	2854
Total number of workers	3742	3608	3017

Notes:
1. Maximum March 2011 occupational exposure was 670.4 mSv.
2. Maximum April 2011 occupational exposure was 69.3 mSv.
3. Maximum May 2011 occupational exposure was 41.6 mSv.
4. One mSv is equal to 0.1 rem.

Source: Wada et al, Occupational and Environmental Medicine, 2012 August; 69(8): p. 600 (Ref. C.26).

To estimate the monthly total occupational radiation exposure received by all workers, a high estimate, base case, and low estimate were calculated based on the maximum category value, the midpoint category value, and the first quartile category value. The results are tabulated in Table 48.

Table 48 Estimated Immediate Accident Occupational Monthly Exposure at Fukushima

Radiation Exposure (mSv)	Best Estimate Category Radiation Exposure (mSv)			High Estimate Category Radiation Exposure (mSv)			Low Estimate Category Radiation Exposure (mSv)		
	March 2011	April 2011	May 2011	March 2011	April 2011	May 2011	March 2011	April 2011	May 2011
≥ 250	460.2			670.4			355.1		
200 - 249	224.5			249			212.25		
150 - 199	174.5			199			162.25		
100 - 149	124.5			149			112.25		
50 - 99	74.5	69.3		99	69.3		62.25	62.25	
20 - 49	34.5	34.5	34.5	49	49	41.6	27.25	27.25	27.25
10 - 19	14.5	14.5	14.5	19	19	19	12.25	12.25	12.25
< 10	5	5	5	10	10	10	2.5	2.5	2.5
Total Monthly Dose	90,200	23,600	17,000	125,600	42,200	32,100	72,500	14,200	9,400
Avg Worker Dose	24.1	6.5	5.6	33.6	11.7	10.6	19.4	3.9	3.1

The immediate accident occupational exposure for a SFP accident shown in Table 49 is estimated based on the Fukushima data and the following assumptions:

- The immediate accident period lasts for 1 year.
- The workforce during the immediate accident period is 3,700 workers.
- The average worker radiation exposure remains constant at the May 2011 value from May 2011 through February 2012.

Table 49 Immediate Accident Occupational Exposure for a Spent Fuel Pool Fire

Case	Immediate Accident Occupational Exposure (averted person-rem)
Low Estimate	18,070
Best Estimate	28,380
High Estimate	48,880

After the immediate response to a SFP accident, a long process of cleanup and refurbishment or decommissioning will follow. The Fukushima Nuclear Accident Analysis Report states, “The average value for 5,128 people in April of 2012 was 1.07 mSv per worker because of decreasing trends in environment dose rates (Ref. C.24, p 415). The NRC assumes that the process of cleanup and refurbishment or decommissioning will begin 1 year after the accident and will take 7 years to complete. During those 7 years, the NRC assumes that each occupational worker at the damaged reactor site will be exposed to 1.07 mSv per month (0.107 rem per month) for the duration of the cleanup and refurbishment or decommissioning. Assuming the average value for 5,128 workers would remain for the duration yields a cumulative long-term occupational dose of 46,000 person-rem.

In NUREG/CR-5281 (Ref. C.20), Jo et al. (1989) conducted what essentially amounted to a regulatory analysis of a non-reactor nuclear fuel cycle facility using Heaberlin, et al 1983 Handbook (Ref. C.27) as guidance. The accidental occupational exposure was assumed to be similar to that from TMI-2, which is 4,580 person-rem.

As described in the RA Handbook (Ref. C.21, p 5.30), the DOE (1987) summarized results on the collective dose received by the populace surrounding the Chernobyl accident. Average dose equivalents of 3.3 rem per person, 45 rem per person, and 5.3 rem per person were estimated for residents within 3 km, between 3 km and 15 km, and between 15 km and 30 km of Chernobyl, respectively (Ref. C.28, p. A-5). Assuming 1,000 workers and a 4.2 multiplier, an estimate radiation exposure of 14,000 person-rem results.

Site worker exposures following a SFP accident could be greater than that of a reactor core melt accident. This is because a SFP stores significantly more fuel assemblies than a reactor core. Given the uncertainties in existing data and variability in severe accident parameters and worker response, Table 50 provides the long-term occupational dose used in this analysis to analyze SFP accidents.

Table 50 Long-Term Accident Occupational Exposure for a Spent Fuel Pool Fire

Case	Long-Term Accident Occupational Exposure (averted person-rem)
Low Estimate	4,580
Best Estimate	14,000
High Estimate	46,000

C.2.10 Spent Fuel Pool Release Fractions

The SFP release fractions used in this analysis is based on the results of the SFPS for Group 1 as well as previous SFP studies. Table 51 shows a comparison of the release fractions between the SFPS and previous studies that demonstrates that cesium release fractions are generally less in the SFPS when compared to previous studies, and the timing of the release is generally longer.

The range of release fractions for this analysis is shown in Table 52. The Group 1 high SFP loading release fractions are based on the high-density cases in the SFPS with the low estimate representing cases where the reactor building remains intact, the base case reflects cases with significant air oxidation as a result of substantial damage to the refueling bay, and the high estimate represents a bounding case with large scale damage and relocation of the spent fuel assemblies and subsequent interaction of the fuel debris with the concrete floor. The Group 1 low SFP loading release fractions represent the low-density cases from the SFPS. For the other groups, the range of release fractions is consistent with past studies, but the high estimate is 90 percent based on insights from the SFPS regarding the molten core concrete interaction sensitivity study. The low SFP loading release fractions in Groups 2, 3, & 4 are assumed the same as in Group 1 since the releases are dominated by the recently discharged fuel.

Table 51 Comparison of Release Fractions from Current and Previous Spent Fuel Pool Analyses

Resolution of GI-82: NUREG-1353 (Ref. C.12), NUREG/CR-4982 (Ref. C.29), NUREG/CR-5281 (Ref. C.20)	NUREG-1738 (Ref. C.10)	Spent Fuel Pool Study (Ref. C.7)
• 10 to 100% cesium release (100% assumed for cases 1 and 2) • Release over 8 hours for a propagating SFP zirconium fire (assumed) • 0.25 (BWR) or 1.0 (PWR) conditional probability if fuel becomes uncovered	• 75% cesium release (assumed from NUREG-1465 (Ref. C.30) • Instantaneous draindown for large seismic event • 2 to 14 hour heatup depending on fuel age (see Ref. C.10, Table A1-1)	• Less than 1% to 49% cesium release • Draindown to uncovery ranges from 2.5 to 43 hours (when leak exists) • Start of release ranges between 8 hours to greater than 72 hours

Table 52 Estimated Cumulative Cesium Inventory Release Fraction Given a Spent Fuel Pool Fire

SFP Group	SFP loading	Low Est.	Base Case	High Est.
Group 1	High-density	3%	40%	90%
	Low-density	0.5%	3%	5%
Group 2, 3 & 4	High-density	10%	75%	90%
	Low-density	0.5%	3%	5%

C.2.11 Atmospheric Modeling and Meteorology

The atmospheric transport and dispersion model used in this analysis are based on the Peach Bottom MACCS2 results described in Section 7.1.2 of the SFPS (Ref. C.7), which uses a straight-line Gaussian plume segment dispersion model. As described in this study, the atmospheric release of radionuclides is discretized into (at longest) 1-hour plume segments. This accounts for variations in the release rate, as well as for changes in wind direction. More plume segments increase the resolution of the dispersion modeling to the point the resolution corresponds to the time resolution of the weather data, because each segment can travel in a compass direction representative of the actual weather data at the time the plume segment is released.

Two important parameters and variables required to model a SFP site are 1) the population density and distribution and 2) the site meteorology. The radionuclide inventory, source term (i.e., release fraction, release start time, and release duration), initial plume dimensions (related to the system geometry), and plume heat content were described.

C.2.12 Population and Economic Data

Population densities and distributions characteristics for SFP sites are examined to provide perspective on site demographic characteristics important to this cost-benefit analysis. Based on the review performed, site population densities near SFPs have the following statistical characteristics:

Table 53 Population Density within a 50 Mile Radius of U.S. Nuclear Power Plant Sites

Case	Statistical Parameter	Average Population Density within 50 miles (No. of people per square mile)	Representative Site Demographics
High estimate	90th percentile	722	Peach Bottom
Mean estimate	Mean	303	Surry
Median estimate	Median	183	Palisades
Low estimate	20th percentile	102	Point Beach

Source: 2010 census. Population density calculations do not correct the area within the radius that is water

Representative site demographics were selected to represent the 90th percentile, the mean, the median, and the 20th percentiles. For each representative site, the site population and economic data was created for 16 compass sectors and then interpolated onto a 64 compass-sector grid for better spatial resolution for the consequence analysis. Site population data is projected to the year 2011 using the latest version of the computer code SECPOP2000 (Ref. C.31). SECPOP2000 uses 2000 census data and applies a multiplier to account for population growth and an economic multiplier to account for the value of the dollar to create site data for the MELCOR Accident Consequence Code System (MACCS2). A multiplier value of 1.1051 from the U.S. Census Bureau was used to account for the average population growth in the U.S. from 2000 to 2011. Consistent with the approach used in the SFPS, the economic values from the database in SECPOP2000 (which uses an economic database based on the year 2002) were scaled to account for price escalation between the years 2002 and 2011. A scaling factor of 1.250 was derived based on the Consumer Price Index.

Population Demographic Sensitivity

The base case and the three additional site population densities and distributions near spent fuel pool locations discussed above were used as additional inputs into the MACCS2 calculations. Although the results provided in Appendix section C.2.12 provides insight into the analysis sensitivity to site population demographics in the United States, the results are not representative of any specific site because site specific meteorology for these additional sites is not used.

Table 54 Sensitivity of Public Health (Accident) Base Case Results to Population Demographics within 50 Miles

SFP Group	Site Population	Dose (averted person-rem per pool)	Benefits (2012 million dollars)		
			2% NPV	3% NPV	7% NPV
1	Low	469	$0.73	$0.65	$0.44
	Median	1097	$1.71	$1.53	$1.02
	Average (base case)	1739	$2.72	$2.42	$1.62
	High	2172	$3.39	$3.03	$2.02
2	Low	652	$0.98	$0.86	$0.55
	Median	1421	$2.14	$1.88	$1.20
	Average (base case)	2109	$3.18	$2.79	$1.79
	High	2684	$4.04	$3.55	$2.27
3	Low	1046	$1.09	$0.82	$0.34
	Median	2360	$2.46	$1.86	$0.78
	Average (base case)	3616	$3.77	$2.85	$1.19
	High	4560	$4.75	$3.59	$1.50
4	Low	751	$1.16	$1.03	$0.68
	Median	1586	$2.46	$2.18	$1.44
	Average (base case)	2284	$3.54	$3.14	$2.08
	High	2933	$4.54	$4.03	$2.67

Variations in population densities given the underlying assumptions stated above have the following net change on the averted public health (accident) attribute as summarized in Table 55.

Table 55 Net Percent Change in Public Health (Accident) Base Case Results for Variations in Population Densities within 50 Miles

Site Population Case	Statistical Parameter	Average Population Density within 50 miles (No. of people per square mile)	Net Percent Change in Public Health (Accident) Base Case (within 50 miles)
High estimate	90th percentile	722	25% – 28%increase
Mean estimate	Mean	303	No change
Median estimate	Median	183	21% - 37% decrease
Low estimate	20th percentile	102	67% - 73% decrease

Because a spent fuel pool fire could result in impacts to public health that extend beyond 50 miles, this case evaluates the sensitivity of averted public health exposures extending beyond 50 miles from the site, using the base case assumptions and the standard and sensitivity value for the person-rem conversion factor. Table 56 shows the sensitivity on public health (accident) benefits of extending the consequence analysis beyond 50 miles for the base case.

Table 56 Sensitivity of Public Health (Accident) Benefits for Expedited Transfer Alternative–Low-density Spent Fuel Pool Storage extending beyond 50 miles (Base case with $2,000 and $4,000 per person-rem)

SFP Group	Case	Dose conversion factor ($/person-rem)	Dose (averted person-rem per pool	Benefits (2012 million dollars)		
				2% NPV	3% NPV	7% NPV
1	Alternative 2 - Low-density storage	$2,000	11,120	$17.37	$15.49	$10.37
		$4,000		$34.73	$30.98	$20.73
2	Alternative 2 - Low-density storage	$2,000	13,680	$20.61	$18.10	$11.58
		$4,000		$41.22	$36.21	$23.17
3	Alternative 2 - Low-density storage	$2,000	22,730	$23.67	$17.90	$7.47
		$4,000		$47.33	$35.80	$14.94
4	Alternative 2 - Low-density storage	$2,000	15,880	$24.57	$21.83	$14.44
		$4,000		$49.14	$43.66	$28.88

Sensitivity of Offsite Property Cost Offset Results to Population Demographics

Certain metrics such as property use, the number of displaced individuals (either temporarily or permanently), and the extent to which such actions may be needed are affected by the population size and the amount of economic activity in the vicinity of the postulated accident.

This section provides a basis for understanding the nature and the extent of the relationship between population densities, distributions characteristics, and property values near spent fuel pool sites. This examination provides a perspective on how important changes to these site demographic variables are for this regulatory analysis. The base case and the three additional site population densities, distributions, and economic characteristics near spent fuel pool locations are discussed above. These population and economic characteristics were used as additional inputs into the MACCS2 calculations that otherwise still used the SFPS reference plant specific values. Although the results provided in Table 57 provide insight into the analysis sensitivity to site population demographics in the U.S., the results are not representative of any specific site because site specific meteorology for these additional sites is not used. These measures are also subject to large uncertainties, as it is difficult to model the impact of disruptions to many different aspects of local economies, the loss of infrastructure on the general U.S. economy, or the details of how long-term protective actions would be performed.

Table 57 Sensitivity of Offsite Property Cost Offset Results to Population Demographics within 50 Miles (Base Case using EPA Intermediate PAG Criterion)

SFP Group	Site Population	Offsite Property Cost Offsets (2012 million dollars)		
		2% NPV	3% NPV	7% NPV
1	Low	$1.29	$1.15	$0.77
	Median	$4.19	$3.73	$2.50
	Average (base case)	$7.65	$6.83	$4.57
	High	$12.55	$11.19	$7.49
2	Low	$2.04	$1.79	$1.14
	Median	$6.75	$5.93	$3.79
	Average (base case)	$11.50	$10.10	$6.46
	High	$13.43	$11.80	$7.55
3	Low	$2.09	$1.58	$0.66
	Median	$6.84	$5.18	$2.16
	Average (base case)	$12.07	$9.13	$3.81
	High	$17.08	$12.91	$5.39
4	Low	$2.60	$2.31	$1.53
	Median	$8.69	$7.72	$5.11
	Average (base case)	$14.35	$12.75	$8.44
	High	$16.14	$14.34	$9.48

Because a spent fuel pool fire under certain scenarios and environmental conditions could result in impacts to offsite property located beyond 50 miles from the postulated accident site, this case evaluates the sensitivity of offsite property cost offsets for damages occurring beyond 50 miles from the site, using the base case assumptions and the intermediate EPA PAG criterion. Table 58 shows the sensitivity on offsite property cost offsets of extending the consequence analysis beyond 50 miles for the base case.

Table 58 Sensitivity of Offsite Property Cost Offset Results to Consequences beyond 50 Miles (Base Case using EPA Intermediate PAG Criterion)

SFP Group	Case	Offsite Property Cost Offsets (2012 million dollars)			
		2% NPV	3% NPV	7% NPV	% increase
1	Base case - within 50 miles	$8.96	$7.99	$5.35	
	Sensitivity - beyond 50 miles	$16.36	$14.59	$9.76	83%
2	Base case - within 50 miles	$9.03	$7.93	$5.08	
	Sensitivity - beyond 50 miles	$28.79	$25.29	$16.18	219%
3	Base case - within 50 miles	$11.45	$8.66	$3.61	
	Sensitivity - beyond 50 miles	$27.17	$20.55	$8.57	137%
4	Base case - within 50 miles	$9.81	$8.71	$5.76	
	Sensitivity - beyond 50 miles	$39.62	$35.20	$23.29	304%

C.2.13 Long-Term Habitability Criteria

The long-term phase is the period following the 7-day emergency phase and is modeled for 50 years to calculate consequences from exposure of the average person. Radiation exposure during this phase is mainly from external radiation from trace contaminants that remain after the land is decontaminated, or in lightly contaminated areas where no decontamination was required. Internal radiation exposures may also occur during this period, including inhalation of resuspended radionuclides and ingestion of food and water with trace contaminants. Depending on the relevant protective action guides (PAGs) and the level of radiation, food, and water below a certain limit could be considered adequately safe for ingestion, and lightly contaminated areas could be considered habitable.

A long-term cleanup policy for recovery after a severe nuclear power plant accident does not currently exist. The actual decisions regarding how land would be recovered and populations relocated after an accident would be made by a number of local, State, and Federal jurisdictions and would most likely be based on a long-term cleanup strategy, which is currently being developed by the NRC, U.S. Environmental Protection Agency (EPA), and other Federal agencies. Furthermore, a cleanup standard may not have an explicit dose level for cleanup. Instead, the cleanup strategy may give local jurisdictions the ability to develop localized cleanup goals after an accident, to allow for a number of factors that include sociopolitical, technical, and economic considerations.

Site-specific values are used to determine long-term habitability. For habitability, most States adhere to EPA intermediate phase protective action guides that allow a dose of 2 rem in the first year and 500 mrem each year thereafter (Ref. C.32). This habitability criterion was used in previous SFP studies, which used 4 rem in 5 years to represent these PAG levels (e.g., 2 rem in year one, followed by 0.5 rem each successive year). The nationally and internationally recommended upper bound for dose in a single year from man-made sources, excluding medical radiation, is 500 mrem per year to the whole body of individuals in the general population. The EPA states “these recommendations were not developed for nuclear incidents … [and] also not appropriate for chronic exposure” (Ref. C.32, p. E-12). However, some States, such as the State of Pennsylvania, has adopted a habitability criterion of 500 mrem beginning in the first year (and each following year) as determined by the Pennsylvania Code Title 25 Section 219.51 (Ref. C.33). The use of this long-term habitability criterion reduces the predicted long-term population doses and health effects and increases the costs associated with interdiction, decontamination, and condemnation.[29]

Given the uncertainties in which long-term habitability criterion would be used, Table 60 provides the long-term phase habitability criterion used in this analysis to analyze the consequences of SFP accidents on public health (accident).

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[29] Interdiction and condemnation refer to the relocation of people from contaminated areas according to the habitability criterion. Interdiction is the temporary relocation of the affected population while decontamination, natural weathering, and radioactive decay reduce the contamination levels. Condemnation is the permanent relocation of the affected population if decontamination, natural weathering, and radioactive decay cannot adequately reduce contamination levels to habitability limits within 30 years.

Table 59 Long-Term Habitability Criterion

Case[30]	Long-Term Habitability Criterion	Protective Action Basis
Low Estimate	500 mrem annually	Pennsylvania dose limit to the public
Base Case	2 rem in the first year and 500 mrem each year thereafter	EPA intermediate phase PAGs
High Estimate	2 rem annually	EPA intermediate phase PAG: first year

MACCS2 computer runs were run for each of the protective action levels listed in Table 59 to calculate averted dose and offsite property damage using the representative plant site demographics listed in Table 53.

Different habitability criteria given the underlying assumptions stated above has the following net change on the averted public health (accident) attribute as summarized in Table 60.

Table 60: Sensitivity of Public Health (Accident) Benefits to Habitability Criteria (within 50 Miles)

SFP Group	Habitability Criteria	Dose (averted person-rem per pool)	Benefits (2012 million dollars)		
			2% NPV	3% NPV	7% NPV
1	Low (500 mrem annually)	770	$1.21	$1.08	$0.72
	Base Case (4rem / 5years)	1,740	$2.72	$2.42	$1.62
	High (2 rem annually)	1,980	$3.09	$2.75	$1.84
2	Low (500 mrem annually)	900	$1.36	$1.20	$0.77
	Base Case (4rem / 5years)	1,630	$2.45	$2.15	$1.38
	High (2 rem annually)	2,480	$3.74	$3.29	$2.10
3	Low (500 mrem annually)	1,580	$1.64	$1.24	$0.52
	Base Case (4rem / 5years)	3,020	$3.14	$2.37	$0.99
	High (2 rem annually)	4,180	$4.36	$3.29	$1.37
4	Low (500 mrem annually)	960	$1.49	$1.33	$0.88
	Base Case (4rem / 5years)	1,690	$2.62	$2.33	$1.54
	High (2 rem annually)	2,730	$4.23	$3.76	$2.49

The use of these habitability criteria also affects the values of offsite property damage used in this analysis. Certain metrics such as offsite property damage, the number of displaced individuals (either temporarily or permanently) and the extents to which such actions may be needed are inversely proportional to changes in collective dose resulting from changes in habitability criteria.

[30] Cases are defined as low and high estimate based on the effect that different long-term habitability criteria have on averted radiation exposure.

These criteria provide a benchmark for understanding the nature and the extent of the relationship between collective dose, economic consequences, and habitability criteria following a severe SFP accident. These measures are subject to large uncertainties, as it is difficult to model the impact of disruptions to many different aspects of local economies, the loss of infrastructure on the general U.S. economy, or the details of how long-term protective actions would be performed.

Table 61 Net Percent Change in Public Health (Accident) Base Case Results for Variations in Population Densities within 50 Miles

Habitability Criterion Case	Habitability Criterion	Net Percent Change in Public Health (Accident) Base Case (within 50 miles)
High estimate	2 rem annually	14% – 20% increase
Base case	2 rem first year, 500 mrem thereafter (4 rem / 5 years)	No change
Low estimate	500 mrem annually	56% – 58% decrease

Offsite Property Costs Sensitivity to Habitability Criteria

A long-term cleanup policy for recovery after a severe nuclear power plant accident does not currently exist. The actual decisions regarding how land would be recovered and populations relocated after an accident would be made by a number of local, State, and Federal jurisdictions and would most likely be based on a long-term cleanup strategy, which is currently being developed by the NRC, EPA, and other Federal agencies. Furthermore, a cleanup standard may not have an explicit dose level for cleanup. Instead, the cleanup strategy may give local jurisdictions the ability to develop localized cleanup goals after an accident, to allow for a number of factors that include sociopolitical, technical, and economic considerations. Given the uncertainties in which long-term habitability criterion would be used, Table 62 provides a low and high value for the long-term phase habitability criterion for use in a sensitivity analysis to analyze the effect on the costs for offsite property damage.

Table 62: Sensitivity of Offsite Property Damage Cost Offsets within 50 Miles to Different Habitability Criteria

SFP Group	Habitability Criteria	Offsite Property Cost Offsets (2012 million dollars)		
		2% NPV	3% NPV	7% NPV
1	Low Est. (500 mrem annually)	$12.83	$11.44	$7.66
	Base Case (4rem / 5years)	$7.65	$6.83	$4.57
	High Est. (2 rem annually)	$7.19	$6.41	$4.29
2	Low Est. (500 mrem annually)	$16.56	$14.54	$9.31
	Base Case (4rem / 5years)	$11.50	$10.10	$6.46
	High Est. (2 rem annually)	$11.10	$9.75	$6.24
3	Low Est. (500 mrem annually)	$18.71	$14.15	$5.90
	Base Case (4rem / 5years)	$12.07	$9.13	$3.81
	High Est. (2 rem annually)	$11.50	$8.70	$3.63
4	Low Est. (500 mrem annually)	$19.28	$17.13	$11.33
	Base Case (4rem / 5years)	$14.35	$12.75	$8.44
	High Est. (2 rem annually)	$14.02	$12.45	$8.24

This sensitivity analysis uses three protective action levels—the Pennsylvania PAG of 500 mrem annually for the low estimate, the EPA intermediate phase PAG level of 2 rem in the first year, and 500 mrem annually thereafter for the base case, and 2 rem annually for the high estimate—to evaluate post-accident collective dose and offsite property costs. As discussed in Appendix section C.2.12, offsite property costs are inversely proportional to changes in collective dose resulting from changes in habitability criteria (i.e., lower PAG guidelines result in lower collective dose value and higher offsite property costs). These results show the cost offsets increase by up to 67 percent (7 percent net present value) than those in the Group 1 base case result when the 500 mrem annual limit is used. Conversely, offsite property damage cost offsets decrease by up to 6 percent (7 percent net present value) than those in the Group 1 base case result when the 2 rem annual limit is used.

C.2.14 Emergency Response Modeling

This cost-benefit analysis uses the emergency response model contained in the Reference Plant-specific MACCS2 results described in Section 7.1.2 and Appendix A of the SFPS. The extended loss of ac power is assumed to be limited to the plume exposure pathway emergency planning zone (EPZ) (approximately 16 kilometers or 10 miles) because of the assumption that the strength of the seismic event is from the proximity of the seismic event to the site, rather than being a wider impact from a larger magnitude. See Section 7.1.4 of the SFPS for additional details.

A summary of the evacuation timing and speeds for each cohort modeled in the SFPS and reproduced here is provided in Table 63. This evacuation timing and speeds is used to produce the consequence analyses results for this analysis.

Table 63 Evacuation Model 1: Plume Exposure Pathway EPZ Evacuation

Population			Response Delays (hours)				Phase Duration (hours)		Evacuation Travel Speeds (mph)		
Cohort		Population Fraction	Siren (OALARM)	Delay to Shelter	Delay to Evacuation	Total (Depart time)	Early (DURBEG)	Middle (DURMID)	Early (ESPEED)	Middle (ESPEED)	Late (ESPEED)
1	0 to 10 miles Early Evacuees	0.3	1	0	0	1	1	0.5	20	15	5
	10 to 20 miles Shadow			2	1	4					
2	0 to 10 miles General Public	0.417	1	1	1	3	0.25	3	5	2	20
3	0 to 10 miles Special Facilities	0.006	1	0	4	5	0.5	0.5	2	15	20
4	0 to 10 miles Evacuation Tail	0.1	1	2	3	6	0.5	0.5	2	15	20
5	0 to 10 miles Schools	0.172	1	0	0.5	1.5	1	0.5	20	15	20
6	0 to 10 miles Nonevacuating Public	0.005	1	-	-	-	-	-	-	-	-

Meteorological data used to calculate offsite consequences for this analysis consisted of 1 year of hourly meteorological data (8,760 data points for each meteorological parameter) for the Peach Bottom site evaluated in the SFPS (Ref. C.7) and in NUREG-1935 (Ref. C.34). The Peach Bottom site provided 2 years of weather data, including directly measured hourly

precipitation data. Stability class data were derived from temperature measurements at two elevations on the site meteorological towers. The specific year of meteorological data chosen for the Peach Bottom site was 2006, which was based on data recovery (greater than 99 percent being desirable) as documented in NUREG/CR-7009 (Ref. C.35). Different trends (e.g., wind rose pattern and hours of precipitation) between the years were estimated to have a relatively minor (less than 25 percent) effect on the results. More specific details of the weather data can be found in NUREG/CR-7009.

The wind rose shown in Figure 15 shows the Peach Bottom site wind direction (direction the wind blows toward) data that were used in the consequence analyses for this analysis. The wind rose in the figure below suggests that the predominant wind direction is to the south and east and a secondary direction in terms of likelihood is to the northwest to north.

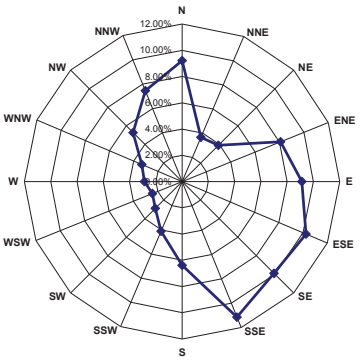

Figure 15 Reference plant wind rose
Source: SFPS (Ref. C.7, p. A-3)

Although using a single plant's emergency response modeling and consequence analyses introduces uncertainty, the conditional individual risk measures near the site are expected to be relatively insensitive to site-specific characteristics (i.e., emergency response measures). This is because the relatively delayed and prolonged releases as predicted by the SFPS and the lack of short-lived radionuclides allow time for effective protective actions, in both the early and long term phases, to limit exposures to the public particularly in the event of large releases. This is consistent with previous studies in which individual early and latent fatality risks were projected to be low. Therefore, the resulting individual risk measures near the site can used for comparisons to the quantitative health objectives represent risk to the average individual within 1.6 and 16 kilometers (1 and 10 miles) of the plant.

C.2.15 Uniform Fuel Pattern during an Outage Sensitivity

The base case of this regulatory analysis assumes that each licensee has prearranged the spent fuel pool such that discharged assemblies can be placed directly into a 1x4 arrangement for the discharges of the last two outages. This approach is consistent with the requirements discussed in Section 9.3 of the SFPS. However, those requirements do allow for the fuel to be stored in a less favorable configuration for some time following discharge if other considerations prevent prearrangement. To capture the effects of nonbeneficial arrangement of discharged fuel, this regulatory analysis evaluates the situation in which the discharged spent fuel is uniformly arranged during the outage to evaluate the effect of this aspect on the public health (accident) attribute.

For the offsite consequence analysis, the sequences with recently discharged fuel in a uniform configuration were binned in a similar manner to the low-density and high-density (1x4) loading scenarios. Because licensees are required to move their recently discharged fuel to a more favorable configuration after a certain amount of time, this sensitivity assumes that the high-density uniform case becomes identical to the high-density (1x4) case by the end of operating cycle phase 2 (OCP 2) or within 25 days.

Table 64 provides a comparison of the effect on the public health (accident) attribute if a plant operator initially places discharged spent fuel in a uniform pattern and achieves the 1x4 pattern by the end of OCP 2 (i.e., within 25 days) versus placing the fuel directly into the 1x4 pattern.

Table 64: Sensitivity of Public Health (Accident) Benefits (within 50 Miles) to Initial Loading Pattern of Discharged Fuel

SFP Group	Initial Loading Pattern of Discharged Fuel	Dose (averted person-rem per pool)	Benefits (2012 million dollars)		
			2% NPV	3% NPV	7% NPV
1	Base Case - 1x4	1,740	$2.72	$2.42	$1.62
	Uniform fuel pattern	2,040	$3.18	$2.84	$1.90
2	Base Case - 1x4	1,630	$2.45	$2.15	$1.38
	Uniform fuel pattern	1,840	$2.77	$2.44	$1.56
3	Base Case - 1x4	3,020	$3.14	$2.37	$0.99
	Uniform fuel pattern	3,310	$3.45	$2.61	$1.09
4	Base Case - 1x4	1,690	$2.62	$2.33	$1.54
	Uniform fuel pattern	1,980	$3.07	$2.73	$1.80

The placement of the discharged fuel directly into a 1x4 pattern reduces the estimated averted dose within 50 miles of the site between 10 percent and 17 percent discounted at 7 percent compared to the cases when achieving this fuel pattern is delayed for up to 25 days at the end of OCP 2. These effects are bounded by the assumption of the unavailability of natural circulation air cooling for the base case and high estimate.

Offsite Property Cost Offset Sensitivity

Table 65 provides a comparison of the effect on the offsite property cost offsets if a plant operator initially places discharged spent fuel in a uniform pattern and achieves the 1x4 pattern by the end of OCP 2 (i.e., within 25 days) versus placing the fuel directly into the 1x4 pattern.

Table 65 Sensitivity of Offsite Property Cost Offsets within 50 Miles to Initial Loading Pattern of Discharged Fuel

SFP Group	Initial Loading Pattern of Discharged Fuel	Offsite Property Cost Offsets (2012 million dollars)		
		2% NPV	3% NPV	7% NPV
1	Base Case - 1x4	8.96	7.99	5.35
	Uniform fuel pattern	9.86	8.80	5.89
2	Base Case - 1x4	9.03	7.93	5.08
	Uniform fuel pattern	14.82	13.02	8.33
3	Base Case - 1x4	11.45	8.66	3.61
	Uniform fuel pattern	15.56	11.77	4.91
4	Base Case - 1x4	9.81	8.71	5.76
	Uniform fuel pattern	18.50	16.44	10.87

C.3 Implementation Assumptions

C.3.1 Dry Storage Occupational Exposure (Routine)

Routine occupational exposure associated with dry storage of spent fuel includes worker dose associated with additional DSC loading, unloading and handling activities; additional ISFSI operations, maintenance, and surveillance activities; additional DSC storage at an ISFSI; and additional transportation cask loading, unloading, and handling activities.

Worker dose associated with DSC loading operations vary depending upon the cask technology being loaded, the characteristics of the fuel being loaded (e.g., fuel age and burnup), and fuel loading patterns in the DSC (e.g., the location of short-cooled, high burnup spent fuel or colder spent fuel within DSC baskets using regional loading). For the regulatory baseline, a worker dose of 400 person-mrem per DSC loaded was assumed. This radiation dose is consistent with the exposure value used in EPRI TR-1021049 (Ref. C.36) and in EPRI TR-1018058 (Ref. C.37), which analyzed worker impacts associated with loading spent fuel for transport to the proposed Yucca Mountain repository. Some sites achieve per package dose ranges in the range of 200 to 300 person-mrem per package loaded, while other sites experience higher per package dose rates. For the low-density storage case, each cask loaded in addition to the number required by the regulatory baseline is estimated to result in an incremental 400 person-mrem dose.

There is routine occupational dose associated with ISFSI annual operation and maintenance activities (i.e., inspection, surveillance, and security operations). The regulatory baseline assumes an annual dose of 120 person-mrem per site per year for inspection, surveillance, and security activities and 1,500 person-mrem per site per year for ISFSI operations and maintenance. These estimated radiation doses are consistent with assumptions used by EPRI in EPRI TR-1021049 (Ref. C.36) and TR-1018058 (Ref. C.37). Because additional shielding is assumed to be provided by concrete overpacks, the worker dose associated with ISFSI operations and maintenance is not expected to increase. Therefore, no incremental occupational dose is predicted for performing annual ISFSI operation and maintenance.

There is routine occupational dose associated with the storage of each DSC at an operational ISFSI. The regulatory baseline assumes a worker dose of 170 person-mrem for each additional DSC loaded at an ISFSI site. This estimated radiation dose is consistent with assumptions used by EPRI in EPRI TR-1021049 (Ref. C.36) and TR-1018058 (Ref. C.37). Because additional shielding is assumed to be provided by concrete overpacks, the worker dose associated with each DSC stored at an operational ISFSI is not expected to increase. For the low-density SFP storage case, each cask stored in addition to the number required by the regulatory baseline is estimated to result in an incremental 170 person-mrem dose.

Table 66 summarizes the occupational dose estimates for each activity.

Table 66 Incremental Occupational Dose (Routine) Estimates

Activity	Incremental Occupational Dose (Routine) (person-mrem per activity)
Load a DSC	400
ISFSI Operation and maintenance	0
Loading a DSC at an ISFSI	170
Total	570

C.3.2 Number of Dry Storage Casks

In 2013, the representative Group 1 plant has 3,055 fuel assemblies stored in the SFP in a high-density 1x4 loading configuration. During each refueling outage, 284 assemblies are offloaded from the reactor vessel to the SFP. For the regulatory baseline, the plant is expected to load the required number of DSCs with a 68-assembly capacity each refueling outage to retain sufficient space in the SFP to discharge one full core of fuel (full core reserve). The estimated DSC inventory is shown in Figure 16.

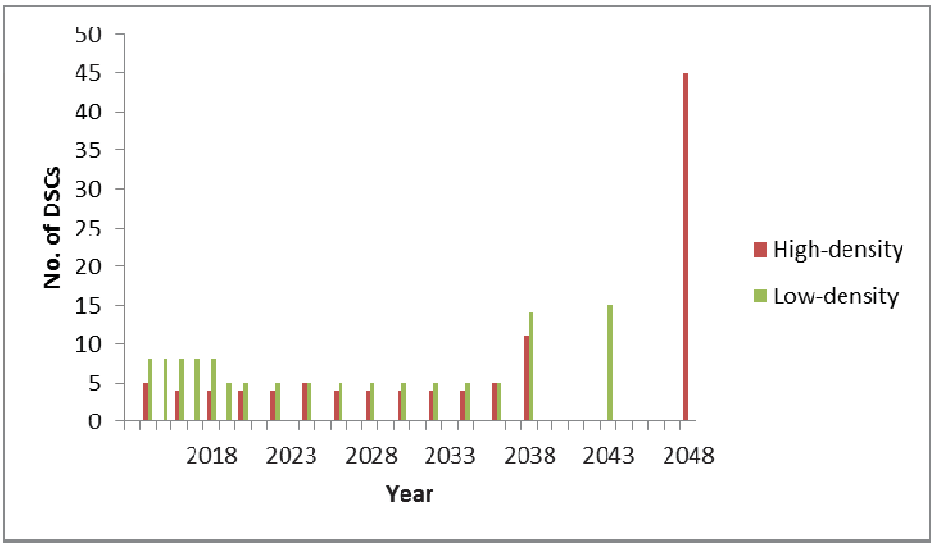

Figure 16 Timing of dry storage cask loading for the representative Group 1 plant

At the expiration of the operating license in 2038, the full core is offloaded into the SFP. The analysis further assumes that the entire SFP inventory will be placed into dry storage by 2048, 10 years after termination of unit commercial operation.

For the low-density SFP storage case, it is assumed that there is an NRC policy decision that requires licensees to offload the spent fuel inventory to dry storage to obtain a low-density configuration within 5 years (e.g., by end of 2019). In this configuration, the representative Group 1 plant SFP stores 852 assemblies, which is equivalent to the discharge from the last three refueling outages. Using the same initial conditions as above, and using the DSC with a 57-assembly derated capacity beginning in year 2020, the inventory model is provided as the low-density chart in Figure 16.

At the expiration of the operating license in 2034, the full core is offloaded into the SFP. The analysis further assumes that the entire SFP inventory will be placed into dry storage by 2048. Additionally, in year 2048, the spent fuel has cooled for a sufficient length of time that the DSC is no longer derated.

Similar calculations were performed for Groups 2, 3 and 4 using the Holtec Hi-Storm FW DSC system for PWR spent fuel. The dry storage cask loading for the representative Group 2 plant is shown in Figure 17.

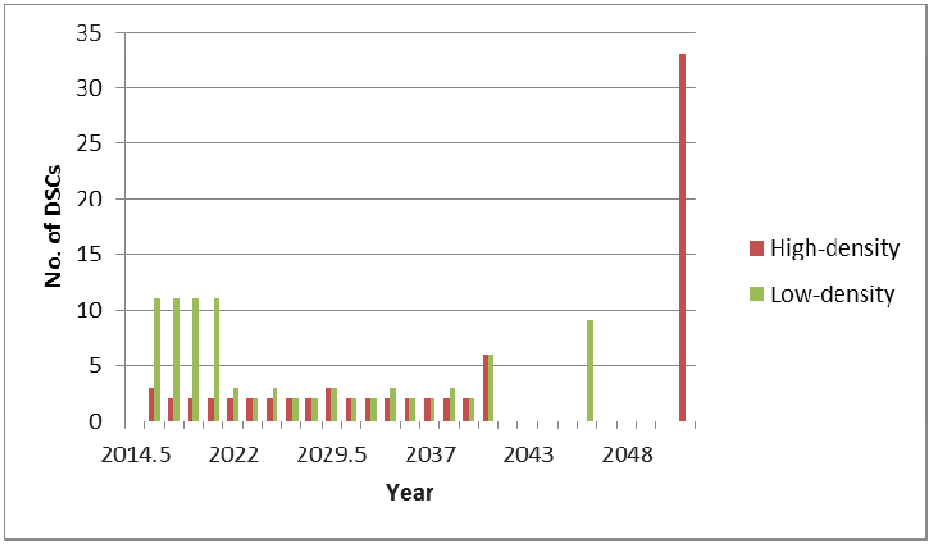

Figure 17 Timing of dry storage cask loading for the representative Group 2 plant

In 2018, the representative Group 3 plant is assumed to begin commercial operation. At this time, there is no spent fuel assemblies stored in the SFP. The unit is assumed to operate on an 18 month refueling cycle, discharging an estimated 69 assemblies per cycle (Ref. C.4, Section 9.1). For the regulatory baseline, the representative new nuclear plant is expected to begin dry storage in 2038 and will load a sufficient number of Holtec Hi-Storm FW casks to maintain its full core offload capability. The estimated timing for DSC loading is shown in Figure 18.

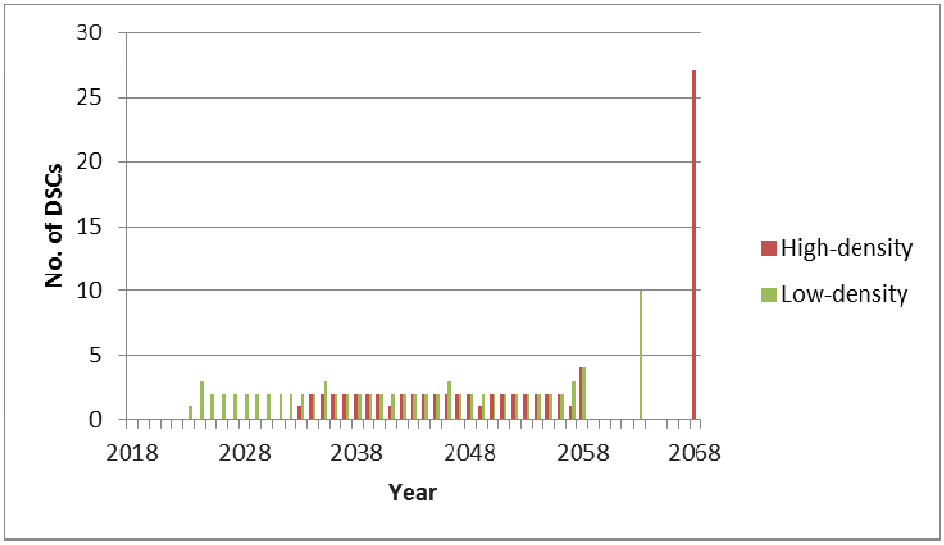

Figure 18 Timing of dry storage cask loading for the representative Group 3 plant

The representative Group 4 SFP which is shared between two PWR units is assumed to have 1,637 fuel assemblies stored in the SFP in a high-density 1x4 loading configuration. Each reactor unit operates on a 24-month refueling cycle and discharges 84 assemblies on a 1-year staggered cycle. The representative shared SFP has already implemented dry storage.

For the regulatory baseline, the Group 4 SFP is expected to load the required number of DSCs with a 37-assembly capacity each refueling outage to retain sufficient space in the SFP to discharge one full core of fuel (full core reserve). For the low-density case, the DSC has a 33-assembly capacity because of the higher heat load of the spent fuel. At the expiration of the operating license in 2038, the full core is offloaded into the SFP. The analysis further assumes that the entire SFP inventory will be placed into dry storage beginning in 2038 and completed by 2048. The estimated timing for DSC loading for the representative Group 4 SFP is shown in Figure 19.

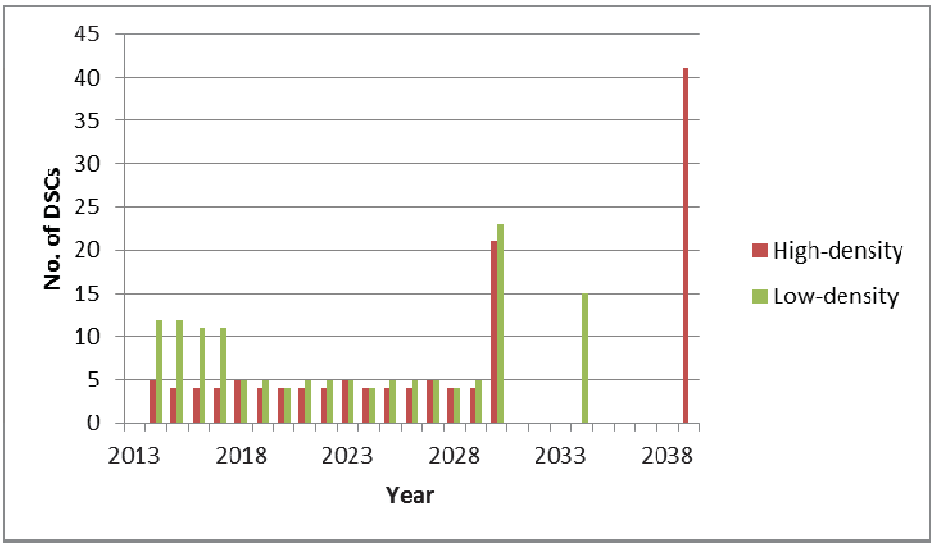

Figure 19 Timing of dry storage cask loading for the representative Group 4 plant

C.4 Cost Assumptions

C.4.1 Generic Costs

Costs presented in this analysis are based on estimates by the author or cited documents. This is a generic cost estimate and should be used accordingly. Site-specific features may result in higher or lower costs than those estimated.

C.4.2 Dry Storage Upfront Costs

Upfront costs include engineering, design, and licensing costs; equipment costs; construction costs; and start up and testing costs. Each of these cost components are further described in EPRI TR-1021048, “Industry Spent Fuel Storage Handbook” (Ref. C.38). As noted in EPRI TR-1025206, “Impacts Associated with Transfer of Spent Nuclear Fuel from Spent Fuel Storage Pools to Dry Storage after Five Years of Cooling, Revision 1” (Ref. C.5), the independent spent fuel storage installation (ISFSI) upfront costs vary widely from site to site and the upfront costs for those in operation vary from several million to tens of millions of dollars (Ref. C.5, p. 2-23). Values for upfront costs were estimated based on two publically available cost estimates that identified the specified number of DSC to be stored. The estimate amortized upfront costs for each site is provided in Table 67.

Table 67 Amortized DSC Upfront Costs

ISFSI Facility	Upfront Cost Estimate (base year)	Upfront Cost Estimate (2012 $)	DSC Storage Capacity	Attributed Upfront Cost per DSC (2012 $)
Monticello	$21.5 million (2005 $)	$25,275,400	30	$842,500
Pilgrim	$22 million (2006 $)	$25,055,800	53	$472,800
Average (Best Estimate)		$25,165,600		$657,700

C.4.3 Incremental Costs Associated with Earlier DSC Purchase and Loading

Incremental costs are the costs associated with the purchase and loading of DSCs on a periodic basis. These costs include the capital costs for the DSC and the loading costs for the storage systems. The unit cost estimates used in this analysis are provided in Table 68. These cost estimates are based on the DSC unit costs that EPRI used for a generic interim storage facility (Ref. C.39) and documented in EPRI TR-1025206 (Ref. C.5). Nuclear power plant licensees may experience incremental DSC purchase and loading costs that are higher or lower than the amount assumed in this cost-benefit analysis.

Table 68 Incremental Unit Cost Estimates

Item	Base Case Unit Cost (Constant $2012)	Adders to load 5-year cooled fuel (Constant $2012)	5-Year cooled fuel Unit Cost (Constant $2012)
Canister	$780,000	$62,400(1)	$842,400
Concrete overpack	$208,000	$41,600(2)	$249,600
Loading of canister-based storage	$312,000	$62,400	$374,400
Total	$1,300,000		$1,466,400

1. The canister cost adder is the product of $780,000 x 40% x 20%.
2. The concrete overpack adder is the sum of the labor adder and the concrete shielding adder (e.g., $208,000 x 40% x 20% + $208,000 x 30% x 40%).

When only five-year cooled, high burnup spent fuel is available for loading into dry storage, there are several potential cost adders to address increased fabrication costs, additional shielding capability in concrete storage overpacks; and higher loading costs because of increased worker dose and work rules that result in longer cask loading durations or the need to utilize additional crews.

Labor costs are approximately 40 percent of the cost of DSCs (Ref. C.5). Assuming that the labor portion of canister and concrete overpack cost increase by 20 percent, this results in a fabrication cost adder of $79,040 per DSC (e.g., 40 percent x $988,000 x 20 percent). This fabrication adder is applied to dry storage incremental costs when five-year cooled inventories are transferred to dry storage.

Concrete shielding costs are approximately 30 percent of the concrete overpack cost (Ref. C.5). Assuming that shielding costs increase by 40 percent, these results in a concrete overpack shielding cost adder of $24,960 per overpack ($208,000 x 30 percent x 40 percent). This shielding adder is applied to dry storage incremental costs when 5-year cooled inventories are transferred to dry storage.

There may be other additional costs associated with amending existing certificates of compliance (CoCs), certifying new designs, or may result from high demand for DSCs in short supply. These costs may be passed on to nuclear plant operators through the price of the DSC systems or may be directly billed to nuclear plant operators if the amended or new designs are specific only to that ISFSI. These additional costs were not estimated given the possibility for a wide range of costs for implementing CoC changes and the possible price swings, which could occur for DSCs if there is limited supply.

Because of the increased costs associated with increased worker dose, longer loading times to comply with work rules, and the need to load more DSCs, and the application of fatigue rules during cask loading operations, the NRC estimates that DSC loading costs increase by 20 percent. This loading cost adder of $62,400 per DSC (e.g., 20 percent times $312,000) is applied when 5-year cooled spent fuel assemblies are loaded into dry storage casks.

C.4.4 Incremental Annual ISFSI Operating Costs

Annual operating costs for an ISFSI during reactor operation include the costs associated with NRC inspections; security; radiation monitoring; ISFSI operational monitoring; technical specification and regulatory compliance, including implementation of new CoC amendments; personnel cost and code maintenance associated with fuel selection for dry storage; personnel costs for spent fuel management and fabrication surveillance activities; electric power usage for lighting and security systems; road maintenance to the ISFSI site; and miscellaneous expenses associated with ISFSI maintenance. NRC license fees for dry storage are included as part of the 10 CFR Part 50 (“Domestic Licensing of Production and Utilization Facilities”) operating license fees and, therefore, are not an incremental cost.

Because most operating nuclear power plants have already implemented dry storage, no incremental annual ISFSI operating costs to implement dry storage at an earlier date is estimated for Group 1, 2, or 4 SFP sites if a policy decision is made to accelerate the transfer of spent fuel stored in SFPs to dry storage.

For the Group 3 SFPs for which the associated reactor is not expected to begin commercial operation until 2018, the NRC estimates that the site would begin transferring fuel to dry storage in 2040. For the expedited transfer alternative, it is expected that the unit would begin transferring fuel to dry storage in 2025 and, therefore, Group 3 sites would incur incremental annual ISFSI operating cost for the earlier ISFSI operating period from 2025 to 2040. EPRI reports a wide variability in published estimates of annual ISFSI operating costs that range from $212,000 to $2 million per year in 2012 dollars and reported their estimate of $1.1 million per year for an ISFSI at an operating nuclear power plant site (Ref. C.5, p. 2–28). This estimate provided in Table 69 is used as the incremental annual Group 3 ISFSI operating cost in this analysis. ISFSIs located at nuclear operating plant sites may experience annual ISFSI operating costs that are higher or lower than this estimated value.

Table 69 Incremental ISFSI Annual Operating Costs

SFP Group	Activity	Incremental ISFSI Annual Operating Cost (2012 dollars)
All	ISFSI operation and maintenance costs	Negligible
3	Early ISFSI operation and maintenance costs	$1,100,00

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APPENDIX D: SENSITIVITY ANALYSIS INFORMATION

D.1 Present Value Calculations

The choice of a discount rate, over long periods of time, raises questions of science, economics, philosophy, and law. Although the discount rate has a large influence on the current value of future damages, there is no consensus about what rates to use in this context.

The NRC traditionally uses constant discount rates of 7 percent for regulatory decisionmaking and 3 percent as a sensitivity value to reflect reliance on a social rate of time preference discounting concept in accordance with OMB Circular A-4. As Circular A-4 acknowledges, however, the choice of discount rate for intergenerational problems raises distinctive problems and presents considerable challenges. After reviewing those challenges, Circular A-4 states, "If your rule will have important intergenerational benefits or costs you might consider a further sensitivity analysis using a lower but positive discount rate in addition to calculating net benefits using discount rates of 3 and 7 percent."

The 3 percent rate is consistent with estimates provided in the economics literature and approximates the real rate of return on long-term government debt which serves as a proxy for the real rate of return on savings. A low discount rate value of 2.0 percent is included, which represents the lower bound for the certainty-equivalency rate in 100 years using the random walk model approach (Ref. D.1) to address the concern that interest rates are highly uncertain over time.

D.2 Dollar per Person-Rem Conversion Factor

The NRC is currently revising the dollar per person-rem averted conversion factor based on recent information regarding the value of a statistical life (VSL). However, until the NRC completes the update and publishes the appropriate guidance documents, the NRC will perform sensitivity analysis to estimate the impact on the calculated results when more current VSL and cancer risk factor are used. The NRC used the EPA's VSL as an interim value in the sensitivity analysis. The EPA's VSL was developed through a rigorous process, reviewing many published academic papers, and includes review from the Scientific Advisory Board, an independent review board.

The EPA's VSL in 2009 dollars is approximately $7.2 million (Ref. D.2, p. 41). The VSL is derived from "using a mixed effects model (random intercept with fixed effects for study characteristics), the authors regressed the VSL estimates on average income, probability of death, and several study design variables" (Ref. D.2, p. 41). Therefore, using the CPI-U based inflator to adjust from 2009 dollars to 2012 dollars yields a VSL of approximately $7.7 million. The International Commission on Radiation Protection (ICRP) updated the mortality risk factor in ICRP Publication No. 103 (Ref. D.3); the updated risk coefficient is 5×10^{-4}. Using the updated ICRP risk coefficient and escalated EPA-based VSL, the dollar per person-rem conversion, rounded to the nearest thousand, is $4,000 per person-rem.

The staff is aware that the $2,000 per person-rem conversion factor may change as a result of ongoing assessments. However, the value of the dollar per person-rem conversion factor is a matter of Commission policy. Therefore, the NRC used the $2,000 per person-rem conversion value for the recommendation and the $4,000 per person-rem conversion value as a sensitivity study for this analysis.

D.3 Replacement Energy Costs

The NRC is currently updating its estimates for replacement energy costs based on a U.S. competitive electricity market area model. The updated model provides the replacement energy costs by day, week, and year, based on market area, in 2010 dollars. For each U.S. power market area, a lowest cost and highest cost replacement energy cost estimate was calculated, normalizing for reactor megawatt rating differences. The estimated replacement energy cost per reactor per year ranges from a high estimate of $54.4 million to a low estimate of $692,000 across all U.S. power markets. The average estimated cost per reactor per year across all U.S. power markets is $9.6 million and the median estimated cost is $6.4 million in 2010 dollars. Using the CPI-U inflator formula and the 2010 CPI-U inflator value from Table 31, the estimated replacement energy costs range from $57.3 million to $729,000 in 2012 dollars. The average estimated cost per reactor per year across all U.S. power markets is $10.1 million and the median estimated cost is $6.7 million in 2012 dollars.

D.4 Consequences Extending Beyond 50 Miles

NUREG/BR-0184 states that in the case of nuclear power plants, changes in public health and safety from radiation exposure and offsite property impacts should be examined over a 50-mile (80-kilometer) distance from the plant site. However, in this circumstance it is beneficial for the analysis to include supplemental information (e.g., analyses and results) that go beyond the guidance provided in this document. The SFPS uses a plume release model that predicts slow deposition of aerosols containing long-lived (i.e., slowly decaying) isotopes that results in public radiation exposures beyond 50 miles from the postulated accident site. While the accuracy of the model decreases with distance, this cost-benefit analysis evaluates the public health and safety and economic consequences estimated by the plume model beyond the 50-mile distance from the plant site as a sensitivity analysis. Refer to section 4.4.1.4 for results of these sensitivity analyses.

D.5 Sensitivity to a Uniform Fuel Pattern during an Outage

The base case of this analysis assumes that the licensee has prearranged the SFP such that discharged assemblies can be placed directly into a 1x4 arrangement for the discharges of the last two outages. This approach is consistent with the requirements discussed in Section 9.3 of the SFPS. However, those requirements do allow for the fuel to be stored in a less favorable configuration for some time following discharge if other considerations prevent prearrangement. A requirement is associated with the time window by which the 1x4 arrangement must be achieved; however, the specific time requirement is not publicly available information. To capture the effects of nonbeneficial arrangement of discharged fuel, this analysis evaluates the situation in which the discharged spent fuel is uniformly arranged during the outage to evaluate the effect of this aspect on the results. Refer to Appendix C, section C.2.15 for results of this sensitivity analysis.

D.6 References

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APPENDIX E: INDUSTRY IMPLEMENTATION MODEL OF MOVING SPENT FUEL TO DRY CASK STORAGE

E.1 Group 1 Spent Fuel Pool

As previously discussed in Appendix section C.4.3, during each refueling outage the representative Group 1 plant discharges 284 fuel assemblies to the SFP. For the regulatory baseline case, the plant is expected to load the required number of DSCs with a 68-assembly capacity each refueling outage to retain sufficient space in the SFP to discharge one full core of fuel (full core reserve). For the expedited transfer alternative, low-density SFP storage case, the representative Group 1 plant SFP stores 852 assemblies, which is equivalent to the discharge from the last three refueling outages. For the expedited transfer alternative, the plant achieves this low-density storage condition within five years and then maintains this storage condition up through cessation of commercial operation. The cumulative DSC implementation costs for a single Group 1 SFP are shown in Figure 20.

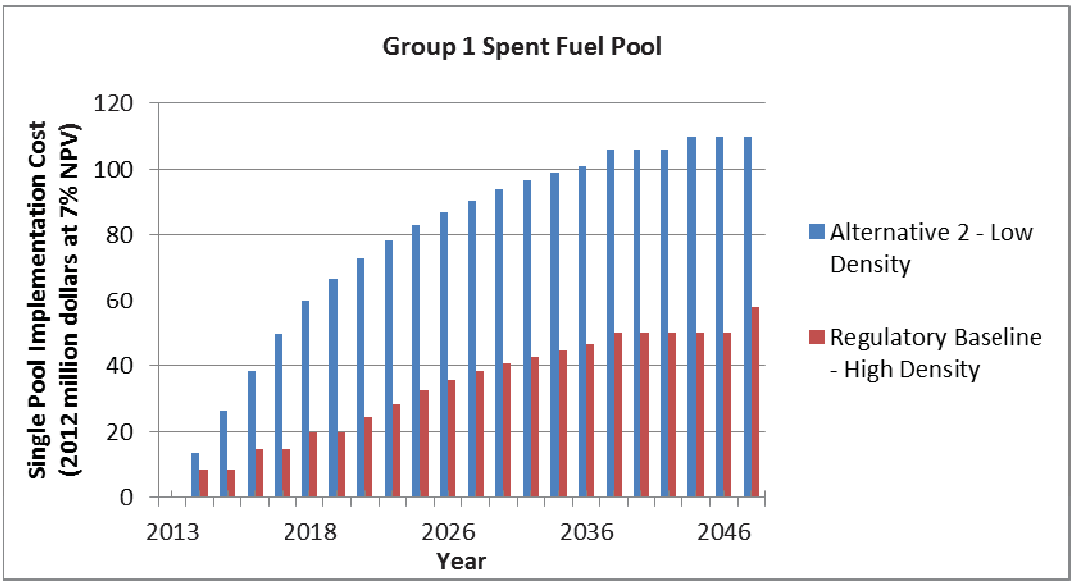

Figure 20 Cumulative dry cask storage implementation costs for a single Group 1 spent fuel pool

E.2 Group 2 Spent Fuel Pool

A similar calculation is performed for the Groups 2 SFPs. As previously discussed in Appendix section C.4.3, every 18-months the representative PWR plant discharges 84 fuel assemblies to the SFP. For the regulatory baseline case, the plant is expected to load the required number of Holtec Hi-Storm FW DSCs with a 37-assembly capacity each refueling outage to retain sufficient space in the SFP to discharge one full core of fuel (full core reserve). For the expedited transfer alternative, low-density SFP storage case, the representative plant SFP stores 312 fuel assemblies, the equivalent to the discharge from the last three refueling outages. The cumulative DSC implementation costs for Group 2 plants are shown in Figure 21.

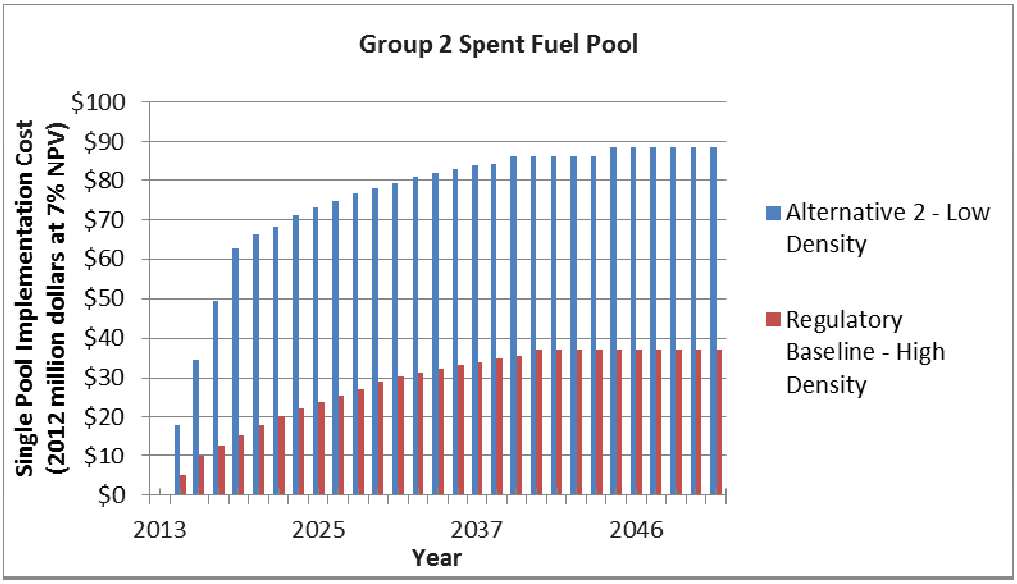

Figure 21 Cumulative dry cask storage implementation costs for a single Group 2 spent fuel pool

E.3 Group 3 Spent Fuel Pool

In 2018, the representative Group 3 plant is assumed to begin commercial operation. At this time, there are no spent fuel assemblies stored in the SFP. The unit is assumed to operate on an 18-month refueling cycle, discharging an estimated 69 assemblies per cycle as discussed in Appendix section C.4.3. For the regulatory baseline, the representative new nuclear plant is expected to begin dry storage in 2038 and will load a sufficient number of Holtec Hi-Storm FW casks to maintain its full core offload capability. The cumulative DSC implementation costs for Group 3 plants are shown in Figure 22.

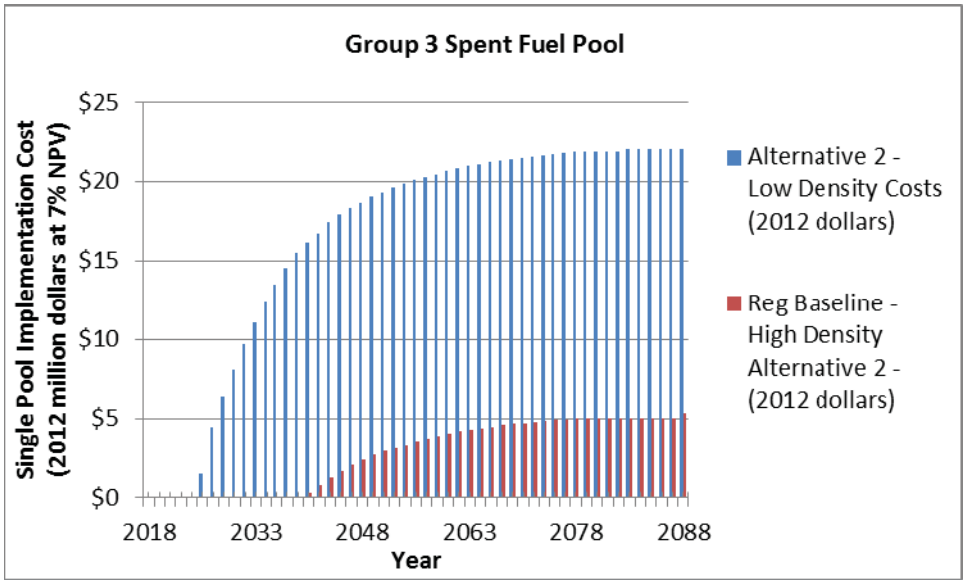

Figure 22 Cumulative dry cask storage implementation costs for a single Group 3 spent fuel pool

E.4 Group 4 Spent Fuel Pool

The representative Group 4 SFP is shared between two PWR units and is assumed to have 1,637 fuel assemblies stored in the SFP in a high-density 1x4 loading configuration. Each reactor unit operates on an 18-month refueling cycle and discharges 84 assemblies during the shoulder months from May through June and September into early November during the same calendar year. For the regulatory baseline, the Group 4 SFP is expected to load the required number of DSCs with a 37-assembly capacity each refueling outage to retain sufficient space in the SFP to discharge one full core of fuel (full core reserve). For the low-density case, the DSC has a 33-assembly capacity because of the higher heat load of the spent fuel. At the cessation of commercial operation, which occurs on average in 2038 for the Group 4 SFP reactors, the full core is offloaded into the SFP. The analysis further assumes that the entire SFP inventory will be placed into dry storage by 2048 for the regulatory baseline and by 2043 for the low-density storage case. The cumulative DSC implementation costs for Group 4 plants are shown in Figure 23.

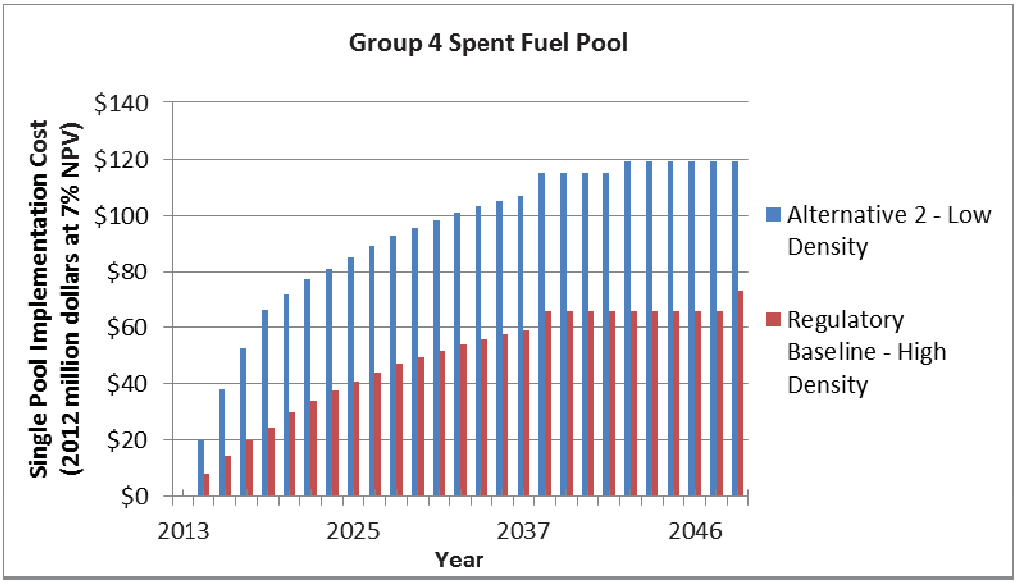

Figure 23 Cumulative dry cask storage implementation costs for a single Group 4 shared spent fuel pool

APPENDIX F: SPENT FUEL DATA AND TABLES

Table 70 Dry Spent Fuel Storage at U.S. Commercial Nuclear Power Plants

Plant Name	Company Name	Fuel Type	Location	License Type	Storage Technology	Year Loaded
Surry 1 & 2	Dominion Generation	PWR	Co-located	Site-specific	CASTOR V/21 MC-10, NAC I-28 CASTOR X/, TN-32	1986
				General	NUHOMS-32PTH	2007
H.B. Robinson	Progress Energy	PWR	Co-located	Site-specific	NUHOMS-07P	1989
				General	NUHOMS-24PTH	2004
Oconee 1, 2, 3	Duke Energy	PWR	Co-located	Site-specific	NUHOMS-24P	1990
				General	NUHOMS-24P NUHOMS-24PHB	2000
Fort St. Vrain (shutdown)	U.S. DOE (Previously owned by Public Service Colorado)	HTGR	–	Site-specific	Foster Wheeler MVDS	1991
Calvert Cliffs 1 & 2	Constellation Energy	PWR	Co-located	Site-specific	NUHOMS-24P NUHOMS-32P	1992
Palisades	Entergy Nuclear Operations	PWR	Co-located	General	VSC-24 NUHOMS-32PT NUHOMS-24PTH	1993
Prairie Island 1 & 2	Xcel Energy	PWR	Co-located	Site specific	TN-40	1993
Point Beach 1 & 2	FPL Energy Point Beach	PWR	Co-located	General	VSC-24 NUHOMS-32PT	1995
Davis Besse	FirstEnergy Nuclear Operating Co.	PWR	Co-located	General	NUHOMS-24P	1995
Arkansas Nuclear One 1 & 2	Entergy Nuclear Operations	PWR	Co-located	General	VSC-24 HI-STORM 24P HI-STORM 32P	1996
North Anna 1 & 2	Dominion Generation	PWR	Co-located	Site-specific	TN-32	1998
				General	NUHOMS-32PTH	2008
Susquehanna 1 & 2	PPL Susquehanna LLC	BWR	Co-located	General	NUHOMS-52B NUHOMS-61BT	1999
Peach Bottom 2 & 3	Exelon Generation	BWR	Co-located	General	TN-68	2000
Dresden 1, 2, 3 (Unit 1 – shutdown)	Exelon Generation	BWR	Co-located	General	HI-STAR 68B HI-STORM 68B	2000
Hatch 1 & 2	Southern Nuclear Operating Co.	BWR	Co-located	General	HI-STAR 68B HI-STORM 68B	2000
Rancho Seco (shutdown)	Sacramento Municipal Utility District	PWR	–	Site-specific	NUHOMS-24P	2001
McGuire 1 & 2	Duke Energy	PWR	Co-located	General	TN-32 NAC UMS	2001
Trojan (shutdown)	Portland General Electric	PWR	–	Site-specific	TranStor Overpack HI-STORM 24P MPC	2002
Oyster Creek	Exelon	BWR	Co-located	General	NUHOMS-61BT	2002

Plant Name	Company Name	Fuel Type	Location	License Type	Storage Technology	Year Loaded
	Generation					
Yankee Rowe (shutdown)	Yankee Atomic Electric Co.	PWR	Stand Alone	General	NAC MPC	2002
Columbia	Energy Northwest	BWR	–	General	HI-STORM 68B	2002
Big Rock Point (shutdown)	Entergy Nuclear Operations	BWR	Stand Alone	General	FuelSolutions W150	2002
FitzPatrick	Entergy Nuclear Operations	BWR	Co-located	General	HI-STORM 68B	2002
Maine Yankee (shutdown)	Maine Yankee Atomic Power	PWR	Stand Alone	General	NAC UMS	2002
Palo Verde 1, 2, 3	Arizona Public Service	PWR	–	General	NAC UMS	2003
San Onofre 1, 2, 3 (Unit 1 – shutdown)	Southern California Edison	PWR	–	General	NUHOMS-24PT	2003
Duane Arnold	FPL Energy.	BWR	Co-located	General	NUHOMS 61BT	2003
Haddam Neck (shutdown)	Connecticut Light & Power	PWR	–	General	NAC MPC	2004
Sequoyah 1 & 2	Tennessee Valley Authority	PWR	Co-located	General	HI-STORM 32P	2004
Millstone 1, 2, 3 (Unit 1 – shutdown)	Dominion Generation	Unit 1 – BWR Unit 2, 3 – PWR	Co-located	General	NUHOMS-32PT	2005
Farley 1 & 2	Southern Nuclear Operating Co.	PWR	Co-located	General	HI-STORM 32P	2005
Browns Ferry 1, 2, 3	Tennessee Valley Authority	BWR	Co-located	General	HI-STORM 68B	2005
Quad Cities 1 & 2	Exelon Generation	BWR	Co-located	General	HI-STORM 68B	2005
River Bend	Entergy Nuclear Operations	BWR	Co-located	General	HI-STORM 68B	2005
Fort Calhoun	Omaha Public Power District	PWR	Co-located	General	NUHOMS-32PT	2006
Hope Creek	PSEG Nuclear	BWR	Co-located	General	HI-STORM 68B	2006
Grand Gulf	Entergy Nuclear Operations	BWR	Co-located	General	HI-STORM 68B	2006
Catawba 1 & 2	Duke Energy	PWR	Co-located	General	NAC UMS	2007
Indian Point 1, 2, 3 (Unit 1 – shutdown)	Entergy Nuclear Operations	PWR	Co-located	General	HI-STORM 32P	2008
Vermont Yankee	Entergy Nuclear Operations	BWR	Co-located	General	HI-STORM 68B	2008
Limerick 1 & 2	Exelon Generation	BWR	Co-located	General	NUHOMS 61BT	2008
St. Lucie 1 & 2	FPL Energy	PWR	Co-located	General	NUHOMS 32PT	2008
Seabrook	FPL Energy	PWR	Co-located	General	NUHOMS 32PT	2008

Plant Name	Company Name	Fuel Type	Location	License Type	Storage Technology	Year Loaded
Monticello	Xcel Energy	BWR	Co-located	General	NUHOMS 61BT	2008
Humboldt Bay (shutdown)	Pacific Gas & Electric	BWR	Co-located	Site-specific	HI-STAR 100	2008
Kewaunee	Dominion Generation	PWR	Co-located	General	NUHOMS-32P	2009
Diablo Canyon 1 & 2	Pacific Gas & Electric	PWR	–	Site-specific	HI-STORM 32P	2009

Source: EPRI TR-1021048, pp. 2-10 to 2-12 (Ref. F.1).

Table 71 Expected Dry Spent Fuel Storage Facility Development at U.S. Commercial Nuclear Power Plants

Plant Name	Company Name	Location	Fuel Type	Approximate Loading Year
Beaver Valley 1	FirstEnergy Nuclear Operating Co.	–	PWR	2013-2014
Brunswick 1 & 2	Progress Energy	Co-located	BWR	2010-2011
Braidwood 1 & 2	Exelon Generation	–	PWR	2011
Byron 1 & 2	Exelon Generation	Co-located	PWR	2010
Clinton	Exelon Generation	–	BWR	2016
Comanche Peak	TXU Generating Company	–	PWR	2014-2016
Cook 1 & 2	Indiana Michigan Power	–	PWR	2011
Cooper	Nebraska Public Power District	Co-located	BWR	2010
Crystal River	Progress Energy	–	PWR	2012
Fermi	Detroit Edison	Co-located	BWR	2010
Ginna	Constellation Energy	Co-located	PWR	2010
LaCrosse (shutdown)	Dairyland Power	–	BWR	2011
LaSalle 1 & 2	Exelon Generation	Co-located	BWR	2010
Nine Mile Point 1 & 2	Constellation Energy	–	BWR	2012
Perry	FirstEnergy	Co-located	BWR	2010
Pilgrim	Entergy Nuclear Operations	–	BWR	2014-2015
Salem 1 & 2	PSEG Nuclear	Co-located	PWR	2010
Summer	South Carolina Electric & Gas	–	PWR	2015-2017
Turkey Point 3 & 4	FPL Energy	–	PWR	2011
Vogtle 1 & 2	Southern Nuclear Operating Co.	–	PWR	2013-2014
Waterford 3	Entergy Nuclear Operations	–	PWR	2011-2012
Watts Bar 1 & 2	Tennessee Valley Authority	–	PWR	2020

Source: EPRI TR-1021048, p. 2-13 (Ref. F.1).

Table 72 Spent Fuel Pool Capacities

Plant Name	Spent Fuel Pool			
	Group[1]	Assoc. Reactor Core Size (no. of assemblies)	Technical Specification Capacity (assemblies/core equivalents)	Estimated Cs-137 Inventory (MCi)
Arkansas Nuclear 1	2	177	968/ 5.5	41.7
Arkansas Nuclear 2	2	177	988/ 5.6	42.8
Beaver Valley 1	2	157	1627/ 10.4	77.6
Beaver Valley 2	2	157	1627/ 10.4	77.6
Braidwood 1	4	193	2984/ 7.7 per unit[2]	142.2
Braidwood 2	4	193		
Browns Ferry 1	1	764	3471/ 4.5[4]	52.3
Browns Ferry 2	1	764	3471/ 4.5[4]	52.3
Browns Ferry 3	1	764	3471/ 4.5	52.3
Brunswick 1	1	560	1803/ 3.2	24.0
Brunswick 2	1	560	1839/ 3.3	24.7
Byron 1	4	193	2984/ 7.7 per unit[2]	142.2
Byron 2	4	193		
Callaway	2	193	2363/ 12.2	114.5
Calvert Cliffs 1	4	217	1830/ 4.2 per unit[4]	79.4
Calvert Cliffs 2	4	217		
Catawba 1	2	193	1421/ 7.3	64.8
Catawba 2	2	193	1421/ 7.3	64.8
Clinton	2	624	3796/ 6.1	61.3
Columbia	1	764	2658/ 3.5	36.6
Comanche Peak 1	2	193	1684/ 8.7[4]	78.7
Comanche Peak 2	2	193	1689/ 8.7[4]	79.0
Cooper	1	548	2651/ 4.8	40.6
Crystal River 3	6	177	1474/ 8.3	68.5
Davis-Besse	2	177	1624/ 9.2	76.4
D.C. Cook 1	4	193	3613/ 9.3 per unit[2]	175.4
D.C. Cook 2	4	193		
Diablo Canyon 1	2	193	1324/ 6.9	59.7
Diablo Canyon 2	2	193	1324/ 6.9	59.7
Dresden 2	1	724	3537/ 4.9	54.3
Dresden 3	1	724	3537/ 4.9	54.3
Duane Arnold	1	368	2829/ 7.7	47.5
Farley 1	2	157	1407/ 9.0	66.0
Farley 2	2	157	1407/ 9.0	66.0
Fermi 2	1	764	4608/ 6.0	74.2
FitzPatrick	1	560	3239/ 5.8	51.7
Fort Calhoun	2	133	1083/ 8.14	50.1
Ginna	2	121	1321/ 10.9	63.3
Grand Gulf 1	2	800	4348/ 5.4	68.5
Hatch 1	1	560	3349/ 6.0[4]	53.9
Hatch 2	1	560	2933/ 5.2[4]	45.8
Hope Creek 1	1	764	4006/ 5.2	62.6
Indian Point 2	2	193	1374/ 7.1	62.3
Indian Point 3	2	193	1345/ 7.0	60.8
Kewaunee	6	121	1205/ 10.0	57.2

Plant Name	Spent Fuel Pool			
	Group[1]	Assoc. Reactor Core Size (no. of assemblies)	Technical Specification Capacity (assemblies/core equivalents)	Estimated Cs-137 Inventory (MCi)
La Salle County 1	1	764	3986/ 5.2[4]	62.2
La Salle County 2	1	764	4078/ 5.3[4]	64.0
Limerick 1	1	764	4117/ 5.4	64.8
Limerick 2	1	764	4117/ 5.4	64.8
McGuire 1	2	193	1463/ 7.6	67.0
McGuire 2	2	193	1463/ 7.6	67.0
Millstone 1	6	–	–	–
Millstone 2	2	217	1346/ 6.2	59.6
Millstone 3	2	193	1860/ 9.6	88.0
Monticello	1	484	2301/ 4.75	35.1
Nine Mile Point 1	1	532	4086/ 7.7	68.6
Nine Mile Point 2	1	764	4049/ 5.3	63.4
North Anna 1	4	157	1737/ 5.5 per unit[2]	79.2
North Anna 2	4	157		
Oconee 1	4	177	1312/ 3.7 per unit[2]	55.2
Oconee 2	4	177		
Oconee 3	2	177	825/ 4.7	34.2
Oyster Creek	1	560	3035/ 5.4	47.8
Palisades	2	204	892/ 4.4	36.3
Palo Verde 1	2	241	1329/ 5.5	57.4
Palo Verde 2	2	241	1329/ 5.5	57.4
Palo Verde 3	2	241	1329/ 5.5	57.4
Peach Bottom 2	1	764	3819/ 5.0	59.0
Peach Bottom 3	1	764	3819/ 5.0	59.0
Perry 1	2	748	4020/ 5.4	63.2
Pilgrim 1	1	580	3859/ 6.7	63.3
Point Beach 1	4	121	1502/ 6.2 per unit[2]	69.7
Point Beach 2	4	121		
Prairie Island 1	4	121	1386/ 5.7 per unit[2]	63.6
Prairie Island 2	4	121		
Quad Cities 1	1	724	3657/ 5.1[4]	56.6
Quad Cities 2	1	724	3897/ 5.4[4]	61.3
River Bend 1	2	624	3104/ 5.0	47.9
Robinson 2	2	157	544/ 3.5	20.4
St. Lucie 1	2	217	1706/ 7.9	78.6
St. Lucie 2	2	217	1491/ 6.9	67.2
Salem 1	2	193	1632/ 8.5	75.9
Salem 2	2	193	1632/ 8.5	75.9
San Onofre 2	6	217	1542/ 7.1	69.9
San Onofre 3	6	217	1542/ 7.1	69.9
Seabrook 1	2	193	1236/ 6.4	55.0
Sequoyah 1	4	193	2091/ 5.4 per unit[2]	95.1
Sequoyah 2	4	193		
Shearon Harris 1	2	157 (PWR)	PWR fuel: 3404 / 21.7 or	167.2
		560 (BWR)	BWR fuel: 4628 / 8.3	73.2
South Texas Project 1	2	193	1969/ 10.2	95.6

Plant Name	Spent Fuel Pool			
	Group[1]	Assoc. Reactor Core Size (no. of assemblies)	Technical Specification Capacity (assemblies/core equivalents)	Estimated Cs-137 Inventory (MCi)
South Texas Project 2	2	193	1969/ 10.2	95.6
Summer 1	2	157	1276/ 8.1	59.1
Summer 2	3	–	–	–
Summer 3	3	–	–	–
Surry 1	4	157	1044/ 3.3 per unit[2]	42.7
Surry 2	4	157		
Susquehanna 1	1	764	2840/3.7[4]	40.1
Susquehanna 2	1	764	2840/ 3.7[4]	40.1
Three Mile Island 1	2	177	1338/ 7.6	61.3
Turkey Point 3	2	157	1395/ 8.9	65.3
Turkey Point 4	2	157	1389/ 8.9	65.0
Vermont Yankee	1	368	3355/ 9.1	57.7
Vogtle 1	2	193	1476/ 7.6[4]	67.7
Vogtle 2	2	193	2098/ 10.9[4]	100.5
Vogtle 3	3	–	–	–
Vogtle 4	3	–	–	–
Waterford 3	2	217	2398/ 11.0	115.1
Watts Bar 1	2	193	1610/ 8.3	74.8
Wolf Creek 1	2	193	2363/ 12.2	114.5
Zion 1	6	–	–	–
Zion 2				

Notes:
1. The Group column corresponds to the SFP groupings discussed in Section 4.1.1.
2. Common pool shared by two reactors. Shared SFPs are required to maintain one full core reserve. However, with the practice that both reactors refuel during the shoulder months of the same year it was judged that shared pools attempt to maintain at least a 1.5 full core reserve in practice.
3. Shearon Harris SFP holds fuel from Robinson and Brunswick.
4. SFPs connected by transfer canal.

Table 73 Cost-Benefit Analysis Inputs Summary

Parameter	Spent Fuel Pool Group 1			Spent Fuel Pool Group 2, 3, & 4		
	Low Est.	Base Case	High Est.	Low Est.	Base Case	High Est.
Seismic hazard initiating event frequency (USGS 2008 model) (per year)						
- Seismic bin 3	1.65E-05	1.65E-05	2.24E-05	1.65E-05	1.65E-05	see Table 43
- Seismic bin 4	4.90E-06	4.90E-06	7.09E-06	4.90E-06	4.90E-06	see Table 43
ac power fragility	100% (bounding value)					
Liner fragility						
- Seismic bin 3	10%	10%	100%	2%	5%	25%
- Seismic bin 4	50%	100%	100%	16%	50%	100%
- Cask drop	100% (bounding value)					
Percent of operating cycle natural circulation cooling is insufficient						
- Seismic bin 3	8%	8%	100%	8%	100% (bounding value)	
- Seismic bin 4	30%	100% (bounding value)		30%	100% (bounding value)	
- Cask drop	8%	100% (bounding value)		8%	100% (bounding value)	
- All other initiators	100% (bounding value)					
Cs-137 release fraction						
- Alternative 1	3%	40%	90%	10%	75%	90%
- Alternative 2	0.5%	3%	5%	0.5%	3%	5%
High-density loading spent fuel pool Cs-137 inventory (MCi)						
- SFP Group 1	40.6	52.7	63.3	-	-	-
- SFP Group 2	-	-	-	57.4	67.9	78.2
- SFP Group 3	-	-	-	33.7	44.4	54.2
- SFP Group 4	-	-	-	63.6	101.1	142.2
Low-density loading spent fuel pool Cs-137 inventory (MCi)						
- SFP Group 1	19.8	22	26.4	-	-	-
- SFP Group 2	-	-	-	15.7	17.4	20.9
- SFP Group 3	-	-	-	15.7	17.4	20.9
- SFP Group 4	-	-	-	31.4	34.8	41.8
Population density within 50 miles of site (people/square mile)	169	317	722	169	317	722
Long-term habitability criteria	500 mrem annually	2 rem first year and 500 mrem each year thereafter	2 rem annually	500 mrem annually	2 rem first year and 500 mrem each year thereafter	2 rem annually
Onsite Property: decontamination, repair, & refurbishment	$303 million	$606 million	$1.82 billion	$303 million	$606 million	$1.82 billion
Short-term occupational exposure (accident) (person-rem)	18,070	28,380	48,880	18,070	28,380	48,880
Long-term occupational exposure (accident (person-rem)	4,580	14,000	46,000	4,580	14,000	46,000
Economic data near site	Palisades	Surry	Peach Bottom	Palisades	Surry	Peach Bottom

APPENDIX F REFERENCES

F.1 EPRI TR-1021048, “Industry Spent Fuel Storage Handbook,” dated July 2010.

APPENDIX G: QUESTIONS RAISED BY THE PUBLIC

The NRC staff conducted two public meeting pertaining to this body of work to gain stakeholder input and feedback on the work conducted and the staff's preliminary conclusions pertaining to the issue. This section addresses some of the questions received during those public meetings.

1) Question

The analysis applies to all sites across the U.S. fleet, so how were the variations in seismicity at the sites considered?

Response

The staff used conservative values for several parameters in the cost-benefit analysis to ensure that design, operational and other site variations among the new and operating reactor fleet were encompassed. For example, the probabilities of exceeding a specified peak ground acceleration at the reference plant fall close to the upper end of each SFP group. However, the amount of conservatism used in the other base case parameters overwhelms the slight non-conservatism in the outlying site seismicity parameter. Therefore, the overall results of the base case are conservative for all plants.

To quantify the effect of exceeding the ground motion estimates used in the base case, a high estimate case was created that conservatively selected the site within each SFP group with the highest plant hazard exceedance frequency for peak ground accelerations greater than 0.6g. The sites selected and the seismic initiating event frequencies values used are listed in Table 37.

2) Question

How were the differences in likelihood of successful mitigation across the sites treated?

Response

Operator diagnosis and recovery are important factors considered in the development of the event frequencies for the successful mitigation of accident events. Success is premised on licensees having taken appropriate actions to understand the potential consequences of spent fuel pool accident events and develop appropriate procedures and mitigating strategies to respond and mitigate the consequences. Specific spent fuel pool loss of water inventory mitigation measures are required under Title 10 of the Code of Federal Regulations (10 CFR) 50.54(hh)(2), which were implemented following the September 11, 2001 attacks. Additionally, the post-Fukushima mitigation required by the NRC in Orders EA-12-051 and EA-12-049 and currently being implemented by all U.S. nuclear power plants should serve to further reduce spent fuel pool accident risk by increasing the capability of nuclear power plants to mitigate beyond-design-basis external events further reducing the frequency of a spent fuel pool accident release. This cost-benefit analysis used a conservative approach to mitigation by crediting successful mitigation to the low-density spent fuel pool storage alternative and assuming no successful mitigation for the high-density spent fuel pool storage regulatory baseline. In this manner the staff biased the results to favor the regulatory action of expediting fuel transfer to dry casks and provided margin to address uncertainties associated with other assumptions.

3) Question

Since the event considered is a large seismic event, how would the accident involving the reactor core affect the study results?

Response

Detailed accident progression analysis were performed in SFPS for high density loading cases including sensitivities to hydrogen combustion (Section 9.1 of the SFPS) and concurrent reactor events (Section 9.4 of the SFPS) leading to the failure of the reactor building. These calculations considered uncertainties associated with hydrogen ignition, and formation of debris leading to blockages at the exit of the assemblies and reduced flow area, and led to a range of release fractions. The base case in the regulatory analysis for high-density loading was based on the average release fractions for small leak scenarios (which result in larger releases than medium leaks) including the uncertainties that resulted in high releases because of hydrogen combustion and significant air oxidation.

A concurrent reactor accident would affect the likelihood of successful implementation of mitigating strategies for the pool. The cost-benefit analysis bounds this effect by assuming that operators were unsuccessful in mitigating the ongoing SFP accident for 72 hours for the high-density case (regulatory baseline) and assumed 100% success rate for mitigation strategies for the low-density case. This biases the results in favor of low-density loading.

Although not considered explicitly in the analysis, the NRC recently issued Orders EA-12-051 and EA-12-049 to all U.S. nuclear power plants which should serve to further reduce core damage risk and SFP accident risk by increasing the capability of nuclear power plants to mitigate beyond-design-basis external events. The staff is currently performing a comprehensive site Level 3 PRA for a U.S. PWR as discussed in SECY-11-0089 and the staff will revisit this issue upon its completion.

4) Question

How was debris generated by hydrogen explosions in the SFP considered?

Response

Based on the structural analyses performed in the SFPS for the reference plant reactor building and overhead crane, no significant debris generated by the seismic event is expected to enter the SFP. Although as stated in Table 3 of the SFPS, some debris could be generated and could fall into the pool as a result of hydrogen combustion. However, the occurrence of a hydrogen combustion event in the SFPS denotes that the fuel in the SFP has already become uncovered and is undergoing a fission product release. The reduction in flow area and losses associated with debris generated from a hydrogen combustion resulting from a reactor accident are explicitly considered in Section 9.4 of the SFPS and described in the response to Question 4 above.

5) Question

The probability of a loss of SFP inventory calculated for the reference was used in the cost-benefit analysis. How were differences in SFP liner failure rates, perhaps due to aging, considered?

Response

The detailed structural analyses performed for the reference plant in the Spent Fuel Pool Study predicted that under the seismic load studied, the liner would fail approximately 10% of the time with either a small or medium sized rupture. To account for any variations in liner material properties, the Tier 3 analysis assumed the liner failure values listed in Table 39. These liner fragility values in combination with other assumed failures provide a conservative estimate for the cost-benefit analysis of expedited transfer of spent fuel.

6) Question

Would the results change if open-frame racks were considered as an option?

Response

For the reference plant studied, the BWR fuel assemblies channel boxes would impede crossflow even with open-frame racks. Furthermore, even for the high-density racking, the study showed that without mitigative actions, fuel is estimated to be air-coolable for at least 72 hours for all but roughly 10% of the operating cycle. Based on the insights from the accident progression analyses in the SFPS, within the first few months after the fuel comes out of the reactor, the decay heat in the freshly unloaded spent fuel is high enough to cause a zirconium fire even in the presence of any additional convective cooling once natural circulation is established (see Figures 90 and 93 in the SFPS for the high-density and low-density pool loadings and a moderate leak). Therefore, open frame racks even with channel boxes removed to allow potential crossflow, would not necessarily prevent a radiological release during this time. In the cost-benefit analysis, values from the SFPS were used to model the BWR Mark I and II SFPs. For the other SFP groups, a simplifying assumption was made to account for the concern. In the base case and high estimate case as shown in Table 41, the analysis assumed that the fuel was never coolable under natural convection of air. This approach bounds any effect of considering open-frame racks.

7) Question

High burnup fuel, which has reduced ductility compared to fresh fuel, is being used across the industry. How would the results change if this reduction in fuel clad ductility were considered?

Response

Seismic loads would be relatively small and loading is slow due to spent fuel rack and fuel assemblies design and widespread damage is not expected even considering the mechanical property changes of high burnup spent fuel cladding. Furthermore, high-density spent fuel storage uses freestanding sliding racks that tend to limit the stresses on the racks and spent fuel assemblies immersed in water, even when the seismic loads increase beyond those calculated for design basis seismic events. Adequate clearances are provided between the

racks and pool walls to avoid, with a margin, impacting of the racks during design basis seismic events. Collisions between racks that might result from seismic loads several times greater than the design basis loads would involve low impact speeds that are expected to be several times smaller than impact speeds in design basis transportation and storage accidents (e.g., 30 feet drop). In addition, some of the impact energy of seismic loads on the fuel would be dissipated by small permanent deformation of the rack structures, which would reduce the shock forces transmitted to the spent fuel relative to those in transportation and storage accidents.

The NRC continues to research the mechanical properties of high burnup cladding relevant for normal conditions of transport, including transportation vibration and fatigue failure. NRC is conducting tests to measure the loads required to fail high burnup spent fuel rods under static loading and a wide range of cycling loading conditions. The loading levels and test speeds are more comparable to seismic loading conditions than those in drop tests for storage and transportation accident conditions. Therefore, the conditions of these tests may be more applicable to assessments of safety margins for high burnup spent fuel assemblies stored in SFPs.

8) Question

Were inadvertent criticality scenarios for spent fuel in the pool considered in the analysis?

Response

Yes, inadvertent criticality scenarios were considered but are not expected to significantly affect this analysis based on the following reasons. Design requirements and related safety analyses ensure fuel stored in the SFP will remain safely subcritical under conditions considered as part of the design basis, but rare conditions beyond the design basis may challenge some measures used to control reactivity. To maintain adequate margin to criticality in U.S. SFPs, the safety analyses credit the geometric configuration of the fuel and a combination of other measures that may include fixed neutron poison material (e.g., Boraflex) and limits on fuel reactivity. In addition, the presence of soluble boron in the coolant of pressurized water reactor SFPs may be credited, but the stored fuel must remain subcritical assuming unborated water is present (10 CFR 50.68). Since these measures may be challenged by a beyond design-basis event, the NRC staff cannot rule out the potential for an inadvertent criticality event. However, the NRC staff judges that the potential consequences of a zirconium fire in the SFP and an associated hydrogen deflagration considered in this analysis would not be significantly affected by an inadvertent criticality event. The NRC staff bases this judgment on the following considerations.

While the earthquakes considered in this analysis are beyond what the fuel was designed to withstand, it is not likely that the fuel would experience sufficient damage to cause significant changes in the geometric configuration of the fuel needed to cause inadvertent criticality.

The necessary moderator would tend to shield and contain the effects of a criticality such that it would primarily pose an on-site rather than off-site hazard.

Criticality requires the presence of a moderator and therefore power would not be sustained as the pool lost inventory due to boiling or draindown. Since the power generated by any inadvertent criticality would be far lower than in the reactor, the inadvertent criticality would have

negligible impact on the long-lived fission product isotope inventory. The additional short-lived isotope inventory would not result in any early fatality risk because of the emergency response as modeled precludes such exposure. This is due in part because of the length of time needed before any fission products are released off-site.

Therefore, any off-site release associated with a criticality would be small relative to potential releases from a zirconium fire.

9) Question

How was the more limiting case of partial draindown considered?

Response

The cost-benefit analysis considered partial draindown events for the plants where this damage state was judged to be the more probable damage state such as SFPs located at grade. The effect was bounded by assuming that the fuel was not coolable by natural convection of air for the base case and high estimate case for these SFP groups as shown in Table 41.

10) Question

How does the study treat variations in population density across the sites?

Response

Since population density varies across the sites, the analysis includes a sensitivity study where the value is varied from low to high population density levels as represented by U.S. operating plant locations. Representative operating reactor site demographics were selected to represent the 90th percentile, the mean, the median, and the 20th percentiles.

11) Question

Since the SFP accident primarily occurs in an air oxidizing environment, how was the release of ruthenium accounted for?

Response

The study uses best estimate ruthenium release rates calculated by the MELCOR code. A model is provided to account for the high volatility of the ruthenium oxides when air ingress is assumed to lead to the formation of a moderately hyperstoichiometric fuel. Details of the modeling approach used for ruthenium release is provided in Section 6.1.5 of the SFPS.

12) Question

Plants may eventually use MOX fuel. Will the results apply to such cases?

Response

Mixed oxide fuel (commonly known as MOX) is not commonly used in the U.S. and very few assemblies of MOX fuel are currently stored. Fission product inventories in MOX fuel do not

differ significantly from those in uranium fuel. In general, since plutonium oxides accumulate in low-enrichment urania fuel as the burnup progresses, large differences in the degradation of MOX fuel and high burnup UO2 fuel would not be expected. Experiments with single MOX fuel pellets indicate higher release of volatile fission products from MOX than from uranium oxide fuel at low temperatures, with release rates converging as the temperature is increased. For SFPs, significant offsite radiological consequences only result from high temperature zirconium fire scenarios. In this study, large releases were associated with small leak scenarios that resulted in very high temperatures and collapse of the fuel rods, which included a range of release fractions depending on the size of the leakage and other factors to reasonably bound the differences between the MOX and uranium fuel types.