NUREG–1353

Regulatory Analysis for the Resolution of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools"

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

E. D. Throm

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ABSTRACT

Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools," addresses the concerns with the use of high density storage racks for the storage of spent fuel, and is applicable to all Light Water Reactor spent fuel pools.

This report presents the regulatory analysis for Generic Issue 82. It includes (1) a summary of the issue, (2) a summary of the technical findings, (3) the proposed technical resolution, (4) alternative resolutions considered by the Nuclear Regulatory Commission, (5) an assessment of the benefits and cost of the alternatives considered, (6) the decision rationale, and (7) the relationships between Generic Issue 82 and other NRC programs and requirements.

Based on this evaluation, the NRC staff concludes that no new regulatory requirements are warranted concerning the use of high density storage racks.

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ABBREVIATIONS AND ACRONYMS

ALARA	As Low As Reasonable Achievable
AR	At-Reactor
ASCE	American Society of Civil Engineers
BNL	Brookhaven National Laboratory
BWR(s)	Boiling Water Reactor(s)
CDF	Core Damage Frequency
CFR	Code of Federal Regulations
DBE	Design Basis Earthquake
DF	Decontamination Factor
DOE	Department of Energy
EPRI	Electric Power Research Institute
FSAR	Final Safety Evaluation Report
GDC(s)	General Design Criterion (Criteria)
GI	Generic Issue
GL	Generic Letter
GSI	Generic Safety Issue
HCLPF	High Confidence of Low Probability of Failure
HEP(s)	Human Error Probability(ies)
IE	Inspection and Enforcement
IEB	Inspection and Enforcement Bulletin
kgm	kilogram
KgU	kilogram of uranium
kw	kilowatt
LLNL	Lawrence Livermore National Laboratory
LPCI	Low Pressure Coolant Injection
LWR(s)	Light Water Reactor(S)
MTIHM	Metric Tons of Initial Heavy Metal
MTHM	Metric Tons of Heavy Metal
MTU	Metric Tons of Uranium
Mw(t)	Megawatts-thermal
NERC	National Electric Reliability Council
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation, NRC
ORNL	Oak Ridge National Laboratory
PRA(s)	Probabilistic Risk Assessment(s)
PWR(s)	Pressurized Water Reactor(s)
RES	Office of Nuclear Regluatory Research, NRC
RHR	Residual Heat Removal
RWST	Refueling Water Storage Tank
SFDF	Spent Fuel Damage Frequency
SFP	Spent Fuel Pool
SNL	Sandia National Laboratory
SRP	Standard Review Plan
SSE	Safe Shutdown Earthquake
SSMRP	Seismic Safety Margins Research Program

PREFACE

This report presents the regulatory analysis, including the decision rationale, for the resolution of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools." The objective of this regulatory analysis is to determine whether the use of high density storage racks for the storage of spent fuel poses an unacceptable risk to the health and safety of the public. As part of this effort, the seismic hazards for two older spent fuel pools were evaluated. The risk change estimates, value/impact and cost-benefit analyses, and other insights gained during this effort, have shown that no new regulatory requirements are warranted in relation to this generic issue.

Edward D. Throm

EXECUTIVE SUMMARY

The risk of beyond design basis accidents in spent fuel storage pools was examined in WASH-1400. It was concluded that these risks were orders of magnitude below those involving the reactor core because of the simplicity of the spent fuel storage pool design: (1) the coolant is at atmospheric pressure, (2) the spent fuel is always subcritical and the heat source is low, (3) there is no piping which can drain the pool and (4) there are no anticipated operational transients that could interrupt cooling or cause criticality.

The reasons for the re-examination of spent fuel storage pool accidents are twofold. First, spent fuel is being stored instead of reprocessed. This has led to the expansion of onsite fuel storage by means of high density storage racks, which results in a larger inventory of fission products in the pool, a greater heat load on the pool cooling system, and less distance between adjacent fuel assemblies. Second, some laboratory studies have provided evidence of the possibility of fire propagation between assemblies in an air cooled environment. Together, these two reasons provide the basis for an accident scenario which was not previously considered.

In addition, in recent years, increasing knowledge in the geosciences has led to a better understanding that, although still highly unlikely, it is more likely that nuclear power plants in the Eastern United States (i.e., east of the Rocky Mountains) could be subjected to earthquake ground motion greater than for which the plants were designed. For this reason, interest has developed in demonstrating that nuclear power plant structures and safety-related systems can safely withstand earthquake ground motion larger than their design earthquake ground motions (post-1973 safe-shutdown earthquake, SSE, or pre-1973 design-basis earthquake, DBE).

Nuclear reactor plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel pool (SFP) and storage racks is to cool the spent fuel assemblies and maintain them in a subcritical array during all credible storage conditions and to provide a safe means of loading the assemblies into shipping casks.

The SFP and components are reviewed to assure conformance with the requirements of 10 CFR Part 50 Appendix A General Design Criteria (GDC) 2, 4, 5, 61, 62, and 63. The review is performed under Section 9.1.2, "Spent Fuel Storage," of the Standard Review Plan (SRP). The SFP water level control system, cleanup system and cooling system are reviewed to assure conformance with the requirements of GDCs 2, 4, 5, 44, 45, 46, 61 and 63 under Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup," of the SRP. In addition, a finding related to 10 CFR Part 20, paragraph 20.1(c) is made as it relates to radiation doses being kept as low as is reasonably achievable (ALARA).

The methods used to provide cooling for the removal of decay heat from the stored assemblies vary from plant to plant depending upon the individual design. The safety function to be performed remains the same: the spent fuel assemblies must be cooled and must remain covered with water during all storage conditions. Assuming that the water is drained, or boiled off, from the spent fuel pool, the fuel rods will heat up until the buoyancy-driven air flow is sufficient to prevent further heatup. If the decay heat level is high enough to heat the fuel rod cladding to about 900 ^{o}C (1650 ^{o}F) the oxidation becomes self-sustaining, resulting in a Zircaloy cladding fire. Propagation of the Zircaloy cladding fire to older adjacent assemblies is likely if the decay heat level in an older adjacent assembly is high enough to heat that assembly to within 100 to

200 °C (200 to 400 °F) of the self-sustaining oxidation temperature. Although propagation of a Zircaloy cladding fire to one to two year old fuel by only thermal radiation can occur, the older fuel would have to be next to the hottest assemblies.

The conditional probability of a Zircaloy cladding fire given a complete loss of water was found to be 1.0 for PWRs and 0.25 for BWRs. The PWR value is based on the use of high density storage racks and the BWR value is selected based on the use of directional storage racks, with the channel box in place. The conditional probability of a Zircaloy cladding fire given a complete loss of water in low density storage racks is estimated to be at least a factor of five less than for the high density configurations. The PWR conditional probability of a Zircaloy fire would be reduced to 0.2 and the BWR conditional probability would be reduced to 0.05. The actual risk reduction achievable may be greater. Open frame racks or cylindrical racks with large inlet holes could result in an greater reduction in risk. The cooling time to preclude a Zircaloy cladding fire could be reduced to less than 20 days, for a conditional probability of 0.05 of a Zircaloy fire for both fuel types.

In addition to implementing the requirements contained in 10 CFR Part 50 Appendix A of the "General Design Criteria," and 10 CFR Part 20, concerning radiation doses being kept as low as is reasonably achievable, licensees should have implemented additional or corrective actions based on the following guidance:

1. IE Bulletin 84-03, "Refueling Cavity Water Seals," issued August 24, 1984.

2. IE Information Notice 84-93, "Potential for Loss of Water From the Refueling Cavity," issued December 17, 1984.

3. Generic Letter 85-11, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants' NUREG-0612," issued June 28, 1985.

4. IE Information Notice 87-13, "Potential for High Radiation Fields Following Loss of Water from Fuel Pool," issued February 24, 1987.

5. IE Information Notice 87-43, "Gaps in Neutron-Absorbing Material in High-Density Spent Fuel Storage Racks," issued September 8, 1987.

6. IE Information Notice 88-65, "Inadvertent Drainages of Spent Fuel Pools," issued August 18, 1988.

7. IE Information Notice 88-92, "Potential For Spent Fuel Pool Draindown," issued November 22, 1988.

The risk from the storage of spent fuel in the spent fuel storage pool at light water reactors is dominated by the beyond design basis earthquake accident scenario. The seismic capacities, or fragility, of two older spent fuel pools indicate that the high confidence of low probability of failure (HCLPF) is about three times the safe shutdown earthquake (SSE) design level. The HCLPF values are estimated to be in the 0.5 to 0.65 g range. The median peak ground acceleration needed to fail these pools is estimated to be in the 1.4 to 2.0 g range, nearly a factor of ten higher than the SSE design value. A report prepared by the American Society of Civil Engineers also concluded that, in general, the seismic design of nuclear facility structures result in median factors of safety on the order of 4 to 19 based on post-1973 design criteria.

The structural capacity of the elevated BWR pool is lower than that for the PWR pool located at the ground level, however the lower conditional probability of a Zircaloy fire for the BWR fuel assembly design offsets the higher seismic failure frequency. The probability of a Zircaloy cladding fire, resulting from the loss of water from the spent fuel pool, is estimated to have a mean value of $2x10^{-6}$ per reactor year for either the PWR or the BWR spent fuel pool. The seismic event contributes over 90% of the PWR spent fuel damage probability, and nearly 95% for the BWR.

The source term for the spent fuel pool accident is not the same as the source term associated with core damage accidents. The consequences of a spent fuel pool accident which results in the complete loss of water are dominated by the long lived isotopes, such as cesium and strontium. The health consequences are dominated by the risk of latent cancer fatalities due to long term exposures.

The best estimate of the consequences of a spent fuel pool accident which results in spent fuel damage to approximately one-third of an equivalent reactor core is $8x10^{6}$ person-rem. This total dose translate to a public health risk from a spent fuel pool accident of 480 person-rem over an average remaining lifetime of 30 years, based on a Zircaloy cladding fire probability of $2x10^{-6}$ per reactor year. The best estimate offsite property damage cost is $4,000 million (1988 $s). The best estimate values are based on a population density of 340 people per square mile within a 50 mile radius from the site and result from the release of radionuclides from the last fuel discharge, 90 days after being discharged. The best estimate of the onsite costs for a SFP accident is $1,180 million (1988 $s), including five years of replacement power to replace the damaged spent fuel pool. Based on an average remaining lifetime of 30 years and a 5% discount rate, the present value of the offsite property damage is estimated to be $124,300 and the present value of the onsite property damage is estimated to be $32,400, based on a Zircaloy cladding fire probability of $2x10^{-6}$ per reactor year.

The value/impact and cost-benefit evaluations for the proposed alternatives for Generic Issue 82 do not indicate that cost effective options are available to mitigate the risk of beyond design basis accidents in spent fuel pools. The option to use low density storage racks for recently discharged fuel has a best estimate value/impact ratio of $32,000 per averted person-rem based on a reduction in spent fuel damage frequency of $2x10^{-6}$ per reactor year. Low density racks would decrease the consequences by a factor of five to ten, but the value/impact ratio is based on 100% reduction in public dose.

The use of post-accident spray systems to mitigate the consequences of a spent fuel pool accident has a best estimate value/impact ratio of $3,300 per averted person rem. This assumes that a post-accident spray system can be designed to withstand the beyond design basis earthquake which causes gross failure of the spent fuel pool structure and has a decontamination factor (DF) of at least 45.

The risks associated with a severe accident in the spent fuel pool are also compared to the objectives and guidance in the Safety Goal Policy Statement. The estimated frequency of a spent fuel pool accident, $2x10^{-6}$ per reactor year, resulting in spent fuel damage meets a target objective of a few percent of a 1×10^{-4} to 5×10^{-5} per reactor year value for overall core damage frequency. The target objective for a "large release" of 1×10^{-6} per reactor year is marginally met, within a best estimate factor of two, but subject to interpretation since the definition of "large release" is still under development. In meeting the societal risk objective of 0.1% of the normally occurring risk to the public given the release frequency of 2×10^{-6} per reactor year, the latent cancer fatality rate from spent fuel pool accidents is estimated to be less than 3% of the target value for the operation of a nuclear power plant.

Therefore, the backfit criteria (10 CFR 50.109) that (1) a substantial increase in the overall protection of the public health and safety is achieved, and (2) the direct and indirect costs of implementation are justified are not met, and Alternative 1 - "No Action" is recommended for the resolution of GI-82.

The risk and consequences of a spent fuel pool accident appear to meet the Safety Goal Policy Statement objectives. They would also meet the proposed 1×10^{-6} per reactor year large-release frequency guideline, at least pending definition of a "large release" by the Commission. Therefore the recommended resolution, Alternative 1 - "No Action," is justified.

Although these studies conclude that most of the spent fuel pool risk is derived from beyond design basis earthquakes, this risk is no greater than the risk from core damage accidents due to seismic events beyond the safe-shutdown earthquake. Therefore, reducing the risk from spent fuel pools due to events beyond the safe-shutdown earthquake would still leave at least a comparable risk due to core damage accidents. Because of the large inherent safety margins in the design and construction of the spent fuel pool, Alternative 1 - "No Action" is justified.

REGULATORY ANALYSIS FOR THE RESOLUTION OF

GENERIC ISSUE 82

"BEYOND DESIGN BASIS ACCIDENTS IN SPENT FUEL POOLS"

1. STATEMENT OF THE PROBLEM

1.1 Historical Background

The risk of beyond design basis accidents in spent fuel storage pools was examined in WASH-1400 (Ref. 1). It was concluded that these risks were orders of magnitude below those involving the reactor core because of the simplicity of the spent fuel storage pool: (1) the coolant is at atmospheric pressure, (2) the spent fuel is always subcritical and the heat source is low, (3) there is no piping which can drain the pool and (4) there are no anticipated operational transients that could interrupt cooling or cause criticality.

The reasons for the re-examination of spent fuel storage pool accidents are twofold. First, spent fuel is being stored instead of reprocessed. This has led to the expansion of onsite fuel storage by means of high density storage racks, which results in a larger inventory of fission products in the pool, a greater heat load on the pool cooling system, and less distance between adjacent fuel assemblies. Second, some laboratory studies have provided evidence of the possibility of fire propagation between assemblies in an air-cooled environment. Together, these two reasons provide the basis for an accident scenario which was not previously considered.

In addition, in recent years, increasing knowledge in the geosciences has led to a better understanding that, although still highly unlikely, it is more likely that nuclear power plants in the Eastern United States (i.e., east of the Rocky Mountains) could be subjected to earthquake ground motion greater than for which the plants were designed. For this reason, interest has developed in demonstrating that nuclear power plant structures and safety-related systems can safely withstand earthquake ground motion larger than their design earthquake ground motions (post-1973 safe-shutdown earthquake, SSE, or pre-1973 design-basis earthquake, DBE).

1.2 Safety Significance

A typical spent fuel storage pool with high density storage racks can hold roughly five times the fuel in the core. However, since reloads typically discharge one third of the core, much of the spent fuel stored in the pool will have had considerable decay time. This reduces the radioactive inventory somewhat. More importantly, after roughly three years of storage, spent fuel can be air-cooled. The spent fuel need not be submerged to prevent melting, although submersion is still desirable for shielding and to reduce airborne activity.

If the spent fuel storage pool were to be drained of water the discharged fuel from the last one or two refuelings, stored in high density storage racks, could still be "fresh" enough to melt under decay heat. The Zircaloy cladding of this fuel could be ignited during heatup. The resulting fire, in a spent fuel storage pool equipped with high density storage racks, might spread to other fuel in the pool.

The heat of combustion, in combination with decay heat, would certainly release considerable gap activity from the fuel and would probably drive "borderline aged" fuel into a molten condition. Moreover, if the fire becomes oxygen-starved (quite probable for a fire located in the bottom of a pit such as the spent fuel storage pool), the hot zirconium would rob oxygen from the uranium dioxide fuel, forming a liquid mixture of metallic uranium, zirconium, oxidized zirconium, and dissolved uranium dioxide. This liquid mixture would allow a release of fission products from the fuel matrix. In addition, although confined, spent fuel storage pools are almost always located outside of the primary containment. Thus, a release to the atmosphere is more likely relative to a release inside primary containment.

The safety significance of "Beyond Design Basis Accidents in Spent Fuel Pools" has been designated as a medium priority issue (Ref. 2), Generic Issue 82. GI-82 applies to all light-water reactor spent fuel storage pools.

2. OBJECTIVES

The general objective of GI-82 is to evaluate the need for additional protective measures for the safe storage of spent fuel in high density storage racks in the spent fuel storage pool at light-water reactor sites.

Both prevention and mitigation are considered. A preventive option is one intended to reduce the frequency of accident sequences potentially conducive to the release of fission products from the spent fuel assemblies. A mitigative option is one intended to reduce the magnitude of the consequences that would result from an accident (environmental radiological releases).

Given the diversity of plant-specific design and construction of spent fuel storage pools, the applicability of any generic analysis of risk reduction measures is limited both with respect to the characterization of risk, and to the cost of implementation for any one plant. The analysis performed for GI-82 is intended to provide a broad evaluation of the value/impact attributes of a given proposed alternative. Plant specific analyses are used for the seismic risk evaluations. In this case, older plants have been selected for the analyses. In general the older plants are more vulnerable to seismic induced failures.

The risk from the storage of spent fuel in spent fuel pools should be a small contributor to the overall risk associated with the operation of a light-water reactor (LWR). On the core damage frequency (CDF) risk level, or more specifically in this case spent fuel damage frequency (SFDF), a target for the resolution of Generic Issue 82, based on the Commission's Safety Goal Policy Statement, is that the contribution from spent fuel pool accidents be a small part (a few percent) of an overall CDF target of $1x10^{-4}$ per reactor year.* Since spent fuel pools are not within the primary containment structure, a target SFDF for spent fuel pool accidents on the order of $1x10^{-6}$ per reactor year may be considered to be compatible with the proposed general performance guidelines given in the Commission's Safety Goal Policy Statement, that is, that the probability of a large release from an operating nuclear power plant should be no greater than $1x10^{-6}$ per reactor year. A more direct comparison of a SFDF target with the policy guidelines requires a definition of a "large release" in the policy statement.

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* More recently, a core damage frequency goal of $5x10^{-5}$ per reactor year has been proposed under the safety goal implementation program. This is a factor of two (2) lower than the $1x10^{-4}$ value used herein, but is within the uncertainty inherent in calculations and assumptions made assessing compliance with either goal, and its adoption in lieu of the $1x10^{-4}$ goal would not affect the recommendations made in this Regulatory Analysis.

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3. ALTERNATIVE RESOLUTIONS

In reaching its proposed resolution of GI-82, the staff considered seven specific alternative courses of action. These are discussed below. The requirements would be applicable to all light-water reactor (LWR) spent fuel storage pools, both in the operating or planned construction stage of licensing. There are 108 spent fuel pools for the 119 operating or planned LWRs at 75 reactor sites in the U.S. The three shutdown units, Dresden 1, Indian Point 1 and Humboldt Bay, are excluded from this accounting.

3.1 Alternative 1 - No Action

This proposed alternative assumes that no additional requirements for the safe storage of spent fuel in the primary spent fuel storage pool are needed. It also assumes that all applicable requirements and guidance to date have been implemented, but no implementation is assumed for related generic issues or other staff requirements or guidance that are still unresolved or still under review.

3.2 Alternative 2 - Require Use of Low Density Racks

This proposed alternative would require the use of low density storage racks for the storage of recently discharged fuel. Also, some reracking from high density to low density racks would be required. As a result, it is expected that additional at-reactor storage of spent fuel would be required to accommodate the lost capacity in the spent fuel storage pool. The use of low density racks shortens the cooling time to preclude a Zircaloy cladding fire by promoting air cooling if water is lost from the spent fuel. The likelihood and the amount of fuel damage would both decrease. This alternative is directed primarily towards prevention of a large release from the spent fuel pool.

3.3 Alternative 3 - Improve Cooling/Make-up Systems

This proposed alternative would require improvements in the spent fuel pool cooling and/or make-up systems, beyond the requirements currently used to license the spent fuel storage pools. Improvements in these systems would reduce the likelihood of fuel damage from loss of cooling events. This alternative is primarily directed towards prevention.

3.4 Alternative 4 - Install Spray Systems

This proposed alternative would require licensees to install post accident spray headers to mitigate the consequences of a Zircaloy cladding fire if the spent fuel storage pool is drained and cannot be reflooded. The likelihood of fuel damage would not change, but the spray systems would remove fission products and lower the consequences of a spent fuel pool accident. This alternative is primarily directed towards risk mitigation.

3.5 Alternative 5 - Modify Spent Fuel Storage Rack Designs

This proposed alternative would require the licensee to compartmentalize the spent fuel storage pool by installing partitions (and individual coolant supply diffusers for each compartment) to limit the extent of the accident, or modify the storage racks to improve air circulation, should the spent fuel storage pool drain. This alternative is directed both towards risk mitigation and prevention.

3.6 Alternative 6 - Cover Fuel Debris With Solid Materials

This proposed alternative would require the development of a contingency plan to dump massive amount of solid materials into a drained spent fuel pool to cover the rubble bed to a depth of several feet. The materials would not be necessarily stockpiled on site, but could also be obtained in a timely manner on an ad hoc basis. The materials (sand, clay, dolomite, boron compounds, lead, etc.) are commonly available in all parts of the country. This alternative would be directed at risk mitigation.

3.7 Alternative 7 - Improve Ventilation Gas Treatment System

This proposed alternative would require the installation of a building ventilation and filter system capable of reducing the concentration of airborne radioactivity before discharge to the environment. This alternative would be directed at risk mitigation.

4. TECHNICAL FINDINGS

4.1 Spent Fuel Pool (SFP) Review Guidelines and Requirements

Nuclear reactor plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel pool (SFP) and storage racks is to cool the spent fuel assemblies and maintain them in a subcritical array during all credible storage conditions and to provide a safe means of loading the assemblies into shipping casks.

The SFP and components are reviewed to assure conformance with the requirements of 10 CFR Part 50 Appendix A General Design Criteria (GDC) 2, 4, 5, 61, 62, and 63. The review is performed under Section 9.1.2, "Spent Fuel Storage," of the Standard Review Plan (SRP) (Ref. 3). The facility and components are reviewed with respect to the following:

(a) The quantity of fuel being stored.

(b) The design and arrangement of the storage racks for maintaining a subcritical array during all conditions.

(c) The degree of subcriticality provided along with the analysis and associated assumptions.

(d) The effects of external loads and forces on the spent fuel storage racks, pool, and liner plate (for example, safe shutdown earthquake, crane uplift forces, missiles, and dropped objects).

(e) Design codes, material compatibility, and shielding requirements.

The SFP water level control system, cleanup system and cooling system are reviewed to assure conformance with the requirements of GDCs 2, 4, 5, 44, 45, 46, 61 and 63 under Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup," of the SRP. In addition, a finding related to 10 CFR Part 20, paragraph 20.1(c) is made as it relates to radiation doses being kept as low as is reasonably achievable (ALARA).

The methods used to provide cooling for the removal of decay heat from the stored assemblies vary from plant to plant depending upon the individual design. The safety function to be performed remains the same: the spent fuel assemblies must be cooled and must remain covered with water during all storage conditions. The capability of the spent fuel pool cooling and cleanup system to provide adequate cooling to the spent fuel during all operating conditions is reviewed on one of two bases. The first basis requires the cooling portion of the system to be designed to seismic Category I, Quality Group C requirements. The second basis allows a non-seismic Category I, Quality Group C spent fuel pool cooling system provided that the following systems are designed to seismic Category I requirements and are protected against tornadoes: the fuel pool make-up water system and its sources; and, the fuel pool building and its ventilation and filtration system. The make-up, ventilation and filtration systems must also withstand a single active failure. The systems are reviewed with respect to the following:

(a) The quantity of fuel being cooled, including the corresponding requirements for continuous cooling during normal, abnormal and accident conditions.

(b) The ability of the system to maintain pool water level.

(c) The ability to provide alternative cooling capability and the associated time required for operation.

(d) Provisions to provide adequate make-up to the pool.

(e) Provisions to preclude loss of function resulting from single active failures or failures of non-safety related components or systems.

(f) The means provided for the detection and isolation of system components that could develop leaks or failures.

(g) The instrumentation provided for initiating appropriate safety actions.

(h) The ability of the system to maintain uniform pool water temperature conditions.

Other functions performed by the system, not related to safety, include water cleanup for the SFP, refueling canal, refueling water storage tank and other equipment storage pools; means for filling and draining the refueling canal and other storage pools; and surface skimming to provide clear water in the SFP.

Load handling in the SFP area is reviewed to assure conformance with GDCs 2, 5, 61 and 62 under Section 9.1.4, "Light Load Handling System (Related to Refueling)," and with GDCs 2, 4, 5 and 61 under Section 9.1.5, "Overhead Heavy Load Handling Systems," of the SRP. In addition the requirements identified in the resolution of GSI A-36, "Control of Heavy Loads Near Spent Fuel," as specified in Generic Letter 85-11 (Ref. 4), "Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants (NUREG-0612)," are reviewed to assure that the implementation of Phase I of NUREG-0612 (Ref. 5) "has provided sufficient protection such that the risk associated with potential heavy load drops is acceptably small." Adequate justification is provided by means of (1) a single-failure proof crane, (2) operator training and procedures, maintenance and inspection procedures, safe load paths and mechanical or electrical stops to prevent movement of heavy loads over irradiated fuel, and/or (3) load drop analyses. The staff concluded, in GL 85-11, "that satisfaction of the Phase I guidelines assures that the potential for a heavy load drop is extremely small."

For reference, the titles of the various review and acceptance criteria are provided in Tables 4.1.1, 4.1.2 and 4.1.3.

Table 4.1.1
10 CFR Part 50 Appendix A, "General Design Criteria"

2	-	"Design Bases for Protection Against Natural Phenomena."
4	-	"Environmental and Missile Design Bases."
5	-	"Sharing of Structures, Systems, and Components."
44	-	"Cooling Water."
45	-	"Inspection of Cooling Water System."
46	-	"Testing of Cooling Water System."
61	-	"Fuel Storage and Handling and Radioactivity Control."
62	-	"Prevention of Criticality in Fuel Storage and Handling."
63	-	"Monitoring Fuel and Waste Storage."

Table 4.1.2
Regulatory Guides

1.13	-	"Design Objectives for Light-Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations."
1.26	-	"Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components at Nuclear Power Plants."
1.29	-	"Seismic Design Classification."
1.52	-	"Design, Testing, and Maintenance Criteria for Engineering-Safety-Feature Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants."
1.115	-	"Protection Against Low-Trajectory Turbine Missiles."
1.117	-	"Tornado Design Classification."
8.8	-	"Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable."

Table 4.1.3
Other Guidelines/References

ANS 57.1/ANSI N208,	"Design Requirements for Light Water Reactor Fuel Handling System."
ANS 57.2/ANSI N210-1976,	"Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations."
NUREG-0554,	"Single-Failure-Proof Cranes for Nuclear Power Plants."
NUREG-0612,	"Control of Heavy Loads at Nuclear Power Plants."

4.2 Spent Fuel Storage Pool Design Features

At some multi-unit sites a single pool is used for both units. Table A.1 of Appendix A identifies 11 dual unit pools, eight sites which have transfer canals between pools and three sites which can use transfer casks to move fuel between pools. The estimated maximum storage capacities (allowing for a full core reserve and other non-fuel reserve areas) and other plant specific information for all spent fuel pools in operation or planned are also provided in Table A.1. There are 108 spent fuel pools for the 119 operating or planned plants at 75 reactor sites in the U.S. (excluding Dresden 1, Humboldt Bay and Indian Point 1, which are now shutdown; and the Ft. St. Vrain HTGR).

The spent fuel pool floor and walls are lined with 1/8 to 1/4 inch thick stainless steel liner plates. The plates are welded to each other by seam welds. Under the seam welds, leak detection (control) channels are provided.

The design features of spent fuel storage pools keep the likelihood of loss of pool water and recriticality small. These features are:

(a) The fuel building concrete structure, the spent fuel storage pool, the spent fuel storage racks, the SFP cooling system, and the supports for the spent fuel handling trolley are designed to withstand seismic forces so that an earthquake as large as the safe shutdown earthquake will not cause loss of water or recriticality.

(b) Fuel storage racks are designed to keep the fuel widely enough separated so that stored fuel will not achieve criticality or, in high density storage racks, poison material is added to the rack structure for criticality control.

(c) The SFP is designed to prevent inadvertent loss of water from the fuel region by drainage through connected piping systems. Although a pool cooling system is connected to the pool for decay heat removal, it is designed to prevent siphoning of the water. A connection exists between the SFP and the reactor pressure vessel head region through the fuel transfer pathway (refueling canal) which is provided with physical barriers to prevent SFP drainage when not in use. The pools are generally sized so that the fuel remains nearly completed covered if the transfer pathway is inadvertently opened.

(d) Should the water inventory in the pool fall below a pre-set level or increase in temperature, multiple water level, water temperature and radioactivity monitors would actuate alarms in the control room. A make-up water system is provided to keep up with small leaks.

(e) Procedures and interlocks are provided to keep the crane from passing over the pool with heavy loads.

(f) The fuel building and the SFP are designed to accommodate the forces which might result from winds and missiles that might be generated by a tornado. Further, the spent fuel storage racks and the SFP cooling system are protected by structures designed to withstand these forces.

For reference, the physical parameters and crane capacities of some typical PWR and BWR spent fuel pools are provided in Table 4.2.1 (from Ref. 6). Some sites share a single pool for multiple units, such as North Anna 1 and 2, Surry 1 and 2, and Oconee 1 and 2. At other sites a transfer canal exists between pools to allow for spent fuel movement, such as Browns Ferry 1 and 2, Calvert Cliffs 1 and 2, and Hatch 1 and 2. At a few sites, a spent fuel shipping cask is used, or available, to transfer spent fuel between pools (San Onofre 1,2, and 3, and Turkey Point 3 and 4).

Table 4.2.1
Typical Pool Dimensions

Plant	Pool Dimensions L(ft)	W(ft)	H(ft)	Cask Area (sq. ft.)	Other Areas (sq.ft.)	Crane Capacity	Minimum Clearance (ft)
PWRs							
Ginna	43	22.2	41.7	116	29	25 T	32
Indian Point 3	33	27	37	n/a	n/a	40 T	n/a
Maine Yankee	41	37	38	100	230	125 T	24
North Anna 1 & 2	56.5	29.3	46.5	12 x 12	96	125 T	29
Oconee 1 & 2	71.3	15	38	131	0	100 T	29
Oconee 3	47.1	13.9	39	192	117	105 T	28
Palisades	38.8	14.7	38	9 x 9	8	100 T	28
Robinson 2	31	33.5	38.3	8.8x8.8	0	125 T	32
San Onofre 1	14	39	39	n/a	n/a	100 T	n/a
San Onofre 2	44	23	46	n/a	n/a	100 T	n/a
St. Lucie 1	33	37	40.5	10 x 12	0	105 T	28
Surry 1 & 2	72.5	27.3	38.5	12 x 12	0	125 T	29
Turkey Point 3	41.3	25.3	40	n/a	n/a	105 T	28
Turkey Point 4	41.3	25.3	40	n/a	n/a	105 T	28
BWRs							
Brunswick 1	56	34	38.8	160	200	125 T	28
Brunswick 2	56	34	38.8	160	200	125 T	28
Fitzpatrick	40	31	37.8	15 x 12	0	125 T	27
Millstone 1	30.5	40.3	38.8	53	310	110 T	22
Monticello	40	26	38	50	20	85 T	23
Oyster Creek	39	27	40	153	21	100 T	24
Peach Bottom 2	40	35.3	40	10 x 10	0	125 T	32
Peach Bottom 3	40	35.3	40	10 x 10	0	125 T	32
Pilgrim 1	32.8	26.1	38.8	77	0	100 T	26
Vermont Yankee	40	26	37.8	49	0	110 T	18

4.3 Spent Fuel Pool Structures

4.3.1 BWR Mark I and Mark II Plants

The spent fuel pool is located at the operating floor level, about 100 to 150 feet above grade. The pool floor and wall are designed for dead load and live load, hydrostatic pressure load, seismic load, thermal loads and loads resulting from the accidental drop of heavy objects. The thickness of the pool walls and floor is on the order of 4 to 6 feet. The horizontal and vertical loads from the pool floor are transmitted to the two longitudinal walls which are designed as deep girders supported at the peripheral wall of the reactor building.

4.3.2 PWR and BWR Mark III Plants

The spent fuel pool is located at the ground level. The physical dimensions and design loads of the the pool are similar to the BWR Mark I and Mark II designs. Due to the lower elevation, the seismic response is relatively low in comparison to the elevated pools in the BWR Mark I and Mark II plants. The vertical and horizontal loads from the pool floor are transmitted to the ground for plants with free standing or floor mounted fuel storage racks. For plants with laterally braced racks, the horizontal seismic loads from the fuel storage racks are transmitted to the pool wall at either the base level or at the base and upper seismic bracing level (about 14 feet above the base level).

4.4 Spent Fuel Storage Rack Descriptions

The following descriptions of spent fuel storage rack configurations are provided:

4.4.1 Low Density Racks (Cell Pitches 20 to 30 Inches)

The subcritical configuration is achieved by the physical separation of the assemblies in an open-frame aluminum of steel structure. Structurally the racks can be laterally braced at the upper and lower levels or they could be bolted to the floor at four corners with the upper and lower grids connected by cross bracing.

4.4.2 Medium Density Racks (Cell Pitches 9 Inches (BWR) to 13 Inches (PWR))

The subcritical configuration is achieved by the flux trap principle. The assemblies are surrounded by stainless steel cans or cells which prevent the neutrons in the water region between the cells from returning to the fuel assemblies. The typical wall thickness is 1/8 inch. Structurally the racks can be laterally braced at the upper and lower grid levels, bolted to the floor at the four corners with the upper and lower grids connected by cross-bracing, or they can be cantilever cells (2x2 or 4x4 modules to reduce flexibility) welded to a base structure.

4.4.3 High Density Racks (Cell Pitches 6 Inches (BWR) to 9 Inches (PWR))

Subcriticality is achieved by the addition of neutron absorbing poison material between the fuel assemblies. The poison is in the form of boron containing material such as boron-carbide, borated stainless steel, or borated aluminum. The storage cell walls have poison containing pockets. Structurally the racks are mostly free standing or laterally braced at the lower level. The honeycomb construction provides structural integrity. The cell walls are typical 0.09 inches thick. The cells are attached to each other by fusion or spot welds.

BWR high density configuration can also be in the form of directional storage racks. In this configuration the BWR assemblies are stored in 6 inch center-to-center racks, with a 5.3 inch open space between rows. No additional neutron absorber material is required in the rack structure for criticality control.

4.4.4 Consolidated Fuel Racks

The fuel assembly is disassembled and stored in a fuel canister. The canister is then stored in high density racks. The consolidation ratio can be 2 to 1. Two fuel assemblies can be compacted into the same physical dimensions of a single assembly. The non-fuel bearing material (such as grid spacers, guide tubes, etc.) is also compacted and stored. The compaction ratios for the non-fuel bearing material are estimated to be 10:1 for PWRs and 20:1 for BWRs (Ref. 7). Since the non-fuel bearing material can take up room in the spent fuel pool, the consolidation ratio may be as low as 1.5 with a weighted average consolidation ratio of 1.63.

4.5 Evaluation of Spent Fuel Cladding Failure

The results of work performed by Sandia (Ref. 8, Ref. 9) suggested that in certain fuel racking configurations (a) a self-sustaining zirconium-air oxidation reaction can be initiated, and (b) this self-sustaining reaction can propagate from one region of the pool to another.

These results were based on both experimental simulation and computer modeling. A computer program was developed by Sandia, called SFUEL1W, to evaluate conditions under which a self-sustaining Zircaloy reaction would occur and under what condition the Zircaloy fire would propagate to older stored fuel assemblies. Large uncertainties, associated with the phenomenology of Zircaloy oxidation and its propagation in spent fuel assemblies, were identified in the Sandia studies.

The SFUEL1W computer program was partially validated by Sandia (Ref. 9) and was also further validated by BNL (Ref. 10). The calculated results of the SFUEL1W models were compared with existing Sandia National Laboratory (SNL) small-scale experimental results. The CLAD computer program, a modified version of SFUEL (an early version of SFUEL1W), was used. CLAD was developed by Sandia to model the experimental test results (Ref. 9). Sandia had performed some verification studies against the experimental tests, but did not complete the work before funding ended. The BNL calculations with CLAD tended to result in an over-prediction of the peak cladding temperatures.

The NRC staff performed an independent verification of SFUEL1W using the CLAD computer program (Ref. 11). The BNL verification program included some modifications to CLAD. These modifications included the addition of helium properties to model the initial test conditions and a switch from helium to air flow, and an energy balance model to force conservation of energy on each gas control model. In the staff program, additional modifications were made to CLAD. The most significant modification was the inclusion of the SFUEL1W gas heatup model.

This NRC staff modified version of CLAD was verified against two of the Sandia air tests and the results compared favorable with the available experimental data. The peak cladding temperatures calculations were in good agreement with the data. It was concluded that the SFUEL1W fuel, cladding and gas heatup models are satisfactory.

The reaction rate equation for the oxidation of Zircaloy cladding in air used in the SFUEL1W computer program was also subject to uncertainties. BNL (NUREG/CR-4982) performed a literature search related to the oxidation rate of Zircaloy in air. Based upon the current state-of-the-art understanding of the associated phenomena and by performing sensitivity studies on the Zircaloy-air reaction rate correlation, it was concluded by BNL that the oxidation rate model used by Sandia in the SFUEL1W computer program is acceptable for the evaluation of spent fuel damage. For temperatures in the 800-1150 °C range (1470-2100 °F) the available data indicates that the Sandia correlation is valid, for exposure periods of 30 minutes. For longer periods the correlation may be non-conservative. At a constant temperature the rate of oxidation may increase with exposure time. This does not alter the findings concerning the initiation and/or propagation of a self-sustaining Zircaloy fire. Initiation is not influenced by the oxidation rate equation, and propagation can occur before cladding failure and relocation of the fuel rods occurs.

The uncertainties in the Zircaloy oxidation propagation calculations under inadequate room ventilation conditions (most typical of the spent fuel storage pool structures) were further studied by BNL (Ref. 10) using the SFUEL1W computer program. A sensitivity study covering hot spent fuel decay power in the 20 to 90 Kw/MTU range was performed. For reference, 90 Kw/MTU is the decay heat generation rate of fuel five to seven days following shutdown of a 3000 Mw(t) reactor. After one year the decay power level for a BWR is about 6 Kw/MTU and 11 Kw/MTU for a PWR. After two years, the decay power levels are estimated to be 4 Kw/MTU and 6 Kw/MTU respectively.

The SFUEL1W computer program is a finite difference solution of the transient equation for heating of the fuel rods considering:

- The heat generation rate from the decay heat and oxidation of the cladding,

- Radiation to adjacent assemblies and pool walls, and

- Convection to buoyancy-driven air flows.

The key assumptions and limitations of SFUEL1W are:

- The water drains instantaneously from the pool,

- The geometry of the fuel assemblies and racks remains undistorted,

- Temperature variations across the fuel rods are neglected,

- The air flow patterns are one-dimensional, and

- The spaces between adjacent holders are assumed to be closed to air flow.

With respect to the limitation concerning instantaneous draining of the pool, this assumption simplifies the heatup model in the SFUWL1W computer program and is not intended to be representative of any accident sequence other than perhaps the catastrophic failure of the spent fuel structure from a beyond design base seismic event.

After the water is drained from the spent fuel pool, the rods heat up until the buoyancy-driven air flow is sufficient to prevent further heatup. If the decay heat level is sufficient to heat the rods to about 900 °C (1650 °F) the oxidation becomes self-sustaining.

BNL has concluded (Ref. 10) that:

- The likelihood of cladding fire initiation is not very sensitive to the oxidation rate equation,

- The oxidation rate equation in SFUEL1W is a reasonable representation of the available data, and

- The likelihood of cladding fire initiation is most sensitive to the decay heat level and the storage rack configuration (which controls the extent of natural convection cooling).

It was also concluded that the oxidation propagation to older adjacent assemblies is likely if the decay heat level of the older adjacent assembly is high enough to heat that assembly to within 100 to 200 °C (200 to 400 °F) of the self-sustaining oxidation temperature. The radiation heat transfer from the burning assemblies then could be sufficient to raise the temperature of the older adjacent assembly to the self-sustained oxidation limit.

The following descriptions of spent fuel storage rack configurations are provided and are representative of the geometries used by both Sandia (Ref. 8) and BNL (Ref. 10) to determine spent fuel storage configuration which can result in Zircaloy fires and propagation of the fire to older stored fuel:

(1) High density PWR configuration: In this configuration, the fuel assemblies are tightly packed with neutron absorber material used in the rack structure to replace the reduced water moderator for criticality control. The center-to-center assembly spacing is 10.25 inches, the open gap between assemblies is 0.7 inches. This configuration is in use in nearly all PWRs, and is referred to as high density storage.

(2) Cylindrical PWR configuration: This configuration is typical of the early rack designs, used before at-reactor storage of spent fuel was required. The center-to-center assembly spacing is 12.75 inches, in a closed cylindrical stainless steel rack. The typical cross sectional area of a PWR assembly is 8.4 by 8.4 inches. This is referred to as low density storage.

(3) Cylindrical BWR configuration: This configuration is typical of early BWR spent fuel storage rack designs. The center-to-center assembly spacing is 8.5 inches. The typical cross sectional area of a BWR assembly is 5.3 inches. This is referred to as low density storage.

(4) Directional BWR configuration: In this configuration the BWR assemblies are stored in 6 inch center-to-center racks, with a 5.3 inch open space between rows. No additional neutron absorber material is required in the rack structure for criticality control. This is considered to be a high density storage configuration for BWRs.

Because of limitations in the SFUEL1W computer program, BNL limited the BWR spent fuel analyses to the low density cylindrical configuration. The SFUEL1W computer program does not account for air flow between adjacent holders, an assumption which was based on the storage rack design. The self-sustaining oxidation analysis is governed by the BWR channel box design, the air flow through the assembly. The SFUEL1W results are not significantly influenced by the BWR rack design.

The estimated likelihood of self-sustaining oxidation for various spent fuel rack configurations is provided in Table 4.5.1, based on a 12 month fuel cycle which is typical for most PWRs. BWRs typically operate on an 18 month fuel cycle. Additional calculated results from the earlier Sandia work (Ref. 8) are also provided. In Table 4.5.1 the Critical Cooling Time is defined as the decay time to reduce the internal heat generation rate to a low enough value, the Minimum Decay Power, to preclude the Zircaloy cladding temperature from exceeding the self-sustained oxidation limit under air cooling. Considering a fuel cycle of not less than one year, the cooling time can be converted to a conditional probability of fire initiation.

The conditional probability of propagation of the Zircaloy fire to older stored spent fuel was evaluated by BNL (Ref. 10). The results are provided in Table 4.5.2. In Table 4.5.2 the High Power Level is the decay heat generation of the recently discharged fuel and the Adjacent Power Level is the decay heat generation rate of the older fuel at the Approximate Decay Time. Perfect ventilation is assumed for these two studies. The impact of no ventilation, which would result in the depletion of the oxygen from the air, is summarized in Table 4.5.3. The High Power Level and Adjacent Power Level are the same as described for Table 4.5.2.

After an extensive review of the SFUEL1W computer code and comparison to the SNL small scale experiments, BNL concluded that the code provides a valuable tool for assessing the likelihood of self-sustaining clad oxidation for a variety of spent fuel storage configurations (Ref. 10). Additional studies performed by the staff supported the BNL conclusion.

For the purpose of evaluating the risk from beyond design basic accidents in spent fuel pools, the conditional probability of a Zircaloy cladding fire given a complete loss of water will be assumed to be 1.0 for PWRs and 0.25 for BWRs. The PWR value is based on the use of high density storage racks. The BWR value is selected based on the use of directional storage racks, with the channel box in place.

For the proposed alternative to require recently discharged spent fuel be stored in low density storage racks, the risk reduction is estimated to be a factor of five (Ref. 10). This level of risk reduction, resulting from the decrease in the cooling time needed to preclude a Zircaloy fire, is seen to be equivalent to the use of low density cylindrical storage racks with three inch inlet holes. The PWR conditional probability of a Zircaloy fire is reduced to 0.2 and the BWR conditional probability is reduced to 0.05. The actual risk reduction achievable may be greater than assumed. Open frame racks or cylindrical racks with larger inlet holes would result in an increased reduction in risk. The cooling decay time could be reduced to less than 20 days, for a conditional probability of 0.05 for both fuel types.

Table 4.5.1
Estimated Likelihood of Self-Sustaining Zircaloy Clad Oxidation for Various Spent Fuel Rack Configurations and Decay Heat Levels

Spent Fuel Rack Configuration	Inlet Orifice Diameter (inches)	Minimum Decay Power (Kw/MTU)	Critical Cooling Time (days)	Conditional Probability (per year)
BNL (Ref. 10) Last Discharge Only				
High Density PWR	10	11	360	1.0
	5	6	700	1.0
Cylindrical PWR	5	90	10	~0.0
	3	45	50	0.14
	1.5	15	250	0.7
Cylindrical BWR	3	30	30	0.08
	1.5	14	180	0.5
SNL (Ref. 8) Full Core Discharge				
Directional BWR with channel box	5	n/a	90	0.25
Directional BWR without channel box	5	n/a	30	0.08
PWR Open Frame	-	n/a	10	~0.0
Cylindrical PWR	5	n/a	20	0.05
	3	n/a	120	0.33
	1.5	n/a	250	0.7

Notes: Conditional probability estimated from NUREG/CR-0649 for maximum peak cladding temperatures less than 600 °C. Self-sustaining Zircaloy oxidation onset is approximately 900 °C.

n/a - Not Available. In the Sandia studies, the decay power level was not reported, however these analyses were performed for a full core discharge situation.

Table 4.5.2
Estimated Likelihood of Propagation of Zircaloy Fire to Older Spent Fuel for Various Spent Fuel Rack Configurations and Decay Heat Levels

Results for High Density PWR Racks With Large Inlet Holes (10" Diameter) and Perfect Ventilation

High Power Level (kw/MTU)	Adjacent Power Level (kw/MTU)	Approximate Decay Time (Days)	Propagation (Yes/No)
11.0	5.9	365	Yes
19.2	5.9	365	Yes
90	5.9	365	Yes
90	4.0	730	No

Results for Cylindrical PWR Racks With 3" Diameter Inlet Holes and Perfect Ventilation

High Power Level (kw/MTU)	Adjacent Power Level (kw/MTU)	Approximate Decay Time (Days)	Propagation (Yes/No)
90	11	365	No
90	19	180	Yes(1)

Results for Cylindrical PWR Racks With 1.5" Diameter Inlet Holes and Perfect Ventilation

High Power Level (kw/MTU)	Adjacent Power Level (kw/MTU)	Approximate Decay Time (Days)	Propagation (Yes/No)
90	11	365	Yes
90	5.9	730	Yes
90	3	1100	No
15	11	365	Yes
15	5.9	730	No

Note: (1) This is unlikely situation, assumes recent spent fuel discharge six months after previous discharge. Fuel cycles are typically 12 to 18 months.

Table 4.5.3
Summary of Radial Oxidation Propagation Results for Various PWR Spent Fuel Rack Configurations and No Ventilation

Spent Fuel Rack Configuration	High Power Level (kw/MTU)	Adjacent Power Level (kw/MTU)	Propagation (Yes/No)
Cylindrical 1.5" hole	90	5.9	Yes
	90	3	No(1)
Cylindrical 3.0" hole	90	5.9	No
	19.2	11	Yes
High Density 10" hole	90	4	No(1)

Note: (1) Without ventilation the fire becomes oxygen starved. Oxygen depletion prevents propagation.

4.6 Quantification of Accident Sequences in Spent Fuel Pools

4.6.1 Structural Failure Due to Missiles

High energy missiles which might impact with the spent fuel pool might result in sufficient structural damage to prevent cooling of the spent fuel. Missiles generated by tornadoes or from a turbine failure are considered during plant licensing. As indicated in Section 4.1, these accident sequences are reviewed by the NRC staff to assure compliance with the General Design Criteria.

In WASH-1400, the probability of a turbine failure and missile generation has been estimated to be on the order of 1×10^{-4} per reactor year, and the limiting strike probability for the spent fuel pool has been estimated to be 4.1×10^{-3}, given an energetic missile. The probability of a turbine missile hitting the spent fuel pool is therefore estimated to be 4.1×10^{-7} per reactor year.

The probability of a beyond design basis tornado striking a reactor site has been estimated to have a mean value of about 5×10^{-6} per reactor year (WASH-1400, Ref. 1), with a mean probability for all tornadoes of 5×10^{-4} per reactor year. A typical reactor site is estimated to be about 620 acres (one square mile), or greater (Ref. 6). The plan area of a spent fuel pool (50 feet wide by 60 feet long) is 1×10^{-4} square miles. The probability of a beyond design basis tornado striking the spent fuel pool is therefore on the order of 5×10^{-10} per reactor year. The probability of a tornado missile striking the spent fuel pool has been estimated based on the Zion site to be on the order of 1×10^{-6} per reactor year (Ref. 16).

The likelihood of a missile, turbine or tornado generated, damaging the spent fuel pool and resulting in an unrecoverable loss of water is estimated to be less than 0.01 per demand (Ref. 16). The missile would have to cause sufficient damage to prevent filling or repair of the spent fuel pool. Given the estimated combined likelihood of a missile strike on the order of $1x10^{-6}$ per reactor year, the estimated probability of the structural failure of the spent fuel pool from a missile resulting in a loss of cooling of the spent fuel is less than $1x10^{-7}$ per reactor year, on the order of $1x10^{-8}$ per reactor year.

4.6.2 Structural Failure Due to Aircraft Crashes

The probability of an aircraft striking the spent fuel pool is proportional to the vulnerable area of the structure, the aerial crash density of an aircraft and the number of operations on applicable runways. SRP Section 3.5.1.6 is used to derive the hit frequency for a reactor site.

The probability of structural failure of the spent fuel pool as a result of an aircraft crash has been estimated using Zion PRA results (Ref. 16). The mean hit frequency is estimated to be $6x10^{-9}$ per reactor year with 5% and 95% confidence bounds of $5x10^{-9}$ to $2x10^{-8}$ per reactor year. The probability of structural failure of the spent fuel pool resulting in significant spent fuel damage was estimated to be less than $1x10^{-10}$ per reactor year in NUREG/CR-4982 (Ref.10) and in EPRI NP-3365 (Ref.16).

4.6.3 Structural Failure Due to Heavy Loads Drop (Shipping Cask)

Load handling in the SFP area is reviewed to assure conformance with GDCs 2, 5, 61 and 62 under Section 9.1.4, "Light Load Handling System (Related to Refueling)," and with GDCs 2, 4, 5 and 61 under Section 9.1.5, "Overhead Heavy Load Handling Systems," of the SRP. In addition, the requirements identified in the resolution of GSI A-36, "Control of Heavy Loads Near Spent Fuel," as specified in Generic Letter 85-11 (Ref. 4), "Completion of Phase II of Control of Heavy Loads at Nuclear Power Plants (NUREG-0612)," are reviewed to assure that the implementation of Phase I of NUREG-0612 (Ref. 5) "has provided sufficient protection such that the risk associated with potential heavy load drops is acceptably small." Adequate justification is provided by means of (1) a single-failure proof crane, (2) operator training and procedures, maintenance and inspection procedures, safe load paths and mechanical or electrical stops to prevent movement of heavy loads over irradiated fuel, and/or (3) load drop analyses. The staff concluded, in GL 85-11, "that satisfaction of the Phase I guidelines assures that the potential for a heavy load drop is extremely small."

The estimated probability of structural failure of the spent fuel from a shipping cask drop was estimated by BNL based on a cask handling assumption of two fuel shipments per week, similar to that used in WASH-1400. The estimated probability of a shipping cask being dropped on the spent fuel pool wall, without consideration of the requirements from GSI A-36, was estimated by BNL to be $3.1x10^{-4}$ per reactor year (Ref. 10). The likelihood of pool damage was estimated by BNL (Ref. 10) to be 0.1 per demand, one-in-ten drops causing sufficient damage to completely drain the pool, with an uncertainty range of 0.01 to 1.0. The estimated reduction in the probability of a shipping cask drop for a plant which complies with the resolution of GSI A-36 (Ref. 4) has been estimated by BNL to be a factor of 0.001, for a revised probability of $3.1x10^{-8}$ per reactor year, including the 0.1 conditional probability of failure given a shipping cask drop.

A more detailed analysis of the resultant damage to a spent fuel pool structure as a result of a shipping cask drop was performed by LLNL (Ref. 17). A BWR and PWR spent fuel pool were analyzed for a variety of cask weights and drop heights. The results of the LLNL analysis indicate that the pool wall could suffer severe damage as a result of a cask drop. The indicated regions of potential reinforcing steel yield are quite extensive and while the integrity of the pool liner is difficult to predict, it was concluded by LLNL that it seems likely that the liner would be severely damaged. The estimated probability of the structural failure of a spent fuel pool resulting from a dropped shipping cask is therefore considered to be equivalent to the probability of dropping the cask, 3.1×10^{-7} per reactor year, given a cask handling rate of twice per week (104 per reactor year).

At the present time, spent fuel is mostly being accumulated in spent fuel pools. At a few facilities, the older fuel assemblies are being transferred to dry storage areas on site. To estimate the probable number of cask handling operations per year the following assumptions are made:

(1) The spent fuel pool capacity, with a full core reserve, has reached a licensing limit (either structurally or due to cooling capacity restrictions).

(2) The capacity, maximum number of allowable assemblies, is maintained and excess assemblies are transferred to either an onsite dry storage area or the DOE repository.

(3) Based on a 12 month fuel cycle in PWRs with 200 assemblies in the core, about 70 assemblies would have to be removed annually from the spent fuel pool to accommodate reloads.

(4) Based on a 18 month fuel cycle in BWRs with 800 assemblies in the core, about 130 assemblies would have to be removed annually from the spent fuel pool.

(5) The weight of a PWR assembly is approximately twice that of a BWR assembly, 657.9 kg (1450 lbs) versus 319.9 kg (700 lbs).

(6) Based on TN-24P (Ref. 18) and MC-10 (Ref. 19) dry storage cask designs, twenty-four PWR, or 48 BWR, assemblies can be moved per cask. The cask weight is in the 100 ton range.

The estimated number of transfers per reactor year is therefore estimated to be about a factor of ten lower as compared to the WASH-1400 rate of 104. The probability of structural damage to the spent fuel pool as a result of a dropped shipping cask is estimated to be 3.1×10^{-8} per reactor year (best estimate) for a reasonable cask handing rate. The upper bound estimate is taken from NUREG/CR-4982 (Ref. 10) as 3.1×10^{-7} per reactor year.

4.6.4 Reactor Cavity and Transfer Gate Pneumatic Seal Failures

Inflatable, pneumatic seals are used during refueling operations in PWRs to seal the gap between the reactor pressure vessel flange area and the biological shield walls. This permits flooding of the reactor pressure vessel cavity above the core to allow for the safe handing of the fuel. In BWRs, the reactor cavity seals are typically permanent stainless steel expansion bellows, and not subject to the failure modes associated with the pneumatic designs. Pneumatic seals are also used to partition areas of the spent fuel pool, for example, between the shipping cask handling area or fuel transfer tube and the main spent fuel storage area. Ten reported cases of pneumatic seal failures resulting in actual or potential loss of water from spent fuel pools are listed in Table 4.6.1, three involving the refueling cavity seal and seven involving other pneumatic seals.

Three of the ten reported events involved the failure of the refueling cavity seal. In one case, no fuel was in the spent fuel pool and the failure occurred during installation and testing. In the other two events, the fuel transfer canal was closed at the time and no actual drainage from the spent fuel pool would have occurred.

At Surry 1 in May 1988 following the water loss from the refueling cavity seal failure, the plant operator opened the fuel transfer canal path to aid in reflooding the reactor cavity. Personnel inside containment, on the fuel crane bridge, had to leave containment as a result of high radiation levels. Operators did not enter appropriate procedures for a loss of refueling cavity level, and the existing procedures provided inadequate guidance to operations personnel on a rapid loss of cavity water (Ref. 20). Procedures had been developed at Surry to address cavity water loss in response to IEB 84-03 (Ref. 21), but were inadvertently omitted from revised procedures in 1987. A review of the design by the seal vendor (Presray), who has stated that they manufacture and continue to supply most of the refueling cavity seals used throughout the industry, determined that the design at Surry is unique and inadequate. A seal backup plate should have been provided to prevent movement of the seal. The IEB 84-03 review by the licensee failed to identify the weakness in the seal design, although procedures were developed to address seal failure.

In the remaining seven reported instances, pneumatic seal failures have occurred which could result in the draining of the spent fuel pool. The event at Hatch (see Table 4.6.1) is considered to be unique, and in two of these seven cases there was no spent fuel in the pool at the time of the event.

For the purpose of evaluating the potential for spent fuel damage from pneumatic seal failures, the event frequency was initially estimated by BNL to be 0.01 per reactor year and is generally applicable to PWRs. Based on advances in seal designs, increased awareness and surveillance resulting from IEB 84-03, BNL estimated the present failure rate to be an order of magnitude less, 0.001 per reactor year. In addition to the seven events identified in NUREG/CR-4892, two more events related to seal failures where identified (Ref. 22) for a total of nine events in about 900 years of experience. The Surry 1 event, which occurred after NUREG/CR-4982 was published, would not alter the event frequency (ten events in 1,000 years of experience).

Not all seal failures will lead to loss of water from the spent fuel pool. A failure of the refueling cavity seal must occur coincident with an open fuel transfer canal. Even under this assumption, the spent fuel pools are designed to preclude significant (a few inches) fuel uncovery due to the leakage. The transfer canal is either located above the top of the storage racks or a weir is used to prevent lowering the level below the top of the fuel.

Seal failures coincident with fuel handling operations are being addressed as a separate issue, Generic Issue 137, as discussed in Section 4.9.3.

Table 4.6.1
Events in Which Pneumatic Inflated Seals Have Failed

Date	Plant	Seal Location	Cause	Result
9/72	Pt. Beach 1(1)	Transfer gate	Failure of air supply	11,689 gal leak
10/76	Brunswick 2	Inner pool gate	Air leak in seals an compressor power supply failure	5" level drop
6/80	Trojan	Transfer gate	Not inflated prior to draining refueling cavity, level alarm also failed	10" below T.S.
2/81	Davis-Besse	Transfer gate	Low seal pressure	15" level drop
5/81	ANO-2	Transfer gate	Maintenance error air supply shutoff	1000 gpm leak
8/84	Haddam Neck(2)	Cavity seal	Design weakness in 20 minutes	200,000 gal leak
10/84	San Onofre 2(1)	Gate seal	Air compressor fails	20,000 gal leak
11/84	San Onofre 2(1)	Cavity seal(3)	Manufacturing defect seal ruptured	19.5" level drop
12/86	Hatch(4)	Pool canal flexible joint	Air supply valve closed	141,000 gal leak
5/88	Surry 1(5)	Cavity seal	Maintenance error and design weakness	30,000 gal leak

Notes: (1) No spent fuel in pool at time of event.
(2) Fuel transfer canal closed. No water loss from pool.
(3) Failure during installation and leak testing
(4) Make-up system cycled to maintain level, undetected for about 7.5 hours. Hatch has interconnected transfer canal between two pools. Seal required for SSE considerations. Unique design.
(5) Licensee had reviewed seal design following Haddam Neck event, and determined design to be adequate. Procedures developed to address seal failure subsequently omitted. Maintenance error isolated air supply to seal, combined with low backup nitrogen accumulator resulted in seal failure. Passive backup seal failed due to improper installation and design. Design determined to be unique to Surry. Maximum leakage 6,500 gpm for 4 minutes.

In response to IEB 84-03, the analyses supplied by licensees indicated that the failure of a pneumatic refueling cavity seal in most PWR plants would not result in massive leakage because of the relatively narrow gap to be sealed and the geometric shape of the seal (the Haddam Neck design was determined to be unique because of the large gap sealed). Also, leaks from seal failures in the transfer canal gates would be limited, in most cases, because the leakage would be into a confined volume, for example from the pool into a drained up-ender sump. This volume is small in comparison to the spent fuel pool volume and the level in the spent fuel pool would decrease slightly.

Licensee responses to IEB 84-03 have been reviewed by the NRC staff to determine the credible leakage that could result from pneumatic seal failures, in particular the refueling cavity seal. Although BWRs do not generally use pneumatic seals (permanent stainless steel bellows are used), some licensees did provide estimates for credible leakage in the highly unlikely case these seals were to fail. The seal failure leakage rates provided by licensees are listed in Table 4.6.2, and are representative of the maximum flow rates achieved for a fully flooded refueling cavity. As the water level drops, the hydrostatic pressure decreases and the flow rate will also decrease. For example, in the Summer submittal (Virgil C. Summer response to IEB 84-03, October 16, 1984, Docket No. 50-395), even though the maximum flow rate is 5,500 gpm, it was estimated that it would require 160 minutes to drain the cavity to the level of the seal ring with the transfer tube open. If the transfer tube is closed, the spent fuel pool will not drain. The Watts Bar submittal (Watts Bar response to IEB 84-03, December 6, 1984, Docket No. 50-390) estimated the time to drain to the reactor vessel flange region was about 95 minutes. The Catawba/McGuire submittal (William B. McGuire 1 and 2 and Catawba 1 and 2 responses to IEB 84-03, November 11, 1984, Docket Nos. 50-369, -370, -413, and -414) provided the most comprehensive assessment of postulated leak rates. The times to drain the refueling cavity, considering the hydrostatic pressure changes as level decreases, ranged from 12 minutes for 100% gross failure to 414 minutes for a 1/16 inch gap around the entire seal circumference. For the 25% gross seal failure, the time was estimated to be 65 minutes. This case was reported to be identical to the Haddam Neck seal failure event.

The catastrophic complete failure of the refueling cavity seal is not considered to be credible. The Haddam Neck seal failure is considered to the most limiting case. Even if a catastrophic failure occurred, with one to two feet of water remaining above the fuel in the spent fuel pool, there would be at least two hours available for the operator to take emergency actions to provide cooling for the spent fuel before the water covering the spent fuel boils off. This time estimate is based on Table 4.6.3, which shows that the maximum rate of boiling following a full core discharge five days after shutdown is equivalent to one foot every two hours (based on 49 hours to boil off 23 feet of water). For a normal refueling, assuming 1/3 core discharge, the recovery time would be five hours. An example of a procedure developed to address this highly unlikely situation was provided in the Catawba/McGuire response to IEB 84-03. The alternate cooling method is to recirculate water from the containment sump to the refueling water storage tank and then to the spent fuel pool using a residual heat removal pump.

The fuel transfer canal structure, located between the fuel transfer canal in the reactor vessel refueling pool and the fuel storage pool in the fuel storage building, is typically equipped with a metal expansion joint to accommodate flexure. In the event the fuel transfer tube tube expansion joint failed, the outer sleeve slip joints would limit the leakage flow. The maximum calculated leakage through a slip joint was estimated to be 400 gpm (Indian Point 2 supplemental response to IEB 84-03, March 31, 1987, Docket No. 50-247).

BNL estimated (Ref. 10) that the frequency of a serious loss of pool water inventory resulting from a pneumatic seal failure would be on the order of one in one hundred events (0.01). The combined probability estimate was 1×10^{-5} per reactor year for a serious loss of pool water event (Ref. 10). This estimate does not include credit for recovery actions to mitigate or stop the draining event from resulting in spent fuel damage. A conditional probability of failure to recovery of 0.05 was used by BNL, based on previous studies (Ref. 27), resulting in a frequency estimate of 5×10^{-7} per reactor year for the seal failure accident sequence.

To assure that the BNL estimate is appropriate for this event, the NRC staff examined the human error probability (HEP) of failure to diagnose this event in sufficient time to take mitigative action. As discussed above, licensee responses to IEB 84-03 demonstrate that there is considerable time to respond to a seal failure event, even for postulated leakage on the order of several thousand of gallons per minute. The refueling cavity seal would have to fail while the transfer canal was open. For the protection of the spent fuel, the transfer canal gate valves can be closed and make-up water provided to restore the pool level if the level in the refueling cavity falls below the reactor pressure vessel flange region. Using the nominal HEP screening model (Ref. 21) to estimate the HEP for failure to diagnose the event, the median joint HEP for a one to four hour time period is 1×10^{-4} to 5×10^{-5}. The error factor on the median HEP is 30, therefore the HEP value for failure to diagnose a serious seal failure is estimated to be in the 3×10^{-3} to 1.5×10^{-4} range.

On December 17, 1984 an IE Information Notice, IN 84-93, "Potential for Loss of Water From the Refueling Cavity," was issued (Ref. 24). In this notice the staff concluded that "Adequate emergency procedures and properly calibrated refueling cavity water level instrumentation are considered to be important in the mitigation of any loss-of-cavity-water accident."

Based on the heightened awareness to refueling cavity seal designs, installation, testing and maintenance of the seals, and to the need for adequate procedures to address seal failures, as identified in IEB 84-03 and in IN 84-93, and considering the time available to diagnose a serious seal failure on the order of one hour, the staff updated the estimate of the frequency of loss of spent fuel pool water resulting in fuel damage. Given a serious seal failure frequency estimate of 1×10^{-5} per reactor year with an HEP conditional failure probability of 3×10^{-3} (median value for one hour with error factor) to diagnose the seal failure, the best estimated frequency is 3×10^{-8} per reactor year of a seal failure resulting in spent fuel damage. The upper estimate for this event is 5×10^{-7} per reactor year, based on the BNL evaluation in NUREG/CR-4982 (Ref. 10).

Table 4.6.2
Refueling Cavity Seal Leak Rates Following Seal Failure

Plant	Type	Leak Rate (gpm)	Assumptions
Point Beach 1/2	PWR	62	break area of 0.5 square inches
Turkey Pt. 3/4	PWR	50	
Comanche Peak	PWR	100	break area of 1.0" diameter
Oconee	PWR	50	total dislodged inner seal
TMI	PWR	4,700	maximum, major gasket failure
Vogtle	PWR	175	8" section of gasket
Watts Bar	PWR	3,176	1/16" gap around seal
Summer	PWR	5,500	60 mil gap around seal
Prairie Island 1/2	PWR	4,200	2" gap
Watts Bar	PWR	3,200	
Haddam Neck	PWR	10,000	actual seal failure data
Catawba/McGuire	PWR	103,642	gross failure, 100% of seal
		20,467	gross failure, 25% of seal
		3,210	1/16" gap around seal
LaSalle	BWR	185	
Vermont Yankee	BWR	500	
FitzPatrick	BWR	370	inner bellow seal
		7,100	outer bellow seal

Note: Leak rates are maximum flows with full refueling cavity water level. As level decreases, flow rates will decrease.

4.6.5 Inadvertent Draining of the Spent Fuel Pool

There are other mechanism for draining the spent fuel, in addition to the seal failures discussed previously. Pipe breaks in the cooling system or heat exchangers, or siphoning paths could result in loss of water. There have been a number of recent events resulting in partial draining of spent fuel pools. These have been identified in IE Information Notice 88-65, "Inadvertent Drainages of Spent Fuel Pools." (Ref. 25)

At Wolf Creek (12/22/87) a valve in a return line to the refueling water storage tank (RWST) was left open following use of the spent fuel pool cleanup system to clean the RWST. The spent fuel pool level indicator and low level alarm were both inoperable in the control room. Successive tripping of the spent fuel pool cooling system pump alerted the operators to a problem. The minimum level in the spent fuel pool was 22 feet above the fuel.

At River Bend (9/20/87) an antisiphoning device in the purification system suction line was plugged in the upper spent fuel pool. River Bend is a Mark III BWR with an upper spent fuel pool near the reactor pressure vessel. Fuel movement to the primary spent fuel pool is not necessary unless the spent fuel is being completely discharged from the reactor. The upper pool was intentionally drained below the level indicator range to accommodate placement of the steam dryer in the pool. When using the condensate storage tank (CST) to refill the upper pool, valve misalignment result in a siphoning effect. High radiation alarms and a level increase in the CST alerted the operators to the problem. The manual valves in the purification line were closed to stop the drainage.

At San Onofre 2 (6/22/88) a siphoning path was present in the purification system, and approximately 9,000 gallons were siphoned. Although siphon breakers, check valves, and locked valves were installed, the administrative controls were not established allowing alignment of the system which led to the siphoning event.

Operating procedures for the interconnected systems associated with the spent fuel pools either were not sufficiently detailed or were incorrect and failed to prevent alignments causing unintentional drainage. At Wolf Creek procedures did not exist. Also surveillance procedures were not implemented to ensure the operability of all instrumentation and control equipment at Wolf Creek.

At Turkey Point 4 (8/16/88) approximately 3,100 gallons of water was released from the spent fuel pool through a vent valve on a failed pump. This event occurred after IE Information Notice 88-65 was issued.

At Surry Unit 1 (10/2/88) a small leak developed in a pneumatic seal in the fuel transfer system as a result of the an accidental pinhole puncture in the single air supply line. The leak was promptly detected and stopped before seal integrity was lost. The reactor cavity seal was not installed at the time and if the seal failed, the loss of water could have lowered the spent fuel pool level to within 13 inches above the top of the stored fuel. IE Information Notice 88-92, "Potential for Spent Fuel Pool Draindown," was issued on November 22, 1988 (Ref. 26) to alert licensees to potential problems resulting from failure of the fuel transfer canal door seal.

The inadvertent draining of a spent fuel pool should be precluded by design or administrative procedures. Antisiphoning devices, or approved system alignment procedures should be in place to assure that the primary safety function of the spent fuel pool is not compromised. To this end, instrumentation to alert operators to potential problems should be operable. IE Information Notice 88-65 identified these issues and all holders of OLs and CPs are expected to review the information and consider appropriate actions to avoid similar problems.

The frequency of a siphoning event was estimated (Ref. 27) to be 0.001 per reactor year, based on a break in the cooling system. An 0.01 conditional failure probability of the cooling system to isolate was assumed. Further it was assumed that the conditional failure probability of the backup make-up system was 0.015. The frequency of a siphoning event was estimated to be 1.5×10^{-7} per reactor year (Ref. 27). Based on a conditional probability of an anti-siphoning check valve failure of 0.08 (Ref. 27), the frequency of this scenario resulting in spent fuel damage is estimated to be 1.2×10^{-8} per reactor year.

The operational experiences concerning spent fuel pool component performance indicates that partial pool draining resulting from non-seal failure related causes, such as inadvertent siphoning, do not result in a significant loss of water from the spent fuel pool (Ref. 28). At San Onofre 2, for example, the 9,000 gallons is equivalent to about 19.5 inches of water. At Davis-Besse, on June 6,1980, an improperly calibrated level alarm did not actuate until the level was 10.25 inches below the Technical Specification limit. At Trojan, June 10,1982, misalignment of the spent fuel pool purification system as a result of not closing the door between the spent fuel pool and the fuel transfer canal resulted in a lowering of the level to 21 feet 9 inches above the top of the stored fuel. The Technical Specification limit is 23 feet. The rate of draining was 132 gpm.

As a result of increased awareness concerning the development and use of proper administrative procedures for system alignments and the use of anti-siphoning devices and the need for spent fuel pool level indication through the issuance of IE Information Notice 88-65, the staff concludes that the frequency of spent fuel damage from inadvertent draining of the spent fuel pool is less than $1x10^{-7}$ per reactor year, and the best estimate value is $1.2x10^{-8}$ per reactor year based on previous estimates (Ref. 27).

4.6.6 Loss of Cooling/Make-Up

The acceptance criteria associated with the general design and operation of the spent fuel pool and it's related support systems, primarily the cooling and make-up systems, are based on the long time interval available to the plant operators to diagnose and correct failures in these systems. In WASH-1400, for example, it was assumed that the likelihood of failure to recover from loss of cooling was $1x10^{-6}$ per event. With an assumed loss of cooling event frequency of 0.1 per reactor year, the probability of damage to the spent fuel was judged to be extremely small, $1x10^{-7}$ per reactor year.

In WASH-1400 the assumed spent fuel pool inventory was limited to about 2/3 of a full core. A pool loading of 1/3 of a core with 150 days decay and 1/3 with three days decay was assumed for the limiting condition. Approximately nine days (216 hours) would be available before the 50,000 cubic feet of water in the spent fuel pool would be completely boiled off. The average pool loading assumption resulted in 3.8 weeks (640 hours) available for the repair of the cooling system and/or water make-up to be accomplished. In WASH-1400 it was believed that spent fuel shipping would be occurring on a weekly basis and the likelihood of failure to recover was estimated to be $1x10^{-6}$.

However, spent fuel is not being shipped within the United States and the spent fuel pool inventories are larger than assumed in WASH-1400. To determine the available time for recovery, a simplified calculation was performed. A 3000 Mw thermal plant is assumed to have discharged 1/3 of a full core annually for a period of 20 years, for a spent fuel pool heat load of 3.5 million BTU/hr of decay heat. Older fuel would not increase the heat load significantly. Based on the pool data provided in Table 4.2.1, it is assumed that there is 32,000 cubic feet of water covering the spent fuel, a 30 foot by 40 foot surface with a depth of 23 feet over the spent fuel. Heatup of only the water is assumed. The results of the calculation are provided in Table 4.6.3 for the case when 1/3 of the core is recently discharged or the case when the full core is discharged. Also provided in the table is the needed make-up rate, in gallons per minute, to match the boil off rate.

Based on Table 4.6.3, the time available to recovery from loss of cooling can be estimated as at least 24 hours for the most limiting case - full core discharge five days following reactor shutdown. At this time the water level covering the spent fuel (approximately ten feet) is adequate to provide shielding to maintain the radiation levels in the spent fuel storage area to a low enough value to permit limited operator access to the SFP area as required to establish make-up. The make-up capacity is less than 100 gpm and the assumption that the fire system can be used to provide make-up appears to be reasonable. It is also noted that the frequency of this limiting condition is estimated to be less than 5% of the lifetime of the facility. Full core offloading is an unusual occurrence, except during ten year inservice inspections of the reactor pressure vessel. Assuming four inspections per life, and eight additional unanticipated offloads, the estimated frequency of having a full core in the spent fuel pool (assuming the full core is in the pool for a period of 30 days) is 1 year per 40 year life or 2.5% of the time. For the typical expected condition of a 1/3 core discharge five days after shutdown, the time available to restore cooling and/or establish make-up is three days. For recovery actions which would not require operators to enter the potentially high radiation area, the available time to recovery cooling would be between two days, the most limiting case, and five days, for a normal refueling case. If the loss of cooling event occurs 30 days after discharge, the recover time intervals nearly double.

The spent fuel pool is equipped with temperature and level instrumentation to warn the operator of a degrading condition. Although these instruments are not safety grade, they do alarm in the control room. The spent fuel pool storage area also contains radiation monitors to alert the operator to degrading situations. These instruments provide the operator with the information necessary to initiate appropriate safety actions to assure that the safety function of the spent fuel pool, to maintain the spent fuel assemblies in a safe and subcritical array during all credible storage conditions, is not compromised. The issuance of IE Information Notice 88-65 (see Section 4.6.5) has emphasized the importance of assuring that the spent fuel pool level instrumentation is operable and that surveillance procedures are in place to assure the the instrumentation is properly maintained.

Operator performance, the human error probability (HEP), is estimated from NUREG/CR-1278 Chapter 12, "Diagnosis of Abnormal Events." (Ref. 23) The nominal screening model is used. Table 4.6.4 lists the median joint HEPs and the error factors for a variety of assumptions for failure to diagnose a loss of cooling event. Typical repair time estimates, as well as failure rates, for the components of the cooling system are provided in Table 4.6.5.

The failure rates in Table 4.6.5 are for all failure modes. For a series system which has one pump, one heat exchanger, one level indicator and four valves, the estimated failure rate of the system is 0.15 per reactor year. The mean time to repair the system is about 34 hours. This is representative of the minimum cooling system allowed under current requirements, and supports the estimated failure rate of a single cooling system of 0.1 per reactor year previously used in WASH-1400 and in NUREG-0933 (Ref. 27). Based on the mean time to repair the spent fuel pool cooling system, 34 hours, it is seen that there is adequate time to repair the cooling system before the spent fuel pool level decrease to a level where spent fuel damage would occur and the make-up rates needed to match the boil off rates are not excessive.

Four "generic" fuel pool cooling and make-up systems have been examined to estimate the possible range of failure frequencies for these systems. The four representative systems, based on current SRP acceptance criteria, are:

System A: Minimum cooling and make-up system requirements. One full capacity cooling train with redundant active components (i.e., valves, pumps, etc.). One Category I make-up system and one backup pump or system (not required to be Category I) which can be aligned to a Category I water supply.

System B: Minimum cooling and make-up system requirements (System A) with credit for make-up from the fire system for recovery.

System C: Typical older system. One cooling train with backup active components (but the backup components are required to supplement cooling about 30% of the time); and, one safety grade make-up system and one non-safety grade make-up system.

System D: Typical older system (System C) with a third make-up train available for recovery (i.e., the fire system).

Systems A and B are not intended to represent actual systems. Rather, they are representative of the minimum requirements in the current SRP.

The failure rates and systems failure frequencies are based on data from WASH-1400 and assumptions used in NUREG-0933 (Ref. 27). Specifically:

1. A 0.1 per reactor year frequency for the initiating event, loss of cooling, is based on WASH-1400 estimates. As discussed above, this is the expected failure frequency for a typical single train system based on typical component failure rates.

2. The conditional failure probability of 0.05 for the second cooling train (Systems A and B) represents a relatively high common mode failure probability. This value was used in NUREG-0933 to assist in the prioritization of this generic issue.

3. The conditional failure probability of 0.3 for the second cooling train (Systems C and D) represents the assumption that both cooling pumps are necessary 30% of the time. Thus a failure of either pump represents a failure of the system.

4. Train 1 of the make-up system is assumed to be independent of the cooling system and is assigned a low common cause contribution. The likelihood of a prolonged station blackout events is assumed to be low. This assumption is supported by a recent study completed by Sandia (Ref. 28). For 63 recorded incidents of loss of off-site power, the longest recovery time reported was about nine hours (for severe weather related losses), with the sample mean recovery time for all causes of 1.2 hours. The conditional probability of 0.015 is used, based on RHR reliability in the LPCI mode (WASH-1400).

5. Train 2 of the make-up system is assigned a conditional failure probability of 0.05. This system is not powered by emergency power buses and may be put out of service by a common mode failure of the spent fuel pool cooling system.

The estimated failure frequency of the spent fuel pool cooling and make-up systems resulting in a heatup of the spent fuel without recovery is summarized in Table 4.6.6 for each of the four systems.

Table 4.6.3
Heatup and Boil Dry Times for a Typical Spent Fuel Pool

	Based on 1/3 Core Discharge				Based on Full Core Discharge			
Days After Shut Down	Q-decay 1/3 core Discharge (BTU/hr)	Heat 125 F - 212F (hrs)	Boil Off Water (hrs)	Make-Up Boil Off (gpm)	Q-decay Full Core Discharge (BTU/hr)	Heat 150F - 212F (hrs)	Boil Off Water (hrs)	Make-Up Boil Off (gpm)
5	1.51×10^7	11.2	125.0	31.9	3.82×10^7	3.1	49.3	81.0
10	1.22×10^7	13.9	154.9	25.8	2.95×10^7	4.1	63.8	62.6
30	8.87×10^6	19.0	212.2	18.8	1.97×10^7	6.1	95.8	41.7
45	7.75×10^6	21.8	242.8	16.4	1.63×10^7	7.4	115.5	34.5
65	6.96×10^6	24.3	270.4	14.8	1.39×10^7	8.6	135.2	29.5
100	6.14×10^6	27.5	306.5	13.0	1.15×10^7	10.5	164.2	24.3
150	5.27×10^6	32.0	357.1	11.2	8.58×10^6	13.6	212.6	18.8
200	4.81×10^6	35.1	391.2	10.2	7.47×10^6	16.1	251.9	15.8
250	4.54×10^6	37.2	414.5	9.6	6.66×10^6	18.1	282.6	14.1
300	4.39×10^6	38.4	428.3	9.3	6.22×10^6	19.3	302.4	13.2
350	4.30×10^6	39.2	437.5	9.1	5.94×10^6	20.2	316.6	12.6
365	4.29×10^6	39.3	438.6	9.1	5.91×10^6	20.4	318.4	12.5

Note: Q-decay includes 3.5×10^6 BTU/hr from 20 years of accumulated discharges.

Table 4.6.4
Nominal HEP Model Estimates for Failure to Diagnose Loss of Cooling

	Following 1/3 Core Discharge			Following Full Core Discharge		
	Time Min	Mean Joint HEP	Error Factor	Time Min	Median Joint HEP	Error Factor
Failure to Diagnosis Loss of Cooling Before Boiling Occurs	600	0.00002	30	180	0.00005	30
Failure to Recovery Before High Radiation Field in SFP Area	>1440	0.00001	30	1440	0.00001	30

Note: The HEP probabilities are based on the nominal HEP model (Ref. 23). Previous estimates of failure to recovery (NUREG-0933, for example) have used a HEP value of 0.05 considering the high temperature and high radiation fields following loss of water from the spent fuel pool, and considering the location of the spent fuel pool in a BWR. Make-up was assumed to be from a fire hose.

Table 4.6.5
Typical Failure Rates and Repair Times for Cooling System Components
(Taken from EPRI NP-3365)

Component	Failure Rate (per hour)	Range (per hour)	Average Repair Time (hours)
Piping (per 10 ft section)	$3x10^{-10}$	$1x10^{-11}$ to $3x10^{-8}$	30
Pump	$1x10^{-5}$	$3x10^{-6}$ to $3x10^{-5}$	40
Heat Exchanger	$3x10^{-6}$	$1x10^{-7}$ to $1x10^{-4}$	30
Valves (per valve)	$1x10^{-6}$	$3x10^{-7}$ to $3x10^{-6}$	24
Instrumentation (per channel)	$1x10^{-6}$	$3x10^{-7}$ to $1x10^{-5}$	6

Table 4.6.6
Failure Frequency of Generic SFP Cooling and Make-Up Systems Without Recovery

System	Cooling Train 1 (per R-y)	Cooling Train 2 (per demand)	Make-up Train 1 (per demand)	Make-up Train 2 (per demand)	Frequency of Heatup (per R-y)
A/B	0.1	0.05	0.015	0.05	$3.8x10^{-6}$
C/D	0.1	0.3	0.015	0.05	$2.2x10^{-5}$

The probability of the complete loss of the cooling and the make-up systems resulting in spent fuel damage is dependent on actions taken by the plant operators to either restore the cooling and/or make-up system or provide an alternative method for make-up or cooling, for example use of the station fire fighting system or use of a portable backup pump. In NUREG-0933, a conditional failure probability of 0.05 was assigned to the operator failure to accomplish recovery actions. The resulting frequency of fuel damage was estimated to be $1.9x10^{-7}$ per reactor year for System B (System A with recovery) and $1.1x10^{-6}$ per reactor year for System D (System C with recovery). The 0.05 conditional failure probability of recovery before fuel damage was based on the assumption that the spent fuel pool area environmental conditions would eventually become severe enough to make it difficult to setup the fire hose for make-up. The NUREG-0933 (Ref. 27) evaluation did not consider the time available to diagnose and take corrective or recovery actions.

Based on the time available to diagnose a loss of cooling event, at least three hours before boiling occurs, the HEP value for failure to diagnose is 0.0015 (median value for full core discharge with error factor). For the 1/3 core discharge case, these values are lower by a factor of about two. The probability of a pool heatup event resulting in boiling of the water in the spent fuel pool is therefore estimated to be $5.7x10^{-9}$ per reactor year for System A ($3.8x10^{-6}$ x $1.5x10^{-3}$) and $3.5x10^{-8}$ per reactor year for System C ($2.2x10^{-5}$ x $1.5x10^{-3}$).

The spent fuel pool cooling and make-up systems are designed to meet one of two basic sets of requirements. The first basis requires the cooling portion of the system to be designed to seismic Category I, Quality Group C requirements. The second basis allows a non-seismic Category I, Quality Group C spent fuel pool cooling system provided that the following systems are designed to seismic Category I requirements and are protected against tornadoes: the fuel pool make-up water system and its sources; and, the fuel pool building and its ventilation and filtration system. The make-up, ventilation and filtration systems must also withstand a single active failure.

Since Generic Issue 82 is concerned with risk from beyond design basis events, LLNL evaluated the probability of a beyond design basis seismic event resulting in a loss of cooling event and subsequent pool heatup transient (Ref. 17).

The cooling and make-up water systems for two pools were reviewed, one for a BWR system and one for a PWR system. Event and fault tress were constructed by LLNL to identify the accident sequences that result from failure of these systems. For components appearing in these accident sequences, seismic fragilities were estimated based on design information and plant walkdowns. Boolean equations developed from the fault tree analysis were quantified using the seismic fragilities and preliminary hazard curves for the two sites. The dominant components to spent fuel pool system failures were found to be similar to components that have been found to contribute to seismic risk in several PRAs concerned with reactor core damage; poorly anchored electrical equipment and tanks. The components which contribute significantly to the seismic induced failure of the spent fuel pool support systems are the non-safety electrical systems in the plant; the motor control centers, switchgear and station service transformers which have relatively low seismic capacities.

The estimated mean probability of a beyond design basis seismic event resulting in loss of cooling, combined with operator failure to properly align normal make-up, was found to be on the order of 1.5×10^{-4} per reactor year for both systems. A sensitivity study without operator failure rates or random failures showed little change in the mean probability estimate (Ref. 17). The median (50-th percentile) probability estimate was found to be about an order of magnitude lower than the mean value. For the seismic induced loss of cooling event to result in damage to spent fuel, the previously determined nominal HEP (with error factor) failure to diagnosis and failure of the second make-up train are used. Using the values of 0.0015 for failure to diagnose and 0.05 for failure of the second make-up train to recover, the estimated probability of fuel damage is therefore $1.5 \times 10^{-4} \times 0.0015 \times 0.05$, or 2.1×10^{-8} per reactor year. Based on the estimated failure frequencies used in Table 4.6.6 above, it would appear that common mode failures in the non-safety electrical systems resulting from a beyond design basis seismic event, at least for the two plants studied, may have been underestimated by a factor of two for System A ($0.1 \times 0.05 \times 0.015$, or 7.5×10^{-5} per reactor year without credit for make-up train 2 as compared to 1.5×10^{-4} per reactor year). The combined best estimate probability of spent fuel damage from loss of cooling, from component failures and beyond design basis seismic events, is therefore estimated to be 4×10^{-8} plus 2×10^{-8} or 6×10^{-8} per reactor year. Previous estimates, based on a conditional probability of failure to recover from a loss of cooling event value of 0.05 without consideration for event diagnosis and using the systems data in Table 4.6.6, provide an upper bound probability estimate of 1.4×10^{-6} per reactor year.

4.6.7 Structural Failure of SFP From Beyond Design Basis Earthquakes

The probability of failure of a structure is related to the functional relationships between the various physical parameters and the variabilities in the parameters themselves. Two types of variability are considered. The first variability is that which is potentially reducible and is called the uncertainty. The additional component, which cannot be practically reduced, is called randomness.

The seismic analysis of a structure includes two parts. The first is the structural capacity and the response of the structure to a seismic event. This is referred to as fragility. The second is the seismic input and site response to an earthquake, the seismic hazard analysis. The hazard analysis is comprised of two parts. The site response (peak ground acceleration, response spectra, and frequency for both horizontal and vertical ground motions) and the annual frequency of exceeding a given peak ground acceleration, the seismic hazard curve. Because of uncertainty, both the fragility and seismic hazard analyses are described by families of curves. Each curve representing different confidence levels. The resulting combination of these two sets of curves in the PRA can yield large differences in the estimated probability of structure failure dependent on the confidence level. While the seismic hazard curves and the parameters used to define the fragility are developed and expressed in terms of median values, the mean value is used in PRA applications.

The NRC Seismic Margins Program has developed an additional measure of importance for assessing seismic risk. This measure is the high confidence of low probability of failure (HCLPF), defined in terms of the peak ground acceleration (g value). This value is derived from the fragility analysis and is independent of the site seismic hazard curves, the annual frequency of exceedance of a given g value. The HCLPF value is defined as the peak ground acceleration at which there is a 95% confidence that failure will not occur. This value may be compared to the safe shutdown earthquake peak ground acceleration used in the deterministic analysis to determine the margin in excess of the SSE for which no structural failure is anticipated.

A comprehensive assessment of uncertainty and conservatism in the seismic analysis and design of nuclear facilities has been prepared by the American Society of Civil Engineers (ASCE) (Ref. 29). The objectives of this study were to:

1. Identify sources of uncertainties present in seismic analysis and design,

2. quantify uncertainties, when possible, and recommend actions where data are missing, and

3. Identify the status of current analysis and design methods relative to the scatter of data for known sources of uncertainty.

The current practices employed to determine the seismic input and site response produce conservative (e.g., 84th percentile or greater), not median values, of the design seismic input. Empirical procedures for structural design provide an additional margin of safety across the entire design response spectrum in earthquake-resistant designs. The ASCE (Ref. 29) concluded that a nuclear power plant having a design seismic input value of 0.25 g may actually be able to withstand much larger values of peak ground acceleration.

Some siting procedures differ in the western United State because there is a strong ground motion data base available. Less extrapolation is required and tectonic faults and structures are also much easier to identify. The design seismic input is still conservatively evaluated in comparison to historical data.

The ASCE estimated the median factors of safety and logarithmic standard deviations associated with the safe shutdown earthquake (SSE), based on post 1973 seismic design methods. These factors are provide in Table 4.6.7 (from Table 9.1, Ref. 29).

The probability of gross structural failure of the spent fuel from a beyond design basis earthquake was estimated by LLNL (Ref. 17) for a typical, although older, elevated BWR spent fuel pool and a typical, older PWR spent fuel pool.

In analyzing the failure of the spent fuel pool structures and systems, LLNL considered the following:

(a) Loss of liner integrity precipitated by structural failure of the spent fuel pool.

(b) Loss of function of the fuel pool support systems (e.g. pool cooling and make-up water capacity) resulting in loss of water through boil-off or drainage.

(c) Damage to fuel racks caused by fuel rack motion.

The failure modes of the spent fuel pool structures were determined by LLNL. For the BWR Mark I and Mark II elevated spent fuel pools, the failure mechanisms which need to be considered are:

1. The failure mode of the pool floor is that of a slab fixed at the four edges. The girders supporting the pool are in reality long walls acting as deep girders and are supported by the peripheral walls of the reactor building.

2. Compressive and shear stresses at the reaction points of the girders (onto the reactor building walls) for transmitting vertical and horizontal seismic loads from the storage pool to the foundation needs to be considered for structural adequacy.

3. Due to large concentrated loads (50 to 70 kips) at each foot of the storage rack, bearing and punching shear stress in the pool floor should also be investigated.

4. For laterally braced high density fuel storage racks, large concentrated loads are transmitted to the pool wall at either the base level or at the base level and the upper seismic bracing level. The effect of concentrated load needs to be investigated.

Although thermal loads are important in the design of the spent fuel pools, their influence on the fragility of the pool is judged not significant by LLNL because the thermal loads are self-relieving.

For BWR Mark III and PWR storage pools, which are general on or below grade, the failure modes for the pool floor are:

1. Punching shear stress due to concentrated loads at the foot of the storage racks.

2. Foundation settlements for soil; soil settlement may only be an issue for piping relative displacements.

3. Failure modes for the walls are similar to that described for the BWR Mark I and II designs.

Possible failure modes for the liner plate identified by LLNL are:

1. Tearing of the liner plate or seam welds at the leak channels due to vertical/horizontal loads from fuel storage racks; this is of concern only if the rack slides and the foot bears on the leak channel.
2. Tearing of the liner plate due to sliding of the rack over any floor depression or wrinkles in the liner plate.

3. For a laterally braced rack, puncturing of the liner plate at the knuckle in the vicinity of the pool floor/walls intersection.

The failure modes of free standing or sliding racks, and for laterally braced high density fuel racks were considered by LLNL. Based on information provided to the NRC staff concerning a reracking amendment by a licensee (Ref. 30), LLNL concluded that the peak ground acceleration would have to exceed 1.5 to 2.0 g before failure of the free standing racks would occur. The median acceleration capacity of the racks for incipient impact with the pool wall is estimated to be 1.0 g, and it would require 1.5 to 2.0 times this value to cause impact and damage. Even then, the fuel rack design is such that the assembly cannot be compressed into a critical mass. LLNL therefore concluded that crushing of fuel and assemblies is not a credible failure mode of the spent fuel pool system. Also, the failure of the spent fuel pool liner plate resulting from movement of the spent fuel storage racks, either from sliding or puncturing, is not expected to result in the sudden or rapid drainage of the water from the spent fuel pool.

The actual potential failure modes of the BWR spent fuel pool studied by LLNL included the following:

(a) Out of plane shear failure of the pool floor slab.

(b) Out of plane bending failure of the pool floor slab.

(c) Punching shear failure of the pool floor slab under the fuel rack support pad.

(d) Out of plane bending failure of the south pool wall.

(e) Bending and shear failure of the girder under the south wall.

(f) Bending and shear failure of the girder under the east pool wall.

(g) Overall transfer of N-S and E-W inertial loads to the reactor building.

The controlling failure mode with the lowest seismic capacity was determined to be the out of plane shear failure of the pool floor slab. The slab was evaluated for out of plane loading resulting from dead weight load plus seismic load. Sources of dead loads are the weights of the slab, grout, water, fuel racks, and attached equipment. Sources of seismic loads are (a) vertical seismic response of the slab and attached masses, (b) fluid impulse and convective mode responses induced by horizontal seismic excitation, and (c) horizontal seismic response of the spent fuel racks.

The PWR spent fuel has two features which are not typical. The storage racks are both low density and high density in design. And to accommodate the region of high density racks, a support column was added beneath the spent fuel pool floor, in the waste gas holdup tank room below the spent fuel structure. The pool floor is actually six feet above grade at Robinson.

The actual potential failure modes of the PWR spent fuel pool studied by LLNL included the following:

(a) Out of plane bending of the East or South wall.

(b) Out of plane shear failure of the East or South wall.

(c) Out of plane shear failure of the pool floor slab.

(d) Out of plane bending failure of the pool floor slab.

(e) Overall seismic stability of the spent fuel.

The out of plane bending failure of the east wall was determined to be the failure mode with the lowest seismic capacity. The East or South wall resists the lateral forces of the old fuel racks. It was modeled by LLNL as a slab fixed on three sides and free on top. The loads considered in the evaluation of the seismic capacity of this wall are (a) hydrostatic loads (normal water level), (b) hydrodynamic loads induced by horizontal and vertical accelerations in earthquakes, (c) wall inertia force, and (d) reaction forces from the old fuel racks.

The fragility of a structure, or component, is expressed in terms of its median factors, A_m, β_R, and β_U. A_m is the median ground acceleration at which the probability of failure is 0.5. β_R and β_U are the random variability and the uncertainty in the median capacity based on a lognormal model. β_R and β_U are the logarithmic standard deviations of the median value. Using the lognormal model a high confidence of low probability of failure (HCLPF) capacity factor is defined, HCLPF capacity = $A_m \exp(-1.64 (\beta_R+\beta_U))$. The HCLPF value is defined as the peak ground acceleration at which there is a 95% confidence that failure will not occur. This value may be compared to the safe shutdown earthquake peak ground acceleration used in the deterministic analysis to determine the margin in excess of the SSE for which no structural failure is anticipated. This fragility model and development of fragility parameters have been utilized in over 25 seismic PRAs.

The median factors of safety and variabilities of the spent fuel pool structure for the elevated BWR were found to be A_m = 1.4 g, with β_R = 0.26 and β_U = 0.39. The HCLPF value is 0.5 g. For the PWR spent fuel pool, A_m = 2.0 g, with β_R = 0.28 and β_U = 0.40. The HCLPF value is 0.65 g. The HCLPF value shows a design margin of a factor of at least three over the SSE design peak ground acceleration for either pool. Typical SSE design values for LWRs are in the 0.15 to 0.2 g range. The BWR used in this study has an SSE value of 0.14 g, and the PWR has an SSE of 0.2 g.

Preliminary seismic hazard curves for the two sites were used to estimate the probability of failure of the spent fuel pool structures from beyond design basis earthquakes. These preliminary curves have not been finally reviewed by the NRC and were used by LLNL only to obtain a better understanding of the seismic induced spent fuel pool failure. The hazard curves may change after NRC review and guidance for their proper use will be developed. They are however expected to be a reasonable representation of the seismic characteristics of the sites. A recently published report (NUREG/CR-5042, Supplement 1, Ref. 31) compares the NRC and EPRI preliminary estimates for the annual probability of exceedance at the SSE earthquake level for nine reactor sites. The differences between the NRC and EPRI estimates are reasonable, with no greater than about one order of magnitude difference between the estimates at any confidence level. The median (50%), 85% and 15% confidence levels were compared.

The resulting annual seismic failure frequencies for the two pools are provided in Table 4.6.8. The results are provided for a variety of confidence levels and for a variety of cutoff values. The use of a cutoff value of less than 100% demonstrates the sensitivity of the analysis to the extreme tails in the seismic hazard curves and fragility curves. LLNL recommends use of the 99% cutoff value based on their experiences with seismic PRAs (Ref. 17). The mean failure frequency at the 99% cutoff value is used for this Regulatory Analysis.

The LLNL study used two representative spent fuel pools. These pools have been designed to the seismic design criteria existing in the late 1960s. Their large seismic capacities lead LLNL to conclude that the pools designed to current seismic standards (post 1973) should have higher seismic capacities and should not contribute significantly to seismic risk. Based on the demonstrated relatively high seismic capacity (the HCLPF capacity of the pool structures are estimated to be more than three times the SSE value), LLNL also concluded that the risk contribution from spent fuel pool structural failures is negligibly small (Ref. 17). In addition, the results obtained for the two pools studied also fall within the margins estimated by the ASCE for nuclear power plant seismic structures. The margins, based on median capacities, are about 8 for the BWR and 10 for the PWR which fall within the estimated structural factor range of 4 to 19 (see Table 4.6.8).

Table 4.6.7
Estimated Median Factors of Safety and Logarithmic Standard Deviations Associated With the Safe Shutdown Earthquake (SSE) (Post 1973 Seismic Design Methods)

Item	Median Factor of Safety	Logarithmic Stnd. Deviation
Structures		
Capacity		
Ultimate Strength vs Code Allowable	1.2 - 2.5	0.16 - 0.20
Inelastic Energy Absorption Capacity	1.8 - 4.0	0.20 - 0.30
Total Capacity Factor (1)	2.5 - 6.0	0.28 - 0.34
Response		
Design Response Spectra	1.2 - 1.6	0.25 - 0.40
Damping Effects	1.2 - 1.4	0.09 - 0.20
Modeling Effects	1.0	0.10 - 0.20
Modal and Component Combination	1.0 - 1.1	0.15 - 0.20
Soil-Structure Interaction	1.1 - 1.5	0.10 - 0.40
Total Response Factor (1)	1.6 - 3.2	0.40 - 0.59
Total Structural Factor (1)	4 - 19	0.52 - 0.65
Mechanical Equipment		
Capacity Factor	1.5 - 8.0	0.28 - 0.34
Building Response Factor	1.6 - 3.2	0.40 - 0.59
Floor Spectra Factor	1.4 - 1.6	0.25 - 0.35
Total Mechanical Equipment Factor	3.5 - 40	0.59 - 0.72

Notes:
(1) These total factors are the product of the preceding individual factors upon which they are based. However, a range is shown for each of the individual factors and it is not reasonable to assume all the individual factors would concurrently be at either their lowest or highest values. Judgment was introduced in establishing the expected range of these factors. For example, the estimated range on the structural median capacity factors of 2.5 to 6.0 is less than would be obtained from concurrently using either the lowest or highest values of the strength and energy absorption which would produce a median capacity factor range of 2.2 to 10.0.

Table 4.6.8
Annual Seismic Failure Frequencies
for Two Representative Spent Fuel Pools

Pool	Fragility Data A_m (g)	β_R (-)	β_U (-)	HCLPF (g)	Cutoff Value (%)	Mean 1/R-y	Failure Frequencies 5% 1/R-y	50% 1/R-y	95% 1/R-y
BWR	1.4	0.26	0.39	0.5	100	$3.8x10^{-5}$	$1.8x10^{-11}$	$7.7x10^{-8}$	$3.6x10^{-5}$
					99	$6.7x10^{-6}$	$3.1x10^{-11}$	$8.3x10^{-8}$	$1.9x10^{-5}$
					97	$3.8x10^{-6}$	$3.1x10^{-11}$	$7.7x10^{-8}$	$1.4x10^{-5}$
PWR	2.0	0.28	0.40	0.65	100	$8.6x10^{-6}$	$6.1x10^{-12}$	$1.3x10^{-8}$	$8.6x10^{-6}$
					99	$1.8x10^{-6}$	$9.9x10^{-12}$	$1.5x10^{-8}$	$5.0x10^{-6}$
					97	$9.9x10^{-7}$	$9.5x10^{-12}$	$1.4x10^{-8}$	$3.5x10^{-6}$

4.7 Summary of Accident Sequence Quantification

The frequency of spent fuel damage resulting from accident sequences which can result in the loss of water from the spent fuel pool through either drainage or through boiling as a result of loss of cooling are summarized in Table 4.7.1. HEP failure to diagnose values, including error factors, based on the nominal HEP model (Ref. 23) have been used to develop the best estimate accident frequencies and are based on the most limiting condition in the spent fuel pool - a full core discharge into a pool containing 20 years of spent fuel. The upper bound frequency values represent previous estimates from WASH-1400, NUREG-0933 and NUREG/CR-4982. In general, these previous studies did not consider the time available for recovery actions but, in general, used intentionally conservative assumptions regarding operator performance.

Table 4.7.1
Summary of SFP Accident Frequencies

Accident Sequence	PWR Frequency		BWR Frequency	
	Best Estimate (per R-year)	Upper Bound (per R-year)	Best Estimate (per R-year)	Upper Bound (per R-year)
Structural Failures				
1. Missiles	$1.0x10^{-8}$	$1.0x10^{-7}$	$1.0x10^{-8}$	$1.0x10^{-7}$
2. Aircraft crashes	$6.0x10^{-9}$	$2.0x10^{-8}$	$6.0x10^{-9}$	$2.0x10^{-8}$
3. Heavy Load Drop	$3.1x10^{-8}$	$3.1x10^{-7}$	$3.1x10^{-8}$	$3.1x10^{-7}$
Pneumatic Seal Failures	$3.0x10^{-8}$	$5.0x10^{-7}$	$3.0x10^{-8}$(1)	$5.0x10^{-7}$(1)
Inadvertent Drainage	$1.2x10^{-8}$	$1.0x10^{-7}$	$1.2x10^{-8}$	$1.0x10^{-7}$
Loss of Cooling/Make-up	$6.0x10^{-8}$(2)	$1.4x10^{-6}$	$6.0x10^{-8}$(2)	$1.4x10^{-6}$
Total	$1.5x10^{-7}$	$2.4x10^{-6}$	$1.5x10^{-7}$	$2.4x10^{-6}$
Seismic Structural Failure	$1.8x10^{-6}$		$6.7x10^{-6}$	
Conditional Probability Of Zircaloy Cladding Fire Given Loss of Water (High Density Storage Racks)	1.0		0.25	

Notes: (1) BWRs do not, in general, use pneumatic refueling cavity seals, but other pneumatic seals are used in the transfer canal.
(2) Includes beyond design basis seismic induced loss of cooling and make-up.

4.8 Radiological Consequences Evaluation

The inventory of radionuclides contained in spent fuel assemblies depends on the operating history and the size of the plant. During refueling, the freshly discharged fuel contains a large inventory of isotopes with short half-lives in the range of approximately one to thirty days. These isotopes decay over the course of a year, until the next refueling outage.

The older fuel contains radionuclides which have longer half-lives. The older fuel approaches a decay rate which is inversely proportion to the decay time. For example, after four years, the spent fuel contains approximately one-forth of the specific activity of one-year old fuel.

During each refueling outage approximately one-third of a PWR core and about one-forth of a BWR core are off-loaded to the spent fuel storage pool. It is noted that releases for an accident involving the reactor core are basically noble gases and halogens, while for a spent fuel storage pool accident the releases are primarily alkali metals (such as cesium, Cs) and alkali earths (such as strontium, Sr). Therefore, it may not be appropriate to directly relate the probability of a spent fuel storage pool accident to a core damage accident because of the different radionuclides involved.

4.8.1 Radionuclide Inventories

The ORIGEN2 computer program (Ref. 12) was used by BNL to determine the radionuclide inventory of the spent fuel as a function of decay time. Separate inventories were calculated for activation products in the fuel assembly hardware and cladding, and for the fissions products and actinides sealed in the fuel elements. The data was obtained for a reference BWR and a reference PWR. Millstone Unit 1 and R. E. Ginna were selected by BNL as the reference plants for the source term evaluation. A comparison of the radionuclide inventory at different decay times, up to the time when the spent fuel storage pool reaches a capacity load, to the equilibrium inventory of a reactor core is provided in NUREG/CR-4982 (Ref. 10).

4.8.2 Radionuclides Potentially Available for Release

The source term for any postulated accident sequence is defined in terms of:

- the amount (curies) of each radionuclide,
- the composition, physical and chemical form of each radionuclide, and,
- the time and the duration of the release of the radioactivity to the environment.

The physical and chemical processes that would take place in a drained spent fuel storage pool are not well characterized at the present time. It was therefore necessary for BNL to use engineering judgment to estimate the source term. The SFUEL1W computer program does not account for relocation of the reaction products (molten un-oxidized cladding, fuel dissolved in molten zirconium, etc). Also the degree to which exposed UO_2 would oxidize to U_3O_8 and reduce the release of less volatile fission products has not been studied. The estimate of the fraction of each radionuclide release was determined by BNL based on available data and on engineering judgment, and is provided in Table 4.8.1 (from NUREG/CR-4982, Table 4.2, Ref. 10).

Table 4.8.1
Estimated Radionuclide Release Fractions During a Spent Fuel Pool Accident Resulting in Complete Destruction of the Fuel Cladding

Chemical Family	Element or Isotope	Release Fractions(1) Value Used	Uncertainty Range	
Noble gases	Kr, Xe	1.0	0	
Halogens	I-129, I-131	1.0	0.5 -	1.0
Alkali metals	Cs, (Ba-137m) Rb	1.0	0.1 -	1.0
Chalcogens	Te, (I-132)	0.02	.002 -	0.2
Alkali earths	Sr, (Y-90), Ba (in fuel)	$2x10^{-3}$	$1x10^{-4}$ -	$1x10^{-2}$
	Sr, Y-91 (in cladding)	1.0	0.5 -	1.0
Transition Elements	Co-58 (assembly hardware)	0.1	0.1 -	1.0
	Co-60 (assembly hardware)(2)	0.12	0.1 -	1.0
	Y-91 (assembly hardware)	0.1	0.1 -	1.0
	Nb-95, Zr-95 (in fuel)	0.01	$1x10^{-3}$ -	$1x10^{-1}$
	Nb-95, Zr-95 (in cladding)	1.0	0.5 -	1.0
Miscellaneous	Mo-99	$1x10^{-6}$	$1x10^{-8}$ -	$1x10^{-5}$
	Ru-106	$2x10^{-5}$	$1x10^{-6}$ -	$1x10^{-4}$
	Sb-125	1.0	0.5 -	1.0
Lanthanides	La, Ce, Pr, Nd, Sm, Eu	$1x10^{-6}$	$1x10^{-8}$ -	$1x10^{-5}$
Transuranics	Np, Pu, Am, Cm	$1x10^{-6}$	$1x10^{-8}$ -	$1x10^{-5}$

Notes: (1) Release fractions of several daughter isotopes are determined by their precursors, e.g., Y-90 by Sr-90, Tc-99m by Mo-99, Rh-106 by Ru-106, I-132 by Te-132, Ba-137m by Cs-137, and La-140 by Ba-140.

(2) Release fraction adjusted to account for 100% release of the small amount of Co-60 contained in the Zircaloy cladding.

4.8.3 Estimated Releases and Consequences for SFP Accidents

The dose equivalent of the release estimates depends on many factors including the location of the spent fuel storage pool and equipment operability (for example, with and without filters in the fuel storage structure). Cesium, for example, is expected to be released as an aerosol and filters may provide an effective removal mechanism. If the fuel storage building structure cracks or if fans fail to function due a seismic event, the release may be substantial. The predicted release to the environment was estimated for each of several accident categories.

The radionuclide inventories for both the BWR and the PWR spent fuel pools were calculated using the ORIGEN2 computer program and the actual operating and discharge histories for a BWR and a PWR. For both plants, the noble gases and halogens in the spent fuel inventory are a small fraction of the inventory in an equilibrium core at shutdown, except for the freshly discharged fuel. The cesium and strontium inventories are more than three times the equilibrium inventory, as a result of the large inventories of spent fuel in the pool (the calculation were based on 11 fuel cycles for the BWR and 16 fuel cycles for the PWR).

A re-evaluation of the cladding fire propagation estimate (given the complete loss of water from the spent fuel pool) indicates that, with the use of high density storage racks, there is a substantial likelihood of propagation to adjacent fuel bundles that have been discharged within the last one or two years. Subsequent propagation to even lower power bundles by thermal radiation is highly unlikely, but with a substantial amount of fuel and cladding debris on the pool floor, the coolability of even these lower power bundles is uncertain.

The fission product release fractions were calculated for two limiting cases in which a Zircaloy cladding fire is assumed to occur. In the first case (1) the cladding combustion is assumed to propagate throughout the entire pool and the entire inventory is involved. In the second case (2) the inventory is limited to only the most recently discharged fuel batch. Parametric studies were performed by BNL to evaluate the consequences as a function of the time of the postulated accident.

In calculating the consequences of a spent fuel pool accident, BNL has assumed no credit for the ventilation and filtration systems in the fuel storage building. While these systems are designed to mitigate the consequences of a fuel handling accident, the design of these systems does not consider the high temperature conditions of a Zircaloy cladding fire (in excess of 2000 ^{o}F). Fission product retention under these conditions is questionable. A sensitivity study with a decontamination factor of 10 was performed by BNL (Ref. 10), to demonstrate the possible affect of fission product retention on structures.

For a Zircaloy cladding fire to occur, the fuel must be recently discharged (between 30 and 180 days in a cylindrical BWR configuration, and between 30 and 250 days in a cylindrical PWR configuration). Since the spent fuel is stored in high density racks, the probability for a Zircaloy fire in a PWR is assumed to be 1.0, given a complete loss of water. For a BWR, which uses directional racks, the probability of a Zircaloy fire is assumed to be 0.25, given a complete loss of water. This value is selected from the full core discharge calculation performed by Sandia (Ref. 8) with the channel box attached (see Table 4.5.1, above), and is also representative of the average values for the cylindrical BWR configuration with different inlet orifice sizes.

For a less severe accident in which the fuel is exposed to air but does not reach temperatures high enough to ignite the Zircaloy cladding, fuel pin failure could occur resulting in a release of the noble gases and halogens. Two cases have been considered by BNL. In the first case the entire pool is assumed to be drained but the decay period is one year since the last discharge and 50% of the pins are assumed to perforate or rupture. In the second case it is assumed that only part of the fuel is uncovered 30 days after the last discharge and all the rods fail. The consequences of either of these two cases are small and result from the release of noble gases.

Sensitivity analyses of the offsite consequences for the Zircaloy fire cases were calculated with the CRAC2 computer program by BNL to study the affect of the source term assumptions on the population dose and interdiction area (Ref.13). The results of this study are provided in Table 4.8.2, along with the results for a case which represents cladding rupture only. The following assumptions were used by BNL (Ref. 10):

- a generalized site surrounded by a constant population density of 100 persons per square mile within a 50 mile radius;

- generalized meteorology (a uniform wind rose, average weather conditions);

- the population in the affected zones are relocated after 24 hours, persons expected to receive more than 25 rem from ground shine in seven days.

There are several unusual characteristics of a severe accident that cause somewhat unexpected results in the radiation exposure calculations. The radiation exposure is relatively insensitive to fairly large variations in the estimated release. This is due principally to the health physics assumptions within CRAC2. This has also been seen in calculation related to fission product releases from core damage accidents. The CRAC2 code assumes that decontamination will limit the exposure of each person to 25 rem. For the long lived isotopes (predominately cesium), the exposure is due mainly to exposure after the area is decontaminated and people return to their homes. Thus, for this type of release the long term whole body dose is limited by the population in the affected sectors (about 0.8 million people in three of the 16 sectors downwind of the release within a 50 mile radius) or about 3,000,000 person-rem (Table 4.8.2).

Another measure of the consequences of a spent fuel pool accident is the interdiction area (the area with such a high level of radiation that it is assumed that it cannot be decontaminated). The interdiction area is sensitive to the source term as shown in Table 4.8.2.

Additional consequences calculations were performed by BNL (Ref. 13) using the MACCS computer program and are provided in Table 4.8.3. The MACCS computer program models are described in NUREG-1150, Appendix O (Ref. 14). The source term data developed by BNL in NUREG/CR-4982 (Ref. 10) was used for these new calculations. These calculations were performed for the value impact studies in Section 5 based on the a site population density of 340 people per square mile, the mean population density around nuclear power plant sites projected for the year 2000 (Ref. 15), Case 1. In addition, the offsite property damage cost for a spent fuel pool accident was also calculated with MACCS for use in the value impact studies in Section 5. A worst case analysis was also performed, assuming the entire spent fuel pool inventory is released at a high population site (Zion, Illinois, 860 people per square mile population density), Case 2.

A direct comparison of the consequences of a severe accident in a spent fuel storage pool to the consequences of a severe core accident can be misleading. For the spent fuel pool accident, there are no "early" fatalities and the risk of early injury is negligible. For a severe core damage accident, early fatalities and early injury are part of the risk due to the presence of the shorter lived isotopes.

Table 4.8.2
Offsite Consequences of Spent Fuel Pool Accidents - CRAC2 Results

Case	Description	Whole Body Dose (person-Rem per-Event)	Interdiction Area (square miles)
1A.	Total inventory 30 days afte last discharge	$2.6x10^6$	244
1B.	Total inventory 90 days after last discharge	$2.6x10^6$	215
2A.	Last discharge 90 days after last discharge	$2.3x10^6$	44
2B.	Last discharge 90 days after last discharge, DF of 10 reduction	$1.1x10^6$	4
3.	50% of all fuel rods leak 1 year after last discharge	4.0	0

Note: Sensitivity study based on population density of 100 people per square mile within a 50 mile radius of the site, from Reference 10.

Table 4.8.3
Offsite Consequences of Spent Fuel Pool Accidents - MACCS Results

Case	Description	Whole Body Dose (person-Rem per-Event)	Offsite Property Damage (1983 $s)
1.	Best Estimate Consequences Last discharge (1/3 core) 90 days after discharge 50 mile radius Based on 340 people/square mile	$8.0x10^6$	$3.4x10^9$
2.	Worst Case Estimate Consequences Total inventory 30 days after discharge, 50 mile radius Based on 860 people/square mile	$2.6x10^7$	$2.6x10^{10}$

4.8.4 Summary Conclusions on Fuel Damage and Consequences

The conditional probability of a Zircaloy cladding fire, given a complete loss of water from a spent fuel pool, is estimated to 1.0 for PWRs with high density storage racks and 0.25 for BWRs with directional storage racks (with the channel box in place in the assembly).

The propagation of the fire to older stored spent fuel assemblies is predicted to occur for spent fuel that has been stored less than two years, under some conditions. Propagation can occur as a result of radiative heat transfer from the hottest fuel assemblies to the older assemblies if the decay heat level of the older assemblies is sufficient to heat the cladding to within 100 to 200 ^oC (200 to 400 ^oF) of the self-sustaining oxidation temperature of 900 ^oC (1650 ^oF).

The best estimate of the consequences of a spent fuel pool accident which results in fuel damage is $8.0x10^6$ person-rem with an offsite property damage estimate of $3,400 million (1983 $s). The best estimate is based on a population density of 340 people per square mile within a 50 mile radius from the site and is a result of the release of radionuclides from the last fuel discharge (1/3 of a reactor core), 90 days after being discharged. Although propagation of a Zircaloy cladding fire to one-to-two year old fuel by thermal radiation can occur, the older fuel would have to be next to the hottest assemblies. Subsequent propagation to even lower power assemblies by thermal radiation is highly unlikely, but with a substantial amount of fuel and cladding debris on the pool floor, the coolability of even these lower power bundles is uncertain.

A worst case estimate of the consequences is based on a population density of 860 people per square mile within a 50 mile radius from the site and is a result of the release of radionuclides from the entire pool inventory, with the last fuel discharge being 30 days old. The consequences are estimated to be $2.6x10^7$ person-rem with an offsite property damage estimate of $26,000 million (1983 $s).

4.9 Other Issues Concerning Use of High Density Storage Racks

4.9.1 Gaps in Neutron-Absorbing Materials

Board Notification 87-011 (Board Notification Regarding Anomalies in Boraflex Neutron Absorbing Material, June 15, 1987, Ref. 32) and IE Information Notice 87-43 (Gaps in Neutron-Absorbing Material In High-Density Spent Fuel Storage Racks, September 8, 1987, Ref. 33) have been issued by the NRC to alert licensees of anomalies in boraflex neutron absorbing material used in the construction of high density storage racks in the Quad Cities Unit 1 spent fuel pool. The gaps were inferred from anomalies in "blackness" testing results and confirmed by underwater neutron radiography. The material supplier and the Electric Power Research Institute (EPRI) have undertaken research programs to collect data and information on gap formation, including the effects of rack fabrication methods and irradiation damage mechanisms.

Boraflex is also used in the Turkey Point spent fuel pool racks and in the Point Beach spent fuel pools. At Point Beach, some deterioration of the samples inserts were noticed during surveillance testing.

These anomalies do not impact on the finding concerning loss of water from a spent fuel pool.

4.9.2 Potential for High Radiation Fields

IE Information Notice 87-13, "Potential for High Radiation Fields Following Loss of water From Fuel Pool," February 24, 1987 (Ref. 34), was issued to alert licensees of the potential for high radiation fields following the inadvertent loss of water from the spent fuel pool or transfer canal. Following the seal leakage at Hatch, the licensee determined that potentially high radiation fields could exit in the spent fuel area as a result of irradiated control blades being stored in the spent fuel pool on short hanger. Some of the control blades could have been completely uncovered if the water level dropped to the bottom of the transfer canal.

In determining the frequency of loss of cooling events and in evaluating HEP diagnosis and recovery actions, the assumptions concerning loss of water from a spent fuel pool used conservative upper bound failure rates for failure to diagnose prior to pool boiling and therefore address this concern. In the highly unlikely situation of draining the spent fuel pool to the transfer canal level, licensee responses to IEB 84-03 (Ref. 19), concerning pneumatic seal failures, indicate that emergency procedures have been considered which would not require entry into the spent fuel pool area - the high radiation fields from the spent fuel alone would likely restrict access.

4.9.3 Refueling Cavity Seal Failure During Fuel Assembly Handling (GI 137)

Generic Issue 137, titled "Refueling cavity Seal Failure," is considering the issue of spent fuel damage resulting from a reactor cavity seal failure during fuel assembly handling (Ref. 35). The risk of failure of a single fuel assembly and the potential risk to plant personnel, not the general public, is being addressed.

The likelihood of uncovering stored spent fuel as a result of a seal failure and the risk to the general public are addressed in this Regulatory Analysis.

5. VALUE/IMPACT ANALYSIS

5.1 Alternative 1 - No Action

This alternative assumes that no additional action is necessary based on the evaluation of the current risk associated with the use of high density racks for the storage of spent fuel in spent fuel pools at LWRs. It is also assumed that all applicable requirements and guidance approved to date have been implemented.

In addition to implementing the requirements contained in 10 CFR Part 50 Appendix A of the "General Design Criteria," and 10 CFR Part 20, concerning radiation doses being kept as low as is reasonably achievable, licensees should have implemented additional or corrective actions based on the following guidance:

1. IE Bulletin 84-03, "Refueling Cavity Water Seals," issued August 24, 1984. (Ref. 21)

2. IE Information Notice 84-93, "Potential for Loss of Water From the Refueling Cavity," issued December 17, 1984. (Ref. 24)

3. Generic Letter 85-11, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants' NUREG-0612," issued June 28, 1985. (Ref. 4)

4. IE Information Notice 87-13, "Potential for High Radiation Fields Following Loss of Water from Fuel Pool," issued February 24, 1987. (Ref. 34)

5. IE Information Notice 87-43, "Gaps in Neutron-Absorbing Material in High-Density Spent Fuel Storage Racks," issued September 8, 1987. (Ref. 33)

6. IE Information Notice 88-65, "Inadvertent Drainages of Spent Fuel Pools," issued August 18, 1988. (Ref. 25)

7. IE Information Notice 88-92, "Potential For Spent Fuel Pool Draindown," issued November 22, 1988. (Ref. 26)

No costs are usually attributed to a "No Action" alternative because the future cost of accidents are conventionally counted as benefits or averted costs in the assessment of the alternative actions. However, a spent fuel pool accident would result in cleanup and repair costs. In addition, replacement power costs could occur during the cleanup and repair period. Reactor operations without a safe place to store spent fuel could not continue. If the accident also results in a large release of radioactivity offsite, the costs of relocating people, restricting food and water, cleanup of contamination, and health consequences would add to these costs. Occupational exposure due to a spent fuel pool accident could also be considered on a monetary basis. BNL has evaluated these attributes (Ref. 13). The following paragraphs summarize this assessment of risk and the cost associated with a representative (base case) spent fuel pool accident, based on an estimated probability of spent fuel damage of 2×10^{-6} per reactor year.

5.1.1 Occupational Exposure (Accidental)

Exposure to plant personnel associated with post-accident cleanup of a major spent fuel pool accident is expected to be similar to those associated with the cleanup activities at TMI-2. For this accident, BNL estimates the occupational radiation dose from cleanup is less than 4,580 person-rem (Ref. 13). Since the potential offsite dose impact (per accident) ranges from 8 to 26 million person-rem, further refinement of this estimate is not warranted.

5.1.2 Onsite Property Damage

The spent fuel pool accident sequence involves (1) failure of the pool due to seismic or load drop events resulting in the complete loss of water inventory, or loss of cooling resulting in boiling dry the pool, (2) Zircaloy fire initiation of recently discharged fuel and the potential propagation of the fire to older spent fuel assemblies stored in the pool, and (3) loss of confinement since the spent fuel pool building is assumed to be breached as a result of a seismic event or as a consequence of the highly exothermic Zircaloy fire.

The consequences of these scenarios are expected to be similar to the Category II accident defined in NUREG/CR-3568 (Ref. 36), 50% clad melting and contamination. For this case cleanup and decontamination are estimated to be approximately $192 million (1988 $s). Plant outage times were estimated by BNL based on the time estimates to license, construct and test a replacement pool (Ref. 37), and range from five to seven years. The cost (1988 $s) for a replacement pool, $54 million for a 400 MTU pool, and the cost of permanent disposal of damaged fuel, $30 million, were also estimated from reference 37.

The cost of replacement power is approximated by (from NUREG/CR-3568):

$$C = (0.13 * R + 0.12)10^6 \$/\text{MW-year}$$

where R is the fraction of replacement energy by oil fired or non-economical power purchases from a given National Electric Reliability Council (NERC) region. This formula includes credit for the avoided variable fuel cycle cost of a shutdown reactor. An R value of 0.41 was used for the best estimate (national average) and a value of 0.9 (highest NERC region) was used for the worst case estimate for replacement power costs (based on 1981 $s). A plant capacity of 65% is factored into the above equation and a 1,000 MW(t) generic plant size was assumed.

The BNL best estimate replacement power cost (in 1983 $s) for a 1,000 Mw(t) plant for five years is $867 million. For the worst case estimate the seven year replacement power cost is $1,660 million.

Using NUREG/CR-4568 (Ref. 38) data, the cost of replacement power (1984 $) for the national average cost of 0.026 $/Kw-hr is $740 million for the best estimate and, using a high NERC region cost of 0.035 $/Kw-hr, the worst case estimate is $1,400 million. Assuming a constant 5% inflation rate over a four year period (to 1988), the current values would be $900 million and $1,700 million respectively. These values are used in this regulatory analysis, and indicated that replacement power costs are not sensitive to modeling assumptions (2% to 4% uncertainty in comparing BNL 1983 values to staff 1988 values).

The onsite costs are calculated from (NUREG/CR-3568):

$$V_{on} = N \times \Delta F \times U_{on}$$

where:

V_{on}	- value of avoided onsite property damage
N	- number of affected facilitities
ΔF	- change in accident frequency

and

$$U_{on} = \frac{(C_c + C_r + C_{rp})}{m} \frac{e^{-rti}}{r^2} (1 - e^{-r(tf-ti)}) (1 - e^{-rm})$$

where:

U_{on}	- present value of onsite property damage conditional upon release
C_c	- cleanup and decontamination costs
C_r	- repair, replacement of spent fuel pool, disposal of damaged fuel
C_{rp}	- replacement power costs
tf	- average years remaining till end of reactor life, 30 years
ti	- year plant starts operating, taken to be 0 years
r	- discount rate
m	- years plant is out of service

The onsite property damage costs per accident, V_{on}, are summarized in Table 5.1.1, based on a ΔF of 2×10^{-6} per reactor year.

5.1.3 Offsite Health and Property Damage

The offsite health and property damage estimates were obtained by BNL from the MACCS computer program (Section 4.6, Table 4.6.2), and are summarized in Table 5.1.2.

The offsite costs are calculated using the NUREG/CR-3568 methodology and discounting the cost of the 30 year remaining life of a typical facility:

$$V_{off} = N \times \Delta F \times U_{off}$$

where:

V_{off}	- value of avoided offsite property damage
N	- number of affected facilitities
ΔF	- change in accident frequency

and

$$U_{off} = C_{off} \times \frac{e^{(-rti)} - e^{(-rtf)}}{r}$$

where:

U_{off}	- present value of offsite property damage, conditional upon release
C_{off}	- offsite property damage cost
tf	- average years remaining till end of reactor life, 30 years
ti	- year plant starts operating, taken to be 0 years
r	- discount rate

The offsite damage costs per accident, V_{off}, are summarized in Table 5.1.2, based on a ΔF of 2×10^{-6} per reactor year.

Table 5.1.1
Onsite Property Damage Costs Per SFP Accident (1988 $s)

Item	Units	Best Estimate	Worst Estimate
Cleanup and Decontamination	$1,000,000	192	192
Repair Pool and Dispose of Fuel	$1,000,000	84	84
Replacement Power	$1,000,000	900	1,700
Average Number of operating years remaining	years	30	30
Years plant is out of service	years	5	7
Present Value (V_{on}) At 10% Discount rate	1988 $s	17,600	27,000
Present Value (V_{on}) At 5% Discount Rate	1988 $s	32,400	51,800

Table 5.1.2
Offsite Health and Property Damage Estimates (1988 $s) Per SFP Accident

Case Description	Whole Body Dose (person-Rem per-Event)	Offsite Property Damage ($s)	Present Value (V_{off}) At 10% Discount ($s)	At 5% Discount ($s)
Best Estimate Consequences Last discharge 90 days after discharge, 50 mile radius Based on 340 people/square mile	8.0×10^6	4.0×10^9	76,000	124,000
Worst Case Estimate Consequences Total inventory 30 days after discharge, 50 mile radius Based on 860 people/square mile	2.6×10^7	3.0×10^{10}	580,000	940,000

5.1.4 Potential Consequences and Cost of SFP Accidents

The probability of a spent fuel pool accident which would result in spent fuel damage is estimated to be 1.5×10^{-7} per reactor year, summed over all accident sequences except for the beyond design basis seismic failure of the spent fuel pool structure. The conditional probability of Zircaloy cladding fire is estimated to be 1.0 for a high density racked PWR and 0.25 for a BWR.

The seismic structural capacity of the spent fuel pool has been shown to have a median ground acceleration in the 1.4 to 2.0 g range. This is a factor of 10 above the typical SSE design values. The high confidence low probability of failure value (0.5 to 0.65 g) shows a margin of safety of three over the SSE design value. That is, there is less than a 5% chance of failure at a confidence level of 95% for a peak ground acceleration three times the SSE design value. The estimated mean seismic probability of a Zircaloy fire in a PWR spent fuel pool is 1.8×10^{-6} per reactor year, and in an elevated BWR spent fuel pool the estimated value is 1.7×10^{-6} per reactor year. The probability of a spent fuel pool accident which would result in spent fuel damage is therefore estimated to be on the order of 2×10^{-6} per reactor year, including the seismic hazard, for a typical LWR spent fuel pool.

For comparison purposes, in a review of the results of seismic core damage and large release frequencies from studies performed as part of USI A-45, "Decay Heat Removal Requirements," the point estimates for seismic core damage frequencies ranged from 1×10^{-5} to 8.3×10^{-5} per reactor year and the seismic release frequencies ranged from 4.6×10^{-6} to 3.7×10^{-5} per reactor year (Ref. 31). The dominant contribution to core damage was found to be from earthquakes in the 0.2 to 0.4 g range (Ref. 31). It is therefore concluded that, in comparison to the probability and consequences of a reactor core damage accident from a seismic event, the likelihood and risk associated with the beyond design basis seismic induced spent fuel pool failure are only a small part of the overall risks associated with the operation of a nuclear power plant.

The best estimate consequences of a spent fuel pool accident are summarized in Table 5.1.3, based on a plant mean probability of a spent fuel pool accident of 2×10^{-6} per reactor year. This value is representative of both the PWR and BWR pool and includes the beyond design basis earthquake accident.

The cost estimate data available is generally based on 1983 values. The offsite property damage from MACCS, the EPRI study on alternative spent fuel storage options, and onsite property damage (cleanup and replacement power) are all monetized to, or represent 1983 costs. These 1983 cost estimates were escalated to 1988 values by using the Gross National Product Price Deflator ratio between 1988 and 1983 (121.4 divide by 104.1, or a factor of 1.17), taken from NUREG/CR-4627 (Ref. 39), Abstract 6.4 "Time-Related Cost Adjustments." As seen in Table 5.1.3, the monetized present value of the offsite health effects, at $1,000 per person-rem, dominates present value estimates when compared to property damage costs at the 5% discount rate. At a 10% discount rate, as recommended in NUREG/BR-0058 (Ref. 40), the onsite and offsite property damage present value cost estimates would be about one-half the values shown in Table 5.1.3. The 5% discount rate is used in this Regulatory Analysis because it is believed to be more representative of the current economical environment than the 10% value. It is therefore concluded that additional refinements in cost estimates concerning onsite and offsite property damage costs are not required and will not affect the value/impact or cost benefit analyses provided. In addition, first order approximation of costs are generally being used in this Regulatory Analysis and any additional adjustment to these estimates is not warranted.

The significance of the potential consequences of this base case spent fuel pool accident evaluation, and the related costs associated with an accident, are discussed in more detail under Section 6, "Decision Rationale." Alternatives actions, other than this "no action" proposal, are considered in the following sections to determine if cost beneficial options are available to reduce the risk or consequences of a spent fuel pool accident.

Table 5.1.3
Best Estimate Consequences of a Spent Fuel Pool Accident

Frequency of Occurrence of Zircaloy Fire (Mean value for PWR or BWR SPF)	$2x10^{-6}$ per reactor year
Consequences, over 30 years based on 340 people per square mile population density within a 50 mile radius of the site	480 person rem
Present Value of Offsite Health Risk Based on $1,000 per person rem	$ 480,000 (1988 $s)
Present Value of Offsite Property Damage 5% discount over 30 years of remaining life	$ 124,300 (1988 $s)
Present Value of Onsite Property Damage 5% discount over a five year repair period	$ 32,400 (1988 $s)
Total Present Value of a SFP Accident	$ 636,700 (1988 $s)

Note: Without the beyond design basis seismic failure, consequences are estimated to be an order of magnitude lower.

5.2 Alternative 2 - Require Use of Low Density Racks

The use of high density storage racks increases the probability of a release of radioactive from stored fuel if water is lost from the spent fuel pool. Studies performed in 1979 (Ref. 8) concluded that the minimum allowable decay time for PWR spent fuel in a well-ventilated room varies from five days, for open-frame storage configurations, to a value of 700 days, for high density, closed-frame configurations with wall-to-wall spent fuel placement. The minimum decay time for BWR fuel varied from five days to 150 days for the configurations studied. In addition, it was determined that the high density storage configuration could allow for the propagation of fuel damage from the recently discharged fuel to older, adjacent stored fuel in the pool. The results of these 1979 studies prompted the identification of Generic Issue 82.

5.2.1 Risk Reduction Estimate

One of the potential means to reduce the risk from loss of water in the spent fuel pool would be to require the use of low density storage racks for the recently discharged fuel. This alternative would reduce the probability of fuel damage, or a Zircaloy fire. The estimated reduction in risk is a result of the decreased decay time required for low density storages racks to preclude spent fuel damage if water is lost, as compared to the high density racks. BNL estimated the reduction in risk to be at least a factor of five, or about an 80% reduction in the consequences (Ref. 10). For the purpose of evaluating this alternative, a 100% reduction in the consequences will be assumed.

5.2.2 Cost of Low Density Storage

Additional storage for spent fuel at reactor sites is required. If the DOE spent fuel repository opens in 2003, it is estimated that the industry will need to provided for between 12,200 MTHM to 20,000 MTHM of additional storage capacity (Ref. 7). If no repository is made available the additional capacity is estimated to be between 32,090 MTHM and 42,450 MTHM (Ref. 7).

The use of low density storage racks would require the need for additional at-reactor-site storage capacity to accommodate the change in the storage configuration, from high density racks. For each PWR low density storage location, three high density assemblies would be displaced. For a BWR, two high density assemblies would be displaced for each low density storage location (Ref. 13).

The evaluation of the amount of additional storage capacity, resulting from a proposal to require low density racks, is provided in reference 13. The fuel cycle information and spent fuel projections where obtained from DOE/RL-87-11, "Spent Fuel Storage Requirements 1987" (Ref. 41). The proposed alternative for low density storage could increase the additional capacity by about 17,000 MTHM.

The cost of additional storage depends on the type of facility to be used. EPRI evaluated four alternative storage facilities in NP-3365 (Ref. 16), "Review of Proposed Dry-Storage Concepts Using Probabilistic Risk Assessment." These concepts are:

1. Additional Pool, separate from existing pool,

2. Cask storage,

3. Caisson, or dry-well, storage, and

4. Vault storage.

The risk associated with these alternative concepts was found to be generally acceptable since the spent fuel would be five years old, or older, and the likelihood of significant fuel damage was found to be low. The consequences were found to be negligible when compared to the operating reactor at the site (Ref. 16).

The costs for alternative storage concepts were evaluated by EPRI in NP-3380, "Cost Comparisons for On-Site Spent-Fuel Storage Options" (Ref. 37). In addition to the primary four alternatives identified above, silo storage, reracking and rod consolidation were addressed in terms of cost. The cost estimates are provided in units of dollars per kilogram of uranium ($/kgU) and vary with the size of the facility. The cost estimates decrease as the storage capacity increases, per unit, because of initial licensing and engineering fees associated with the facility design.

In a recently completed study performed by DOE (Ref. 7), the cost of additional storage for the cask concept and the rod consolidation concept have been reviewed and updated to reflect the limited amount of actual experiences with these methods of providing additional at-reactor-site storage. These methods are considered to be the most practical and represent demonstrated technologies. The costs estimates are provided below, in Table 5.2.1.

The estimated cost (1988 $s) of additional storage by the year 2003 is estimated by DOE to range between $945 million and $1,267 million for 12,200 MTHM, and between $1,545 million and $2,000 million for 20,000 MTHM. If rod consolidation can accommodate the 350 MTHM requirements, the costs estimates would be reduced to $468 million for the 12,200 MTHM case and $793 million for the 20,000 MTHM case. The mean cost estimate for the 17,000 MTHM additional storage which would be needed if low density storage is required is $1580 million, based on the DOE cost estimates. The mean cost (for the 108 pools) is $14.6 million per pool, for a unit cost of $93/Kgm of heavy metal.

The cost estimates for the dry storage concepts were developed assuming that the cost of a low density requirement would be an incremental, additional cost as virtually every spent fuel pool will reach it's capacity limit and out of pool storage will be required before end-of-license if a repository in not available (Ref. 13). Rod consolidation was not considered because it is not known how many pools can accommodate the additional decay heat load and structural loads associated with the in-pool increase and still meet NRC requirements. The unit cost for at-reactor storage decreases with an increase in the total amount, or capacity, of the additional storage required. The total licensing, engineering and fixed facility costs for the initial at-reactor storage facility is estimated to be $0.6 to $1.8 million (1988 $s) per facility, based on metal cask storage (Ref. 7). These one-time costs would not be impacted by this alternative, and are costs which will be incurred by most licensees prior to the availability of the DOE repository. The cost of the additional at-reactor storage which would result from this alternative is also provided, for reference, assuming that the additional storage capacity cost is based on the unit cost estimates for a facility the size of only the additional storage requirement. This is referred to as the lead cost estimate. Table 5.2.2 provides a summary of the incremental capacity increases which would result from this alternative. The associated cost estimates for dry storage alternatives based on these capacities are provided in Table 5.2.3.

The DOE cost estimates based on the incremental cost assumption for the 17,000 MTHM additional storage are $1,510 million, or a mean cost of $14 million per pool (1988 $s). The DOE cost estimates (Ref. 7) level off for capacities in excess of 300 MTHM (see Table 5.2.4 below).

Table 5.2.1
Range of Unit-Cost Estimates for Additional Storage Requirements
(Costs in 1988 $s per kilogram of heavy metal)

Storage Technology	Capacity Increase 100 MTHM	300 MTHM	1000 MTHM
Rod Consolidation(1)	40 - 75	30 - 50	n/a
Metal Cask (10 MTHM)	60 -115	55 -115	55 - 100
Concrete Cask	50 -105	45 - 90	45 - 80
Horizontal Concrete Modules	45 - 65	40 - 55	40 - 55
Modular Vault System	105 -155	70 -105	45 - 70

Notes:

(1) - The unit costs are based on the cost for an additional storage slot created in the storage pool. From 2.6 to 3 spent-fuel assemblies must be consolidated for each storage slot (Ref. 7).

n/a - An increase of 1000 MTHM is not applicable to rod consolidation because at a typical reactor not much more than 300 MTHM of additional spent fuel can be gained through consolidation (Ref. 7).

Table 5.2.2
Additional Incremental Storage Capacity Requirements for Alternative 2

Capacity Range (MTHM)	PWRs Impacted	BWRs Impacted	Total Impacted
0 - 50	0	2	2
50 - 100	8	10	18
100 - 150	15	18	33
150 - 200	23	8	31
200 - 250	15	1	16
250 - 300	3	0	3
300 - 350	4	0	4
350 - 400	1	0	1

Table 5.2.3
Storage Costs Associated With Alternative 2 (1988 $s)

	Cost:	Per Pool ($1,000,000)			For All 108 Pools ($1,000,000)		
	Discount Rate:	0%	5%	10%	0%	5%	10%
Pool	(BNL Incremental Costs)	25.3	19.5	14.9	2,720	2,100	1,610
Drywell	(BNL Incremental Costs)	10.6	9.5	8.0	1,150	1,038	863
Vault	(BNL Incremental Costs)	24.1	19.5	14.9	2,612	2,100	1,610
Cask	(BNL Incremental Costs)	14.0	14.2	12.2	1,516	1,539	1,318
Silo	(BNL Incremental Costs)	18.2	14.2	11.2	1,959	1,539	1,178
Cask	(BNL Lead Costs)	19.7	20.1	17.3	2,134	2,169	1,866
Cask	(DOE Incremental Costs)	14.0	14.2	12.3	1,510	1,530	1,330
Cask	(DOE Lead Costs)	14.6	14.9	12.8	1,580	1,610	1,380

Notes:
Zero % discount rate corresponds to the case where additional storage capacity is built now. The 5% and 10% rates reflect discounted costs in delaying the building of additional capacity until needed.

The difference between the estimated costs for cask storage, in comparing the BNL-based and DOE-based estimates, are due to the difference in the $/Kgm costs estimates based on facility capacity. In Table 5.2.4, the BNL point-estimate cost (based on EPRI NP-3380, Ref. 37) is compared to the DOE lower and upper bound estimates.

Table 5.2.4
Cask Storage Cost Estimates as a Function of Facility Capacity

Capacity (MTHM)	BNL Point Estimate ($/Kgm) (1983 $s)	DOE Lower Estimate ($/Kgm) (1988 $s)	DOE Upper Estimate ($/Kgm) (1988 $s)
25	113.2	105	160
50	113.2	95	140
75	113.2	85	120
100	113.2	80	115
125	113.2	80	110
150	113.2	80	110
200	113.2	80	105
300	99.8	80	100
400	93.4	80	100
500	89.	80	100
600	85.8	80	100
700	84.2	80	100
800	82.9	80	100
900	81.6	80	100
1000	81.	80	100
1200	79.4	80	100
1400	78.8	80	100
1600	77.5	80	100
1800	77.2	80	100
2000	77.2	80	100

5.2.3 Value/Impact Summary

The value/impact, cost-benefit analysis is provided in terms of the mean industry risk from spent fuel pool accidents in Table 5.2.5. The best estimate accident frequencies are used and the best estimate consequences, based on fission product release from 1/3 of a reactor core, are used. The conditional probability of the Zircaloy fire, given the loss of water from the spent fuel pool, is 1.0 for the PWRs and 0.25 for the BWRs. Since the amount of spent fuel which could become involved in the release is uncertain, a sensitivity study using the worst case consequences, full spent fuel pool inventory at a high population site, is also provided.

The risk is comprised of 69 PWR spent fuel pools with a spent fuel damage probability of 1.95×10^{-6} per reactor year (including seismic events and conditional Zircaloy fire probability of 1.0 given loss of water) and 39 BWR spent fuel pools with a spent fuel damage probability of 1.71×10^{-6} per reactor year (including seismic events and conditional Zircaloy fire probability of 0.25 given loss of water). The mean remaining lifetime for the PWR spent fuel pool is 29.8 reactor years, and 27.9 years for the BWR spent fuel pool.

Since this alternative addresses dry storage, and because nearly every utility will require some additional dry storage prior to the start-up of the DOE repository, the NRC development and implementation costs concerning the licensing of a 10 CRF Part 72, "Licensing Requirements for the Independent Storage of Nuclear Fuel and High-Level Radioactive Waste," (amended August 19, 1988, 53 FR 31651), facility are not included as additional costs to be included in the value/impact analysis. These costs will be incurred and are not affected by this alternative.

The industry and NRC operational costs associated with dry storage are also costs which will be incurred and are not impacted by this alternative. The additional storage requirements which would result from this alternative would not affect these costs. Industry and NRC costs which could result from a re-racking amendment to replace high density racks with low density racks as part of this proposed alternative are expected to be small in comparison to the dry-storage costs and have not been quantified. Inclusion of these additional costs would result in an even less favorable value/impact or cost benefit assessment.

Sensitivity studies were performed by BNL to test the assumptions used (Ref. 13). The discount rate applied to property damage costs, the monetary conversion factor for health effects, plant site economics and meteorology, and industry implementation costs were evaluated.

The recommended discount rate, NUREG/BR-0058 (Ref. 40), is 10%, which results in a lower estimate of property damages by a factor of two. Public dose reduction is not affected since it is not discounted.

A major difficulty with the net-benefit method is the evaluation of health effects in monetary units. Sensitivity studies of $500 and $2,000 per person-rem are used to demonstrate uncertainty.

The base line calculation adopted by BNL used the economic factors of the Zion plant site as a best estimate. Zion is somewhat higher than the U.S. average. A sensitivity study based on West Virginia was performed by BNL to evaluate the sensitivity to economic factors. The economics of West Virginia are considered to be much below the national average. Zion weathers conditions were also employed in the base line calculations with MACCS. A review of several severe accidents calculation, by BNL, indicated that varying weather models have a small effect on the public health and offsite consequences.

Since the costs associated with spent fuel storage construction costs and overhead and maintenance are well documented in either the EPRI NP-3380 (Ref. 37) study or in the DOE Dry-Storage study (Ref. 7), the industry implementation costs appear to be well defined and no sensitivity study is performed.

The impact of these sensitivity studies are shown in Table 5.2.6. The best case analysis is used as the reference, base line case. The results of the sensitivity study shows that the dominance of the cost of dry-storage in comparison to the potential damage costs is overwhelming and therefore this alternative is not cost effective. A factor of 10 increase in the probability of a Zircaloy fire in a spent fuel pool would not alter this conclusion.

Table 5.2.5
Summary of Industry Wide Value/Impact Analysis for Alternative 2
Based on 100% Risk Reduction (1988 $s)

Attribute	Dose Reduction (person-Rem)		Cost ($1,000)	
	Best Est.	High Est.[a]	Best Est.	High Est.[a]
Public Health [1]	47,000	153,000	47,000	153,000
Occupational Exposure (Accidental)	negligible	negligible		
Onsite Property Damage (5% discount)	3,500	5,600		
Offsite Property Damage (5% discount)			13,400	101,000
Industry Implementation and Operation Cask Assumption			-1,510,000	-1,510,000
NRC Development/Implementation and Operation			negligible	negligible
Net Benefit			-1,446,000	-1,250,000
Benefit($)/Cost($) [2]			0.042	0.172
Dose Reduction (person-rem)/Million $s [3]			31	100
Value/Impact Ratio [4] ($/Person-rem reduction)			32,130	9,900

Notes: (a) High estimate based on worst case of entire pool inventory at site with 860 people per square mile population density and Zion land use factors.

(1) Cost of health consequences set at $1,000 per person-rem.

(2) Averted costs divided by NRC + Industry implementation and operational costs.

(3) Public dose reduction divided by NRC + Industry implementation and operational cost.

(4) Cost of NRC + Industry implementation and operation divide by public dose reduction.

Table 5.2.6
Benefit/Cost Ratio Sensitivity Analysis for Alternative 2 (1988 $s)

Parameter Under Study	New Value	Net Benefit ($1,000)	Benefit/ Cost Ratio (1)
Baseline Best Estimate		-1,446,000	0.042
Discount Rate	10%	-1,453,000	0.038
Health Effects	$ 500/person-rem	-1,476,000	0.022
	$2,000/person-rem	-1,406,000	0.069
Site Economics	West Virginia	-1,453,000	0.038

Note: (1) None of these assumptions changes the risk reduction in person-rem averted, which remains constant at 47,000 person-rem averted. The implementation cost also is not changed. The value/impact ratio remains $32,130 per averted person-rem.

5.3 Alternative 3 - Improve Cooling/Make-Up Systems

Four generic spent fuel pool cooling and make-up systems have been described in Section 4.7.6. The probability of failure of these systems to fail to provide adequate cooling and make-up to the spent fuel ranges from 2.2×10^{-5} per reactor year to 3.6×10^{-6} per reactor year, without consideration for recovery. With recovery, the probability of damage to spent fuel is estimated to be in the 1×10^{-8} per reactor year range, with an upper bound value of 1×10^{-6} per reactor year.

A beyond design basis earthquake, as a result of the low seismic capacity of non-safety grade electrical components (motor control centers and switchgear), is estimated to result in a probability of damage to spent fuel on the order of about 5×10^{-8} per reactor year. Essential components to either the cooling system or the make-up system, or both systems, are designed to the SSE and deterministically demonstrated to perform the safety function of maintaining the spent fuel in a safe and subcritical configuration for all credible storage conditions.

5.3.1 Risk Reduction Estimate

Although a loss of cooling and subsequent heatup is a very slow event (on the order of several days), analyses have shown that after the spent fuel is uncovered, the remaining water would block air circulation and cladding overheating would occur for fuel which had been cooled for one year (Ref. 8). However, because of the lack of air circulation within the spent fuel holders, the oxidation reaction would be oxygen starved and the cladding would not melt. Thus, BNL concludes that catastrophic failure of the spent fuel would not be expected. Consequence estimates for ruptured fuel pins was performed in NUREG/CR-4982 (Ref. 10), and the resulting offsite consequences were found to be minimal, about 4 person-rem given the accident (see Table 4.6.2).

The economics of such an accident appear to be important. Since the reactor could not operate until the spent fuel pool was available, the cost of replacement power, until the spent fuel pool building was decontaminated and the equipment repaired, could be considerable. It is estimated by BNL (Ref. 13) that repairs and decontamination would take one month to one year depending on the degree of fuel damage and contamination. Replacement power costs estimates were obtained based on the method presented in Section 5.1. The onsite costs range from $19 to $227 million (1988 $s) conditional upon a spent fuel pool accident (Ref. 36). Integrated over the remaining lifetime of a typical plant, 30 years (the industry average), the expected cost associated with the gradual coolant loss sequences (without discount) could be as high as $150,000, based on a $2.2x10^{-5}$ per reactor year event (without credit for recovery actions). The low value is estimated to be $12,500, without discount.

5.3.2 Cost of Improved Cooling/Make-Up Systems

Two alternative systems for improvement of the spent fuel pool cooling system were evaluated by BNL (Ref. 13) to assess the potential cost-benefit for each of the four generic system types:

1. Provide another full capacity pump and associated valves to eliminate the need for running the cooling system without a backup pump (System C and D). The first order approximation of the cost of this option is estimated by BNL to be $50,000 (1983 $), or $60,000 (1988 $s) based on Section 5.1.4 cost escalation factors.

2. Provide a completely independent make-up train, BNL assumed this system to be similar to the primary spent fuel pool supply train. The hardware requires include a Category I water storage tank (200,000 gallon capacity), pumps, controls, and piping. The first order approximation of the cost of the independent make-up train plus overhead and maintenance costs were estimated by BNL to be one million dollars (1983 $s), or $1.2 million (1988 $s).

5.3.3 Value/Impact Summary

The cost-benefit characteristics of these options are summarized in Table 5.3.1. For the additional make-up train, the analyses assume 100% reduction in the initiating frequency, that is complete recover of the potential loss, and the averted costs are calculated at a 5% discount over the remaining average plant life of 30 years. The addition of a full capacity pump to System C or D would result in a risk reduction to the equivalent frequency of System A or B. For example the change in the initiating frequency for System C would be from $2.2x10^{-5}$ to $3.8x10^{-6}$ per reactor year, or a change of $1.8x10^{-5}$ per reactor year.

The only system which might benefit from either of these options is System C without credit for recovery actions. The generalized presentation of the Standard Review Plan requirements indicate that additional requirements, to improve the cooling/make-up system, would not result in a cost-beneficial improvement. When recovery actions are considered, the most appropriate estimate for the cost-benefit ratio is System D.
The current requirements for the design of the spent fuel pool cooling and make-up systems, when credit for operator action to diagnose and recovery from a loss of cooling event is considered, are judged to be satisfactory. This finding is based, in part, on the assumption that,

as a result of IE Bulletins and Information Notices (see Section 5.1 above), licensees are aware of the need to assure that adequate instrumentation is available and maintained to alert the operators to degradation in the spent fuel pool or its support systems.

Table 5.3.1
Value/Impact for Generic Improvements to the SFP Cooling Systems[1]
(5% Discount Rate - 1988 $s)

System	Description (Frequency)	Option	Cost of Option ($1,000)	Averted Cost Range[2] ($1,000) Low	Averted Cost Range[2] ($1,000) High	Benefit/Cost Ratio Range Low	Benefit/Cost Ratio Range High
A	Minimum SRP ($3.8x10^{-6}$/R-y)	1. Add Pump	60	None			-
		2. Make-up train	1,200	1.1	13.1	0.001	0.011
B	Minimum SRP With Credit for Fire Hose ($1.9x10^{-7}$/R-y)	1. Add Pump	60	None			-
		2. Make-up train	1,200	0.1	6.5	0.001	0.005
C	Old System w/ Both trains 30% of time ($2.2x10^{-5}$/R-y)	1. Add Pump	60	3.5	41.4	0.058	0.690[3]
		2. Make-up train	1,200	6.3	75.7	0.005	0.063
D	Old System With Credit for Fire Hose ($1.1x10^{-6}$/R-y)	1. Add Pump	60	0.2	2.1	0.003	0.035
		2. Make-up train	1,200	0.3	3.8	0.001	0.003

Notes: (1) Spent fuel cladding ruptures and releases gaseous fission products, no Zircaloy cladding fire. The offsite consequences are small, 4 person-rem given the loss of cooling cooling/make-up. Value/impact ratio, in $s per averted person-rem, is very large (well in excess of $1,000 per averted person-rem), however economics of spent fuel cladding rupture could be important.

(2) Averted costs of replacement power and cleanup/repair of spent fuel pool. Low estimate is for one month outage, high estimate is for one year outage.

(3) Based on a 10% discount rate, the averted cost estimate is reduced to $24,700 and the benefit/cost ratio is reduced to 0.41. Similar reductions apply to all options at a 10% discount rate.

5.4 Alternative 4 - Install Spray Systems

Post-accident spray systems have been considered as a potentially significant mitigative measure for spent fuel pool accidents. A scoping value/impact assessment was performed by BNL (Ref. 13) to provide some insights into the potential cost effectiveness of installing spray systems. The guidelines outlined in NUREG/CR-3568 (Ref. 36) for "First Approximation of Benefits and Costs" were used.

BNL emphasized that this assessment is scoping in nature due to the many assumptions involved and large uncertainties in data and decontamination factors assumed for spray systems.

5.4.1 Risk Reduction Estimate

The principle reduction effect of the spray systems is achieved by decontaminating radiological releases thus permitting greater retention of fission products in the pool and the pool building. Results of analyses of severe reactor accidents in support of NUREG-1150 (Ref. 42), indicate that containment spray systems can be significantly effective in reducing source terms and severity of consequences of nuclear reactor accidents (Ref. 43).

In this assessment, it is assumed that the major benefit of spray systems results from reduction in the offsite consequences. The onsite property damage is not effected, that is cleanup and repair and replacement power costs would still be incurred due to spent fuel damage and a Zircaloy fire.

The effectiveness of the spray system is measured by the decontamination factor (DF), the amount of radioactive species released to the environment without the spray divided by the amount released with the spray. Decontamination factors for a spent fuel pool spray system are difficult to estimate without detailed calculations, therefore BNL assumed that the DF would be 45 based on NUREG-1150 analysis for the Surry plant containment spray system effectiveness. The effects of a DF of 45 on the results of MACCS consequence calculations are provided in Table 5.4.1, and compared to the previous case without sprays. The effects of a spray system with a DF of 45 has the effect of reducing the offsite consequences to a small fraction of their original levels, therefore this can be considered to be an upper bound measure of the potential benefit of a post-accident spray system.

5.4.2 Cost of Installing Spray Systems

Preliminary construction and industry maintenance costs were estimated by BNL. Assumed hardware requirements included a Category I water storage tank (200,000 gallon capacity) and a spray system including pumps, spray nozzles and associated hardware. The cost, on a first approximation basis, is estimated to be $1.2 million (1988 $s) per spent fuel pool (Ref. 13).

The NRC cost associated with this option is estimated to be $100,000 (1988 $s) per spent fuel pool, roughly equivalent to one staff-year review effort per pool at $75,000 per staff-year (NUREG/CR-4627, Abstract 5.2, Ref. 39) plus $25,000 for the development and approval of a Technical Specification for the control of the administration, surveillance and maintenance of the spray system.

Table 5.4.1
Offsite Health and Property Damage Estimates (1988 $s)
With Pool Spray System (DF = 45)

Case Description	Whole Body Dose (person-Rem per-Event)		Offsite Property Damage (1988 $s)	
	Without Spray	With Spray	Without Spray	With Spray
Best Estimate Consequences Last discharge 90 days after discharge, 50 mile radius Based on 340 people/square mile	7.97×10^6	1.25×10^6	4.0×10^9	7.2×10^7
Worst Case Estimate Consequences Total inventory 30 days after discharge, 50 mile radius Based on 860 people/square mile	2.56×10^7	6.78×10^6	3.0×10^{10}	5.2×10^8

5.4.3 Value/Impact Summary

The value/impact, cost-benefit analysis is provided in terms of the mean industry risk from spent fuel pool accidents in Table 5.4.2. The best estimate accident frequencies are used and the best estimate consequences, based on fission product release from 1/3 of a reactor core, are used. The conditional probability of the Zircaloy fire, given the loss of water from the spent fuel pool, is 1.0 for the PWRs and 0.25 for the BWRs. Since the amount of spent fuel which could become involved in the release is uncertain, a sensitivity study using the worst case consequences, full spent fuel pool inventory at a high population site, is also provided.

The risk is comprised of 69 PWR spent fuel pools with a spent fuel damage probability of 1.95×10^{-6} per reactor year (including seismic events and conditional Zircaloy fire probability of 1.0 given loss of water) and 39 BWR spent fuel pools with a spent fuel damage probability of 1.71×10^{-6} per reactor year (including seismic events and conditional Zircaloy fire probability of 0.25 given loss of water). The mean remaining lifetime for the PWR spent fuel pool is 29.8 reactor years, and 27.9 years for the BWR spent fuel pool.

The dose reduction estimate is derived from the change in the offsite health consequences shown in Table 5.4.1, 6.72×10^6 person-rem per accident for the best estimate case and 1.88×10^7 person-rem per accident for the worst case. The offsite property damage costs are estimated using the revised MACCS values (with sprays), discount at a 5% rate over 30 years. The onsite property damage costs are assumed to be unchanged, cleanup and repair and replacement power costs are incurred.

The best estimate value/impact ratio for this alternative is estimated to be $3,340 per averted person-rem and exceeds the general guideline value of $1,000 per averted person-rem. While the high estimate is seen to be marginally cost effective ($1,200 per averted person-rem), the use of Zion site demography for the high estimate evaluation results is an overly conservative estimate of the risk reduction properties of a given plant modification (860 people per square mile).

Table 5.4.2
Summary of Industry Wide Value/Impact Analysis for Alternative 4
Based on a Spray System DF of 45 (1988 $s)

Attribute	Dose Reduction (person-Rem) Best Est.	Dose Reduction (person-Rem) High Est.[a]	Cost ($1,000) Best Est.	Cost ($1,000) High Est.[a]
Public Health [1]	39,450	110,500	39,450	110,500
Occupational Exposure (Accidental)	negligible	negligible		
Onsite Property Damage			0	0
Offsite Property Damage (5% discount)			13,000	100,000
Industry Implementation and Operation			-130,000	-130,000
NRC Development/Implementation and Operation			- 10,800	- 10,800
Net Benefit			- 88,400	+ 69,700
Benefit($)/Cost($) [2]			0.373	1.50
Dose Reduction (person-rem)/Million $s [3]			280	840
Value/Impact Ratio [4] ($/Person-rem reduction)			3,340	1,200

Notes: (a) High estimate based on worst case of entire pool inventory at site with 860 people per square mile population density and Zion land use factors.
(1) Cost of health consequences set at $1,000 per person-rem.
(2) Averted costs divided by NRC + Industry implementation and operational costs.
(3) Public dose reduction divided by NRC + Industry implementation and operational cost.
(4) Cost of NRC + Industry implementation and operation divide by public dose reduction.

5.5 Alternative 5 - Modify Spent Fuel Storage Rack Designs

This proposed alternative would require the licensee to compartmentalize the spent fuel storage pool by installing partitions (and individual coolant supply diffusers for each compartment) to limit the extent of the accident, or modify the storage racks to improve air circulation, should the spent fuel storage pool drain. This alternative is directed towards risk mitigation, and to a lesser extent prevention.

This alternative was not quantified as part of this value/impact study. The results of the cladding heatup calculation suggest that the only rack geometry that would result in mitigation is low density racks. The probability of a loss of water from the spent fuel pool would not be changed. Compartmental restructuring of the spent fuel is not judged to be feasible without a significant loss in the storage capacity and the resulting need for additional at reactor dry storage is expected to overwhelm any potential risk reduction.

5.6 Alternative 6 - Cover Fuel Debris With Solid Materials

This proposed alternative would require the development of a contingency plan to dump massive amount of solid materials into a drained spent fuel pool to cover the rubble bed to a depth of several feet. The necessary materials would not be stockpiled on site, but could be obtained in a timely manner on an ad hoc basis, the materials (sand, clay, dolomite, boron compounds, lead, etc.) being commonly available in all parts of the country. This alternative would be directed at risk mitigation, not prevention.

This alternative was not quantified as part of this value/impact study. The contingency plan would be concerned with a low frequency event (on the order of $1x10^{-6}$ per reactor year), with potential high consequence event. The results at Chernobyl can be used as a rough gauge of the efficacy of this measure, when carried out on a strictly ad hoc basis with no apparent advanced planning. However, since the dominant risk sequence for the spent fuel pool accident is a beyond design basis earthquake, BNL concludes that it is dubious that the measures could be implemented soon enough to prevent the major release to the environment during the first few hours of the accident (Ref. 13).

5.7 Alternative 7 - Improve Ventilation Gas Treatment System

This alternative would require the installation of a building ventilation and filter system capable of reducing the concentration of airborne radioactivity before discharge to the environment. This alternative would be directed at risk mitigation, not prevention.

This alternative was not quantified as part of this value/impact study. Again the dominant risk contribution results from the beyond design basis seismic failure of the spent fuel pool structure, a low frequency high consequences accident. To be effective, the spent fuel pool building structure would have to maintain its integrity and the system itself would have to be designed to survive the postulated peak ground acceleration which result in the spent fuel pool failure. Additional investigations into this alternative are not considered to be reasonable.

5.8 Relationships With Other Requirements and Activities

5.8.1 Severe Accident Policy Statement

A recently published report by LLNL, "Evaluation of External Hazards to Nuclear Power Plants in the United States - Seismic Hazard," NUREG/CR-5042, Supplement 1 (Ref. 31), summarizes the result of the study of the risk of core damage due to seismic initiated events.

The overall objective of the LLNL study "is to present information that assists the NRC staff in deciding whether seismic vulnerability searches for nuclear power plants should be in the implementation of the Severe Accident Policy Statement." To accomplish this objective, the LLNL report:

> 1. Considers effects of the evolution of design requirements and design practices on plant seismic capacity.
>
> 2. Identifies other specific review area of potential seismic vulnerability, including seismically induced fires and floods, spent fuel pools and seismic common-mode failures.
>
> 3. Identifies programs which address item 1 and/or item 2, and assess the extent to which these programs provide useful information on seismic capacity of nuclear plants.
>
> 4. Recommends incorporating appropriate items from above into the seismic margins program or other seismic vulnerability searches.

The LLNL report considered the results presented in NUREG/CR-4982, "Severe Accidents in Spent Fuel pools in Support of Generic Safety Issue 82" (Ref 10), and concluded that

> "A comparison of the results of the fuel pool analysis with the two figures of merit is difficult since the fuel pool failure does not constitute core damage and any potential release involves long lived radioactive material. In addition, it is difficult to draw conclusions concerning spent fuel pools based on only a single generic analysis. Therefore, any decision on the inclusion of spent fuel pools into the severe accident policy implementation requires more data and analysis, and cannot be concluded at this time."

The first figure of merit considered by LLNL is the core damage frequency. In numerical terms LLNL uses a mean core damage frequency in the range of 1×10^{-5} (or less) per reactor year as meeting the Commissioners stated objective, in the Policy statement on Safety Goals, as:

> "providing reasonable assurance, given consideration to the uncertainties involved, that a core damage accident will not occur at a U.S. nuclear power plant."

The second figure of merit is the frequency of a large release. In the Policy Statement on Safety Goals, the following guidance is given as a general performance guideline:

> "Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring performance of containment systems, the overall mean frequency of a large release of radioactive material to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation."

The current status of this guideline is that the NRC staff is giving detailed consideration to how such a performance guideline can be implemented, including how to define more precisely the definition of a "large release of radioactive material to the environment." In the LLNL study, a large release of radioactive material to the environment has been defined as a release of a substantial fraction of the radioactive core in a time period relatively early in the postulated accident scenario. This definition was derived from Probabilistic Risk Assessment (PRA) literature which has defined a "large early release." Further discussion on the applicability of such guidance to Generic Issue 82 is presented in Section 6.2.

5.8.2 Seismic Design Margins Program

The current objectives of the Seismic Design Margins Program are:

1. To develop and improve guidance for assessing the inherent capability of nuclear power plants to withstand earthquakes above the design level.

2. To provide an effective and efficient means to identify vulnerabilities of nuclear plants to seismic events.

The seismic margins approach has chosen as one of its figures of merit a high confidence of low probability of failure (HCLPF). The HCLPF is a conservative representation of capacity and in simple terms corresponds to the earthquake level at which it is extremely unlikely that failure will occur. Two approaches are recommended for estimating the component HCLPF values: the PRA fragility approach and the Conservative Deterministic Failure Margins (CDFM) approach.

The CDFM HCLPF approach has been developed and used by EPRI in a trial review of the Catawba Nuclear Station, with a seismic margins earthquake (SME) of 0.3 g. The resultant HCLPF for core damage sequences was found to be 0.24 g. An NRC sponsored review panel examined the EPRI work and found the methodology can accomplish its main objective and is reasonably accurate (NUREG/CR-5042, Supplement 1, Section 4, Ref. 31).

The PRA HCLPF approach has been used by the NRC to evaluate core damage sequences at Maine Yankee. The SME was also set at 0.3 g for this study. The HCLPF was found to be 0.21 g, and later revised to 0.27 g after the licensee committed to upgrading the refueling water storage tank.

The HCLPF approach used in the seismic design margins program does not use the seismic hazard curves. That is, the probability of a core damage sequences due to a seismic initiator are not evaluated in the traditional terms of frequency per reactor year used in PRAs. Instead the HCLPF value can be compared to the SME value. A plant HCLPF value greater than or equal to the SME would be considered to have adequate capacity since the SME would be chosen to assure adequacy. If the HCLPF value is less than the SME, then the site specific hazard curve could be used to estimate the recurrence frequency for that level of earthquake.

In NUREG/CR-5042, LLNL concludes that plant HCLPF capacity represents a conservative estimate at which there is a high confidence of a low probability of core damage. A more realistic parameter is the plant median capacity which is more than a factor of two greater than the HCLPF. It has been suggested that two times the plant HCLPF capacity could be used in conjunction with the median site specific hazard curve to obtain a recurrence frequency for comparison with some evaluation criterion. In light of the screening approach used for seismic margin reviews, LLNL goes on to conclude that research is needed to address what may be the appropriate factor that can be used along with the plant HCLPF capacity and what would be an appropriate evaluation criterion.

Until more definitive guidance is developed and approved by the Commission for the assessment of the external seismic hazard risk, the currently accepted guidelines for a regulatory impact analysis are used to define the risk. The mean failure frequency is used. The mean frequency is currently used for external events based on the use of the mean frequency in evaluating risk from internal events. Component and systems failures are described by their estimated mean failure rates.

6. DECISION RATIONALE

The risk from the storage of spent fuel in the spent fuel storage pool at light water reactors is dominated by the beyond design basis earthquake accident scenario. The seismic capacity, or fragility, of two older spent fuel pools indicate that the high confidence of low probability of failure (HCLPF) is about three times the safe shutdown earthquake (SSE) design level. The HCLPF values are estimated to be 0.5 for the BWR and 0.65 g for the PWR spent fuel pools studied. The safe shutdown earthquake (SSE) for the two plants are 0.14 g and 0.2 g, respectively. The median peak ground acceleration needed to fail these pools is estimated to be in the 1.4 to 2.0 g range, a factor of ten higher than the SSE design value. A report prepared by the American Society of Civil Engineers (Ref. 29) also concluded that, in general, the seismic design of nuclear facility structures results in median factors of safety on the order of 4 to 19 based on post-1973 design criteria.

The structural capacity of the elevated BWR pool is lower than that for the PWR pool located at the ground level, however the lower conditional probability of a Zircaloy fire for the BWR fuel assembly design (0.25 as compared to the PWR value of 1.0) offsets the higher seismic failure frequency. The probability of a Zircaloy cladding fire, resulting from the loss of water from the spent fuel pool, is estimated to have a mean value of $2x10^{-6}$ per reactor year for either the PWR or the BWR spent fuel pool. The seismic event contributes over 90% of the PWR probability, and nearly 95% for the BWR.

The source term for the spent fuel pool accident is not the same as the source term associated with core damage accidents. The consequences of a spent fuel pool accident which results in the complete loss of water is dominated by the long lived isotopes, such as cesium and strontium. The health consequences are dominated by the risk of latent cancer fatalities due to long term exposures.

The best estimate of the consequences of a spent fuel pool accident which results in spent fuel damage to approximately one-third of an equivalent reactor core is $8x10^6$ person-rem. This total dose translates to a public health risk from a spent fuel pool accident of 480 person-rem over an average remaining licensed lifetime of 30 years. The best estimate offsite property damage cost is $4,000 million (1988 $s). The best estimate values are based on a population density of 340 people per square mile within a 50 mile radius from the site and result from the release of radionuclides from the last fuel discharge, 90 days after being discharged. The best estimate of the onsite costs for a SFP accident is $1,180 million (1988 $s), including five years of replacement power to replace the damaged spent fuel pool. Based on an average remaining lifetime of 30 years and a 5% discount rate, the present value of the offsite property damage is estimated to be $124,300 and the present value of the onsite property damage is estimated to be $32,400. As an upper bound, worst case, the consequences of the release of the full fuel pool at a high population site (860 people per square mile within a 50 mile radius from the site), $26x10^6$ person-rem, was used to evaluate the sensitivity of the consequences for proposed alternatives. The corresponding estimate in offsite property damage is $30,000 million (1988 $s).

The consequences, in person-rem, from a spent fuel pool accident are relatively insensitive to the quantity of spent fuel assumed to be released during an accident, when the typical assumptions regarding interdiction dose and decontamination are applied. In the MACCS consequence calculations, no planned evacuation was assumed, however, persons expected to receive more than 25 rem from ground shine in seven days were assumed to be relocated in one day. An additional dose limit over 30 years of 25 rem was also used to determine the interdiction level. MACCS also includes a separate interdiction criteria for crops: crops are interdicted if the resulting ingestion doses would exceed 25 millirem per year. This dose rate is the U.S. Environmental Protection Agency allowable chronic environmental dose rate for normal activities.

The amount of contamination, or land interdiction area, is strongly influenced by the quantity of spent fuel assumed to be released. Sensitivity studies have been performed for the release from the last refueling discharge and for release from the full inventory of a spent fuel pool which has accumulated the equivalent of about four cores in spent fuel assemblies. Sensitivity calculations to study the possible effects of fission product retention on structures and to study the possible effects of a spent fuel pool post-accident spray systems were also performed. The results of these analysis (based on the last discharge assumption) indicated that a decontamination factor assumption of ten reduces the consequences by a factor of two, and the interdiction area by a factor of ten (Ref. 10). A decontamination factor of 45 results in a reduction in consequences of a factor of six and a factor of about 55 in the value of offsite property damage (Ref. 13).

6.1 Comparison to the Backfit Criteria (10 CFR 50.109)

The value impact evaluation, presented in Section 5, for the proposed alternatives for Generic Issue 82 does not indicate that cost effective options are available to mitigate the risk of beyond design basis accidents in spent fuel pools. The option to use low density storage racks for recently discharged fuel has a best estimate value impact ratio of $32,000 per averted person-rem. Low density racks would decrease the frequency of a Zircaloy cladding fire by at least a factor of five to ten, and the value impact ratio is based on 100% reduction in public dose. For the worst case, a high population site with the full fuel pool inventory being released, the value impact ratio is $9,900 per averted person-rem. When compared to the general guideline value of $1,000 per averted person-rem, the low density option is not justified.

The use of a post-accident spray system to mitigate the consequences of a spent fuel pool accident has a best estimate value impact ratio of $3,300 per averted person rem, with a worst case estimate of $1,200 per averted person rem. This assumes that a post-accident spray system can be designed to withstand the beyond design basis earthquake which causes failure of the spent fuel pool structure and has a decontamination factor (DF) of at least 45. Other structures and equipment within the spent fuel storage pool building (for example the refueling crane) would also have to be reviewed to assure that their failure would not compromise the proposed spray system. Under the worst case release assumption, full fuel pool inventory at a high population site, this option is marginally cost beneficial but still exceeds the general guideline value of $1,000 per averted person-rem. However the complete spent fuel pool inventory being released is considered to be highly unlikely. Results of cladding fire propagation calculations indicate that only fuel which is one to two years old could be involved in the release. Also, the demographics are a high estimate of the attributes of a typical plant modification (860 people per square mile).

Potential improvements to the spent fuel pool cooling and make-up systems were also examined. The potential risk to the general public is estimated to be very small, on the order of 3 to 4 person-rem, given a loss of cooling event which results in failure of the spent fuel cladding but not a Zircaloy cladding fire. The value/impact ratios are very large, well in excess of the general guideline value of $1,000 per averted person-rem, however the economics could be important if the spent fuel pool is unavailable and the reactor is shutdown until cleanup and repairs are completed. The cost-benefit ratios for either an additional cooling pump or an additional make-up train were found to be less than one.

Three additional alternatives, (1) to modify the spent fuel storage rack designs, (2) to cover the spent fuel debris with solid materials, and (3) to improve the ventilation gas treatment system, were not explicitly quantified. Compartmental modification to the storage rack designs would result in the displacement of fuel from the spent fuel pool to at-reactor storage casks, a costly option as shown in Alternative 2. Considering that the risk from a spent fuel pool accident is a result of a beyond design basis earthquake, it it highly unlikely that materials could be transported to the site to cover the spent fuel debris in time to reduce the releases of radioactive materials from the spent fuel pool. Finally, since the integrity of the spent fuel building structure following a beyond design basis earthquake is questionable, improvements in the ventilation gas treatment system would be difficult to obtain.

Therefore, the backfit criteria (Ref. 44) that (1) a substantial increase in the overall protection of the public health and safety is achieved, and (2) the direct and indirect costs of implementation are justified, are not met for any of the alternatives considered.

6.2 Comparison to the Safety Goal Policy Statement

The frequency of damage to the spent fuel is estimated to be on the order of $2x10^{-6}$ per reactor year, including the beyond design basis seismic earthquake. This value, when compared to a target value of $1x10^{-4}$ (or $5x10^{-5}$) for a core damage accident, represents a small part of the overall frequency of core damage - 2% to 4%.

The frequency of a release of radioactive material to the environment is assumed to be the same as the frequency of spent fuel damage. The underlying assumption is that the spent fuel pool housing (refueling building, auxiliary building or secondary building) fails due to either the dominant seismic event or due the extreme temperature conditions which would accompany a Zircaloy cladding fire and fuel melting scenario. The spent fuel pool housing does not provide a containment barrier similar to the containment structure surrounding the reactor core, especially under the conditions postulated to dominate the release of radioactive materials.

It is difficult to compare the estimated $2x10^{-6}$ per reactor year release frequency due to a spent fuel pool accident to a target value of $1x10^{-6}$ per reactor year for a large release, particularly without a definition for "large release" . The spent fuel pool source term is not similar to the core damage (or melt) source term and the consequences of a spent fuel pool accident are dominated by latent cancer risks. A possible definition is used in current PRA studies; that is, a "large release" is considered to be an "early, large release" associated with an environmental release within a few hours of a core damage accident (presumably from 100% power). Another definition of a "large release" currently being considered by the staff is a release that has a potential for causing an offsite early fatality (see for example NUREG-1150, Ref. 42). Either of these definitions, in particular any consideration for early fatalities, appear to suggest that the spent fuel pool release is not a "large release."

Societal risk to the public is based on the statistically expected number of early and latent cancer fatalities. The Safety Goal Policy Statement (Ref. 45) currently defines the early fatality area calculation as that within one mile from the site boundary. A ten mile radius is defined for calculating latent cancer fatalities. The language of the Policy Statement also requires that the risk from an accident at a nuclear power plant be 0.1% of that normally encountered by the public. Based on recent data (Ref. 46) the total fatality rate from cancer in the U.S. is 189.3 per 100,000 persons, or a risk of 1.9×10^{-3} per year. Therefore it can be inferred that a latent cancer fatality rate for nuclear power plant operations of 2×10^{-6} per reactor year, or less, is consistent with the safety goal.

To meet the general objective for societal risk, the probability of a latent cancer fatality from a spent fuel pool accident should not be more than a relatively small fraction of an overall target value for nuclear power plant operations. The best estimate MACCS calculation for the spent fuel pool source term, for 340 people per square mile over a 50 mile radius, predicts a consequence of 8 million person-rem per event. The dose conversion factor for latent cancer fatalities is in the 150 to 200 latent cancer fatalities per million person-rem range. The expected number of latent cancer fatalities is 1,600 per event, and the latent cancer fatality rate would be 0.0032 per reactor year (1,600 latent cancer fatalities per event times 2×10^{-6} events per reactor year) for the affected population.

The mean population within a 10 mile radius of a reactor site is 57,000 people (based on a mean density of 182 people per square mile), and 2,670,000 people within a 50 mile radius (based on 340 people per square mile) in the year 2,000 (Ref. 47). The expected number of cancer fatalities from all causes in the 50 mile radius is 0.2% of the population, or 5,340 per year. In a 10 mile radius, the expected number of cancer fatalities is 114 per year. Using 0.1% of 10 mile radius value, a target value for latent cancer fatalities from the operation of a nuclear power plant would be less than 0.114 latent cancer fatalities per reactor year. The 0.0032 latent cancer fatalities per reactor year associated with the spent fuel pool accident is less than 3% of the 0.114 per year target value based on the calculation area specified in the Safety Goal Policy Statement, even without correcting for the fact that only a fraction of the 50 mile radius latent cancer fatalities would occur within the 10 mile radius.

The estimated frequency of a spent fuel pool accident, 2×10^{-6} reactor year, resulting in spent fuel damage meets a target objective of a few percent of a 1×10^{-4} to 5×10^{-5} per reactor year value for overall core damage frequency. The target objective for a "large release" of 1×10^{-6} per reactor year is marginally met, within a best estimate factor of two, but subject to interpretation since the definition of "large release" is still under development. In meeting the societal risk objective of 0.1% of the normally occurring risk to the public given the release frequency of 2×10^{-6} per reactor year, the latent cancer fatality rate from a spent fuel pool accidents is estimated to be less than 3% of the target value for the operation of a nuclear power plant.

Therefore, the risk and consequences of a spent fuel pool accident appear to meet the Safety Goal Policy Statement public health objectives. They would also meet the proposed 1×10^{-6} per reactor year large-release frequency guidelines, at least pending definition of a "large release" by the Commission. Therefore, Alternative 1 - "No Action" is justified.

6.3 Other Considerations

In addition to implementing the requirements contained in 10 CFR Part 50 Appendix A of the "General Design Criteria," and 10 CFR Part 20, concerning radiation doses being kept as low as is reasonably achievable, licensees should have implemented additional or corrective actions based on the following guidance:

1. IE Bulletin 84-03, "Refueling Cavity Water Seals," issued August 24, 1984. (Ref. 21)

2. IE Information Notice 84-93, "Potential for Loss of Water From the Refueling Cavity," issued December 17, 1984. (Ref. 24)

3. Generic Letter 85-11, "Completion of Phase II of 'Control of Heavy Loads at Nuclear Power Plants' NUREG-0612," issued June 28, 1985. (Ref. 4)

4. IE Information Notice 87-13, "Potential for High Radiation Fields Following Loss of Water from Fuel Pool," issued February 24, 1987. (Ref. 34)

5. IE Information Notice 87-43, "Gaps in Neutron-Absorbing Material in High-Density Spent Fuel Storage Racks," issued September 8, 1987. (Ref. 33)

6. IE Information Notice 88-65, "Inadvertent Drainages of Spent Fuel Pools," issued August 18, 1988. (Ref. 25)

7. IE Information Notice 88-92, "Potential for Spent Fuel Pool Draindown," issued November 22, 1988. (Ref. 26)

Based on compliance with the GDCs and licensees taking corrective actions identified as a result of reviewing facility designs and operations based on IE Bulletins and Information Notices, the frequency of a spent fuel pool accident resulting in a Zircaloy cladding fire and the release of fission products to the environment from internal events, such as missiles, heavy load drops, loss of cooling or make-up, inadvertent drainage or siphoning and pneumatic seal failures, is estimated to be on the order of $2x10^{-7}$ per reactor year. 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 in response to the concerns identified to prevent similar occurrences, or at least understand the potential consequences of these events and develop appropriate procedures to respond to them and to mitigate the consequences.

The overall frequency of a spent fuel pool accident resulting in a release of radioactive materials to the environment is estimated to be $2x10^{-6}$ per reactor year for a light water reactor spent fuel storage pool when the external seismic hazard is included. The beyond design basis earthquake dominates the risk, 90% to 95% of the total. The HCLPF value is estimated to be three times the safe shutdown earthquake (SSE) value peak ground acceleration value, in the 0.5 to 0.65 g range. The median capacity is estimated to be in the 1.4 to 2.0 g range. 10 CFR Part 100 Appendix III.(c) defines an SSE as:

"that earthquake which is based upon an evaluation of the maximum earthquake potential considering regional and local geology and seismology, and specific characteristics of local subsurface material. It is that earthquake which produces the maximum vibratory ground motion for which certain structures, systems, and components are designed to remain functional. These structures, systems, and components are those necessary to assure: (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shutdown the reactor and maintain it in a safe shut down condition, or (3) the capability to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guideline exposures of 10 CFR Part 100."

In NUREG/CR-5042, Supplement 1 (Ref. 31), LLNL reviewed available PRA literature to determine the seismic hazard contribution to core damage accidents. A review of analyses for A-45, "Decay Heat Removal Requirements," indicates that the dominant earthquake range for core damage falls within the 0.2 to 0.4 g range. A review of Zion and LaSalle Seismic Safety Margins Research Program (SSMRP) analyses also concludes that the 0.2 to 0.4 g range dominates core damage from seismic initiators. The dominant component failures, contributing to core damage, were found to be:

1. Yard Tanks - condensate storage tanks, refueling water storage tanks.

2. Electrical Equipment - batteries, buses, cabinet anchorage, contacts, relays, transformers.

3. Diesel Generator Peripherals - fuel oil tanks, lube oil tanks, coolers.

4. Structural failures - block walls, service water buildings, reactor internals.

5. Equipment Anchorages.

In other words, this type of spent fuel pool accident requires an earthquake larger than that which would result in core damage and the release of radioactive material to the environment. The mean core damage frequencies due to the seismic hazard are in the $3x10^{-6}$ to $1.4x10^{-4}$ per reactor year range based on published PRA results, with seismic related release frequencies in the range of $2x10^{-7}$ to $1.4x10^{-4}$ per reactor year for peak ground accelerations in the 0.2 to 0.4 g range (NUREG/CR-5042, Supplement 1, Table 3-3, Ref. 31). The spent fuel pool accident is estimated to have a frequency on the order of $2x10^{-6}$ per reactor year for a peak ground acceleration in excess of 0.5 g.

In estimating the likelihood of a beyond design basis earthquake resulting in a failure of a spent fuel pool, uncertainty can be introduced into the evaluation when attempting to characterize the seismic hazard of a site. The seismic hazard is a quantification of the probability of exceeding a given peak ground acceleration on an annual basis. As shown in NUREG/CR-5176, and also noted in an NRC memorandum dated December 29, 1988 (Ref. 48), the uncertainty in estimating the seismic risk is about an order of magnitude, and relates to how expert judgment is used in the development of the site characterizations.

For each of the two plants studied by LLNL in NUREG/CR-5176, a family of seismic hazard curves were convolved with a family of plant-level seismic fragility curves to obtain a probability distribution of the frequency of occurrence of the seismic initiated accident under study. At the time this work was performed by LLNL, the complete family of seismic hazards curves were not available for the two plants studied, other than at some selected percentile values. When the seismic hazard curves are grouped in this manner, the specific features of the individual hazard curves (for example, they may intersect one another) are lost. Median and 95 percentile hazard curves were used by LLNL to develop a discrete set of seismic hazard curves for each of the two plants studied. A lognormal distribution was used for the purpose of obtaining approximate risk estimates. The resulting lognormal distribution was cutoff at different percentile values to judge the sensitivity of the results. A cutoff value of 99 percent was recommended for use by LLNL in NUREG/CR-5176. The resultant frequency estimate for spent fuel damage due to a beyond design basis earthquake is 2.0×10^{-6} per reactor year for the LWR spent fuel pools studied in NUREG/CR-5176.

More recently, EQE Engineering, Inc., the same subcontractor employed by LLNL for the NUREG/CR-5176 effort, re-evaluated the seismic risk for the same two plants based on true mean seismic hazard curve data (Ref. 48). EQE provided two sets of results based on the use of two sets of experts, the "5 G-Experts" and the "4 G-Experts." The resultant mean annual frequency of failure of the spent fuel pool structures decreases by a factor of 8.8 for the BWR spent fuel pool and 2.8 for the PWR spent fuel pool by removing one seismic ground motion expert, or "outlier," from the seismic hazard characterization estimate (for example when going from the "5 G-Expert" to the "4 G-Expert" ground motion expert judgment). Similar results were obtained by LLNL in NUREG/CR-5176 in going from a cutoff value of 100 percent to 99 percent, by eliminating a small portion of the tails from the lognormal distribution curves. Since the tail of the lognormal distribution extends to infinity, it might be possible to get values of the probability of exceedance greater than one. Truncation of the lognormal distribution curves at an exceedance value less than one, at 0.99, was used in the LLNL study. The relative magnitudes are similar. For the "5 G-Expert" values, the more recent plant specific BWR seismic failure frequency from the EQE study could be a factor of 5.5 higher than the earlier LLNL evaluation. Similarly, the more recent plant specific PWR seismic failure frequency could be a factor of 2.6 higher than the earlier LLNL evaluation. Based on the "4 G-Expert" values, the earlier LLNL evaluation of the seismic failure frequency is slightly higher than the more recent EQE values for both spent fuel pools studied. The mean seismic failure frequencies for the two methods are summarized in Table 6.3.1.

Due to the skewed nature of the distribution of expert judgment, the mean is a highly unstable estimate of the seismic hazard . In these distributions the most extreme opinion weighs heavily when the mean is calculated. The mean, which is an arithmetic average of all inputs, frequently exceeds the 85th percentile of all the inputs. This problem created by the skewed distribution of expert judgment exists for either method, the actual true arithmetic mean or the lognormal distribution.

A re-evaluation for Alternative 2, the use of low-density storage racks for recently discharged fuel, using these higher seismic failure frequencies results in a best estimate value/impact ratio of $9,500 per averted person-rem. Using the worst case assumptions, the value/impact ratio is

$3,000 per averted person-rem. Alternative 2 is judged to be the most practical option for reducing the risk and the implementation costs are well defined. While a re-evaluation for Alternative 4, the installation of a post-accident spray system, indicates a marginally acceptable best estimate value/impact ratio of $1,050 per averted person-rem, the uncertainty in the implementation cost of this option is large. The implementation cost is based solely on the installation of the spray system and does not consider the potential for the need to reinforce other parts of the spent fuel storage building structures to assure that their failure in a beyond design basis earthquake would not compromise the spray system. Therefore, even with the higher seismic frequencies, the staff would not conclude that any of the options considered would be cost-effective.

Although these studies conclude that most of the spent fuel pool risk is derived from beyond design basis earthquakes, this risk is no greater than the risk from core damage accidents due to seismic events beyond the safe-shutdown earthquake. Therefore, reducing the risk from spent fuel pools due to events beyond the safe-shutdown earthquake would still leave a comparable risk due to core damage accidents. Because of the large inherent safety margins in the design and construction of the spent fuel pool, Alternative 1 - "No Action" is justified.

When taken together, the discussions presented in Sections 6.1, 6.2 and 6.3 form the basis for a decision that no corrective actions are justified. The risk due to beyond design basis accidents in spent fuel pools, while not negligible, are sufficiently low that the added costs involved with further risk reductions are not warranted.

Table 6.3.1
Summary of SFP Seismic Failure Frequency Estimates

Pool Type	NUREG/CR-5176 Results		NRR True Mean Results	
	Cutoff Value (per cent)	Frequency (per R-year)	Expert Group	Frequency (per R-year)
Elevated BWR	100	3.8×10^{-5}	5 G-Experts	3.7×10^{-5}
	99	6.7×10^{-6}	4 G-Experts	4.2×10^{-6}
On Ground PWR	100	8.6×10^{-6}	5 G-Experts	4.7×10^{-6}
	99	1.8×10^{-6}	4 G-Experts	1.7×10^{-6}

Note: The NUREG/CR-5176 frequencies at the 99 per cent cutoff level were used in this Regulatory Analysis as being representative of the best estimate, generic values for an elevated BWR spent fuel pool and a PWR spent fuel pool located at the ground elevation.

7. IMPLEMENTATION

No regulatory action is necessary for the resolution of this issue. This regulatory analysis and the supporting contractor reports have been made publicly available as part of their normal distributions.

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Appendix A

Spent Fuel Data and Storage Requirements

The Department of Energy (DOE), Office of Civilian Radioactive Waste Management (OCRWM) is responsible for the management and ultimate permanent disposal of the civilian spent fuel and high level radioactive waste generated as a result of commercial nuclear power plant operations in the U.S. This responsibility is prescribed under the provisions of the Nuclear Waste Policy Act of 1982 (NWPA) (as amended).

The greatest portion of the radioactive waste covered under this government responsibility will be spent nuclear fuel discharged from commercial nuclear power plants. Because most of the spent fuel that will ultimately require disposal has not yet been generated, planning for the management and disposal of this spent fuel must be largely based on projections of future spent fuel discharges from commercial nuclear power plants.

The OCRWM plans for management and disposal of spent fuel are based on the DOE Energy Information Administration (EIA) nuclear energy projections. These data are used for this Regulatory Analysis, in support of resolution of Generic Issue 82, "Beyond design Basis Accidents in Spent Fuel Pool," to estimate the additional cost of at-reactor storage of spent fuel.

The data source is DOE/RL-87-11, "Spent Fuel Storage Requirements 1987," September 1987, United States Department of Energy, Richland Operations Office. Data used by the NRC, taken from this report, are provided in Tables A.1 through A.4 of this Appendix.

Table A.1. Nuclear Power Plant Data

	PLANT NAME	UTILITY NAME	STATE	REAC TYPE	DESIGN NET (MWE)	STARTUP DATE	SHUTDOWN DATE	PRESENT(a) CAPAC.	MAX.(a) CAPAC.	REACTOR VENDOR	FULL CORE SIZE (a)	MTIHM
	ARK NUCLEAR 1	ARK PWR & LGT CO.	AR	PWR	850	1974	2008	968	968	BW	177	82
	ARK NUCLEAR 2	ARK PWR & LGT CO.	AR	PWR	912	1980	2012	988	988	CE	177	74
	BEAVER VALLEY 1	DUQUESNE LIGHT COMPANY	PA	PWR	835	1976	2016	833	833	WE	157	73
	BEAVER VALLEY 2	DUQUESNE LIGHT COMPANY	PA	PWR	857	1987	2026	1088	1088	WE	157	72
	BELLEFONTE 1	TENNESSEE VALLEY AUTHORITY	AL	PWR	1235	1992(f)	2028	1058	1058	BW	205	93
	BELLEFONTE 2	TENNESSEE VALLEY AUTHORITY	AL	PWR	1235	1995(f)	2030	270	1058	BW	205	93
	BIG ROCK 1	CONSUMERS PWR CO.	MI	BWR	72	1965	2001	441	441	GE	84	11
A	BRAIDWOOD 1	COMMONWEALTH EDISON COMPANY	IL	PWR	1175	1987	2026	1050	1050	WE	193	82
A	BRAIDWOOD 2	COMMONWEALTH EDISON COMPANY	IL	PWR	1175	1988	2027	0	0	WE	193	82
B	BROWNS FERRY 1	TENNESSEE VALLEY AUTHORITY	AL	BWR	1065	1974	2014	3471	3471	GE	764	140
B	BROWNS FERRY 2	TENNESSEE VALLEY AUTHORITY	AL	BWR	1065	1975	2014	3133	3471	GE	764	140
	BROWNS FERRY 3	TENNESSEE VALLEY AUTHORITY	AL	BWR	1065	1977	2017	2353	3471	GE	764	140
	BRUNSWICK 1	CAROLINA POWER & LIGHT COMPANY	NC	BWR(c)	821	1977	2010	1767	1803	GE	560	103
	BRUNSWICK 2	CAROLINA POWER & LIGHT COMPANY	NC	BWR(c)	821	1975	2010	1325	1839	GE	560	102
A	BYRON 1	COMMONWEALTH EDISON COMPANY	IL	PWR	1120	1985	2024	1050	1050	WE	193	82
A	BYRON 2	COMMONWEALTH EDISON COMPANY	IL	PWR	1120	1987	2026	0	0	WE	193	82
	CALLAWAY 1	UNION ELEC COMPANY	MO	PWR	1171	1984	2024	1340	1340	WE	193	86
B	CALVERT CLF 1	BALTIMORE GAS & ELEC CO.	MD	PWR	845	1975	2014	830	830	CE	217	84
B	CALVERT CLF 2	BALTIMORE GAS & ELEC CO.	MD	PWR	845	1977	2016	1000	1000	CE	217	84
	CATAWBA 1	DUKE POWER COMPANY	SC	PWR	1145	1985	2025	1419	2615	WE	193	89
	CATAWBA 2	DUKE POWER COMPANY	SC	PWR	1145	1986	2026	1421	2615	WE	193	82
	CLINTON 1	ILLINOIS PWR CO.	IL	BWR	933	1987(f)	2027	2672	2672	GE	624	114
B	COMANCHE PK 1	TEXAS UTILITIES GENERATING CO.	TX	PWR	1150	1989	2030	260	1695	WE	193	89
B	COMANCHE PK 2	TEXAS UTILITIES GENERATING CO.	TX	PWR	1150	1989	2030	0	1687	WE	193	83
A	COOK 1	INDIANA & MICH ELEC CO.	MI	PWR	1030	1975	2009	2048	2270	WE	193	88
A	COOK 2	INDIANA & MICH ELEC CO.	MI	PWR	1100	1978	2009	0	0	WE	193	78
	COOPER STN	NEBRASKA PUB PWR DISTRICT	NE	BWR	778	1974	2008	2366	2366	GE	548	101
	CRYSTAL RVR 3	FLORIDA PWR CORP	FL	PWR	825	1977	2016	676	1157	BW	177	82
	DAVIS-BESSE 1	TOLEDO EDISON CO.	OH	PWR	906	1978	2017	735	735	BW	177	83
	DIABLO CANYON 1	PACIFIC GAS AND ELECTRIC CO.	CA	PWR	1086	1985	2025	270	1324	WE	193	89
	DIABLO CANYON 2	PACIFIC GAS AND ELECTRIC CO.	CA	PWR	1119	1986	2025	270	1324	WE	193	89
	DRESDEN 1	COMMONWEALTH EDISON COMPANY	IL	BWR	200	1960	1984	720	720	GE	464	47
	DRESDEN 2	COMMONWEALTH EDISON COMPANY	IL	BWR	794	1970	2008	3537	3537	GE	724	125
	DRESDEN 3	COMMONWEALTH EDISON COMPANY	IL	BWR	794	1971	2006	3537	3537	GE	724	125
	DUANE ARNOLD	IOWA ELEC LIGHT & POWER CO.	IA	BWR	538	1975	2010	2050	2050	GE	368	67
	ENRICO FERMI 2	DETROIT EDISON COMPANY	MI	BWR	1093	1987(f)	2025	2305	2305	GE	764	140
	FARLEY 1	ALABAMA POWER COMPANY	AL	PWR	829	1977	2012	1407	1407	WE	157	73
	FARLEY 2	ALABAMA POWER COMPANY	AL	PWR	829	1981	2012	1407	1407	WE	157	73
	FITZPATRICK	PWR AUTHORITY OF STATE OF NY	NY	BWR	821	1975	2015	2244	2854	GE	560	103
	FORT CALHOUN	OMAHA PUB PWR DIST	NE	PWR	486	1973	2008	729	729	CE	133	47
	FT ST VRAIN	PUB SVC CO OF COLORADO	CO	HTG	330	1979	2007	504	504	GA	1482	16
	GINNA	ROCHESTER GAS & ELEC CORP	NY	PWR	490	1970	2006	1016	1016	WE	121	43
	GRAND GULF 1	SYSTEM ENERGY RESOURCES, INC.	MS	BWR	1250	1985	2022	3124	3124	GE	800	145

Table A.1. Nuclear Power Plant Data (con't)

	PLANT NAME	UTILITY NAME	STATE	REAC TYPE	DESIGN NET (MWE)	STARTUP DATE	SHUTDOWN DATE	PRESENT(a) CAPAC.	MAX.(a) CAPAC.	REACTOR VENDOR	FULL CORE SIZE (a)	MTIHM
	HADDAM NECK	NORTHEAST UTILITIES	CT	PWR	582	1968	2007	1168	1168	WE	157	64
	HARRIS 1	CAROLINA POWER & LIGHT COMPANY	NC	PWR	940	1987(f)	2026	480	3351	WE	157	73
B	HATCH 1	GEORGIA PWR COMPANY	GA	BWR	777	1974	2009	3025	3181	GE	560	103
B	HATCH 2	GEORGIA PWR COMPANY	GA	BWR	784	1979	2012	2765	2845	GE	560	104
	HOPE CREEK	PUBLIC SERV. ELEC AND GAS CO.	NJ	BWR	1118	1987(f)	2026	1078	3976	GE	764	141
	HUMBOLDT BAY	PACIFIC GAS AND ELECTRIC CO.	CA	BWR	65	1963	1976	486	486	GE	184	13
	INDIAN PT 1	CONSOLIDATED EDISON CO.	NY	PWR	265	1962	1980	756	756	BW	120	23
	INDIAN PT 2	CONSOLIDATED EDISON CO.	NY	PWR	873	1974	2006	980	980	WE	193	88
	INDIAN PT 3	PWR AUTHORITY OF STATE OF NY	NY	PWR	965	1976	2015	840	1317	WE	193	89
	KEWAUNEE	WISCONSIN PUBLIC SERVICE CORP	WI	PWR	535	1974	2014	603	963	WE	121	46
	LACROSSE	DAIRYLAND PWR COOP	WI	BWR	50	1969	2002	440	440	AC	72	8
B	LASALLE CTY 1	COMMONWEALTH EDISON COMPANY	IL	BWR	1122	1982	2022	1080	1080	GE	764	140
B	LASALLE CTY 2	COMMONWEALTH EDISON COMPANY	IL	BWR	1122	1984	2023	1080	1080	GE	764	140
	LIMERICK 1	PHILADELPHIA ELEC CO.	PA	BWR	1055	1986	2024	2040	2040	GE	764	141
	LIMERICK 2	PHILADELPHIA ELEC CO.	PA	BWR	1055	1990	2029	2040	2040	GE	764	140
	MAINE YANKEE	MAINE YANKEE ATOMIC PWR CO.	ME	PWR	825	1972	2008	1476	1476	CE	217	80
	MCGUIRE 1	DUKE POWER COMPANY	NC	PWR	1180	1981	2021	1359	1463	WE	193	89
	MCGUIRE 2	DUKE POWER COMPANY	NC	PWR	1180	1984	2023	1421	1463	WE	193	89
	MILLSTONE 1	NORTHEAST UTIL SVC CO.	CT	BWR	660	1970	2010	2184	2184	GE	580	103
	MILLSTONE 2	NORTHEAST UTIL SVC CO.	CT	PWR	870	1975	2015	1112	1112	CE	217	88
	MILLSTONE 3	NORTHEAST UTIL SVC CO.	CT	PWR	1150	1986	2025	756	1836	WE	193	89
	MONTICELLO	NORTHERN STATES PWR COMPANY	MN	BWR	545	1971	2007	2217	2237	GE	484	86
	NINE MILE PT 1	NIAGARA MOHAWK POWER CORP	NY	BWR	620	1969	2005	2362	2776	GE	532	94
	NINE MILE PT 2	NIAGARA MOHAWK POWER CORP	NY	BWR	1080	1987	2026	2530	4049	GE	764	140
A	NORTH ANNA 1	VIRGINIA POWER	VA	PWR	907	1978	2018	1737	1737	WE	157	72
A	NORTH ANNA 2	VIRGINIA POWER	VA	PWR	907	1980	2020	0	0	WE	157	73
A	OCONEE 1	DUKE POWER COMPANY	SC	PWR	887	1973	2013	1298	1312	BW	177	82
A	OCONEE 2	DUKE POWER COMPANY	SC	PWR	887	1974	2013	0	0	BW	177	82
	OCONEE 3	DUKE POWER COMPANY	SC	PWR	886	1974	2014	818	825	BW	177	82
	OYSTER CRK 1	GPU NUCLEAR	NJ	BWR	650	1969	2004	2600	2600	GE	560	98
	PALISADES	CONSUMERS PWR CO.	MI	PWR	805	1971	2011	798	798	CE	204	80
	PALO VERDE 1	ARIZONA PUBLIC SERVICE CO.	AZ	PWR	1270	1986	2024	665	1329	CE	241	99
	PALO VERDE 2	ARIZONA PUBLIC SERVICE CO.	AZ	PWR	1270	1986	2025	665	1329	CE	241	99
	PALO VERDE 3	ARIZONA PUBLIC SERVICE CO.	AZ	PWR	1270	1987	2026	665	1329	CE	241	99
	PEACHBOTTOM 2	PHILADELPHIA ELEC CO.	PA	BWR	1065	1974	2008	3814	3814	GE	764	140
	PEACHBOTTOM 3	PHILADELPHIA ELEC CO.	PA	BWR	1065	1974	2008	3819	3819	GE	764	140
	PERRY 1	CLEVELAND ELEC ILLUM CO.	OH	BWR	1265	1987	2026	4020	4020	GE	748	138
	PILGRIM 1	BOSTON EDISON CO.	MA	BWR	655	1972	2008	2320	2320	GE	580	103
A	POINT BEACH 1	WISCONSIN ELEC PWR CO.	WI	PWR	497	1970	2007	1502	1502	WE	121	46
A	POINT BEACH 2	WISCONSIN ELEC PWR CO.	WI	PWR	497	1972	2008	0	0	WE	121	45
A	PRAIRIE ISL 1	NORTHERN STATES PWR CO.	MN	PWR	530	1973	2008	1386	1386	WE	121	44
A	PRAIRIE ISL 2	NORTHERN STATES PWR CO.	MN	PWR	530	1974	2008	0	0	WE	121	40

Table A.1. Nuclear Power Plant Data (con't)

	PLANT NAME	UTILITY NAME	STATE	REAC TYPE	DESIGN NET (MWE)	STARTUP DATE	SHUTDOWN DATE	PRESENT(a) CAPAC.	MAX.(a) CAPAC.	REACTOR VENDOR	FULL CORE SIZE (a)	MTIHM
B	QUAD CITIES 1	COMMONWEALTH EDISON COMPANY	IL	BWR	789	1973	2007	3657	3657	GE	724	129
B	QUAD CITIES 2	COMMONWEALTH EDISON COMPANY	IL	BWR	789	1973	2007	3897	3897	GE	724	126
	RANCHO SECO 1	SACRAMENTO MUNICIP UTIL DISTR	CA	PWR	918	1975	2008	1080	1080	BW	177	82
	ROBINSON 2	CAROLINA POWER & LIGHT COMPANY	SC	PWR	700	1971	2007	544	544	WE	157	66
	RVR BEND 1	GULF STATES UTILITIES	LA	BWR	936	1986	2025	3172	3172	GE	624	116
	SALEM 1	PUBLIC SERV. ELEC. AND GAS CO.	NJ	PWR	1115	1977	2016	1133	1170	WE	193	89
	SALEM 2	PUBLIC SERV. ELEC. AND GAS CO.	NJ	PWR	1115	1981	2020	1140	1170	WE	193	89
E	SAN ONOFRE 1	SOUTHERN CALIF EDISON CO.	CA	PWR	436	1968	1999	216	216	WE	157	58
E	SAN ONOFRE 2	SOUTHERN CALIF EDISON CO.	CA	PWR	1070	1983	2012	800	800	CE	217	91
E	SAN ONOFRE 3	SOUTHERN CALIF EDISON CO.	CA	PWR	1080	1984	2013	800	800	CE	217	90
	SEABROOK 1	NHY DIVISION OF PSNH	NH	PWR	1150	1987(f)	2031	660	1236	WE	193	89
A	SEQUOYAH 1	TENNESSEE VALLEY AUTHORITY	TN	PWR	1148	1981	2021	1381	1381	WE	193	89
A	SEQUOYAH 2	TENNESSEE VALLEY AUTHORITY	TN	PWR	1148	1982	2022	0	0	WE	193	89
	SHOREHAM	LONG ISL LGT CO.	NY	BWR	849	1988(b)	2027	2176	2685	GE	560	102
	SOUTH TEXAS 1	HOUSTON LIGHTING & POWER CO.	TX	PWR	1250	1987	2027	196	1969	WE	193	105
	SOUTH TEXAS 2	HOUSTON LIGHTING & POWER CO.	TX	PWR	1250	1989	2028	0	1969	WE	193	104
	ST LUCIE 1	FLORIDA PWR & LGT CO.	FL	PWR	830	1976	2010	728	728	CE	217	81
	ST LUCIE 2	FLORIDA PWR & LGT CO.	FL	PWR	804	1983	2023	1076	1076	CE	217	81
	SUMMER 1	SOUTH CAROLINA ELEC & GAS CO.	SC	PWR	900	1984	2024	1276	1276	WE	157	72
A	SURRY 1	VIRGINIA POWER	VA	PWR	788	1972	2012	1044	1044	WE	157	72
A	SURRY 2	VIRGINIA POWER	VA	PWR	788	1973	2013	1764	1764(e)	WE	157	72
B	SUSQUEHANNA 1	PENNSYLVANIA PWR & LGT CO.	PA	BWR	1065	1983	2022	2840	2840	GE	764	137
B	SUSQUEHANNA 2	PENNSYLVANIA PWR & LGT CO.	PA	BWR	1065	1985	2024	2840	2840	GE	764	137
	THREE MILE ISL 1	GPU NUCLEAR	PA	PWR	819	1974	2008	752	1401(g)	BW	177	82
	TROJAN	PORTLAND GENERAL ELEC	OR	PWR	1130	1976	2011	1408	1408	WE	193	89
E	TURKEY PT 3	FLORIDA PWR & LGT CO.	FL	PWR	693	1972	2007	1376	1404	WE	157	72
E	TURKEY PT 4	FLORIDA PWR & LGT CO.	FL	PWR	693	1973	2007	614	636	WE	157	72
B	VOGTLE 1	GEORGIA POWER COMPANY	GA	PWR	1069	1987	2027	288	1117(g)	WE	193	89
B	VOGTLE 2	GEORGIA POWER COMPANY	GA	PWR	1069	1988	2028	288	1117(g)	WE	193	89
	VT YANKEE 1	VT YANKEE NUCLEAR PWR CORP	VT	BWR	514	1972	2012	1690	2870	GE	368	68
	WASH NUCLEAR 2	WASH PUB PWR SUPPLY SYSTEM	WA	BWR	1100	1984	2023	2658	2658	GE	764	140
	WATERFORD 3	LOUISIANA POWER & LIGHT	LA	PWR	1104	1985	2024	1088	1366	CE	217	89
A	WATTS BAR 1	TENNESSEE VALLEY AUTHORITY	TN	PWR	1165	1989(f)	2025	1294	1294	WE	193	89
A	WATTS BAR 2	TENNESSEE VALLEY AUTHORITY	TN	PWR	1165	1990(f)	2027	0	0	WE	193	89
	WOLF CREEK 1	WOLF CREEK NUCLEAR OPERATING CO.	KS	PWR	1150	1985	2025	1327	1340	WE	193	89
	YANKEE-ROWE 1	YANKEE ATOMIC ELEC CO.	MA	PWR	175	1961	2001	440	721	WE	76	18
A	ZION 1	COMMONWEALTH EDISON COMPANY	IL	PWR	1085	1973	2008	2079	2079	WE	193	89
A	ZION 2	COMMONWEALTH EDISON COMPANY	IL	PWR	1085	1974	2008	0	0	WE	193	89

A INDICATES COMMON POOL SHARED BY TWO REACTORS

B INDICATES POOLS CONNECTED BY TRANSFER CANAL: CAPACITIES AND INVENTORIES ARE COMBINED WITH ONLY ONE FULL CORE RESERVE

E INDICATES POOLS REQUIRING CASK TRANSFER: CAPACITIES AND INVENTORIES ARE COMBINED WITH ONLY ONE FULL CORE RESERVE

Table A.1. Nuclear Power Plant Data (con't)

	STORAGE SITES	UTILITY NAME	STATE	TYPE FUEL STORED	STARTUP DATE	SHUTDOWN DATE	PRESENT(a) CAPAC.	MAX.(a) CAPAC.
(c)	BRUNSWICK 1 PWR	CAROLINA POWER & LIGHT COMPANY	NC	PWR	1977	2010	160	160
(c)	BRUNSWICK 2 PWR	CAROLINA POWER & LIGHT COMPANY	NC	PWR	1975	2010	144	144
	DOE ID (INEL) (EG&G)	DEPARTMENT OF ENERGY	ID	BWR			(h)	(h)
	DOE ID (INEL) (EG&G)	DEPARTMENT OF ENERGY	ID	HTG			(h)	(h)

	STORAGE SITES	UTILITY NAME	STATE	TYPE FUEL STORED	STARTUP DATE	SHUTDOWN DATE	PRESENT(a) CAPAC.	MAX.(a) CAPAC.
	DOE ID (INEL) (EG&G)	DEPARTMENT OF ENERGY	ID	PWR			(h)	(h)
	DOE OH (BATTELLE)	DEPARTMENT OF ENERGY	OH	PWR			(h)	(h)
	DOE WA (HANFORD)	DEPARTMENT OF ENERGY	WA	BWR			(h)	(h)
	DOE WA (HANFORD)	DEPARTMENT OF ENERGY	WA	PWR			(h)	(h)
(d)	HARRIS 1 BWR POOL	CAROLINA POWER & LIGHT COMPANY	NC	BWR	1987	2026	0	2057(g)
	MORRIS-BWR/PWR	MORRIS OPERATION (AFR)	IL	BWR		2002	3735(i)	3775(i)
	MORRIS-BWR/PWR	MORRIS OPERATION (AFR)	IL	PWR		2002	1660(i)	1660(i)
	WEST VALLEY	WEST VALLEY DEMONSTRATION PRJ.	NY	BWR			(h)	(h)
	WEST VALLEY	WEST VALLEY DEMONSTRATION PRJ.	NY	PWR			(h)	(h)
(j)	OTHER			PWR			(h)	(h)

BWR PLANTS TOTAL :	39	CURRENTLY OPERATING:	32	TOTAL MWE:	33705	CURRENTLY OPERATING MWE:	26312
PWR PLANTS TOTAL :	80	CURRENTLY OPERATING:	63	TOTAL MWE:	75800	CURRENTLY OPERATING MWE:	56375
HTGR PLANTS TOTAL :	1	CURRENTLY OPERATING:	1	TOTAL MWE:	330	CURRENTLY OPERATING MWE:	330
OPERATING & PLANNED PLANTS TOTAL :	120	CURRENTLY OPERATING:	96	TOTAL MWE:	109835	CURRENTLY OPERATING MWE:	83017
RETIRED PLANTS TOTAL :	3			TOTAL MWE:	530		

NOTE: UTILITY DATA AS OF 12/31/1986

(a) IN ASSEMBLIES

(b) SHOREHAM ISSUED A LICENSE IN 1983 BUT HAS NOT OPERATED. 1988 STARTUP ESTIMATED BASED ON PROJECTED FIRST DISCHARGE IN 1989.

(c) SOME ROBINSON 2 PWR FUEL IS STORED AT THE BRUNSWICK (BWR) REACTORS.

(d) IN 1985, HARRIS 1 IDENTIFIED SPACE FOR THE FUTURE STORAGE OF BWR FUEL. (HARRIS 1 IS A PWR.)

(e) INCLUDES STORAGE CAPACITY OF DRY STORAGE INSTALLATION (ISFSI).

(f) STARTUP DATE BASED ON PROJECTED YEAR OF FIRST DISCHARGE.

(g) CURRENT AS OF 12/31/86

(h) CAPACITY FOR STORAGE UNKNOWN.

(i) POOL CAN HOLD BOTH FUEL TYPES. CAPACITY SHOWN REFLECTS ENTIRE POOL IN USE FOR ONE TYPE OF FUEL ONLY.

(j) ONE ROBINSON ASSEMBLY HAS BEEN SENT TO A LOCATION WHICH DOES NOT HAVE AN EIA ID.

Table A.2. Projected Cumulative Storage Requirements--Maximum AR Capacity, Assemblies

ASSEMBLIES

POOL		1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005
ST LUCIE 1	PWR	42	122	122	194	270	270	346	422	422	498	574	574	650	726	726	802	878	878	954
MILLSTONE 1	BWR	128	128	324	324	520	520	716	716	912	912	1108	1108	1304	1304	1500	1500	1696	1696	1892
PALISADES	PWR	0	19	19	87	87	155	155	223	291	291	359	359	427	427	495	563	563	631	631
OCONEE 1&2	PWR	0	0	62	182	242	302	422	482	602	662	782	842	962	1022	1082	1202	1322	1322	1442
OCONEE 3	PWR	0	0	1	1	61	121	121	181	241	241	301	361	421	421	481	541	541	601	661
ROBINSON 2	PWR	0	0	0	40	88	149	149	197	245	245	306	354	402	402	463	511	511	559	620
BRUNSWICK 1	BWR	0	0	0	157	157	345	345	533	717	717	905	1093	1093	1277	1277	1465	1653	1653	1837
LASALLE CTY 1&2	BWR	0	0	0	144	364	584	1024	1244	1464	1904	2124	2344	2784	3004	3224	3664	3884	4104	4544
BRUNSWICK 2	BWR	0	0	0	0	37	37	225	413	413	597	785	785	973	973	1157	1345	1345	1533	1717
CALVERT CLF 1&2	PWR	0	0	0	0	0	5	101	197	293	389	485	581	677	773	869	965	1061	1157	1253
LACROSSE	BWR	0	0	0	0	0	13	37	61	85	85	109	133	157	181	205	205	205	205	205
PILGRIM 1	BWR	0	0	0	0	0	0	160	160	356	356	548	548	740	740	740	936	936	1128	1128
PRAIRIE ISL 1&2	PWR	0	0	0	0	0	0	22	102	182	222	302	382	462	542	582	662	742	822	902
BYRON 1&2	PWR	0	0	0	0	0	0	0	99	99	267	435	435	603	771	771	939	1107	1107	1275
INDIAN PT 2	PWR	0	0	0	0	0	0	0	17	17	85	153	153	221	221	289	357	357	425	425
OYSTER CRK 1	BWR	0	0	0	0	0	0	0	68	68	200	348	348	488	628	628	768	912	912	912
FORT CALHOUN	PWR	0	0	0	0	0	0	0	8	8	53	98	98	143	188	188	233	278	278	323
ZION 1&2	PWR	0	0	0	0	0	0	0	0	71	143	287	359	431	575	647	719	863	935	1007
BIG ROCK 1	BWR	0	0	0	0	0	0	0	0	15	35	35	55	75	95	95	95	95	95	95
LIMERICK 1	BWR	0	0	0	0	0	0	0	0	152	152	372	592	592	812	1032	1032	1240	1456	1456
SAN ONOFRE 1,2,&3	PWR	0	0	0	0	0	0	0	0	139	300	409	570	679	945	1054	1163	1272	1381	1490
SEQUOYAH 1&2	PWR	0	0	0	0	0	0	0	0	40	40	200	280	360	520	600	680	760	920	1000
DAVIS-BESSE 1	PWR	0	0	0	0	0	0	0	0	9	9	70	130	130	190	250	250	310	370	370
POINT BEACH 1&2	PWR	0	0	0	0	0	0	0	0	49	113	177	241	305	369	433	497	561	625	689
ARK NUCLEAR 1	PWR	0	0	0	0	0	0	0	0	0	17	17	77	77	137	197	197	257	257	317
BRAIDWOOD 1&2	PWR	0	0	0	0	0	0	0	0	0	99	99	267	435	435	603	771	771	939	1107
BEAVER VALLEY 1	PWR	0	0	0	0	0	0	0	0	0	25	94	94	163	163	232	232	301	370	370
MAINE YANKEE	PWR	0	0	0	0	0	0	0	0	0	48	121	121	194	267	267	340	413	413	486
NINE MILE PT1	BWR	0	0	0	0	0	0	0	0	0	152	152	332	332	516	516	700	700	884	884
HADDAM NECK	PWR	0	0	0	0	0	0	0	0	0	0	3	3	56	108	161	161	213	266	266
ENRICO FERMI2	BWR	0	0	0	0	0	0	0	0	0	0	135	135	427	427	715	715	1007	1295	1295
COOPER STN	BWR	0	0	0	0	0	0	0	0	0	0	6	122	234	346	458	570	682	794	902
MILLSTONE 2	PWR	0	0	0	0	0	0	0	0	0	0	51	116	116	177	242	242	307	368	368
PEACHBOTTOM 2	BWR	0	0	0	0	0	0	0	0	0	0	216	216	444	672	672	900	1128	1128	1356
PEACHBOTTOM 3	BWR	0	0	0	0	0	0	0	0	0	0	41	41	261	481	481	701	701	921	1141
FITZPATRICK	BWR	0	0	0	0	0	0	0	0	0	0	134	134	310	486	486	662	838	838	1014
SALEM 1	PWR	0	0	0	0	0	0	0	0	0	0	47	47	127	207	207	287	367	367	447
DRESDEN 2	BWR	0	0	0	0	0	0	0	0	0	0	0	67	67	225	383	383	541	699	699
COOK 1&2	PWR	0	0	0	0	0	0	0	0	0	0	0	61	149	317	397	485	565	653	821
GRAND GULF 1	BWR	0	0	0	0	0	0	0	0	0	0	0	104	372	372	640	908	908	1176	1444
LIMERICK 2	BWR	0	0	0	0	0	0	0	0	0	0	0	132	132	348	564	564	780	780	996
WASH NUCLEAR2	BWR	0	0	0	0	0	0	0	0	0	0	0	110	286	426	582	750	898	1066	1210
ARK NUCLEAR 2	PWR	0	0	0	0	0	0	0	0	0	0	0	0	21	89	89	157	225	225	293

Table A.2. Projected Cumulative Storage Requirements--Maximum AR Capacity, Assemblies (con't)

		ASSEMBLIES																		
POOL		1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005
DUANE ARNOLD	BWR	0	0	0	0	0	0	0	0	0	0	0	0	118	246	246	366	486	486	614
VT YANKEE 1	BWR	0	0	0	0	0	0	0	0	0	0	0	0	12	12	144	276	276	408	540
NORTH ANNA 1&2	PWR	0	0	0	0	0	0	0	0	0	0	0	0	119	183	247	375	439	503	631
KEWAUNEE	PWR	0	0	0	0	0	0	0	0	0	0	0	0	24	61	98	135	172	209	246
YANKEE-ROWE 1	PWR	0	0	0	0	0	0	0	0	0	0	0	0	36	36	36	36	36	36	36
DRESDEN 3	BWR	0	0	0	0	0	0	0	0	0	0	0	0	0	61	219	219	377	535	535
HATCH 1&2	BWR	0	0	0	0	0	0	0	0	0	0	0	0	0	130	326	718	914	1110	1502
SUSQUEHANNA 1&2	BWR	0	0	0	0	0	0	0	0	0	0	0	0	0	96	560	792	1024	1488	1720
GINNA	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	27	59	91	123	155	187
ST LUCIE 2	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	46	118	118	190	262
SALEM 2	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	71	155	155	239	323
BROWNS FERRY3	BWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	161	161	389	617	617
WATTS BAR 1&2	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	51	211	211	371	451
TURKEY PT 3&4	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	48	144	192	240
BROWNS FERRY1&2	BWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	34	34	490	718
TROJAN	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	37	85
INDIAN PT 3	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	4	80
RANCHO SECO 1	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	43	104
PALO VERDE 1	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	12
QUAD CITIES 1&2	BWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	23
MCGUIRE 1	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	29
CRYSTAL RVR 3	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	72
CALLAWAY 1	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	37
PWR ASSEMBLIES		42	141	204	504	748	1002	1316	1928	2708	3747	5370	6505	8390	10299	11933	14125	15943	17848	20312
BWR ASSEMBLIES		128	128	324	625	1078	1499	2507	3195	4182	5110	7018	8399	11201	13858	17011	20429	23649	27497	30996
TOTAL ASSEMBLIES		170	269	528	1129	1826	2501	3823	5123	6890	8857	12388	14904	19591	24157	28944	34554	39592	45345	51308

Table A.3. Projected Cumulative Storage Requirements--Maximum AR Capacity, MTIHM

POOL		1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005
										METRIC TONS										
ST LUCIE 1	PWR	15	46	46	73	102	102	131	160	160	189	218	218	247	276	276	305	335	335	364
MILLSTONE 1	BWR	23	23	58	58	92	92	127	127	162	162	197	197	231	231	266	266	301	301	336
PALISADES	PWR	0	7	7	34	34	61	61	88	115	115	142	142	170	170	197	224	224	251	251
OCONEE 1&2	PWR	0	0	29	84	112	140	195	223	279	307	362	390	445	473	501	557	612	612	668
OCONEE 3	PWR	0	0	0	0	28	56	56	84	112	112	139	167	195	195	223	250	250	278	306
ROBINSON 2	PWR	0	0	0	17	38	63	63	84	105	105	130	151	172	172	197	218	218	238	263
BRUNSWICK 1	BWR	0	0	0	29	29	64	64	99	134	134	169	204	204	238	238	274	309	309	343
LASALLE CTY 1&2	BWR	0	0	0	26	66	106	186	226	266	347	387	427	507	547	587	667	707	747	827
BRUNSWICK 2	BWR	0	0	0	0	7	7	42	77	77	112	147	147	182	182	216	251	251	287	321
CALVERT CLF 1&2	PWR	0	0	0	0	0	2	38	74	110	145	181	213	249	285	321	357	393	429	465
LACROSSE	BWR	0	0	0	0	0	1	4	7	9	9	12	14	17	20	22	22	22	22	22
PILGRIM 1	BWR	0	0	0	0	0	0	28	28	63	63	97	97	131	131	131	165	165	199	199
PRAIRIE ISL 1&2	PWR	0	0	0	0	0	0	8	36	65	79	107	135	164	192	206	235	263	291	320
BYRON 1&2	PWR	0	0	0	0	0	0	0	42	42	113	184	184	255	326	326	397	468	468	539
INDIAN PT 2	PWR	0	0	0	0	0	0	0	8	8	38	69	69	100	100	130	161	161	192	192
OYSTER CRK 1	BWR	0	0	0	0	0	0	0	12	12	36	62	62	87	112	112	137	163	163	163
FORT CALHOUN	PWR	0	0	0	0	0	0	0	3	3	19	35	35	51	67	67	83	99	99	115
ZION 1&2	PWR	0	0	0	0	0	0	0	0	32	65	131	164	197	263	296	329	394	427	460
BIG ROCK 1	BWR	0	0	0	0	0	0	0	0	2	5	5	7	10	12	12	12	12	12	12
LIMERICK 1	BWR	0	0	0	0	0	0	0	0	27	27	66	105	105	144	183	183	220	258	258
SAN ONOFRE 1,2,&3	PWR	0	0	0	0	0	0	0	0	56	119	163	226	270	372	416	460	504	548	592
SEQUOYAH 1&2	PWR	0	0	0	0	0	0	0	0	18	18	92	129	166	239	276	313	350	423	460
DAVIS-BESSE 1	PWR	0	0	0	0	0	0	0	0	4	4	33	61	61	89	117	117	145	173	173
POINT BEACH 1&2	PWR	0	0	0	0	0	0	0	0	18	41	64	87	110	133	156	179	203	226	249
ARK NUCLEAR 1	PWR	0	0	0	0	0	0	0	0	0	8	8	36	36	64	91	91	119	119	147
BRAIDWOOD 1&2	PWR	0	0	0	0	0	0	0	0	0	42	42	113	184	184	255	326	326	397	468
BEAVER VALLEY 1	PWR	0	0	0	0	0	0	0	0	0	12	44	44	76	76	108	108	140	172	172
MAINE YANKEE	PWR	0	0	0	0	0	0	0	0	0	18	46	46	74	102	102	130	158	158	186
NINE MILE PT1	BWR	0	0	0	0	0	0	0	0	0	26	26	57	57	89	89	120	120	152	152
HADDAM NECK	PWR	0	0	0	0	0	0	0	0	0	0	1	1	20	39	59	59	78	97	97
ENRICO FERMI2	BWR	0	0	0	0	0	0	0	0	0	0	25	25	78	78	130	130	183	236	236
COOPER STN	BWR	0	0	0	0	0	0	0	0	0	0	1	22	43	63	83	104	124	145	164
MILLSTONE 2	PWR	0	0	0	0	0	0	0	0	0	0	19	44	44	68	92	92	117	140	140
PEACHBOTTOM 2	BWR	0	0	0	0	0	0	0	0	0	0	38	38	79	119	119	160	200	200	240
PEACHBOTTOM 3	BWR	0	0	0	0	0	0	0	0	0	0	7	7	46	85	85	124	124	163	202
FITZPATRICK	BWR	0	0	0	0	0	0	0	0	0	0	24	24	55	86	86	118	149	149	180
SALEM 1	PWR	0	0	0	0	0	0	0	0	0	0	22	22	58	95	95	132	168	168	205
DRESDEN 2	BWR	0	0	0	0	0	0	0	0	0	0	0	11	11	38	64	64	91	117	117
COOK 1&2	PWR	0	0	0	0	0	0	0	0	0	0	0	28	64	136	173	208	245	281	353
GRAND GULF 1	BWR	0	0	0	0	0	0	0	0	0	0	0	18	66	66	113	160	160	207	255
LIMERICK 2	BWR	0	0	0	0	0	0	0	0	0	0	0	23	23	62	100	100	138	138	177
WASH NUCLEAR2	BWR	0	0	0	0	0	0	0	0	0	0	0	19	50	75	102	132	158	188	213
ARK NUCLEAR 2	PWR	0	0	0	0	0	0	0	0	0	0	0	0	9	37	37	66	94	94	122
DUANE ARNOLD	BWR	0	0	0	0	0	0	0	0	0	0	0	0	21	44	44	65	87	87	109

Table A.3. Projected Cumulative Storage Requirements--Maximum AR Capacity, MTIHM (con't)

METRIC TONS

POOL		1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005
VT YANKEE 1	BWR	0	0	0	0	0	0	0	0	0	0	0	0	2	2	26	49	49	73	96
NORTH ANNA 1&2	PWR	0	0	0	0	0	0	0	0	0	0	0	0	55	85	114	173	203	232	292
KEWAUNEE	PWR	0	0	0	0	0	0	0	0	0	0	0	0	9	23	37	51	65	79	94
YANKEE-ROWE 1	PWR	0	0	0	0	0	0	0	0	0	0	0	0	8	8	8	8	8	8	8
DRESDEN 3	BWR	0	0	0	0	0	0	0	0	0	0	0	0	0	10	37	37	63	90	90
HATCH 1&2	BWR	0	0	0	0	0	0	0	0	0	0	0	0	0	24	60	133	170	206	279
SUSQUEHANNA 1&2	BWR	0	0	0	0	0	0	0	0	0	0	0	0	0	17	97	137	177	257	297
GINNA	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	9	21	32	43	54	66
ST LUCIE 2	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	18	46	46	74	102
SALEM 2	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	33	71	71	111	150
BROWNS FERRY3	BWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	29	29	71	112	112
WATTS BAR 1&2	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	24	97	97	171	208
TURKEY PT 3&4	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	22	66	88	110
BROWNS FERRY1&2	BWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	6	6	89	130
TROJAN	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	17	39
INDIAN PT 3	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	2	36
RANCHO SECO 1	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	20	48
PALO VERDE 1	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	5
QUAD CITIES 1&2	BWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	4
MCGUIRE 1	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	12
CRYSTAL RVR 3	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	33
CALLAWAY 1	PWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	16
PWR MTIHM		15	53	83	209	315	424	553	802	1126	1550	2234	2706	3489	4279	4973	5899	6665	7476	8528
BWR MTHIM		23	23	58	113	194	271	452	577	752	919	1261	1505	2005	2474	3033	3646	4221	4905	5535
TOTAL MTIHM		38	76	140	322	509	695	1004	1379	1879	2469	3495	4211	5494	6753	8006	9545	10886	12382	14063

Table A.4. 1986 Inventory and Projected Annual Reactor Discharges, Assemblies

REACTOR		INV. (a) 1986	ASSEMBLIES 1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005
ARK NUCLEAR 1	PWR	448	0	60	0	60	60	0	60	0	60	60	0	60	0	60	60	0	60	0	60
ARK NUCLEAR 2	PWR	288	0	68	68	0	68	68	0	68	68	0	68	0	68	68	0	68	68	0	68
BEAVER VALLEY 1	PWR	283	73	0	69	0	69	69	0	69	0	69	69	0	69	0	69	0	69	69	0
BEAVER VALLEY 2	PWR	0	0	0	37	0	73	0	73	73	73	0	73	0	73	73	0	73	73	0	73
BELLEFONTE 1	PWR	0	0	0	0	0	0	0	0	64	72	0	84	84	0	84	0	84	84	0	84
BELLEFONTE 2	PWR	0	0	0	0	0	0	0	0	0	0	64	72	0	84	84	0	84	0	84	84
BIG ROCK 1	BWR	188	22	22	20	20	20	20	20	20	20	20	0	20	20	20	84	0	0	0	0
BRAIDWOOD 1	PWR	0	0	0	88	88	0	88	88	0	84	84	0	84	84	0	84	84	0	84	84
BRAIDWOOD 2	PWR	0	0	0	0	88	0	88	88	0	88	84	0	84	84	0	84	84	0	84	84
BROWNS FERRY1	BWR	1328	0	0	0	0	0	228	0	228	0	228	228	0	228	228	0	228	0	228	228
BROWNS FERRY2	BWR	1192	0	0	284	0	220	0	224	228	0	228	0	228	228	0	228	228	0	228	0
BROWNS FERRY3	BWR	1004	0	0	0	268	228	0	228	0	228	228	0	228	228	0	228	0	228	228	0
BRUNSWICK 1	BWR	840	188	0	188	184	0	188	0	188	184	0	188	188	0	184	0	188	188	0	184
BRUNSWICK 1	PWR	160	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
BRUNSWICK 2	BWR	756	0	188	188	0	184	0	188	188	0	184	188	0	188	0	184	188	0	188	184
BRUNSWICK 2	PWR	144	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
BYRON 1	PWR	0	88	88	0	88	88	0	84	84	0	84	84	0	84	84	0	84	84	0	84
BYRON 2	PWR	0	0	88	0	88	88	0	88	84	0	84	84	0	84	84	0	84	84	0	84
CALLAWAY 1	PWR	84	96	0	80	84	0	84	84	0	84	84	0	84	84	0	84	84	0	84	84
CALVERT CLF 1	PWR	618	0	96	0	96	0	96	0	96	0	96	0	96	0	96	0	96	0	96	0
CALVERT CLF 2	PWR	432	88	0	96	0	96	0	96	0	96	0	96	0	96	0	96	0	96	0	96
CATAWBA 1	PWR	64	68	0	69	68	72	72	0	72	73	73	0	72	72	72	72	0	73	72	72
CATAWBA 2	PWR	0	0	65	68	64	64	69	0	60	80	72	0	72	73	72	0	73	72	72	72
CLINTON 1	BWR	0	0	140	192	0	160	176	0	172	168	0	172	172	0	172	172	0	172	172	0
COMANCHE PK 1	PWR	0	0	0	0	64	64	64	0	68	64	64	68	64	64	68	64	64	68	64	64
COMANCHE PK 2	PWR	0	0	0	0	0	68	64	64	68	64	64	68	64	64	68	64	64	68	64	64
COOK 1	PWR	546	80	0	80	80	0	80	80	0	80	0	80	80	0	80	80	0	80	0	80
COOK 2	PWR	424	0	88	0	0	88	88	0	88	88	0	88	0	88	88	0	88	0	88	88
COOPER STN	BWR	648	0	136	116	116	116	116	120	112	116	116	112	116	112	112	112	112	112	112	108
CRYSTAL RVR 3	PWR	302	93	0	81	0	72	0	72	0	72	0	72	0	72	0	72	0	72	0	72
DAVIS-BESSE 1	PWR	197	0	65	61	0	61	61	0	61	61	0	61	60	0	60	60	0	60	60	0
DIABLO CANYON 1	PWR	51	0	68	0	84	0	85	0	85	0	85	0	85	0	85	0	85	0	85	0
DIABLO CANYON 2	PWR	0	51	68	0	84	0	85	0	85	0	85	0	85	0	85	0	85	0	85	0
DRESDEN 1	BWR	683	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
DRESDEN 2	BWR	1606	0	168	158	0	158	158	0	158	158	0	158	158	0	158	158	0	158	158	0
DRESDEN 3	BWR	1456	152	0	160	0	158	158	0	158	158	0	158	158	0	158	158	0	158	158	0
DUANE ARNOLD	BWR	696	128	120	0	120	128	0	120	120	0	128	120	0	120	128	0	120	120	0	128
ENRICO FERMI2	BWR	0	0	0	232	292	0	276	0	296	0	292	288	0	292	0	288	0	292	288	0
FARLEY 1	PWR	410	0	68	65	0	68	65	0	68	65	0	68	65	0	68	65	0	68	65	0
FARLEY 2	PWR	256	64	0	68	65	0	68	65	0	68	65	0	68	65	0	68	65	0	68	65

Table A.4. 1986 Inventory and Projected Annual Reactor Discharges, Assemblies (con't)

		INV. (a)	ASSEMBLIES																		
REACTOR		1986	1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005
FITZPATRICK	BWR	1012	188	164	0	176	180	0	176	176	0	180	176	0	176	176	0	176	176	0	176
FORT CALHOUN	PWR	334	45	45	0	45	45	0	45	45	0	45	45	0	45	45	0	45	45	0	45
GINNA	PWR	470	36	32	32	32	32	32	32	32	32	32	32	32	32	32	32	32	32	32	32
GRAND GULF 1	BWR	264	288	0	268	268	0	268	268	0	268	268	0	268	268	0	268	268	0	268	268
HADDAM NECK	PWR	594	57	0	48	53	52	0	53	52	0	53	52	0	53	52	53	0	52	53	0
HARRIS 1	BWR	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
HARRIS 1	PWR	0	0	53	0	52	52	0	53	52	0	52	53	0	52	52	0	53	52	0	52
HATCH 1	BWR	1107	240	196	0	196	196	0	196	196	0	196	196	0	196	196	0	196	196	0	196
HATCH 2	BWR	745	0	184	184	196	0	196	196	0	196	196	0	196	196	0	196	196	0	196	196
HOPE CREEK	BWR	0	0	232	232	0	232	232	0	232	232	0	232	232	0	232	232	0	232	232	0
HUMBOLDT BAY	BWR	390	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
INDIAN PT 1	PWR	160	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
INDIAN PT 2	PWR	464	68	0	68	0	68	68	0	68	0	68	68	0	68	0	68	68	0	68	0
INDIAN PT 3	PWR	292	76	0	76	76	0	76	0	76	76	0	76	0	76	76	0	76	0	76	76
KEWAUNEE	PWR	369	45	37	33	41	37	45	37	37	37	37	37	37	37	37	37	37	37	37	37
LACROSSE	BWR	261	24	24	0	24	24	24	24	24	24	0	24	24	24	24	24	72	0	0	0
LASALLE CTY 1	BWR	132	324	0	188	220	0	220	220	0	220	220	0	220	220	0	220	220	0	220	220
LASALLE CTY 2	BWR	0	224	232	0	220	220	0	220	220	0	220	220	0	220	220	0	220	220	0	220
LIMERICK 1	BWR	0	268	272	0	224	0	228	216	0	220	0	220	220	0	220	220	0	208	216	0
LIMERICK 2	BWR	0	0	0	0	328	220	0	208	0	224	212	0	216	0	216	216	0	216	0	216
MAINE YANKEE	PWR	793	73	76	0	73	73	0	73	73	0	73	73	0	73	73	0	73	73	0	73
MCGUIRE 1	PWR	219	69	72	0	72	72	72	0	73	72	72	72	0	73	72	72	0	72	72	73
MCGUIRE 2	PWR	186	73	69	64	69	0	60	80	72	0	73	72	72	72	0	72	73	72	0	72
MILLSTONE 1	BWR	1536	196	0	196	0	196	0	196	0	196	0	196	0	196	0	196	0	196	0	196
MILLSTONE 2	PWR	474	0	77	85	52	0	68	0	65	61	0	64	65	0	61	65	0	65	61	0
MILLSTONE 3	PWR	0	84	0	84	84	0	84	84	0	84	84	0	84	84	0	84	84	0	84	84
MONTICELLO	BWR	428	116	0	128	116	0	124	116	0	120	0	120	120	0	120	120	0	120	120	0
MORRIS	BWR	2047	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
MORRIS	PWR	350	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
NINE MILE PT1	BWR	1444	0	200	0	192	0	196	0	176	0	188	0	180	0	184	0	184	0	184	532
NINE MILE PT2	BWR	0	0	0	296	0	276	0	264	0	284	0	268	0	280	0	276	0	276	0	276
NORTH ANNA 1	PWR	294	61	65	0	65	65	0	65	65	0	65	65	0	65	64	0	64	64	0	64
NORTH ANNA 2	PWR	235	69	0	65	65	0	65	65	0	65	65	0	65	65	0	64	64	0	64	64
OCONEE 1	PWR	590	57	0	60	60	60	0	60	60	60	0	60	60	60	0	60	60	60	0	60
OCONEE 2	PWR	381	49	0	60	60	0	60	60	0	60	60	60	0	60	60	0	60	60	0	60
OCONEE 3	PWR	529	0	60	60	0	60	60	0	60	60	0	60	60	60	0	60	60	0	60	60
OYSTER CRK 1	BWR	1392	0	128	136	0	128	172	0	152	0	132	148	0	140	140	0	140	144	0	556
PALISADES	PWR	545	0	68	0	68	0	68	0	68	68	0	68	0	68	0	68	68	0	68	0
PALO VERDE 1	PWR	0	80	77	0	93	77	0	93	77	0	93	77	0	93	77	0	93	77	0	93
PALO VERDE 2	PWR	0	0	92	85	0	85	77	0	93	77	0	93	77	0	93	77	0	93	77	0
PALO VERDE 3	PWR	0	0	0	92	85	0	85	77	0	93	77	0	93	77	0	93	77	0	93	77

Table A.4. 1986 Inventory and Projected Annual Reactor Discharges, Assemblies (con't)

REACTOR		INV. (a) 1986	ASSEMBLIES 1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005
PEACHBOTTOM 2	BWR	1462	272	0	308	0	284	0	264	220	0	228	228	0	228	228	0	228	228	0	228
PEACHBOTTOM 3	BWR	1496	268	0	0	216	236	0	220	216	0	224	220	0	220	220	0	220	0	220	220
PERRY 1	BWR	0	0	220	0	244	200	0	228	224	0	224	224	0	224	224	0	224	224	224	0
PILGRIM 1	BWR	1320	0	196	0	0	192	0	192	0	196	0	192	0	192	0	0	196	0	192	0
POINT BEACH 1	PWR	446	32	32	32	32	32	32	32	32	32	32	32	32	32	32	32	32	32	32	32
POINT BEACH 2	PWR	408	32	32	32	32	32	32	32	32	32	32	32	32	32	32	32	32	32	32	32
PRAIRIE ISL 1	PWR	386	45	40	40	41	40	0	40	40	40	40	40	40	40	40	0	40	40	40	40
PRAIRIE ISL 2	PWR	415	0	40	40	40	40	40	40	40	40	0	40	40	40	40	40	40	40	40	40
QUAD CITIES 1	BWR	1393	172	0	160	160	0	160	172	0	160	160	0	160	160	0	160	160	0	160	160
QUAD CITIES 2	BWR	1428	0	168	160	0	160	160	0	160	160	0	160	160	0	160	160	0	160	160	0
RANCHO SECO 1	PWR	267	0	57	65	0	69	69	0	61	61	0	61	0	57	61	0	61	0	57	61
ROBINSON 2	PWR	270	48	61	0	48	48	61	0	48	48	0	61	48	48	0	61	48	0	48	61
RVR BEND 1	BWR	0	164	0	224	180	0	192	204	0	188	196	0	192	192	0	196	192	0	192	192
SALEM 1	PWR	344	83	0	85	85	101	0	81	85	0	80	80	0	80	80	0	80	80	0	80
SALEM 2	PWR	174	0	73	89	0	97	101	93	0	85	84	0	84	84	0	84	84	0	84	84
SAN ONOFRE 1	PWR	146	0	52	52	0	0	52	0	52	0	52	0	52	0	157	0	0	0	0	0
SAN ONOFRE 2	PWR	147	109	0	109	0	109	0	109	0	109	109	0	109	0	109	0	109	0	109	0
SAN ONOFRE 3	PWR	147	0	109	0	109	0	109	0	109	109	0	109	0	109	0	109	0	109	0	109
SEABROOK 1	PWR	0	0	64	0	64	64	64	64	64	0	64	64	64	64	64	0	64	64	64	64
SEQUOYAH 1	PWR	212	0	0	80	80	0	80	80	0	80	0	80	80	0	80	80	0	80	80	0
SEQUOYAH 2	PWR	136	80	0	80	0	80	80	0	80	80	0	80	0	80	80	0	80	0	80	80
SHOREHAM	BWR	0	0	0	204	0	176	160	0	172	184	0	184	184	0	184	184	0	184	184	0
SOUTH TEXAS 1	PWR	0	0	0	56	54	52	52	52	52	52	52	52	52	52	52	52	52	52	52	52
SOUTH TEXAS 2	PWR	0	0	0	0	56	54	52	52	52	52	52	52	52	52	52	52	52	52	52	52
ST LUCIE 1	PWR	444	109	80	0	72	76	0	76	76	0	76	76	0	76	76	0	76	76	0	76
ST LUCIE 2	PWR	164	93	0	72	72	0	72	72	0	72	72	0	72	72	0	72	72	0	72	72
SUMMER 1	PWR	112	68	68	0	68	68	0	68	68	0	68	68	0	68	68	0	68	68	0	68
SURRY 1	PWR	488	0	69	49	56	0	53	53	0	53	53	0	53	53	0	52	52	0	52	52
SURRY 2	PWR	385	0	65	49	0	57	53	0	53	53	0	53	53	0	53	52	0	52	52	0
SUSQUEHANNA 1	BWR	488	240	0	248	228	0	236	232	0	232	232	0	232	232	0	232	232	0	232	232
SUSQUEHANNA 2	BWR	324	0	236	228	0	232	232	0	232	232	0	232	232	0	232	232	0	232	232	0
THREE MILE ISL 1	PWR	284	0	73	0	73	73	69	0	73	0	73	73	0	73	73	0	72	72	0	72
TROJAN	PWR	379	57	47	48	49	48	48	48	48	48	48	48	48	48	48	48	48	48	48	48
TURKEY PT 3	PWR	424	77	60	0	48	0	52	48	48	0	48	48	0	48	48	0	48	48	0	48
TURKEY PT 4	PWR	446	0	52	52	0	48	48	48	0	48	48	0	48	48	0	48	0	48	48	0
VOGTLE 1	PWR	0	0	84	84	0	84	84	0	84	84	0	84	84	0	84	84	0	84	84	0
VOGTLE 2	PWR	0	0	0	84	0	84	84	0	84	84	0	84	84	0	84	84	0	84	84	0
VT YANKEE 1	BWR	1322	136	0	132	132	0	132	132	0	132	132	0	132	132	0	132	132	0	132	132
WASH NUCLEAR2	BWR	128	148	168	160	148	156	156	156	172	140	156	172	144	176	140	156	168	148	168	144
WATERFORD 3	PWR	92	0	88	88	0	88	88	0	88	88	0	88	88	0	88	88	0	88	88	0

Table A.4. 1986 Inventory and Projected Annual Reactor Discharges, Assemblies (con't)

REACTOR		INV. (a) 1986	ASSEMBLIES 1987	1988	1989	1990	1991	1992	1993	1994	1995	1996	1997	1998	1999	2000	2001	2002	2003	2004	2005
WATTS BAR 1	PWR	0	0	0	0	64	72	0	80	80	0	80	0	80	80	0	80	80	0	80	0
WATTS BAR 2	PWR	0	0	0	0	0	64	72	0	80	0	80	80	0	80	80	0	80	0	80	80
WEST VALLEY	BWR	85	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
WEST VALLEY	PWR	40	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
WOLF CREEK 1	PWR	52	52	76	0	76	76	0	76	76	0	76	76	0	76	76	0	76	76	0	76
YANKEE-ROWE 1	PWR	341	36	40	0	36	0	40	36	0	40	36	0	40	36	0	76	0	0	0	0
ZION 1	PWR	574	0	76	72	0	72	72	0	72	72	0	72	72	0	72	72	0	72	72	0
ZION 2	PWR	503	80	72	0	76	72	0	72	72	0	72	72	0	72	72	0	72	72	0	72
FT ST VRAIN	HTG	0	0	240	0	282	0	240	0	240	0	240	0	240	0	240	0	1482	0	0	0
RESEARCH SITES	PWR	97	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
RESEARCH SITES	BWR	4	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
RESEARCH SITES	HTG	720	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0	0
PWR ASSEMBLIES		20309	2644	3155	3200	3577	3677	3873	3171	4010	3647	3593	3967	3259	4121	3974	3225	3954	3502	3599	3935
BWR ASSEMBLIES		30605	3758	3394	4990	4468	4680	4408	5000	4440	4640	4788	5024	4380	5088	4276	5032	4488	4588	5292	5192
HTG ASSEMBLIES		720	0	240	0	282	0	240	0	240	0	240	0	240	0	240	0	1482	0	0	0
TOTAL ASSEMBLIES		51634	6402	6789	8190	8327	8357	8521	8171	8690	8287	8621	8991	7879	9209	8490	8257	9924	8090	8891	9127

(a) PERMANENTLY DISCHARGED SPENT FUEL. THIS INCLUDES SOME SPENT FUEL, APPROXIMATELY 140 MTIHM, PHYSICALLY RESIDENT IN THE REACTOR CORE ON DECEMBER 31, 1986 WHICH IS NOT PLANNED TO UNDERGO ANY FUTURE IRRADIATION.

NRC FORM 335 (2-84) NRCM 1102, 3201, 3202 U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** SEE INSTRUCTIONS ON THE REVERSE.	1. REPORT NUMBER (Assigned by TIDC, add Vol. No., if any) NUREG-1353
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5. AUTHOR(S) Edward D. Throm	6. DATE REPORT ISSUED MONTH April YEAR 1989
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12. SUPPLEMENTARY NOTES

13. ABSTRACT (200 words or less)

Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools," addresses the concerns with the use of high density storage racks for the storage of spent fuel, and is applicable to all Light Water Reactor spent fuel pools.

This report presents the regulatory analysis for Generic Issue 82. It includes (1) a summary of the issue, (2) a summary of the technical findings, (3) the proposed technical resolution, (4) alternative resolutions considered by the Nuclear Regulatory Commission, (5) an assessment of the benefits and cost of the alternatives considered, (6) the decision rationale, and (7) the relationships between Generic Issue 82 and other NRC programs and requirements.

Based on this evaluation, the NRC staff concludes that no new regulatory requirements are warranted concerning the use of high density storage racks.

14. DOCUMENT ANALYSIS – a. KEYWORDS/DESCRIPTORS	15. AVAILABILITY STATEMENT
Spent Fuel Pools, Generic Safety Issue, Value/Impact Analysis, Probabilistic Risk Assessment, Zircaloy Cladding Fire, PWRs, BWRs, Seismic Hazard, Fragility, Reracking	Unlimited
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