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A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants

Prepared by
R. J. Travis, R. E. Davis, E. J. Grove, M. A. Azarm

Brookhaven National Laboratory

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A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants

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ABSTRACT

An evaluation of the nuclear power plant regulatory basis is performed, as it pertains to those plants that are permanently shutdown (PSD) and awaiting or undergoing decommissioning. Four spent fuel storage configurations are examined. Recommendations are provided for those operationally based regulations that could be partially or totaliy removed for PSD plants without impacting public health and safety.

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EXECUTIVE SUMMARY

The long-term availability of less expensive power and the increasing plant modification and maintenance costs have caused some utilities to re-examine the economics of nuclear power. As a result, several utilities have opted to permanently shutdown their plants. Each licensee of these permanently shutdown (PSD) plants has submitted plant-specific exemption requests for those regulations that they believe are no longer applicable to their facility. The preparation and subsequent review of these exemption requests represents a large level of effort for both the licensees and the NRC staff. This experience has indicated the need for an explicit regulatory treatment of PSD nuclear power plants.

This report presents a regulatory assessment for generic BWR and PWR plants that have permanently ceased operation in support of NRC rulemaking activities in this area.

After the reactor vessel is defueled, the traditional accident sequences that dominate the operating plant risk are no longer applicable. The remaining source of public risk is associated with the accidents that involve the spent fuel. Previous studies have indicated that complete spent fuel pool drainage is an accident of potential concern. Certain combinations of spent fuel storage configurations and decay times, could cause freshly discharged fuel assemblies to self heat to a temperature where the self sustained oxidation of the zircaloy fuel cladding may cause cladding failure.

Spent Fuel Configurations

This study has defined four spent fuel configurations which encompass all of the anticipated spent fuel characteristics and storage modes following permanent shutdown. Spent fuel which (due to a combination of storage geometry, decay time, and reactor type) can support rapid zircaloy oxidation is designated as Spent Fuel Storage Configuration 1 - "Hot Fuel in the Spent Fuel Pool." Configuration 1 encompasses the period commencing immediately after the offload of the core to a point in time when the decay heat of the hottest assemblies is low enough such that no substantial zircaloy oxidation takes place (given the pool is drained), and the fuel cladding will remain intact (i.e., no gap releases).

After this point, the fuel is considered to be in Configuration 2 - "Cold Fuel in the Spent Fuel Pool." The fuel can be stored on a long-term basis in the spent fuel pool, while the rest of the plant is in safe storage or decontaminated (partial decommissioning). Alternatively, after decay heat loads have declined further, the fuel can be moved to an ISFSI (designated as spent fuel storage Configuration 3). This would allow complete decommissioning of the plant and closure of the Part 50 license. Spent fuel storage Configuration 4 assumes all spent fuel has been shipped offsite. This configuration assumes the plant Part 50 license remains in effect only because the plant has not been fully decontaminated and cannot be released for unrestricted public access.

A representative accident sequence was chosen for each configuration. Consequence analyses were performed using these sequences to estimate onsite and boundary doses, population doses and economic costs.

Regulatory Assessment

After a plant is permanently shutdown, awaiting or in the decommissioning process, certain operating based regulations may no longer be applicable. A list of candidate regulations was identified from a screening of 10 CFR Parts 0 to 199. The continued applicability of each regulation was assessed within the context of each spent fuel storage configuration and the results of the consequence analyses. The regulations that are no longer fully applicable to the permanently shutdown plant are summarized below:

The set of regulations that are designed to protect the public against full power and/or design basis accidents are no longer applicable and can be deleted for all spent fuel storage configurations of the permanently shutdown plant. These regulations include combustible gas control (50.44), fracture prevention measures (50.60, 50.61), and ATWS requirements (50.62).

Other regulations, although based on the operating plant, may continue to be partially applicable to the permanently defueled facility. This group of requirements includes the Technical Specifications (50.36, 36b), the fire protection program (50.48) and Quality Assurance (50.54(a) and Part 50 Appendix B).

The requirements for emergency preparedness (50.47, 50.54(q) and (t), and Part 50 Appendix E), onsite property damage insurance (50.54(w)) and offsite liability insurance (Part 140), were evaluated using the accident consequence analysis. Since the estimated consequences of the Configuration 1 representative accident sequence approximate those of a core damage accident, it is recommended that all offsite and onsite emergency planning requirements remain in place during this period, with the exception of the Emergency Response Data System requirements of Part 50, Appendix E. Subject to plant specific confirmation, the offsite emergency preparedness (EP) requirements are expected to be eliminated for Configuration 2, on the basis of a generic boundary dose calculation. Part 50 offsite EP requirements can also be eliminated for Configurations 3 and 4 because the spent fuel has been transferred to an ISFSI (subject to Part 72 requirements) or transported offsite. Without spent fuel, the plant is not a significant health risk. It is recommended that the onsite property damage and the offsite liability insurance levels remain at operating reactor levels for the duration of Configuration 1. The consequence analyses support reduced insurance requirements for the remaining configurations (2,3, and 4).

FOREWORD

The information in this report is being considered by the U.S. Nuclear Regulatory Commission (NRC) staff in the development of amendments to its regulations for permanently shutdown nuclear power reactors in the process of decommissioning. The NRC has undertaken a number of initiative to reduce the regulatory burden for licensees that are in the process of permanently removing nuclear facilities from service. This report provides baseline data to the NRC for evaluating which regulations may be considered for amending to enhance the regulatory effectiveness during decommissioning.

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ACRONYMS

AC	alternating current
AEC	Atomic Energy Commission
ANI	American Nuclear Insurers
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ATWS	anticipated transient without scram
BNL	Brookhaven National Laboratory
BWR	boiling water reactor
BWST	borated water storage tank
C	Celsius
DECON	decontamination
DOE	Department of Energy
ECCS	emergency core cooling systems
EDE	effective dose equivalent
EDG	emergency diesel generator
EOF	emergency operations facility
EP	emergency planning and preparedness
EPA	Environmental Protection Agency
EPRI	Electric Power Research Institute
EPZ	emergency planning zone
EQ	environmental qualification
FSAR	final safety analysis report
GWD	gigawatt day
H	high release factors
HLW	High Level Waste
HVAC	heating, ventilation and air conditioning
IAEA	International Atomic Energy Agency
ISFSI	independent spent fuel storage installation
ISI	inservice inspection
IST	inservice testing
KW	kilowatt
L	low release factors
LOCA	loss of coolant accident
LWR	light water reactor
MACCS	Melcor Accident Consequence Code System
MAERP	Mutual Atomic Energy Reinsurance Pool
MRS	monitored retrievable storage
MTU	metric ton of uranium
MW, MWe	electrical megawatt
NEIL	Nuclear Electric Insurance Limited

ACRONYMS (Cont'd)

PAGs	protective action guides
PNL	Pacific Northwest Laboratory
POL	possession only license
PRA	probabilistic risk assessment
PSD	permanent shutdown
PWR	pressurized water reactor
QA	quality assurance
RG	regulatory guide
RPS	reactor protection system
RSS	Reactor Safety Study
SAFSTOR	safe storage with deferred decontamination
SBO	station blackout
SSCs	structures, systems and components
SFP	spent fuel pool
SFUEL1W	a spent fuel heatup code
SHARP	spent-fuel heatup code
SMN	special nuclear material
SMUD	Sacramento Municipal Utility District
SNL	Sandia National Laboratory
SRO	senior reactor operator
TMI	Three Mile Island
TSC	Technical Support Center
US	United States

1 INTRODUCTION

The long-term availability of less expensive power, compounded by the increasing plant modification and maintenance costs, have caused some utilities to re-examine the economics of nuclear power. As a result, several plants with years, some with decades, left on their operating licenses have opted to permanently shutdown their facilities.

At present, six (6) nuclear power plants* are permanently shutdown in various stages of the decommissioning process. The absence of a clearly defined regulatory path for these licensees has become apparent. Each of the permanently shutdown (PSD) licensees has submitted plant-specific exemption requests for those regulations that they believe are no longer applicable to their facility. The lack of a regulatory roadmap for the permanently defueled plant has resulted in a large effort for both the licensees and the NRC staff attributable to the development and review of plant-specific exemption requests. This experience has established the need for an explicit regulatory treatment of PSD nuclear power plants, including:

- the clarification of the regulations for decommissioning nuclear power plants,
- the activities that are permissible for major phases of the decommissioning process,
- the specification of those Part 50 regulations that are applicable only to plants authorized to operate.[1]

Brookhaven National Laboratory (BNL) has undertaken a program (FIN L-2590) "Safety and Regulatory Issues Related to the Permanent Shutdown of Nuclear Power Plants Awaiting Decommissioning," to support the last NRC goal stated above, i.e., "to determine the extent and types of safety criteria that should remain as part of the decommissioning regulations to assure that the health and safety of public is protected when a licensee enters the permanent shutdown condition in preparation for plant decommissioning."

This NUREG/CR documents the results of this program.

The remainder of this report is structured as follows:

Section 2, "Background" presents a brief discussion of the changes that are likely to take place when a licensee permanently ceases operation of a nuclear power plant. As the primary source of public risk, the focus of this discussion is the storage alternatives for the spent fuel. This section, in conjunction with Appendix A, "Previous Examinations of Spent Fuel Pool Accidents," also summarizes the assumptions and conclusions of earlier studies in this area. This information can be helpful as it provides the necessary context for the assessment of the present study's assumptions and conclusions.

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*Fort St. Vrain, Rancho Seco, San Onofre Unit 1, Three Mile Island 2, Trojan, and Yankee Rowe, are undergoing decommissioning. Shoreham has completed the process and the license has been terminated.

Section 3, "Input Assumptions," provides detailed information and assumptions (such as accident initiator, timing, source terms, meteorology, population, etc.) that are necessary to support the accident consequence analyses. In support of potential rulemaking, the calculation assumptions of Section 3 have been developed to envelope the end of life plant shutdowns that are anticipated in the future. Thus, this study considers spent fuel pools that are full to capacity with high burnup fuel, and an offsite population density that is consistent with end of plant life. As such, these assumptions tend to be conservative with regard to those plants that are currently shutdown awaiting, or in various stages of, decommissioning.

Section 4, "Results of the Consequence Analyses," presents the estimated accident consequences for each spent fuel storage configuration, including societal dose, condemned land area, and accident cost. Multiple cases were evaluated using different inventory and source-term assumptions. BNL has chosen a "best estimate" case for each configuration.

Section 5, "Regulatory Assessment Summary," and Appendix B present the evaluation of the current operating plant based NRC regulations as applied to the permanently shutdown nuclear power plant. The applicability of each candidate regulation is assessed for each spent fuel storage configuration, based on the likely status of the physical plant and the consequence analysis of the preceding section.

Sections 6 and 7, respectively, summarize the report and provide the necessary references.

2 BACKGROUND

Once a decision is made to permanently cease operation of its nuclear power plant, the licensee will defuel the reactor vessel. In parallel (or perhaps in anticipation of permanent shutdown) the licensee will apply for an NRC license amendment to withdraw the authority to operate the plant. It also provides a basis to remove the regulatory requirements that are no longer necessary to protect the health and safety of the public. Thus, the amendment to remove the authority to operate provides a basis for a licensee to begin eliminating personnel, equipment, and activities pursuant to 10CFR 50.59 analyses, license amendments and exemption requests. The regulatory ambiguity regarding the permanently shutdown nuclear power plant has prompted the NRC to develop further guidance in this area*. However, the basis for any regulatory relief must ultimately address the potential impact on public health and safety. Previous decommissioning studies[3-4] have shown that the offsite doses associated with decommissioning accidents that do not involve spent fuel are negligible. Therefore, this study has focused on the spent fuel storage alternatives after a plant has been permanently shutdown and the potential public risk associated with each alternative.

After the reactor vessel is defueled the traditional accident sequences that dominated the operating plant risk are no longer applicable. The remaining source of public risk is associated with the accidents that involve the spent fuel stored in the spent fuel pool (SFP). As discussed in Appendix A, accidents involving spent fuel, although limited to the 1/3 core offloads associated with refueling were considered as part of the spectrum of nuclear power plant risk as early as the Reactor Safety Study (WASH 1400). More recently, Sandia National Laboratories (SNL) studies[5-6] have indicated that complete spent fuel pool drainage, with certain combinations of spent fuel storage configurations and decay times, could cause freshly discharged fuel assemblies to self heat to a temperature where the oxidation of the zircaloy fuel cladding may become self sustaining. Follow-up efforts by BNL[7] applied simplified PRA analyses to quantify the frequency of initiating events that could compromise the SFP integrity; the conditional probability of subsequent system failure, fuel failure probability; the magnitude of radionuclide releases to the environment and the consequences of those releases.

A 1989 BNL report,[8] describes a value/impact assessment of various proposed options intended to reduce the risk posed by potential accidents occurring in commercial nuclear power plant spent fuel pools. As was the case with previous efforts, attention was limited to an operating plant. The risk dominant accidents, source terms and inventory considered in this later effort were identical to those investigated by Sailor, et al. in Reference 7. Major differences in the estimation of the off-site consequences exist between these two studies which are primarily attributable to the higher population density assumptions of the later report.

This study has defined four (4) spent fuel configurations which encompass all anticipated spent fuel characteristics and storage modes following permanent shutdown. Spent fuel which, due to a combination of storage geometry, decay time, and reactor type, can support rapid zircaloy oxidation is designated as Spent Fuel Storage Configuration 1 - "Hot Fuel in the Spent Fuel Pool." Configuration 1 encompasses the period commencing immediately after the offload of the core to a point in time when the decay heat

*Although a licensee is prohibited from making changes that materially affect costs, methods, or options for decommissioning the facility, the extent of permissible decommissioning activities has been clarified by issuance of final rule (61 FR 39278) amending regulations on decommissioning procedures.

of the hottest assemblies is low enough such that no zircaloy oxidation takes place, and the fuel cladding will remain intact (i.e., no gap releases).

At this point the fuel is considered to be in Configuration 2 - "Cold Fuel in the Spent Fuel Pool." The fuel can be stored on a long-term basis in the spent fuel pool, while the rest of the plant is in SAFSTOR* or decontaminated (partial decommissioning). Alternatively, after decay heat loads have declined further,** the fuel can be moved to an ISFSI (designated as spent fuel torage Configuration 3). This would allow complete decommissioning of the plant and closure of the Part 50 license.

Given the present unavailability of a permanent geological high level waste repository, or an interim Monitored Retrievable Storage (MRS) facility the fuel is expected to remain onsite for an indefinite time period.

At some point in the future, a MRS facility or a high level waste repository will become available. Spent fuel storage Configuration 4 assumes all spent fuel has been shipped offsite. This configuration assumes the plant Part 50 license remains in effect only because the plant has not been fully decontaminated and cannot be released for unrestricted public access.

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*Safe storage followed by deferred decontamination.

**Limits are placed on the burnup, decay time, enrichment and decay heat of the spent fuel assemblies to ensure the ISFSI design heat load is not exceeded. Although 10CFR Part 72 specifies a minimum of one year pool decay time, plant ISFSI technical specifications specify minimum decay times up to 10 years.

3 SPENT FUEL STORAGE CONFIGURATION INPUT ASSUMPTIONS

The purpose of this section is to define the input assumptions for each spent fuel storage configuration to support the consequence analyses of the next section. A set of assumptions was developed that is used in Section 4 to provide an estimate of the accident consequences that envelope future end of life nuclear power plant shutdowns, as well as plants that have prematurely ceased operation. However, an effort has been made to avoid unduly pessimistic assumptions or combinations of assumptions. The accident consequences thus obtained, are believed to be reasonably bounding for present and future closures and are not so overly conservative as to clearly represent some high (but unspecified) percentile result.

The input assumptions for each configuration will be discussed for PWRs and BWRs, respectively. Table 3.1 presents a summary of this section.

3.1 Configuration 1 - Hot Fuel in the Spent Fuel Pool

Spent fuel storage Configuration 1 commences immediately after the permanently shutdown facility has completed the reactor vessel defueling. This configuration models the potential consequences of rapid zircaloy oxidation resulting from an event which has caused the draining of the spent fuel pool. After a suitable time period, dependent on assembly burnup and racking geometry, the decay heat is low enough to preclude the rapid oxidation phenomenon. The end of this configuration is defined as that point in time when the fuel decay heat is low such that the cladding remains intact upon extended exposure to the air.

The consequence analysis input assumptions for Configuration 1 are provided below in the form of generic PWR and BWR plant configurations.

3.1.1 Representative Plant and Fuel Pool Data

The representative PWR* chosen for this study is a single 1130 MWe unit with 193 assemblies in the core. The corresponding 1155 MWe BWR has 764 assemblies. In accord with the industry trend to maximize storage capacity, both plants have high density fuel racking geometries.** The PWR spent fuel racks have a 10.40 inch cell to cell pitch and a five inch orifice at the bottom of each cell.[9] The BWR spent fuel racks a 6.255 inch pitch. Each BWR cell has a 4-inch orifice.[10] Variation in these parameters exist among various rack designs and manufacturers. These values were chosen to represent typical attributes.

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*The representative PWR and BWR geometries and spent fuel data were developed from a review of a limited set of plant information. They are generally the most conservative values from that set of information and are viewed as reasonably conservative, but not necessarily the most limiting configurations.

**Previous studies of the spent fuel rapid oxidation phenomenon have assumed a low density racking configuration for BWRs. (See Appendix A).

The spent fuel pool storage capacities were 1460 intact assemblies for the generic PWR and 3300 assemblies for the generic BWR. These are the average pool capacities of the current 193 assembly PWRs and 764 assembly BWRs. In order to envelope end of life shutdowns, this analysis assumed that the pools are full. The last full core offload was assumed to contain high burnup fuel (60,000 and 40,000 megawatt days per metric tons of heavy metal (MWD/MTU), PWR and BWR, respectively), to reflect the current trend to increase burnup. The earlier refueling discharges began at 20,000 MWD/MTU and increased linearly with each subsequent discharge to the ultimate assumed burnup. Consistent with Regulatory Guide 4.7, an exclusion boundary of 0.4 miles was assumed for each plant.

3.1.2 Accident Initiator and Timing

The accident initiator was a composite of events that can cause draining or boiloff of the spent fuel pool and expose the relatively hot spent fuel assemblies to an air environment. The initiator includes beyond design basis seismic events, spent fuel cask drop events, and other less dominant events such as spent fuel pool loss of cooling/makeup.

The composite initiator frequency of 2E-6 (PWR) and 7E-6 (BWR) events per year is adapted from the NUREG-1353 "best estimate" with modifications to reflect a higher spent fuel cask drop contributor associated with a higher assumed spent fuel transfer rate for the permanently shutdown plant. For the purposes of the offsite liability insurance discussion in Appendix B, the initiator frequency is equivalent to the release frequency.

The accident timing considered the minimum in-core decay requirements of the Standard Technical Specifications (about 4 days) and industry experience of several weeks to fully offload a core during refueling outages. For this study, the Configuration 1 accident initiator was assumed to occur 12 days following final shutdown.

3.1.3 Critical Decay Time

Previous studies[5-7] have defined the critical decay time as the duration, measured with respect to reactor shutdown, when the most recently discharged set of fuel assemblies have sufficient decay heat, that if the fuel pool were to completely drain, would heat to the point that clad oxidation would become self sustaining and eventually result in extensive clad failure with fission product release. This time is a function of the reactor type, spent fuel storage rack geometry and fuel burnup.

To be conservative, this effort chose to examine high density rack geometries for both PWR and BWR plants. In the time frame of the previous studies, high density racking was not widely used by in BWR plants. The previous efforts, therefore, do not provide results for this case.

The PWR high density racking geometry with a 5-inch orifice (albeit with low burnup fuel) was examined in NUREG/CR-4982. A 700 day critical decay time was estimated, using the SFUEL1W[5,6] code. based on a minimum decay power of 6 KW/MTU.

Table 3.1 Spent Fuel Storage Configuration Matrix

Parameters	Configuration 1 Hot Fuel in Spent Fuel Pool		Configuration 2 Cold fuel in spent fuel pool		Configuration 3 Spent fuel stored in ISFSI	Configuration 4 All spent fuel removed from site
	PWR	BWR	PWR	BWR		All Fuel Removed
Representative Plant Data	Single Unit 1130 MWe 193 Fuel Assemblies	Single Unit 1155 MWe 764 Fuel Assemblies	Same as Config. 1	Same as Config. 1	N/A	N/A
Spent Fuel Pool Storage Cask Data	Capacity 1460 intact fuel assemblies All slots filled Cask laydown area in pool	Capacity 3300 intact fuel assemblies All slots filled Cask laydown area in pool	Same as Config. 1	Same as Config. 1	Storage Casks (metal & concrete) 28 PWR Fuel assemblies 56 BWR Fuel assemblies NUHOMS 24 PWR Fuel assemblies 52 BWR Fuel assemblies	N/A
Spent Fuel Storage Rack Design Data	2 Region design Stainless steel max capacity 10.40 inch pitch 5 inch dia. bottom orifice hole (each cell)	Single region Stainless steel max capacity 6.25 inch pitch 4 inch dia. bottom orifice hole (each cell)	Same as Config. 1	Same as Config. 1	N/A	N/A
Fuel Assembly Burnup	Burnup range 20-60 GWD/MTU	Burnup range 20-40 GWD/MTU	Same as Config. 1	Same as Config. 1	5 year fuel decay time prior to storage in ISFSI High burnup fuel placed in ISFSI (60 GWD/MTU for PWR, 40 GWD/MTU for BWR)	N/A
Accident Initiator	Spent fuel pool fire resulting from a spent fuel pool draindown	Same as PWR	Fuel handling accident resulting in gap release		Tornado driven missile which results in breach of both primary & secondary seals, fuel cladding failure & fission product release	No credible decommissioning accidents can be postulated that have significant health effects
Accident Timing	12 days after final reactor shutdown		3.5 years after final shutdown	2 years after final shutdown	5 years after final shutdown	5 years after final shutdown
Exclusion Zone	0.4 miles		Same as Config. 1		100 meters	N/A
Offsite Meteorological Data	Mean weather attributes from continental U.S.		Same as Config. 1		A high wind speed is assumed to approximate a tornado's dispersion	N/A
Population Distribution	Population density of 1000 persons/sq. mi. out to 30 miles.		Same as Config. 1		Same as Config. 1	N/A
Accident Inventory and Source Term	Four accident inventories are examined ranging from fire involving full pool to gap release of last core offload. High (H) and low (L) release fractions based on NUREG/CR-4982 as modified by more recent studies of gap release and high burnup fuel are examined for each inventory.		Accident inventory is one fuel assembly. High and low gap release fractions examined		Two accident inventories examined: one assembly or entire cask. High and low gap releases for each	N/A

It should be stressed that there are uncertainties associated with this SFUEL1W calculation. The authors of the present study fully agree with the code limitations presented in NUREG/CR-4982 report. The SFUEL1W code provides a stylized analysis of the progression of events following the complete loss of spent fuel pool coolant and as such, does not have the ability to realistically model actual spent fuel pool configurations.

In response to the need to accurately predict the likelihood of reaching critical clad temperatures with realistic spent fuel pool configurations, BNL has developed the SHARP code (Spent-fuel Heatup: Analytical Response Program.)[45]

This code has been used, in conjunction with the Configuration 1 spent fuel data from Table 3.1 to develop maximum clad temperature as a function of decay time, given a loss of all spent fuel pool water. These relationships are presented as Figures 3.1 and 3.2 for the PWR and BWR representative geometries.

The end of Configuration 1 has been defined as the decay time that is necessary to ensure that the fuel rod cladding remains intact given a loss of all spent fuel pool water. The previous study[7] defined 650°C as a maximum temperature for cladding integrity. The Workshop on Transportation Accident Scenarios[47] estimated incipient clad failure at 565°C with expected failure at 671°C, presumably based on expert opinion. Given that the large seismic event is the dominant contributor to the configuration 1 initiator, it is likely that it would take a prolonged period of time to retrieve the fuel, repair the spent fuel pool or establish an alternate means of long-term spent fuel storage. Therefore, we presume there will be a significant period of time that the fuel will be exposed to air. On this basis, BNL has chosen a temperature of 565°C as the critical cladding temperature. This results in critical decay times of about 17 months for the representative PWR and 7 months for the representative BWR.

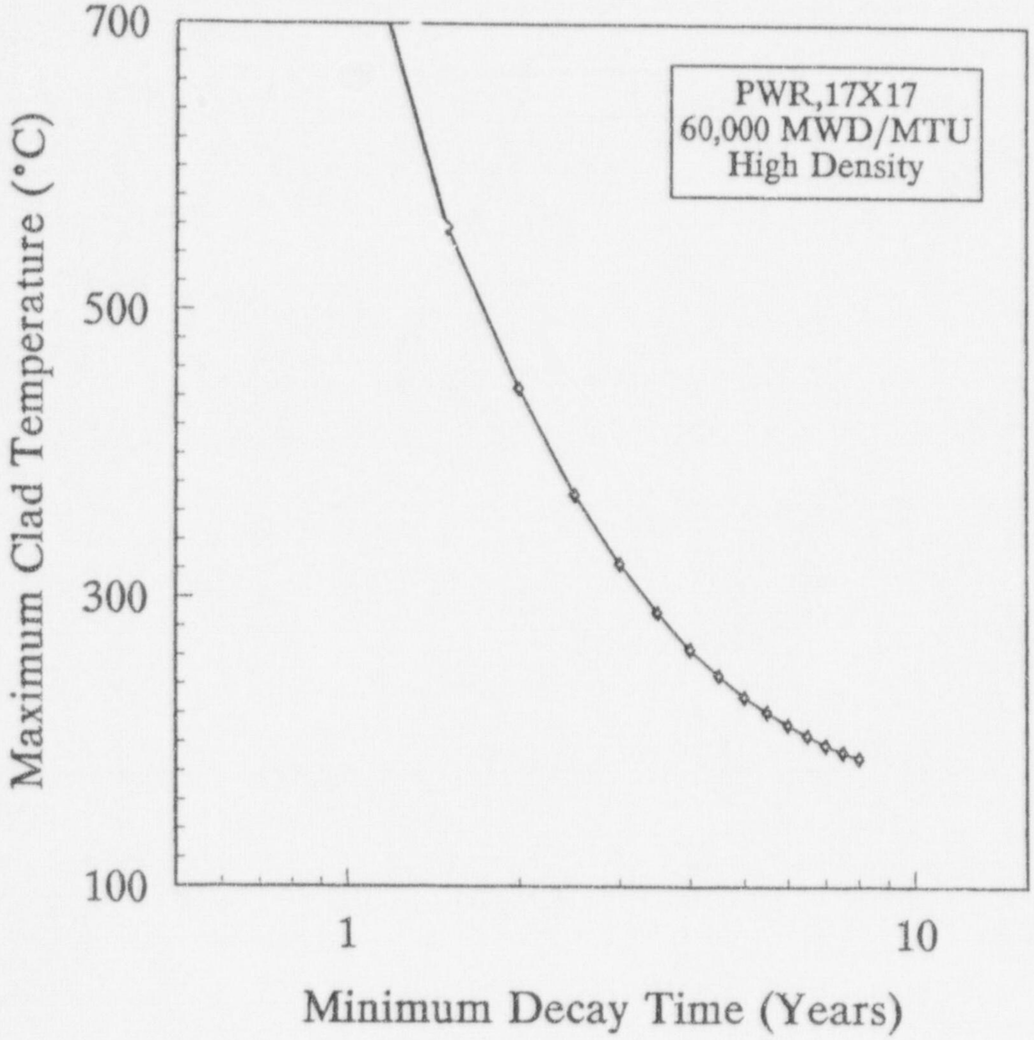
3.1.4 Meteorological and Population Data

Weather and its variability play an important role in the estimation of consequences that may result from a release of radioactivity to the environment. The prevailing weather conditions at the time of release will influence: the extent of downwind transport and lateral dispersion; the atmospheric concentration; and the extent and severity of land contamination. The SNL Siting Study, NUREG/CR-2239[13] and a BNL reassessment[14] were utilized to develop a representative meteorology for the continental United States composed of: mean weather attributes (wind speed, stability, class occurrence total hours, and amount of rain for Omaha, NB); a generic mean wind rose; and an average mixing height.

This study has adopted a generic population distribution within a 500 mile radius of the site that will reasonably envelope the majority of the current reactor sites* and account for future population growth over the life of the plant.

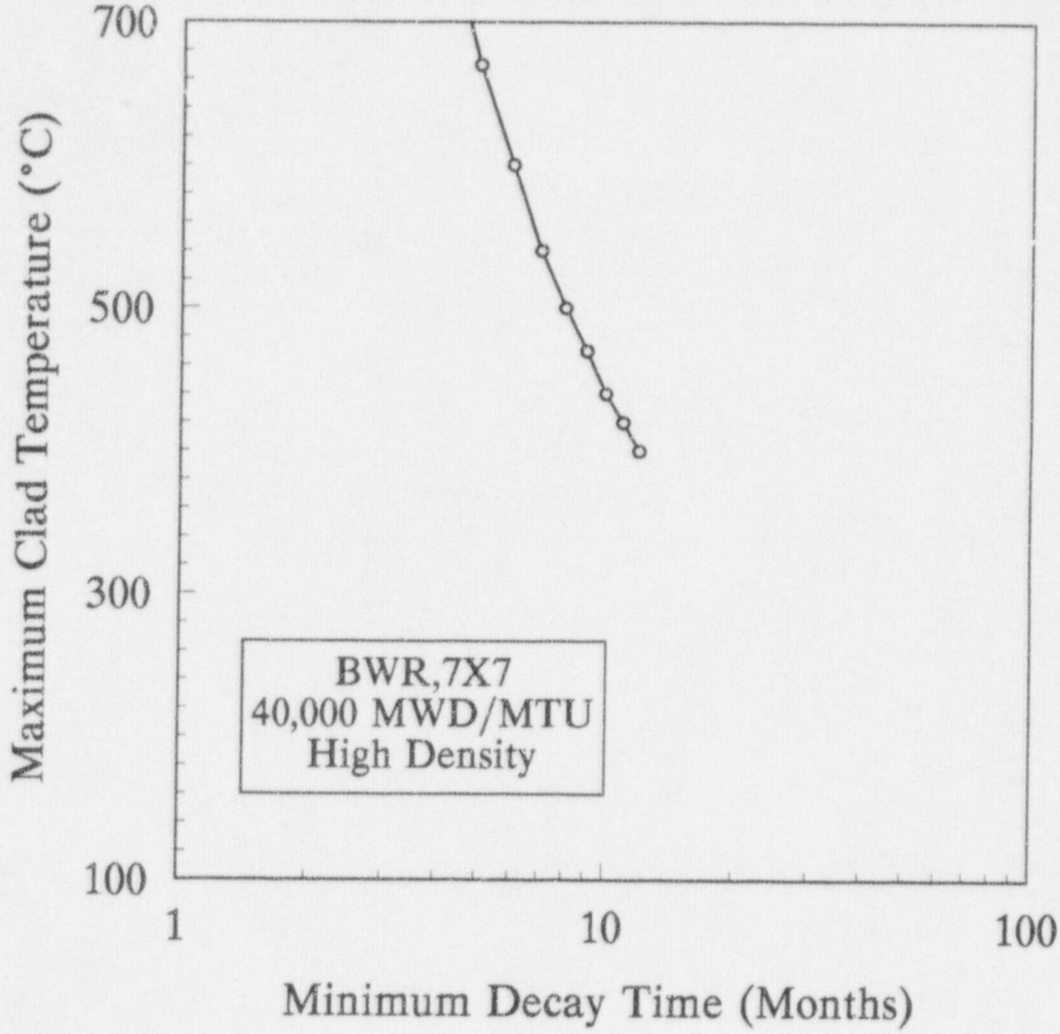
*There are several existing plant sites (i.e., Indian Point, Limerick, and Zion) that precede the issuance of R.G. 4.7 and exceed the site population distributions generally considered acceptable by current NRC policy.

A uniform population distribution (0-30 miles) of 1000 persons per square mile has been specified based on the end of life average population density from Regulatory Guide 4.7. Between 30 to 50 miles, we have assumed a large city of 10 million and a uniform population density of 280 persons/mile^2 for the remaining land in this region.[14] A uniform population density of 200 persons/mile^2 (twice the current average of the 48 contiguous states) was assumed for the area 50 to 500 miles from the plant.



(Adapted from Reference 46)

Figure 3.1 Spent fuel temperature as a function of time for the representative PWR configuration



(Adapted from Reference 46)

Figure 3.2 Spent fuel temperature as a function of time for the representative BWR configuration

3.1.5 Accident Inventory and Source Term

The spent fuel pool inventory at accident initiation is a function of the ages and burnups of the spent fuel discharges that occurred over the life of the plant. The DOE High Level Radioactive Waste Management Database[15] was used as the source of the generic spent fuel inventory data for discharges one year or older.

The inventory of material at risk 12 days after reactor shutdown (i.e., at the beginning of Configuration 1) was developed from both the DOE Spent Fuel Data Base and the default reactor core inventories provided in the MELCOR Accident Consequence Code System (MACCS).

MACCS Version 1.5.11.1[16-17] was used in the next section to model the postulated accident consequence. Like other consequence codes, MACCS models radionuclide releases that occur shortly after reactor shutdown. The code has a default set of risk dominant radionuclide species that is consistent with the premise of a release within days of shutdown. In contrast, the inventory of the spent fuel pool, including the last core offload, has had sufficient time for the short lived isotopes, which have important dose contributions, to decay away. The concern is that perhaps the MACCS default set of isotopes might not accurately model long lived isotopes that are relatively insignificant for short-term releases, but rise in prominence for spent fuel pool accidents. The code default isotopes set was spot checked with the DOE database[15] inventory for two offloads. It was determined that the MACCS code will capture greater than 90% of the activity in the spent fuel. Therefore, it was not necessary to revise the code's default isotope set to include any additional radionuclide species.

The atmospheric source term is a set of characteristics describing the radionuclide release to the environment. These characteristics include: the number of plume segments released, the associated timing duration and release height of each segment, the emergency response warning time and the radionuclide release fractions.

This study examined four cases for Configuration 1. The assumptions for each case are described below:

Case 1 Complete draining of the spent fuel pool occurs twelve days after shutdown. Rapid cladding oxidation starts in the last full core discharge and propagates throughout the pool.

Case 2 Complete pool drainage occurs, again at twelve days. The rapid zircaloy oxidation is limited to the last full core discharge (plus the last refueling offload for PWRs).

Case 3 Complete pool drainage occurs one year after shutdown. The lowered decay heat does not cause rapid oxidation, however the assemblies reach high temperatures and 50 percent of the fuel rods in the pool fail, resulting in a gap release.

Case 4 Partial pool drainage occurs at twelve days, exposing the upper portion of the fuel assemblies. This case assumes all fuel rods in the last full core discharge experience cladding failure, again resulting in a gap release.

This study used the release fractions of NUREG/CR-4982, as modified by studies associated with gap inventory and high burnup fuel.[18-22,33] Table 3.2 provides the source terms developed for the present study.

The majority of the high release fractions for Cases 1 and 2 were largely adopted from NUREG/CR-4982. However, the lanthanum (La) and cerium (Ce) groups have been adjusted slightly to reflect the observed release of fuel fines as part of the gap release in high burnup fuel. The low release fractions for Cases 1 and 2 assumed a decontamination factor (DF) of 10 for all fractions except noble gases and iodine. Cases 3 and 4 heat the fuel cladding to failure, but do not result in fire. The gap release fractions developed for this work differs markedly from the previous efforts. The noble gas fraction, 0.4, was based on high burnup/high linear power calculation and is therefore believed to be conservative. The fractions for the cesium (Cs), iodine (I), and tellurium (Te) groups were based on experimental observation. In the case of the high gap release, these were increased by a factor of ten to reflect evidence that these fractions may increase for high burnup fuels. For both the high and low gap releases, the Te fractions were corrected for the interaction observed to occur with the cladding, since unoxidized cladding will be present. The fractions for the remaining groups are established by the release of fuel fines.

Table 3.2 Configuration 1 Release Fractions

Case	Release Characterization	NG	I	Cs	Te	Sr	Ru	La	Ce	Ba
1H, 2H	Fire/High release	1.0	1.0	1.0	2E-2	2E-3	2E-5	6E-6	6E-6	2E-3
1L, 2L	Fire/Low release	1.0	0.5	0.1	2E-3	2E-4	2E-6	6E-7	6E-7	2E-4
3H, 4H	Gap/High release	0.4	3E-2	3E-2	1E-3	6E-6	6E-6	6E-6	6E-6	6E-6
3L, 4L	Gap/Low release	0.4	3E-3	3E-3	1E-4	6E-7	6E-7	6E-7	6E-7	6E-7

For the set of low gap releases (Cases 3 and 4), all release fractions were reduced by an order of magnitude (DF=10) with the exception of noble gases (NG).

3.1.6 Emergency Response and Other Data Requirements

The MACCS code can model various emergency response actions such as evacuation, sheltering, and post accident relocation (including dose criteria). Consistent with NUREG/CR-5281,[8] this study assumed a short-term emergency response of no planned evacuation, followed by relocation at one day if projected doses are unacceptable. Long-term protective actions include permanent relocation, crop interdiction, and land decontamination or condemnation. The dose threshold for these actions are the MACCS default values which were also utilized in NUREG-1150.[23]

The code also considers land usage and economic data for the region surrounding the reactor site to estimate accident cases. The national average value of farmland of $2094/hectare and a mean value of $73,750/person for non-farm wealth was assumed.[23] The Omaha, Nebraska region, also used for the mean meteorology, was used to model the code's agricultural data block, including the growing season and the fraction of land used for farming.

These estimated accident costs will be used to analyze the insurance issues for permanently shutdown nuclear power reactors.

3.2 Configuration 2 - Cold Fuel in the Spent Fuel Pool

Spent fuel storage Configuration 2 models the continued storage of the fuel in the spent fuel pool. Time has reduced the decay heat, and the rapid clad oxidation or clad rupture events of Configuration 1 are not likely. This section summarizes the input assumptions, such as accident initiator, and source terms that differ from those of the previous section. Other parameters (i.e., spent fuel pool data, rack design, and fuel burnup) remain consistent with the Configuration 1 baseline. A summary of each spent fuel configuration is provided in Table 3.1.

3.2.1 Accident Initiator and Timing

By definition, Configuration 2 eliminates the pool drainage accident scenarios of Configuration 1 from consideration. The prolonged exposure of the low-decay heat fuel in air is not expected to cause fuel rod clad failures. BNL has adopted the traditional fuel handling accident analysis of Regulatory Guide 1.25, with modifications. The present study assumed a single assembly is dropped in the spent fuel pool, resulting in damage to 100 percent of the rods in the affected assembly.

The estimated initiator frequency of 3E-4 events per year* was developed from industry refueling outage data reported in Reference 48, modified to reflect a higher assumed spent fuel transfer rate.

The accident was assumed to occur after the transition from Configuration 1, one to two years after final reactor shutdown.

3.2.2 Accident Inventory and Source Terms

The accident inventories for the Configuration 2 accident cases consist of a single two year old PWR fuel assembly or a single one year old BWR assembly. As before, the DOE spent fuel database**[15] was used to assemble the isotope quantities for the MACCS default set of nuclides.***

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*This is also the estimated release frequency.

**At 60,000 and 40,000 MWD/MTU burnup for the PWR and BWR cases, respectively.

***In both reactor types the MACCS default, risk dominant nuclides represent about 89 percent of the total activity in the fuel.

The source term is composed of the single assembly gap release. In addition to partial releases of the noble gases and iodine (if present), small releases of the remaining nuclide groups are expected on the basis of experimentally observed releases of fuel fines. The Configuration 2 high gap release fractions are the same as Case 3H of Table 3.2 in the previous section. The low gap source term assumes a DF of 100 to credit the scrubbing effect of the water overlying the spent fuel and the retention of the building.

3.3 Configuration 3 - All Fuel Stored in an Independent Spent Fuel Storage Installation (ISFSI)

As discussed in Section 2, after a sufficient decay period, long-term spent fuel storage outside the spent fuel pool becomes a possibility. The decision to apply for a Part 72 license and to transfer all fuel to an onsite ISFSI is a licensee decision that is based, in part, on such plant-specific factors as the timing and method of plant decommissioning,* the preexistence of a licensed ISFSI, and the anticipated start of fuel shipments to a DOE facility. This section discusses the supporting assumptions for Configuration 3 that differ from the previous spent fuel storage configurations.

3.3.1 Accident Initiator and Timing

The Configuration 3 accident initiator** is assumed to be a tornado driven missile that pierces one cask of the ISFSI. An initiator frequency is developed, for the purposes of the offsite liability discussion in Appendix B. The Electric Power Research Institute document, EPRI NP 3365, "Review of Proposed Dry Storage Concepts Using PRA,"[49] developed an initiator frequency of 6E-6 events per year for the extremely severe tornado (windspeed of 567 miles/hour) that would be necessary to generate a missile that could pierce an ISFSI cask. The report conservatively assumes the probability of missile generation,*** missile strike and impact orientation are unity. In addition, the windspeed and the missile speed are considered to be equal; no slippage is considered. Therefore, the extremely severe tornado initiator frequency is also the ISFSI cask release frequency.

BNL believes there are also additional conservatisms embedded in the development of the severe tornado initiator frequency. The frequency was based on a Zion PRA[52] initiator frequency of 1E-3 tornados/mile^2-year for all tornados. According to Regulatory Guide 1.76,[53] the Zion plant is in tornado Region I. Tornado Region I has the most severe design conditions. It comprises over 50% of the land area of the

*Partial DECON or SAFSTOR could allow long-term utilization of the spent fuel pool without significant impact on the facility decommissioning plan. Complete DECON would require fuel transfer to permit decommissioning of the spent fuel pool and supporting equipment.

**Current licensing documents for spent fuel casks and modular concrete vaults do not postulate any credible accident scenarios which will breach the ISFSI.[24-25]

***The vast majority of missiles do not have the rigidity, shape, or weight to pierce the ISFSI cask.

contiguous United States, or in excess of 1,560,000 square miles. Everything else being equal, we would expect to see an average of:

1,560,000 miles^2 x 6E-6 extremely severe tornados/mile^2 - year = 9 extremely severe tornado events per year.

Although windspeeds have been estimated that are in excess of 450 mph,[54] to the best of our knowledge, there has never been a tornado of the magnitude that would be necessary to fail an ISFSI cask.

The equation used in the EPRI report to estimate the annual probability of exceeding a velocity V at a site is:

$$P(V) = \lambda \left(\frac{V_d}{V} \right)^{\frac{1}{k}} R'(V)$$

where λ = local mean rate of occurrence of tornadoes per square mile per year.

V_d = gale velocity

k = 0.5 to 1.6 a parameter value depending on a given storm, and conservatively recommended as 1.6 until such time as additional data becomes available

R′ (V) = 17.4 exp (-0.014v for V ≥ 290 mph,

(As developed in Reference 54.)

The factor R′(V) is an approximation (based on tornado data) that accounts for the relative frequency of different tornado events, with their respective peak velocities and correlated path dimensions. Since a tornado of the magnitude of the ISFSI initiator exceeds the information that was used to develop (R′(V), the use of this equation is suspect.

On the bases of the frequency discussion, we believe that the initiator frequency of this extremely severe tornado is overstated. In our judgement, the frequency should be at least 2 orders of magnitude less.*

Table 3.3 Configuration 3 Release Fractions

NG	I	Cs	Te	Sr	Rw	La	Ce	Ba
0.40	1.5E-5	2.25E-5	1.5E-5	1.5E-5	1.5E-5	1.5E-5	1.5E-5	1.5E-5

*This judgement is supported by NUREG/CR-4461, "Tornado Climatology of the Contiguous United States,"[55] which identifies the windspeed of 10^{-7} probability of tornado strike for all of the U.S. to be significantly less than that required to pierce an ISFSI cask. The staff has referenced NUREG/CR-4461 in the advanced reactors evaluations and is using the same to develop new guidance with less maximum windspeeds for tornado design criteria.

With regard to accident timing, although 10CFR Part 72 allows a minimum in-pool decay time of one year the current vendor requirements and license submittals specify five-to-ten year minimum decay times.[24-25] This study assumed accident initiation at five years after final shutdown.

3.3.2 Exclusion Area and Meteorology

In accordance with 10CFR72.106, this study assumes the distance from the ISFSI to the exclusion area is 100 meters. The onsite weather modeling assumes "A" stability weather with a high wind speed (30 meters/second), approximating the rapid dilution associated with a tornado to develop an estimated dose at the exclusion boundary. The offsite dose model uses the MACCS code. As discussed in Section 4, the use of MACCS under these conditions adds additional uncertainty, but the authors believe the results obtained beyond the exclusion boundary are a conservative approximation.

3.3.3 Accident Inventory and Source Term

The storage capacity varies for each ISFSI type. A metal or concrete storage cask can accommodate 28 PWR or 56 BWR fuel assemblies. Each NUHOMS unit has a slightly smaller design capacity of 24 PWR or 52 BWR assemblies.[26] This study utilized the higher capacity cask inventories and further assumed the high burnup of the previous configurations, 60,000 (PWR) and 40,000 (BWR) MWD/MTU.* The DOE spent fuel database[15] was again used to assemble the quantities of radionuclides for input into the MACCS code.

Licensed ISFSIs are substantial engineered enclosures. The catastrophic failure of the current designs is not believed to be credible. Any damage to the ISFSI and the contained fuel is expected to be limited. Therefore, the accident inventory assumes that all of the fuel rods in one assembly are breached.

The best estimate release fractions for Configuration 3 were developed by a peer group.[39] The group reviewed published information,[21,40-42] and considered the effect of high burnup on the particulate release fractions to the cask. Since the ISFSI design pressure is slightly above atmospheric (~0.4 bar), there could be a slight driving force to the environment if the cask integrity is compromised. A bounding calculation was performed to estimate the fission product retention. Assuming isentropic expansion of the gas within the ISFSI and an environmental pressure associated with a tornado, a decontamination factor (DF) of about 2 was obtained. The Configuration 3 release fractions are presented in Table 3.3.

3.4 Configuration 4 - All Fuel Removed from the Site

In the future, when a DOE MRS (or a high level waste repository) becomes operational, the option of offsite storage (or disposal) of spent fuel will become available. At that time, the DOE will begin accepting spent fuel shipments with a minimum of five years decay.[27] In order to envelope future plant

*Although presently limited to a maximum burnup level of 40,000 MWD/MTU it is anticipated that future ISFSI storage concepts will be licensed for high burnup fuel.

shutdowns when the offsite shipment of fuel can be accommodated, this configuration assumed a five year onsite decay prior to the start of Configuration 4.

Publicly available literature[3-4] was reviewed to identify potential accidents that could occur during the decommissioning of nuclear power plants.

After all the spent fuel has been removed from the site, the estimated inventory that remains, although considerable, is primarily attributable to activated reactor components and structural materials. There are no credible accident sequences that can mobilize a significant portion of this activity. As a result, the potential accidents that could occur during the decommissioning of a nuclear power reactor in Configuration 4 have negligible offsite and onsite consequences. In order to develop onsite property damage insurance recommendations for Configuration 4, a rupture of the borated water storage tank is postulated.[43,44] To support the offsite liability insurance discussions of Appendix B a tank rupture initiator was developed assuming a seismic induced failure. The initiator frequency is approximately 2E-7 events per year based on a tank fragility from Reference 50 and a seismicity curve representative of the eastern United States from Reference 51. Although the health effects are negligible, the cleanup costs are significant.

4 RESULTS OF THE CONSEQUENCE ANALYSES

The MELCOR Accident Consequence Code System, MACCS[16-17] was used in this study to model offsite consequences. The principal phenomena considered in MACCS are atmospheric transport, mitigative actions based on dose projection, dose accumulation by a number of pathways (including food and water ingestion), early and latent health effects, and economic costs.

The prediction of onsite consequences (occupational doses) has traditionally been estimated through deterministic calculation of dose rate(s), dose(s) and contamination level(s), generally of a scoping or bounding character. Typical of these methods, was the guidance provided by Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."[28] A typical application of this method was documented in NUREG/CR-5771.[29]

In this study, a variety of deterministic methods were applied. These included the standard method as outlined in relevant Reg. Guides, and/or alternate methods, such as the Ramsdell model,[30] for estimating the concentration of material entrained in the building wake. The methods are important for predicting on-site consequences, a region generally not modelled adequately by the MACCS code.

4.1 Configuration 1 - Results

A series of MACCS code calculations were performed to quantify the postulated accidents cases for the Configuration 1 conditions described in Section 3.1. For each accident, Cases 1 through 4, and each generic reactor type, two calculations were performed: one using the set of high release fractions (H) and a second employing the set of low release fractions (L). The latter generally included a DF of 10 for particulates to reflect potential for retention of activity in structures. The results are tabulated in Tables 4.1 and 4.2.

A case by case comparison of the results for Configuration 1 indicates that the generic PWR and BWR results are very similar. Generally, the results are within 20 percent of one another, although in a few comparisons the differences may be somewhat larger. This similarity would be expected on the basis of *identical* site assumptions, weather conditions, interdiction criteria, and source term fractional releases adopted for both reactor evaluations. PWR inventories were generally larger than corresponding BWR inventories. The higher PWR consequences were attributable to the assumed higher burnup, the inclusion of the last normal refueling discharge in cases where the last core discharge was considered, and the relatively larger PWR pool size in the cases that considered full pool involvement.

Table 4.1 Mean PWR Consequences

Accident	Inventory	Distance (miles)	Prompt Fatalities	Societal Dose (person-rem $x10^6$)	Latent Fatalities	Condemned Land (sq. miles)	Total Cost ($\$x10^9$)**
Case 1H	full pool	0-50 0-500	70 95	74 339	31,300 143,000	467 2790	287 566
Case 1L	full pool	0-50 0-500	1.2 1.2	62 130	25,300 53,800	297 869	100 117
Case 2H	last core*	0-50 0-500	29 33	81 226	33,200 94,600	286 776	186 274
Case 2L	last core*	0-50 0-500	0.3 0.3	42 70	16,800 28,800	156 188	56 59
Case 3H	50% pool	0-50 0-500	0 0	32 48	13,200 20,400	25 25	25 25
Case 3L	50% pool	0-50 0-500	0 0	6 8	2,400 3,400	2 2	1.1 1.1
Case 4H	last core*	0-50 0-500	0 0	24 36	10,100 15,400	15 15	15 15
Case 4L	last core*	0-50 0-500	0 0	4 5	1,500 2,300	1 1	0.8 0.8

* The "last core" also includes the last normal refueling discharge.
** excludes health effects

A limited comparison can be made of the results obtained in this effort with those of previous investigations. The consequence estimates obtained here are generally higher. For example, the societal dose commitment (0 to 50 miles) for the worst case accident (fire, full pool involvement, high release fractions) reported by Sailor[7] was 2.6 million person-rem; Jo[8] reported 25.6 million person-rem; while in the present work 75.3 million person-rem (BWR) was obtained. As discussed in Appendix A, these early efforts used identical inventory and source term assumptions. The differences observed were primarily due to the population assumptions. The average population density (0-50 miles which includes the large city) used herein was about 1800 persons per square mile. This would support an approximate increase of a factor of two over the dose reported by Jo. The second major reason the consequences are greater is the radionuclide inventory used here. The assumptions made for reactor power, end of plant life fuel burnup and fuel pool capacity, resulted in an inventory which has substantially higher quantities of the long lived radionuclides than previous studies. For example, the total BWR pool inventory of Cs-137 was about a factor of 3 greater than developed by Sailor for the Millstone plant. Thus, the limited comparisons would indicate that the consequences determined in this study were generally higher than the former studies. The consequences are consistent with earlier work, when gross differences in the underlying assumptions are taken into account.

Table 4.2 Mean BWR Consequences

Accident	Inventory	Distance (miles)	Prompt Fatalities	Societal Dose (person-rem x10^6	Latent Fatalities	Condemned Land (sq. miles)	Total Cost x10^9$**
Case 1H	full pool	0-50 0-500	74 101	75 327	31,900 138,000	456 2170	280 546
Case 1L	full pool	0-50 0-500	1.3 1.3	58 120	23,600 49,800	286 784	97 113
Case 2H	last core	0-50 0-500	24 26	81 207	33,000 86,400	262 521	167 234
Case 2L	last core	0-50 0-500	0.2 0.2	38 62	15,300 25,700	140 159	48 51
Case 3H	50% pool	0-50 0-500	0 0	29 45	12,200 18,900	23 23	23 23
Case 3L	50% pool	0-50 0-500	0 0	5 7	2,100 3,000	2 2	1.0 1.0
Case 4H	last core	0-50 0-500	0 0	20 30	8,300 12,700	13 13	12 12
Case 4L	last core	0-50 0-500	0 0	3 4	1,300 1,900	1 1	0.7 0.7

** excludes health effects

The total costs of fuel pool accidents observed in this study were found to rise more sharply than the societal dose. This reflects the tradeoffs of protective (interdiction and relocation) actions. These actions are, of course, intended to limit public exposure to the released radioactivity, but at the increased cost of primarily population dependent interdiction and relocation expenses. Again the major obvious factors, which will drive costs up in comparison to earlier studies, are the larger population at risk and the larger inventory of material considered in this study. This observation is supported by a comparison of the condemned land. Comparing Case 1H in Table 4.1 or 4.2 with case 1A of Table A.2, it can be seen that the condemned area has doubled. Although, Table A.2 identifies this as interdicted area, which might be subject to a different interpretation given the usage of this term by the MACCS code, the text of the Sailor study clearly stated "... interdicted area (the area with such a high level of radiation that it is assumed that it cannot ever be decontaminated)." Condemned land is defined as farmland permanently removed from production, as such it does not account for the population affected area. However, the condemned area for case 1H in the present study clearly indicates a more extensive contamination of all lands when compared to the former study. This increase translates into increased costs.

Table 4.3 PWR Core Melt Accident Results

Accident	Inventory	Distance (miles)	Prompt Fatalities	Societal Dose (person-rem x10⁶)	Latent Fatalities	Condemned Land (sq. miles)	Total Cost ($x10⁶)
RZ1 with evacuation	3800 Mwt core	0-500	88	70.*	35,000	2000	NR
RZ1 no evacuation	3800 Mwt core	0-500	160	220.	110,000	2000	NR
* Doses that were not reported, have been estimated from the number of latent fatalities and the BEIR-V recommended risk coefficient of 5.0E-4 fatalities per person-rem.							

(Reproduced from Reference 14)

For perspective, it is interesting to provide some comparison to core melt accidents. A major core melt accident (RZ1, large early release) was selected from the results reported in Reference 14. This study employed many of the assumptions, i.e., population distribution and weather conditions, that were employed in the present analysis, thus allowing for reasonable comparison. The core melt accident source term was 100% of the noble gases, 27% of the iodine group, 21% of the cesium group, 10% of the tellurium group, 12% of the barium and strontium groups, 0.52% of the ruthenium group, 0.2% of the lanthanum group and 0.6% of the cerium group. Table 4.3 summarizes the reported results.

The core melt accident results are provided for two emergency protective actions: one in which a representative evacuation was modelled along with long term protective actions; and a no evacuation, no long term protective action case. The later case, while unrealistic, provides a very conservative bounding estimate of the consequences. A case with protective actions identical to this study was not reported. However, the results of such an analysis would have provided results intermediate to those reported (with the exception to condemned land which is not affected by emergency response). Comparison with the results shown in Tables 4.1 and 4.2 clearly indicates that for worst case assumptions, i.e., full pool involvement and large source term, the postulated Configuration 1 spent fuel pool accident may have *comparable* consequences to a major core melt accident.

Previous studies have elected to quantify the risks and costs of fuel pool accidents using either Case 1 or Case 2 results. In their final analysis, Sailor, et al.,[7] chose the last refueling offload/maximum source term accident results. In Jo, et al.,[8] a worst case (full pool/maximum source term accident) and a best estimate case (last refueling/maximum source term accident) were explored. For the present evaluation, BNL recommends that the estimated consequences for case 2L be used. This case assumes that the accident is limited to the last full core discharge (plus the last normal refueling discharge in the case of a PWR) and the lower release fractions, that reflect some credit for fission product retention.

This recommendation has been made for the following reasons. As discussed in NUREG/CR-4982, there is a large degree of uncertainty associated with the fire propagation throughout the entire pool. Additionally, mitigative options such as rack modifications,[5-6] (i.e., increased hole size) and fuel management practices (including checkerboarding of fresh assemblies and the use of regions in the SFP) are all possible. Thus, it is possible to reduce the likelihood of propagation into the older assemblies. Regarding the lower fractional releases in the recommended case, BNL considered the implications of the accident that occurred at the Chernobyl Unit-4 power plant in the Ukraine.[31] Although Chernobyl is clearly not an analog of the accidents treated in this section, several similarities exist which have relevance to the fuel pool accident. These include oxidation of the clad, failed reactor structure and the availability of air. (There are of course many dissimilarities, such as the burning of the graphite moderator which provided additional heating and the expulsion of fuel fragments to the environment during the violent steam explosion.) Nonetheless, it is difficult to envision that the spent fuel pool accident(s) could result in much greater release. The estimated Chernobyl release, as a fraction of core inventory, was 1.0 of the noble gases, 2.0E-1 of the iodine, ~1.3E-1 of the cesium and tellurium, 4.0E-2 of the strontium, 5.6E-2 of the barium, and approximately 3.0E-2 of the ruthenium, cerium and lanthanum group nuclides.

A comparison with the source terms in Table 3.2, shows better agreement for the noble gas (NG), I and Cs groups with the low (Case 2) release source term. In contrast, the Chernobyl releases for Te and the nonvolatiles greatly exceed any of the releases shown. There are two justifications for the lower Te and nonvolatile group releases used in this study. In the case of Te, the formation of an intermetallic compound with Zr in the clad is known to suppress Te release until the clad is completely oxidized. At Chernobyl, complete oxidation of the clad probably occurred in the rubble bed that the reactor became. In the spent fuel pool accident, Sailor et al. believed that cladding would melt prior to complete oxidation, relocate and be quenched on the floor of the pool. The cladding material would thus retain Te.

4.2 Configuration 2 - Results

The offsite consequences for Configuration 2, "Cold Fuel in the Spent Fuel Pool," were modeled with the MACCS code using the input assumptions of Section 3.2. The deterministic treatment outlined in Reg. Guide 1.25 was not pursued because it provided a limited description of the consequences.*

The estimated offsite consequences for each reactor type and assumed environmental release is shown in Table 4.4.

As expected, these results indicate a far lower level of offsite consequences than the Configuration 1 cases. The much lower inventory is the obvious reason for the low level of predicted accident consequences. In no case is prompt fatalities indicated. Societal doses are very much lower than those developed for Configuration 1 accidents. These low doses are reflected in the low numbers of latent fatalities estimated. For either reactor type a very small area of farmland is predicted to be permanently condemned, only when the high gap release fractions (worst case assumptions) are employed. These lands are well within 10 miles of the plant. When the low gap release fraction (central estimate) was

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*The Reg. Guide 1.25 methodology is limited to noble gases and iodine. The extension of this methodology to address the small fraction of particulates postulated for Configuration 2 is beyond the scope of this program.

employed, the condemnation of land was not predicted. The estimated total off site cost, excluding health costs, range from 28 million dollars to negligible, dependent on reactor type and release assumptions. These costs are very much lower than the Configuration 1 accident.

Table 4.4 Mean Offsite Consequences - Configuration 2

Generic Plant Type	Release Characterization	Distance (miles)	Prompt Fatalities	Societal Dose (person-rem)	Latent Fatalities	Condemned Land (sq. miles)	Total Cost** $\$x10^6$
PWR	High gap	0-50 0-500	0 0	2.E+5 3.E+5	100 134	0.03*	28
PWR	Low gap	0-50 0-500	0 0	3.E+3 4.E+3	1 2	0.0	neg.
BWR	High gap	0-50 0-500	0 0	7.E+4 9.E+4	31 40	0.002*	6
BWR	Low gap	0-50 0-500	0 0	8.E+2 1.E+3	0.4 0.5	0.0 0.0	neg.
* Indicates within 10 miles of plant, ** excludes health effects, "neg" denotes negligible							

To estimate the dose at the site boundary (0.4 miles beyond the point of release) the MACCS calculations were repeated, since centerline dose was not predicted for the "relocation only" emergency response. The code requires an evacuation model to calculate centerline dose. To maximize time in the plume, BNL chose a ten hour delay to start the evacuation. Thus, individuals near the site boundary were exposed for ten hours to the release, then evacuated. The lifetime whole body effective dose equivalent for this exposure was calculated. Both the high and low source terms assumed for Configuration 2 were evaluated. As calculated by the MACCS code, these doses included exposure from all direct pathways.

In the mean, the doses at the site boundary were estimated to be 930 and 0.9 mrem, for the high and low PWR Configuration 2 release assumptions. The BWR doses were estimated to be about a factor of 4 lower.

For the purpose of regulatory requirement analysis, it is recommended that the consequences developed with low fractional releases be employed. The consequences estimated with the high gap releases should be viewed as an upper limit, as no credit is taken for retention in the pool or in the undamaged housing structure. Clearly, some level of fission product retention in the pool and in the structure is to be expected. The low fractional releases therefore would appear to provide a more reasonable estimate of the actual releases that could occur.

Configuration 2 - Onsite Consequences

Onsite dose assessments were performed with the Ramsdell model[30] and the model provided in Reg. Guide 1.145.[32] These deterministic analyses, which take into account the entrainment of the release into a building wake, were performed for two polar weather conditions to provide an indication of the range of anticipated dose(s). Descriptions of these dispersion/dose models are provided in Reference 30. For the Ramsdell model, unstable A and stable G weather conditions were evaluated at a 1 meter/sec wind speed. For the Reg. Guide 1.145 model, Class A and F weather were evaluated. The release was assumed to occur at a height of 10 meters and the reactor structure had an effective area of 1500 square meters which enters into the description

Table 4.5 Configuration 2 Estimates of the Committed 50 Year Dose to a Worker

Model	Weather Stability	Committed Dose (rem)	
		PWR	BWR
Ramsdell	A G	0.88 1.23	0.24 0.33
Reg. Guide 1.145	A F	0.60 4.24	0.16 1.14

of the building wake. The integral 50 year effective whole body dose commitment from cloudshine and inhalation were estimated 100 meters downwind of the release. The necessary dose conversion factors were taken from the MACCS code DOSDATA file.[16] These calculations conservatively assumed an individual is immersed in the release plume for the entire 2 hour duration of the release.

Table 4.5 provides the estimated on site ("parking lot") dose assessment. Only the lower release for each generic reactor type was evaluated.

The range of dose is dependent on both the assumed weather conditions at the time of release and the model that was employed to arrive at the result. In all cases, the estimated doses for the single assembly fuel handling accident are relatively low.

Since the Ramsdell model has been developed more recently than the regulatory guidance and since it has been based on the results of experimentation, the authors were inclined to place more confidence in its estimates. Thus assuming stable weather condition G at the time of release for a degree of conservatism, the onsite worker dose from the postulated fuel handling accident were estimated at 1.2 and 0.3 rem, PWR and BWR, respectively.

The cleanup and decontamination costs for the Configuration 2 fuel handling accident were estimated using the cost estimates provided in a study performed by Pacific Northwest Laboratories (PNL).[34] Three reactor accident regimes were considered in the PNL study. The least severe of these regimes, assumed that the accident involved a 10% cladding failure, no fuel melting, moderate contamination of structures and no significant damage to the physical plant. While the extent of assumed fuel damage was greater than the single assembly fuel handling accident, several similarities are observed. The cleanup and decontamination of the plant structure(s) to bring the plant the site to a safe condition will require damaged fuel removal, water cleanup, and surface decontamination of walls, floors, etc. Since a release of fuel fines for a mechanical disruption of the fuel cladding is postulated, and complete retention in the pool coolant is not assured, potential fission product contamination of the interior of the structure housing the spent fuel pool must be assumed. As such, the estimate developed by PNL provides a basis for estimating the cleanup cost of a fuel pool accident. The costs were $98 and 72 million (1981$) for BWR and PWR plants, respectively. If we assume that the extent of contamination and complexity of cleanup and decontamination are proportional to material at risk in the respective accidents and the cleanup cost escalates at 5% per year, the BWR and PWR costs for a fuel handling accident are $2.7. and 7.8 million dollars, respectively. Since these costs may not be totally elastic, a contingency factor of three has been added. This places the total onsite cost at approximately $9 to 24 million dollars. These costs are relatively small and further quantification is not believed to be necessary for this analysis.

4.3 Configuration 3 - Results

Offsite consequences were again modelled with the MACCS code. The identical set of assumptions that were employed in the Configuration 1 and 2 analyses were used for Configuration 3 with the following exceptions: the exclusion boundary was 100 meters; the release height was 1 meter; and the height and effective width of the ISFSI were 2 and 6 meters, respectively. The appropriate Configuration 3 inventories and source terms were used. The use of the MACCS code, or for that matter any Gaussian dispersion model, at a distance of 100 meters is debatable. It is generally agreed that the experimentally determined dispersion parameters, and more importantly, the analytical expressions used within the MACCS code to summarize this data, provided a better picture of plume behavior at a distance greater than several hundred meters. Thus, the estimated results of the MACCS code close to the point of release are subject to an additional degree of uncertainty, whereas results beyond several hundred meters are not. However, this limitation is minor in comparison to the limitation discussed below.

The standard treatment of estimating offsite consequences with the MACCS code, and in particular sampling representative weather conditions, is in conflict with the assumed accident scenario. The accident was assumed to be initiated by a tornado driven missile with resultant very rapid release of material. The weather conditions at the time of release are therefore more accurately described as high turbulence with very high velocity winds. Accurate treatment of these conditions is beyond the capabilities of the MACCS code. However, the results obtained with the code executed in the typical fashion of accident analysis, should provide a conservative estimate of the accident consequences. (It can be stated that the anticipated dispersion occurring in the wake of a tornado would be much greater than that predicted for practically all other weather conditions).

The estimated offsite consequences for each type of reactor fuel is presented in Table 4.6.

The offsite consequence estimates provided in Table 4.6 are qualitatively comparable to those obtained for Configuration 2, and low in comparison to Configuration 1.

To obtain an estimate of the dose at the site boundary (for Configuration 3 the site boundary was placed at 100 meters beyond the point of release), the MACCS calculations were not repeated as was the case for Configuration 2. The results of the Reg. Guide 1.145 treatment,[32] which were intended to assess worker exposures, also serve as a reasonable estimate of the dose at the site boundary, since the ISFSIs were located 100 meters from the exclusion boundary in this study. The 50 year committed doses are 472 millirem for the PWR and 82 millirem for the BWR. The difference in estimated committed doses is primarily attributable to the greater nuclide inventory and the higher burnup associated with the PWR assembly.

Table 4.6 Mean Offsite Consequences - Configuration 3

Generic Plant Type	Distance (miles)	Prompt Fatalities	Societal Dose (person-rem)	Latent Fatalities	Condemned Land (sq. miles)	Total Cost ($x10^6)*
PWR	0-50 0-500	0 0	5.9E+2 6.9E+2	1.8E-1 2.2E-1	0.002	neg.
BWR	0-50 0-500	0 0	1.2E+2 1.5E+2	4.2E-2 5.1E-2	0.0	neg.
"neg" denotes negligible, * excludes health effects						

Onsite costs for Configuration 3 are estimated to be the sum of the replacement cost of the damaged cask and of the removal and disposal cost of contaminated soil. The cost of an ISFSI cask is $0.75 to 1 million dollars. The onsite area that is contaminated is estimated to be 0.002 square miles. Assuming the affected soil is removed to a depth of 3 inches and a disposal cost of $320.00 per cubic foot, the soil cleanup costs are approximately 5 million dollars. The total estimated costs are about 12 million dollars, including a contingency factor of about two.

4.4 Configuration 4 - Results

After all the spent fuel has been removed from the site, the radionuclide inventory that remains, although considerable, primarily consists of activated reactor components and structural materials. There are no credible accidents that can mobilize a significant portion of this activity. Previous studies[3,4] have estimated that routine and postulated accident releases to the environment were in the range of μCi to 10 mCi. Releases of this magnitude are also expected to result in negligible onsite accident worker doses and negligible onsite contamination.

For the purpose of estimating onsite accident cost one could consider an accident at a power plant similar to the postulated borated water tank rupture accident that was discussed in the Rancho Seco exemption request.[35] This scenario postulated that the most severe accident was the postulated rupture of the borated water storage tank (BWST) which could release about 450,000 gallons of slightly radioactive water onto the plant grounds. The level of released activity was small, but it was assumed that a cleanup of the grounds would be required. The cost of cleanup is driven by the volume of liquid and not directly by the level of activity in the water. This is illustrated by Tables 4.7 and 4.8 which present the expected concentration of radioisotopes in the BWST. Table 4.7 presents the expected level of short-lived radioisotopes, while Table 4.8 provides the level of long lived radioisotopes at selected times after shutdown. Most of the radioisotopes listed in Table 4.7 decay to nothing within 120 days, and virtually all are gone after 1 year.

At Rancho Seco, the BWST has a capacity of 450,000 gals. The activity of this water is extremely low, and after 5 years is primarily due to tritium with an activity of 5000 curies, (a soft beta emitter) and approximately 60 mCi of Cs-137. This amount of radioactivity is generally considered to be a trace contamination; all the shorter half-lived nuclides, shown on Table 4.8, have decayed away. The cleanup estimate developed by the Sacramento Municipal Utility District (SMUD) for the Rancho Seco plant primarily consisted of the removal and disposal of 18 inches of gravel and two feet of the underlying soil in the vicinity of the BWST. This would result in the disposal of about 150,000 ft^3 of soil. SMUD assumed a 1991 waste disposal cost of $150.00 per cubic foot. Waste transportation costs were neglected.

BNL modified the Rancho Seco plant specific estimate to make it more generic by using the 1995 disposal cost of $320.00/$ft^3$ for the Barnwell facility.[36] This results in a cleanup cost of about $54 million.*

However, it is likely that much of this contaminated water would migrate toward the water table and not be captured by the mechanical removal of the surface soil. The contaminated water could reach the water table below the site and result in tritium levels in excess of the maximum concentration limit for drinking water. BNL has calculated that in the time it takes the plume to reach the site boundary, radioactive decay and dispersion could be expected to reduce the tritium concentration below the maximum concentration limit for drinking water, thus it is assumed no treatment would be required.

In order to encompass the cost of onsite groundwater characterization, groundwater monitoring and sample testing over approximately 60 years, the waste disposal estimate of $54 million has been multiplied by a factor of ~2 to $110 million.

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*Consisting of removal, disposal and restoration costs. Waste transportation costs were neglected.

Table 4.7 Activity of the Short-Lived Isotopes in the Boric Acid Concentration Tanks

Isotope	Concentration (μCi/ml)	Max. Activity (μCi in 450,000 gal)
I-131	2.45E-08	50.11965
I-132	0	0
I-135	3.36E-09	6.873552
I-136	1.09E-11	0.22298
Cs-136	6.23E-07	1274.471
Mo-99	1.90E-07	388.683
Y-90	9.20E-09	18.82044
Kr-85m	1.24E-13	0.000254
Kr-88	0	0
Xe-131m	9.52E-09	19.47506
Xe-133	8.57E-07	1753.165
Xe-133m	5.88E-09	12.02872
Xe-135	1.80E-10	0.368226
Y-91	5.34E-08	109.2404

Table 4.8 Activity of the Long-Lived Isotopes i.. the Boric Acid Concentrate Tanks

Isotope	Concentration (μCi/ml)	Activity (Ci) in 450,000 gal.					
		Initial Activity	@120D	@1 yr	@2 yr	@5 yr	@10 yr
H-3	2.5	5110	5020	4830	4570	3860	2910
Cs-137	0.00003	.0610	.0605	.0596	.0582	.0543	.0484
Kr-85⁺	3.30E-08	6.7(-4)*	6.6(-4)	6.3(-4)	5.9(-4)	4.8(-4)	3.5(-4)

⁺Assumed release to atmosphere at time of spill

*6.7(-4) - 6.7×10^{-4}

5 REGULATORY ASSESSMENT SUMMARY

The preceding sections of this report have provided an overview of the processes that are likely to occur when a nuclear power plant permanently ceases operation. The primary focus of this study has been the storage alteratives for the spent fuel. Section 4 examined multiple cases for each spent fuel configuration. A "best estimate" case/consequence analysis was presented for each spent fuel storage configuration including: societal dose, latent fatalities, the amount of condemned land, and the estimated cost of the postulated accident.

After a plant is permanently shutdown, awaiting or in the decommissioning process, certain operating based regulations (or technical issues) may no longer be applicable. The purpose of this section is to present the results of this regulatory assessment.

A list of candidate regulations was identified from a screening of 10CFR Parts 0-199.[37] Each of these technical issues was subjected to a detailed review which included federal register notices, SECYs, NRC policy statements, regulatory guides, standard review plans, NUREGs, NUREG/CRs,
\etc, to develop an understanding of the regulatory
bases. The continued applicability of each technical issue was assessed within the context of each spent fuel storage configuration, the results of the consequence analyses, as well as the expected plant status.

With the possible exception of Part 171, "Annual Fees for Licensees," each regulation is ultimately focussed on the protection of public health and safety. However, a particular regulation may not be applicable to a permanently shutdown plant in general, or a specific spent fuel storage configuration. For example, an exemption from the containment leakage testing requirements of 10CFR50.54(o) for a permanently defueled plant will not impact public health and safety as the plant risk is primarily associated with the spent fuel that is now stored in the spent fuel pool outside the primary containment.

The results of the regulatory assessment are presented in Table 5.1. The detailed recommendations, including regulatory background, specific cites, and regulatory assessment are included as Appendix B to this report.

Table 5.1 Assessment of Continued Regulatory Applicability for Permanently Shutdown Nuclear Power Reactors (Summary)

		Regulatory Applicability[2-5, 30]				
		Configuration				
Technical Issue	**10CFR Reference[1]**	**1**	**2**	**3[6-8]**	**4**	**Notes**
Fitness for Duty	Part 26 55.53(j), (k) 72.194	P F	N N	N N F	N N	9,10
Technical Specifications	50.36, .36a, .36b 72.26, 72.44	P	P	P F	P	11
Combustible Gas Control	50.44	N	N	N	N	
ECCS Acceptance Criteria	50.46	N	N	N	N	
Emergency Planning and Preparedness	50.47, .54(q),(t) App. E 72.32	F	P	P F	P	12,13
Fire Protection	50.48, App. R 72.122	P	P	P F	P	15
Environmental Qualification	50.49	N	N	N	N	
QA Program	50.54(a), App. B Part 72, Subpart G	P	P	P F	P	16
Operator Requalification Program	50.54(i), 55.45, 55.59 72.44(b) Part 72, Subpart I	P	P	N F	N	17
Operator Staffing Requirements	50.54(k), 50.54(m)	N F	N P	N N	N N	18
Containment Leakage Testing	50.54(o), App. J	N	N	N	N	
Security Plan	50.54(p), 70.32, Part 73 Part 73 App. B and C 72.44(e) Part 72 Subpart H Part 73	P	P	N F	N	19
Onsite Property Damage Insurance	50.54(w) Part 72	F	P	P *	P	
Inservice Inspection Requirements	50.55a(g)	P	P	N	N	21
Fracture Prevention Measures	50.60, .61, Apps. G and H	N	N	N	N	
ATWS Requirements	50.62	N	N	N	N	

		Regulatory Applicability[2-5, 30]				
		Configuration				
Technical Issue	10CFR Reference[1]	1	2	3[6-8]	4	Notes
Fitness for Duty	Part 26 55.53(j), (k) 72.194	P F	N N	N N F	N N	9,10
Loss of all AC Power	50.63 72.122(k)	N	N	N F	N	22 23
Maintenance Effectiveness	50.65 POL before 7/10/96 POL after 7/10/96	N P	N P	N N	N N	24
Periodic FSAR Update Requirement	50.71(e) 72.70	P	P	P F	PN	 26
Training and Qualification of Nuclear Power Plant Personnel	50.120	P	P	N	N	
Material Control/Accounting of Special Nuclear Material (including US/IAEA Agreement)	70.51, .53, 74.13(a) (Part 75) 72.72, .76	F	F	N F	N	27 27
Financial Protection Requirements	Part 140 Part 72	F	P	P *	P	
Annual Fees for Licenses	171.15 171.16	P	P	P F	P	29

* See discussion in Appendix B.

NOTES TO TABLE 5.1

1. 10CFR Parts 0 to 199, revised January 1, 1995.

2. All other regulatory requirements applicable to nuclear power reactors and not listed in this table are assumed to remain in effect, unless addressed by a plant-specific exemption.

3. The spent fuel storage configurations are defined in Sections 2 and 3 of this report. Briefly:

   Configuration 1 - hot fuel in the spent fuel pool
   Configuration 2 - cold fuel in the spent fuel pool
   Configuration 3 - all fuel stored in an ISFSI
   Configuration 4 - all fuel shipped offsite

4. Configuration 1 also assumes the licensee has a Possession Only Licensee or that a confirmatory letter has been issued to prevent refueling the vessel without NRC authorization.

5. F-Regulation continues to be fully applicable for this spent fuel storage configuration.
   P-Regulation is assessed to be partially applicable for this configuration.
   N-Regulation is not considered applicable to this configuration.

6. A permanently shutdown nuclear power plant may store its fuel in an Independent Spent Fuel Storage Installation, before, during, and after the plant itself has been decommissioned. As such, Configuration 3 must examine the regulatory requirements for the plant without fuel (similar to Configuration 4) and the ISFSI. This necessitates two (or more) entries in Table 5.1 for Configuration 3. The first (and second, if applicable) pertains to the plant itself prior to the completion of decommissioning. The last entry examines the Part 72 requirements for the ISFSI.

7. The requirements of Configuration 3 remain applicable until all fuel has been removed from the ISFSI and shipped offsite.

8. In addition to the applicable provisions of Part 72 as noted for Configuration 3, Parts 20, 21, 71, and 73 remain applicable to the transportation of spent fuel from the ISFSI to a HLW repository or MRS.

9. Although the Part 26 requirements may no longer be appropriate for certain spent fuel storage configurations, the recordkeeping requirements of Section 26.71 are still applicable.

10. The Part 26, Fitness for Duty requirements remain applicable for Configuration 1. However, the scope of the program can be limited to those personnel with unescorted access to the fuel building.

11. The technical specification requirements are very plant specific. Plant systems and controls necessary for the continued public health and safety will vary from plant to plant. BNL

recommends a plant-specific amendment request to reduce the scope of the operating tech specs or institute defueled tech specs.

12. BNL recommends that all emergency planning and preparedness requirements remain applicable to Configuration 1, with the exception of the Emergency Response Data System (Part 50, Appendix E, VI).

13. BNL recommends site-specific calculations to establish a new smaller EPZ boundary for the plant for Configuration 2. Based on the assumption (subject to plant-specific verification that no members of the public will be exposed in excess of the EPA PAGs, BNL recommends the licensee apply for exemptions from the following Part 50 EP requirements for Configuration 2:

- The early public notification requirements of 50.47(b)(5) and Appendix E.IV.D.3.
- The periodic dissemination of emergency planning information to the public of 50.47(b)(7) and Appendix E.IV.E.8.
- Offsite emergency facilities and equipment such as the EOF, and the emergency news center (50.47(b)(8), Appendix E.IV.E.8).
- Offsite radiological assessment and monitoring capability, including field teams (50.47(b)(9)).
- Periodic offsite drills and exercises (50.47(b)(14), Appendix E.IV.F.3).
- Licensee headquarters support personnel training (50.47(b)(15), Appendix E.IV.F.b.h).

Since decommissioning accidents that do not involve spent fuel have negligible public health consequences offsite EP can also be eliminated for Configurations 3 (plant only) and 4.

14. The emergency planning requirements for ISFSIs that are not associated with an operating nuclear power plant are the subject of a final rule issued on June 22, 1995 [60FR32430].

15. Each licensee has a Fire Protection Program that, in addition to safe shutdown requirements, has training requirements, administrative procedures and controls, and detection/suppression requirements for plant areas that contain radioactive inventories with potential offsite consequences. BNL recommends deleting requirements directly related to safe shutdown capability. Further reductions in the scope of the fire protection program should be on a plant-specific basis.

16. Permanently defueled plants are expected to be able significantly to reduce the scope of their QA program without impacting public health and safety. In accordance with 50.54(a)(3), any proposed changes to the previously accepted QA program must be approved by the NRC.

17. The licensee should submit, per 10CFR50.54(i), a revised operator requalification program limited to fuel handling to reflect the defueled configuration.

18. BNL recommends that at least one licensed SRO be present or readily available on call at all times (see 50.54(m)(l)), for Configurations 1 and 2. Our concern is maintaining fuel cooling

under off normal conditions and the ability to carry out the units' emergency plan (EP), at least in its early stages.

19. In comparison to an operating unit, a permanently defueled plant has less vital equipment and a potentially smaller vital area(s). Accordingly, it is expected that these licensees will continue to apply for exemptions to reduce the scope of the plan.

20. Not used.

21. The scope of the Inservice Inspection Program can be reduced to address only those systems in the existing plan that support spent fuel storage. Some plants do not include spent fuel cooling in their program and may eliminate the Program in its entirety.

22. The intent of the Station Blackout (SBO) Rule is to maintain the risk of fuel damage due to SBO to $\sim 10^{-5}$/reactor year. Permanently shutdown plants meet the intent of 10CFR50.63. BNL recommends existing SBO plant procedures and training be revised to reflect the storage of all fuel in the spent fuel pool.

23. For Configuration 3, offsite power is required for ISFSI security and monitoring systems.

24. The Maintenance Rule does not become effective until July 10, 1996. Plants that request a POL prior to that date should not be subject to this requirement. A facility that is permanently shutdown after that date will have a program to enhance maintenance effectiveness which can be reduced to those systems that support fuel storage and handling, building ventilation and filtering, and radiation monitoring.

25. Not used.

26. ISFSIs are currently required to submit an *annual* FSAR update per 10CFR72.70.

27. The Part 70 license remains in effect until the site is released for unrestricted use. However, an exemption from the special nuclear material (SNM) control and accounting requirements of Parts 70 and 74 and the safeguards requirement of Part 75 can be issued after the SNM has been disposed of. However, please note that an ISFSI has its own requirements under Part 72.

28. Not used

29. Although the current practice is to grant full exemptions from the annual licensing fees for permanently shutdown power reactors, BNL proposes a partial exemption for future years. As the NRC experience with large power reactor decommissioning grows, a fee based on the services provided to these licensees could be applied. Alternatively, Part 171.15 fee that is equivalent to the ISFSI annual fee may be appropriate.

30. This regulatory assessment assumes an onsite, operating spent fuel pool is not necessary to satisfy the fuel retrievability requirement of 72.122(l).

6 SUMMARY AND CONCLUSIONS

Brookhaven National Laboratory (BNL) has undertaken a program (FIN L-2590), "Safety and Regulatory Issues Related to the Permanent Shutdown of Nuclear Power Plants Awaiting Decommissioning." This report summarizes the results of the program, which performed a regulatory assessment for generic BWR and PWR plants that have permanently ceased operation.

Previous studies have concluded that decommissioning accidents that do not involve spent fuel have negligible off-site and on-site consequences. Therefore this study focused on current and future spent fuel storage alternatives for the permanently shutdown facility. Four spent fuel storage alternatives were identified:

Configuration 1 -	Hot fuel in the spent fuel pool
Configuration 2 -	Cold fuel in the spent fuel pool
Configuration 3 -	All fuel stored in an ISFSI
Configuration 4 -	All fuel removed from the site

Each of these configurations was further defined to support the consequence analyses and the regulatory assessment. A set of assumptions was developed to envelope future end of life nuclear power plant shutdowns, as well as plants that have prematurely ceased operation. Thus, this study postulated: higher end of life fuel burnups than presently experienced; spent fuel pools at full capacity; and a high population density to account for future industry and population trends. In addition, this study also differs from previous efforts because the gap release source terms, used herein, are partially based on experimental results and include a small fraction of fuel fines.

Consequence Analyses

Several accident cases, with different inventory and release assumptions, were evaluated for each spent fuel storage configuration. Table 6.1 presents the consequences for the accident cases that were adopted for the regulatory assessment. The Configuration 1 accident postulates an event that causes the draining or boiloff of the water in the fuel pool, exposing the relatively hot spent fuel assemblies to an air environment. The most recently discharged assemblies self heat to a point where the Zircaloy oxidation becomes self sustaining, resulting in extensive clad failure and fission product release. As shown in Table 6.1, the Configuration 1 accident consequences are severe, approximating those of a core melt accident. These results are higher in comparison to previous studies. This is primarily attributable to the higher population assumption used herein. A secondary contributor is the greater radionuclide inventory. The assumptions made for reactor power, end of plant life fuel burnup and fuel pool capacity* resulted in an

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*Does not impact the recommended Configuration 1 accident consequences.

inventory with substantially higher quantities of long lived radionuclides than those assumed in previous studies.*

After sufficient decay time has elapsed and the rapid oxidation phenomenon is not likely, the fuel was considered to be in Configuration 2, "Cold fuel in the spent fuel pool." The accident initiator was the drop of a single assembly, resulting in a gap release. In addition to partial releases of the noble gases and iodine (if present), small releases of the remaining nuclide groups are expected on the basis of experimentally observed releases of fuel fines. The source term for the recommended Configuration 2 accident case includes credit for the scrubbing effect of the water overlying the fuel.

As shown in Table 6.1, the estimated consequences of the bundle drop accident are very much lower than those of Configuration 1. However, the consequences are higher than a Reg. Guide 1.25 analysis which would not consider particulates in the gap release source term.

Although the long term storage of spent fuel in the fuel pool is possible, this study considered the transfer of all fuel to an ISFSI. For accident analysis purposes, the Configuration 3 initiator is a tornado generated missile that pierces one cask of the ISFSI. The recommended accident cases assume one assembly is damaged. A high burnup gap release with a small amount of particulates was again assumed. As shown in Table 6.1, the estimated consequences are generally less than the Configuration 2 results.

After all fuel has been removed from the site, the radionuclide inventory that remains, although considerable, cannot be easily dispersed into the environment. Previous studies have estimated very low accident releases that would have negligible offsite and onsite health effects. For the purpose of estimating an onsite accident cost, this study considered the postulated rupture of the Borated Water Storage Tank. The level of released activity, although small, was assumed to require a cleanup. As shown in Table 6.1, BNL estimated a cleanup cost of 110 million dollars for this accident.

Regulatory Assessment

After a plant is permanently shutdown, awaiting or undergoing decommissioning, certain regulations, which are based on full power operation, may no longer be applicable. BNL identified a list of candidate regulations (or technical issues) from a screening of 10CFR Parts 0-199. Each of these technical issues was subjected to a detailed review which included federal register notices, SECY memos, NRC policy statements, regulatory guides, standard review plans, NUREG reports, NUREG/CR reports, etc. to develop an understanding of the regulatory bases. The continued applicability of each technical issue was assessed within the context of each spent fuel storage configuration, the results of the consequence analyses, as well as, the expected plant configuration.

The public risk associated with a permanently shutdown nuclear power plant is very different from an operating unit, both in magnitude and content. Accident sequences such as LOCAs and ATWs are no

*NUREG/CR-4982 used Millstone and Ginna information (Circa 1987) to develop a "snapshot" of plant specific spent fuel pool radionuclide inventories that have since been exceeded.

longer relevant to the defueled facility. Regulations that are designed to protect the public against full power and/or design basis accidents are no longer applicable. Therefore, it is recommended that the following regulations be deleted for all spent fuel storage configurations of the permanently shutdown plant:

- Combustible Gas control (50.44)
- ECCS Acceptance Criteria (50.46)
- Environmental Qualification (50.49)
- Operator Presence at the Controls (50.54 (k))
- Containment Leakage Testing (50.54(0), Appendix J)
- Fracture Prevention Measures (50.60, 50.61, Appendices G and H)
- ATWS Requirements (50.62)
- Loss of All AC Power (50.63)

Other regulations, although based on the full power operating plant, may continue to be partially applicable to the permanently defueled facility. Typically, the scope of these requirements can be reduced to eliminate those that do not pertain to the safe storage of the spent fuel or are no longer necessary to protect the health and safety of the public. The following regulations have been assessed to remain partially applicable for one or more configurations of the permanently shutdown plant:

- Fitness for Duty (Part 26, 55.63(j),(k))
- Technical Specifications (50.36, .36b)
- Fire Protection Program (50.48, Appendix R)
- Quality Assurance Program (50.54(a), Appendix B)
- Operator Staffing Requirements (50.54(m))
- Operator Requalification Program (50.54(i), 55.45, 55.59)
- Security Plan (50.49(p), 70.32, Part 73, Part 73 Appendices B and C)
- Inservice Inspection Requirements (50.55a(g))
- Maintenance Effectiveness* (50.65)

Several technical issues do not fit into these categories. They are discussed below.

We have recommended the continued application of the periodic FSAR update requirement (50.71(e)) to provide a basis for the 50.59 safety evaluations that will be performed when a plant ceases operation. The special nuclear material control requirements of Parts 70 and 74 should continue as long as fuel remains within the plant. The annual fees for the permanently shutdown plant licensees (171.15) should be adjusted to reflect the generic regulatory costs that are directly applicable to their facility type.

The emergency planning and preparedness requirements (50.47, 50.54(q), (t) and Appendix E) and the insurance issues (50.54(w) and Part 140) were evaluated using the accident consequence analyses of this

*Assumes a formal request for permanent cessation of operation after 7/10/96.

study. The estimated consequences for the Configuration 1 accident approximate those of a core damage accident.

It is recommended that all offsite and onsite emergency planning requirements remain in place, with the exception of the Emergency Response Data System requirements of Part 50, Appendix E, VI.

The offsite emergency planning and preparedness (EP) requirements are expected to be eliminated for Configuration 2, based on the results of the generic PWR calculation which estimated a 9 millirem dose at the exclusion area boundary (see Table 6.1).* Part 50 offsite EP requirements can also be eliminated for Configurations 3 (plant only) and 4 because the spent fuel has been transferred to an ISFSI (Part 72 requirements) or transported offsite. Without spent fuel, the plant is not a significant health risk.

It is recommended that the onsite property damage and the offsite liability insurance levels remain at operating reactor levels for the duration of Configuration 1. The consequence analyses of Section 4 support reduced insurance requirements for the remaining configurations.

*However, since plant specific parameters (such as exclusion areas) can vary we recommend that the licensee perform a plant specific evaluation for Configuration 2.

Table 6.1 Generic PWR Accident Summary[1]

Spent Fuel Storage Config.	Accident Timing (yrs after final SD)	Recommended Accident Case	Offsite Consequences						Dose at Exclusion Boundary (rem)	Onsite Cleanup Cost ($)
			Distance (miles)	Prompt Fatalities	Societal Dose (person-rem)	Latent Fatalities	Condemned Land (sq. miles)	Total Cost ($)		
1	~0	2L[2]	0-50 0-500	0.3 0.3	4.2E+7 7.0E+7	16,800 28,800	156 188	5.6E+10 5.9E+10	NC	NC
2	3.5 (PWR)	Low gap release	0-50 0-500	0 0	3000 4000	1 2	0 0	neg.	.009	3.2E+7
3	5	Single best assembly, estimate release	0-50 0-500	0 0	590 690	0.18 0.22	0.002 0.002	neg.	0.472	1.2E+7
4	5	BWST failure	-	-	-	-	-	-	-	1.1E+8

[1]The accident consequences associated with the generic PWR are more severe than the comparable BWR cases.

[2]Rapid zircaloy oxidation involving the last full core offload (and the last normal offload for PWRs) low release fractions assumed.

NC = not calculated; neg = negligible

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APPENDIX A PREVIOUS ANALYSES OF SPENT FUEL POOL ACCIDENTS

A.1 DISCUSSION

The Reactor Safety Study[1] considered accidents involving spent fuel. The inventory of material that was potentially at risk was limited to one third of a reactor core. This was consistent with the intention of the routine shipping of spent fuel for reprocessing (or disposal). The Reactor Safety Study concluded that the risk associated with spent fuel storage was extremely small in comparison to that associated with the operating reactor core.

During the Carter administration a federal moratorium halted the reprocessing of spent commercial reactor fuel. Given the absence of away-from-reactor storage facilities or a permanent disposal facility, utilities had no alternative but to store spent fuel at the reactor site. This led to increasingly larger inventories of fuel being stored in reactor spent fuel pools. Modified spent fuel storage racks have also been employed to further increase the ultimate capacities of most reactor spent fuel pools.

A.S. Benjamin and others[2-3] published investigations of the probable course of events following the complete draining of a spent fuel pool. A theoretical model and the computer codes SFUEL and SFUEL1W were developed and employed to analyze the thermal-hydraulic behavior of stored spent fuel assemblies on exposure to air. These studies indicated, that for certain combinations of storage configurations and decay times, freshly discharged fuel assemblies could self heat to a temperature where the air oxidation of the zircaloy fuel cladding would become self sustaining. The additional chemical heat released during clad oxidation, which is comparable to the decay heat, then causes a rapid temperature increase with the resultant failure of the cladding. Additionally, these studies further concluded that for certain conditions, the cladding of freshly discharged assemblies would attain a sufficiently high temperature to heat adjacently located assemblies, with lower decay heat, to the point of "ignition" (self sustaining clad oxidation). The possibility of propagation from assembly to assembly with the involvement of the entire spent fuel pool inventory was not ruled out in all cases.

V.L. Sailor, et al.,[4] reported a study of severe accidents in spent fuel pools. Their investigation provided an assessment of the potential risk from possible accidents in spent fuel pools. The authors describe their effort as a "simplified analysis which followed the logic of a typical probabilistic risk assessment (PRA)." To assess the risk Sailor, et al., quantified the frequencies of initiating events that could compromise the integrity of fuel pool, the probability of system failure conditional on the initiating event, fuel failure occurrence, the magnitudes of radionuclide releases to the environment and the consequences which result from those releases as well as the consequences associated with these releases.

In the Sailor study, two plants were primarily selected for examination on the basis of perceived vulnerability to seismic events. A preliminary screening study using RSS methodology indicated seismic initiated pool failure was the dominant risk contributor. The selected plants were the Millstone 1 (BWR) and the Ginna (PWR) plants. The operating histories of these plants were used to model, through application of the ORIGIN code, realistic radionuclide inventories present in their respective spent fuel pools, at the time the study was performed.

The accident initiators considered in Sailor's work were loss of pool heat removal capability, structural failure of the pool due to missiles, seismic events or the drop of a heavy load on the pool wall, and the draining of the pool due to pneumatic seal failure. The study concluded accidents which lead to the complete draining of the spent fuel pool caused by loss of cooling, missiles and pneumatic seal failure were very unlikely. However, failures resulting from seismic events and the drop of a heavy load were concluded to be credible, though the frequencies of these accidents was assessed to be quite uncertain. As part of Sailor's study, BNL performed a review of the SFUEL1W models and code. Limited verifications of the code's prediction with the results of small scale experiments performed at SNL were also made. Sailor, et al., concluded that the SFUEL1W code "provides a valuable tool for assessing the likelihood of self-sustaining clad oxidation for a variety of spent fuel configurations assuming the pool has been drained."

Although BNL made at least one modification to the SFUEL1W code, their predictions of critical decay times,* were in good agreement with the earlier published results of the SNL staff.

To estimate the release of radioactivity from the fuel pins, the authors employed the CORSOR code,[5] using the time-temperature histories obtained with the SFUEL1W code. These results are reproduced in Table A.1. The releases are expressed as sets of fractions, which are applied to the total inventory of material involved in the accident. The initial inventory of radionuclides available for release as noted above was calculated with the ORIGIN code using the operating histories of the selected plants. The calculated inventories were a realistic snapshot of the activity present in the spent fuel pools of the selected plants at the time Sailor's study was completed. These inventories are not presented here for several reasons. Both plants investigated were relatively small: 2011 Mw thermal in the case of the BWR and 1520 Mw thermal for the PWR. Continued operation at these plants has also increased their present spent fuel pool inventories. But more importantly, the last one third core discharge was for a normal refueling, and this would represent a significant underestimation of a full core off-load, which was evaluated in the present study.

Offsite accident consequences in NUREG/CR-4982 were calculated with the CRAC2 computer code.[6] Major assumptions used in the evaluation included: a generic site having uniform population density of 100 persons per square mile (approximately the national average); generalized average weather conditions; and the emergency response action being relocation 24 hours after release (criterion 25 rem whole body projected individual dose commitment). The consequences reported, societal dose and

*The cooling time required to lower the decay heat of freshly discharged fuel assemblies to a point where the self sustaining clad oxidation is unlikely to occur.

interdicted land, are presented in Table A.2. The risk estimates of Sailor's work have been superseded by more recent studies.[7-8] However, it should be noted that to evaluate the risk the authors ultimately selected the consequence results of an accident where the only the last refuelling discharge is involved. In this accident, fire does **not** propagate its way throughout the entire spent fuel pool, but the maximum release fractions were assumed (no credit taken for structures removing activity).

The 1989 report of J. Jo, et al.[7] described a value/impact assessment of various proposed options[2-4] intended to reduce the risk of potential accidents occurring in the commercial nuclear power plant spent fuel pool. As was the case with previous efforts, attention was limited to an operating plant. The risk dominant accidents, source terms and inventory assumptions were identical to those investigated by Sailor, et al. Major differences in the estimation of the offsite consequences existed between these two studies. Jo, et al., used the MELCOR Accident Consequence Code System (MACCS), Version 1.4.[9] This code, developed by Sandia National Laboratory for the NRC, has replaced the CRAC2 code for offsite consequence assessment. The MACCS code has been used exclusively in the preparation of NUREG-1150 and its supporting documentation.[10] Site assumptions which significantly affected the predicted consequences also differed. The Zion site was selected by Jo to represent the "worst" case conditions in regard to population density distributed about a plant site. The actual population distribution, weather conditions, land usage fraction and regional economic data associated with the Zion site were employed. These actual data, coupled with release assumptions of 100 percent pool involvement and the set of maximum fractional releases specified by Sailor, were used to evaluate a **worst** case. For a **best** estimate calculation of accident consequences, the study assumed: only the last refueling discharge is involved in the fire; Zion weather; average land usage and economic data for the state of Illinois; a 95 percent land fraction and a uniform population density of 340 persons per square mile out to 50 miles beyond the plant.* In both cases examined, no planned evacuation was modelled, since this was stated to have only a small effect on total costs and societal doses. However, people were relocated at one day based on projected 7 day dose commitment of 25 rem. (Prior to relocation people were assumed to be engaged in normal activity, which afforded them limited protection from the early dose pathways.) The long term dose limit of 25 rem effective dose equivalent (EDE) employed in this effort was consistent with WASH-1400. The results of these calculations are shown in Table A.3. The public dose and offsite property damage were reported out to 50 miles from the plant. The public doses reported by Jo, et al., are factors of 3.5 and 10 (best estimate and worst case, respectively) higher than those reported by Sailor, et al. The population density assumptions of the latter study (340 and 860 persons per square mile versus the 100 used in the Sailor study) account for 98 and 87 percent, respectively, of the observed increases. As such, and notwithstanding consequence codes differences in the release and health effects modeling, the societal dose results of Sailor and the more recent Jo effort appear to be fairly consistent.

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*The average population density for existing plants, circa 1980.[11]

Table A.1 Estimated Radionuclide Release Fraction During a Spent Fuel Pool Accident Resulting in Complete Destruction of Cladding (Cases 1 and 2)

Chemical Family	Element or Isotope	Release Fraction*	
		Value Used	Uncertainty Range
Noble gases	Kr, Xe	1.00	0
Halogens	I-129, I-131	1.00	0.5-1.0
Alkali Metals	Cs, (Ba-137m) Rb	1.00	0.1-1.0
Chalcogens	Te, (I-132)	0.02	0.002-.02
Alkali Earths	Sr, (Y-90), Ba (in fuel) Sr, Y-91 (in clad)	$2x10^{-3}$ 1.00	10^{-4}-10^{-2} 0.5-1.0
Transition Elements	Co-58 (assembly hardware) Co-60 (assembly hardware)** Y-91 (assembly hardware) Nb-95, Zr-95 (in fuel) Nb-95, Zr-95 (in clad)	0.10 0.12 0.10 0.01 1.00	0.1-1.0 0.1-1.0 0.1-1.0 10^{-3}-10^{-1} 0.5-1.0
Miscellaneous	Mo-99 Ru-106 Sb-125	$1x10^{-6}$ $2x10^{-5}$ 1.00	10^{-8}-10^{-5} 10^{-6}-10^{-4} 0.5-1.0
Lanthanides	La, Ce, Pr, Nd, Sm, Eu	$1x10^{-6}$	10^{-8}-10^{-5}
Transuranics	Np, Pu, Am, Cm	$1x10^{-6}$	10^{-8}-10^{-5}

*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.

**Release fraction adjusted to account for a 100% release of the small amount of Co-60 contained in the zircaloy cladding.

(Reproduced from NUREG/CR-4982)

Table A.2 CRAC2 Results for Various Releases Corresponding to Postulated Spent Fuel Pool Accidents with Total Loss of Pool Water

Case Description		Whole Body Dose (Man-rem)	Interdiction Area (sq. miles)
1A.	Total inventory 30 days after discharge 50 mile radial zone	$2.6x10^6$	224
1B.	Total inventory 90 days after discharge 50 mile radial zone	$2.6x10^6$	215
1C.*	Total inventory 30 days after discharge 500 mile radial zone	$7.1x10^7$	224
2A.	Last fuel discharge 90 days after discharge 50 mile radial zone (maximum release frac tion)	$2.3x10^6$	44
2B.	Last fuel discharge 90 days after discharge 50 mile radial zone (minimum release fraction)	$1.1x10^6$	4
2C.	50% of all fuel rods leak 1 year after discharge 50 mile radial zone	4.0	0.0

*Note that the consequence calculations in NUREG-1150 are based on a 50 mile radial zone. Case 1C is given as a sensitivity result.

(Reproduced from NUREG/CR-4982)

Table A.3 Offsite Consequence Calculations

Case	Characterization	Source Term*	Population	Public Health Dose (person-rem)	Offsite Property Damage ($1983)
1	Average case	Last fuel discharge 90 days after discharge	340 persons/mile^2	7.97×10^6	3.41×10^9
2	Worst case	Entire pool inventory 30 days after discharge	Zion population (roughly 860 persons/mile^2	2.56×10^7	2.62×10^{10}

(Reproduced from NUREG/CR-5281)

Table A.4 Onsite Property Damage Costs Per Accident ($)

Item	Best Estimate	Worst Case
Cleanup and Decontamination	1.65E8	1.65E8
Repair	7.2E7	7.2E7
Replacement power	8.67E8	1.66E9
Total number of operating years remaining	29.8 years	29.8 years
Number of years plant is out of service	5 years	7 years
Expected Dollar loss	8.24E9	1.29E10

(Reproduced from NUREG/CR-5281)

Occupational exposure for a major spent fuel pool accident was assumed in the Jo report to be similar to the estimated occupational exposure, of 4850 man-rem,[11] incurred during the recovery of the Three Mile Island plant. The Jo report stated that "This exposure is small compared to the potential off-site dose impact and more refined quantification appears to be unwarranted."

Onsite property damages were also estimated in the Jo study. The cost of a major spent fuel pool accident was expected to be similar to the cost associated with a Category II severe accident as defined in Reference 13. The estimates provided in the Jo report are reproduced in Table A.4.

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APPENDIX B DETAILED REGULATORY ASSESSMENT

B.1 INTRODUCTION

This section provides a detailed assessment of each of the regulations (or technical issues) that may not be fully applicable to permanently shutdown nuclear power plants. This list of candidate regulations was identified from a screening of 10CFR Parts 0-199[1] and is presented in Table B.1. Each of these technical issues was subjected to a detailed review which included federal register notices, SECY memos, NRC policy statements, regulatory guides, standard review plans, NUREG reports, and NUREG/CR reports to develop an understanding of the regulatory bases. The continued applicability of each technical issue was assessed within the context of each spent fuel storage configuration,* the associated safety hazard analysis results, as well as the expected plant status.

With the possible exception of Part 171, "Annual Fees for Licenses," each regulation is ultimately focussed on the protection of public health and safety. However, a particular regulation may not be applicable to a permanently shutdown plant in general, or to a specific spent fuel storage configuration. For example, an exemption from the containment leakage testing requirements of 10CFR50.54(o) for a permanently defueled plant will not impact public health and safety as the plant risk is primarily associated with the spent fuel that is now stored in the spent fuel pool outside the primary containment.

The remainder of this appendix examines each of the candidate regulations of Table B.1. A short discussion of the regulatory background and objective is provided. Our assessment of the continued applicability to each spent fuel storage configuration is stated with additional supporting information, as necessary.

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The spent fuel retrievability requirements for ISFSIs may perturbate the regulatory assessment presented in this appendix. An ISFSI storage method (i.e., NUHOMS or storage only casks) that is presently not licensed for offsite transportation under 10CFR Part 71, may require an operating onsite spent fuel pool to comply with the retrievability requirement of 72.122(l). The BNL recommendations assume: dual purpose cases are used; a NUHOMS transport cask will be licensed; storage only casks (with modifications) can be licensed for transport; or that fuel transfer methods will be licensed that do not require an operating onsite spent fuel pool.

Table B.1 Assessment of Continued Regulatory Applicability for Permanently Shutdown Nuclear Power Reactors (Summary)

Technical Issue	10CFR Reference[1]	Regulatory Applicability[2-5] Configuration 1	2	3[6-9]	4
Fitness for Duty	Part 26 55.53(j), (k) 72.194	P F	N N	N N F	N N
Technical Specifications	50.36, .36b 72.26, 72.44	P	P	P F	P
Combustible Gas Control	50.44	N	N	N	N
ECCS Acceptance Criteria	50.46	N	N	N	N
Emergency Planning	50.47, .54(q),(t) App. E 72.32	F	P	P F	P
Fire Protection	50.48, App. R 72.122	P	P	P F	P
Environmental Qualification	50.49	N	N	N	N
QA Program	50.54(a), App. B Part 72, Subpart G	P	P	P F	P
Operator Requalification Program	50.54(i), 55.45, 55.59 72.44(b) Part 72, Subpart I	P	P	N F	N
Operator Staffing Requirements	50.54(k). 50.54(m)	N F	N P	N N	N N
Containment Leakage Testing	50.54(o), App. J	N	N	N	N
Security Plan	50.54(p), 70.32, Part 73 Part 73 App. B and C 72.44(e) Part 72 Subpart H Part 73	P	P	N F	N
Onsite Property Damage Insurance	50.54(w) Part 72	F	P	P *	P

*See discussion in the text.

Technical Issue	10CFR Reference[1]	Regulatory Applicability[2-5] Configuration 1	2	3[6-9]	4
Inservice Inspection Requirements	50.55a(g)	P	P	N	N
Fracture Prevention Measures	50.60, .61, Apps. G and H	N	N	N	N
ATWS Requirements	50.62	N	N	N	N
Loss of all AC Power	50.63 72.122(k)	N	N	N F	N
Maintenance Effectiveness	50.65 POL before 7/10/96 POL after 7/10/96	N P	N P	N N	N N
Periodic FSAR Update Requirement	50.71(e) 72.70	P	P	P F	P
Training and Qualification of Nuclear Power Plant Personnel	50.120	P	P	N	N
Material Control/Accounting of Special Nuclear Material (including US/IAEA Agreement)	70.51, .53, 74.13(a), Part 75 72.72, .76	F	F	N F	N
Financial Protection Requirements	Part 140 Part 72	F	P	P *	P
Annual Fees for Licenses	171.15 171.16	P	P	P F	P

NOTES TO TABLE B.1

1. 10CFR Parts 0 to 199, revised January 1, 1995.

2. All other regulatory requirements applicable to nuclear power reactors and not listed in this table are assumed to remain in effect, unless addressed by a plant-specific exemption.

3. The spent fuel storage configurations are defined in Sections 2 and 3 of this report. Briefly:
   Configuration 1 - hot fuel in the spent fuel pool
   Configuration 2 - cold fuel in the spent fuel pool
   Configuration 3 - all fuel stored in an ISFSI
   Configuration 4 - all fuel shipped offsite

4. Configuration 1 also assumes the licensee has permanently caused operation and that a confirmatory letter has been issued to prevent refueling the vessel without NRC authorization.

NOTES TO TABLE B.1 (Cont'd)

5. F - Regulation continues to be fully applicable for this spent fuel storage configuration.
   P - Regulation is assessed to be partially applicable for this configuration.
   N - Regulation is not considered applicable to this configuration.

6. A permanently shutdown nuclear power plant may store its fuel in an Independent Spent Fuel Storage Installation, before, during, and after the plant itself has been decommissioned. As such, Configuration 3 must examine the regulatory requirements for the plant without fuel (similar to Configuration 4) and the ISFSI. This necessitates two entries in Table B.1 for Configuration 3. The first (and second, if applicable) pertains to the plant itself prior to the completion of decommissioning. The last entry examines the Part 72 requirements for the ISFSI.

7. The requirements of Configuration 3 remain applicable until all fuel has been removed from the ISFSI and shipped offsite.

8. In addition to the applicable provisions of Part 72 as noted for Configuration 3, Parts 20, 21, 71, and 73 remain applicable to the transportation of spent fuel from the ISFSI to a HLW repository or MRS.

9. This regulatory assessment assumes an onsite, operating spent fuel pool is not necessary to satisfy the fuel retrievability requirement of 72.122(l). See the introductory section of Appendix B for further information.

B.2 REGULATORY ASSESSMENT

Fitness for Duty Program

Background

The Fitness for Duty Program is contained in Part 26 of Title 10, Code of Federal Regulations. Another reference to the Fitness for Duty can be found in the Operators Licenses Section (10CFR55.53(j),(k)). The licensing requirements for the independent storage of spent nuclear fuel and high level radioactive waste (10CFR72.194) do not require a formal fitness for duty program.

The Fitness for Duty Final Rule was published in the June 7, 1989 Federal Register (54 FR 24468) The Supplementary Information, published with the rule, provided the general background, the need for a rule and a summary of comments on the proposed rule with NRC responses.

The NRC stated that the objective of the rulemaking was to provide reasonable assurance that nuclear power plant personnel were not mentally or physically impaired from any cause which could adversely affect their ability to safely and competently perform their duties. The rulemaking action was taken to significantly increase the assurance of public health and safety. All workers with unescorted assess to the nuclear power reactor protected area, as well as personnel who are physically required to report to the TSC or the EOF under emergency conditions, fall within the scope of this rule.

The associated backfit analysis found that the rule will prove a substantial increase in the overall protection of public health and safety and that the direct and indirect costs of implementation are justified in view of the increased protection. In response to comments on the proposed rule, the NRC reiterated that the Fitness for Duty Rule was limited to nuclear power reactors and they saw no reason to extend the coverage of the rule to other facility types such as non-power test reactors, materials facilities, and special nuclear materials licensees. By extension, one can surmise that the lesser public risk associated with non-power reactors, materials licensees, and independent spent fuel storage installations (ISFSIs) did not warrant the implementation of a fitness for duty program at those facilities.

Assessment

Configuration 1, "Hot Fuel in the Spent Fuel Pool" postulated rapid zircaloy oxidation of the spent fuel rods after the loss of pool water inventory. The safety hazard analyses of Section 4 has estimated consequences that are approximately equal to a severe core damage accident. Given the potential magnitude of the consequences, it is appropriate that a formal fitness for duty program, in accordance with the requirements of 10CFR Part 26, remain in place. In recognition of the defueled status of the permanently shutdown plant, and the lack of significant non-fuel sources of public risk,[2,3] It is recommended to reduce the scope of the program to those personnel with unescorted assess to any area that contains equipment necessary to support and maintain continued safe storage or handling of spent fuel. As shown in Table B.1, the Part 26 requirements should remain fully applicable for licensed operators (10CFR55.53(j),(k)).

Configuration 2, "Cold Fuel in the Spent Fuel Pool," has sufficiently low decay heat loads such that the cladding will remain intact even if all spent fuel pool water is lost. Configuration 2 considers the consequences of a dropped fuel assembly. The safety hazard analysis, as discussed in Section 4, shows minimal offsite consequences. On this basis, it appears that the Part 26 requirements for Configuration 2 can be deleted without a significant impact on the public health and safety.

In lieu of long-term storage in the spent fuel pool, a permanently shutdown nuclear power plant may store its spent fuel in an Independent Spent Fuel Storage Installation (ISFSI), before, during, and after the plant itself has been decommissioned. As such, Configuration 3 must examine the regulatory requirements for the plant without fuel (similar to Configuration 4) and the ISFSI. Although the postulated accident for Configuration 3 does result in offsite consequences, the results are not dependent on human intervention. Other postulated ISFSI accidents found in the literature[4,5] do not result in significant offsite consequences.

As discussed below, decommissioning accidents, not involving spent fuel, do not have offsite consequences. Therefore, a Part 26 program for Configuration 3 would not significantly impact the health and safety of the public. The requirements of 10CFR72.194 regarding the physical condition of certified ISFSI operation personnel govern.

Configuration 4, "All Fuel Removed from the Site," assumes that all spent fuel has been shipped offsite, including any that might have been stored in an ISFSI. As discussed in Section 4, the postulated accidental radioactive releases to the atmosphere during decommissioning do not pose a significant threat to the onsite workers and/or the public.

Based on the limited consequences associated with Configuration 4, a Part 26 Fitness for Duty program would not have a significant effect on public health and safety.

Although the Fitness for Duty Program requirements may no longer be appropriate for certain spent fuel storage configurations, the record keeping requirements of section 26.71 are still applicable.

Technical Specifications

Background

Section 50.50 of 10 CFR, "Issuance of Licenses and Construction Permits" provides that each operating license for a nuclear power plant issued by the NRC will contain such conditions and limitations that the Commission deems appropriate and necessary. Operating technical specifications, imposed by Section 50.36 in the interest of the health and safety of the public, are included as Appendix A of the operating license.

Under 10CFR50.36b non radiological environmental technical specifications to protect and monitor the plant's impact on the environment can be included as Appendix B to the license.

Each applicant for an operating license proposes technical specifications for its plant which are then reviewed by the NRC and modified, as necessary,. This process results in a set of plant-specific technical specifications that reflect plant-specific design and siting characteristics. Additional changes, in the form of license amendments, may be granted by the NRC over the operating life of the plant, as appropriate.

Assessment

Very few plants have a defueled mode in their technical specifications. After a permanent cessation of operations issued, the existing technical specifications can be modified to include a permanently defueled mode to reflect the more limited range of postulated accident and radiological consequences associated with a permanently shutdown nuclear power plant. The defueled mode will represent a significant scope reduction in comparison to the operating plant technical specifications requirements. For example, shutdown margin calculations, (normally required for all tech spec modes) and cooling tower drift or noise monitoring programs would no longer be necessary from a health and safety or an environmental impact perspective.

Since the technical specifications can be very plant specific, it is recommended that the licensee submit an amendment request to reduce the scope of the operating technical specifications and the environmental technical specifications* (or institute a permanently defueled mode) after permanent cessation of operations. Subsequent amendments to the plant technical specifications may be appropriate as the spent fuel decay heat declines (Configuration 2) or if all fuel is moved to an ISFSI** or removed from the site (Configurations 3 and 4, respectively).

Combustible Gas Control

Background

The combustible gas control requirements are found in 10CFR50.44. These requirements were instituted to "improve hydrogen management in LWR facilities and to provide specific design and other requirements to mitigate the consequences of accidents resulting in a degraded reactor core" [46 FR 58484, 12/2/81].

Assessment

The requirements focus on the capability for: measuring hydrogen concentrations, ensuring a mixed atmosphere and controlling combustible gas mixtures, post LOCA. The concern is that hydrogen generation due to metal water reaction or the radiolytic decomposition of water during a LOCA could result in a detonation or deflagration that could fail primary containment.

Obviously, the post LOCA control of combustible gases inside containment is an operating plant issue. The permanently shutdown plant stores all of its fuel outside containment; the reactor pressure vessel and the primary containment are no longer necessary fission product barriers. Therefore, it is recommended that the requirements of 10CFR50.44 be removed for all four spent fuel configurations for the permanently shutdown nuclear power plants.

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*The technical specifications on effluents for nuclear power reactors (50.36a and Appendix A) continue to remain fully applicable to permanently shutdown plants.

**ISFSIs have their own technical specification requirements under 72.26 and 72.44.

ECCS Acceptance Criteria

Background

The acceptance criteria for emergency core cooling systems (ECCS) for light water reactors is found in 10CFR50.46. This section requires that the ECCS be designed to limit post LOCA peak cladding temperature, clad oxidation and hydrogen generation to specified values and provide for long-term cooling. Acceptable ECCS evaluation models must address the sources of heat during a postulated LOCA, clad swelling or rupture, blowdown phenomena, etc. Although this section is primarily addressed during the design phase, operating license holders are required to estimate the effect of a change or an error in the ECCS evaluation model or the model application. Section 50.46(a)(3) specifies the reporting and reanalysis requirements, which are dependent on the magnitude of the error or change.

Assessment

The purpose of these requirements is to ensure that the ECCS design can, and continues to be able to, mitigate the design basis LOCA throughout the operating life of the plant. Without fuel in the vessel, a permanently shutdown plant could make changes to its ECCS systems without a significant public health and safety impact, yet an ECCS re-evaluation could be required. Therefore, the ECCS acceptance requirements of 10CFR50.44 may be deleted for all spent fuel storage configurations of the permanently shutdown plant.

Emergency Planning

Background

The emergency preparedness requirements for nuclear power reactors are contained under 10CFR50.54, "Conditions of Licenses." Paragraph (q) requires that a licensee, authorized to possess and operate a nuclear power reactor, follow and maintain in effect emergency plans which meet the standards of Section 50.47(b) and Appendix E to Part 50. Paragraph (t) of 50.54 emphasizes the revision and maintenance of the emergency preparedness program and requires an annual independent review. Section 50.47(b) presents sixteen requirements for offsite and onsite emergency response. Appendix E to Part 50 generally augments the requirements of 50.47(b).

Due to the lower inherent risk to the public, other facilities licensed by the NRC typically have less stringent emergency preparedness (EP) requirements than nuclear power reactors. For example, research reactors and special nuclear materials licensees are also subject to the requirements of Appendix E to Part 50. However, the size of the emergency planning zone for these facilities and the degree of compliance to the requirements of Appendix E are determined on a case by case basis. Materials license applicants, under 10CFR30.32(i) with quantities of radioactive material in excess of Appendix C to Part 30 must furnish either:

- An evaluation showing that the maximum dose to a person offsite due to a radioactive release would not exceed one rem effective dose equivalent or five rems to the thyroid.

- An emergency plan for responding to the release of radioactivity.

Assessment

The estimated offsite consequences of a rapid zircaloy oxidation event in the spent fuel pool dictate the continuance of all nuclear power reactor emergency preparedness regulatory requirements* for Configuration 1, "Hot Fuel in the Spent Fuel Pool."

Section 4 of this report developed consequence estimates based on generic BWR and PWR plant parameters, source term assumptions and recommended accident cases. The recommended accident case for Configuration 2 had an estimated dose at the exclusion area boundary (0.4 miles) of 9 millirem for the generic PWR. This dose is well below the EPA Protective Action Guide (PAG) whole body dose of 1 rem at the exclusion area boundary. Since this dose estimate is based on generic plant assumptions (such as the exclusion area boundary, it is recommended that the permanently shutdown plant perform a plant specific evaluation for Configuration 2 and specify sufficiently sized emergency planning zone (EPZ) so that the EPA PAGs are not exceeded at the EPZ boundary. Based on our generic calculations for Configuration 2 Section 4.2, BNL believes a permanently shutdown plant EPZ can be reduced so that it resides entirely within the former full power exclusion zone, i.e., within the site boundary.

Section 4 has also stated that decommissioning accidents that do not involve spent fuel do not pose a significant health risk to the public. Therefore, offsite emergency planning is not required for Configurations 3 (plant only) and 4.

It is recommended that the permanently shutdown licensee apply for exemptions from the following *offsite* emergency planning requirements for Configurations 2,3, (plant only) and 4:

- The early public notification requirements of 50.47(b)(5) and Appendix E.IV.D.3.
- The periodic dissemination of emergency planning information to the public (50.47(b)(7) and Appendix E.IV.E.8).
- Offsite emergency facilities and equipment such as the EOF, and the emergency news center (50.47(b)(8), Appendix E.IV.E.8).
- Offsite radiological assessment and monitoring capability, including field teams (50.47(b)(9)).
- Periodic offsite drills and exercises (50.47(b)(14), Appendix E.IV.F.3).
- Licensee headquarters support personnel training (50.47(b)(15), Appendix E.IV.F.b.h).

*except the Emergency Response Data System Requirements of Part 50, Appendix E, VI.

The NRC has recently issued a final rule [60 FR 32430, 6/22/95]. The emergency planning requirements for a typical, storage only ISFSI are provided in paragraphs 72.32 (a), (c) and (d).

Onsite emergency planning requirements should remain applicable for all spent fuel storage configurations.

Fire Protection

Background

Section 50.48 of 10CFR states, "each operating nuclear power plant must have a fire protection plan that satisfies Criteria 3 of Appendix A of this part." Criterion 3 states that fire detection and fighting systems of appropriate capacity and capability are required to minimize the effects of fires on structures, systems, and components important to safety. Section 50.48 further states that basic fire protection guidance provided in two documents: Branch Technical Position APCSB 9.5-1 and its Appendix A. The appropriate document is dependent on the plant's status as of July 1, 1976. The Branch Technical Position (BTP) APCSB 9.5-1 is applicable to new plants docketed after that date, while Appendix A to the BTP addresses older plants that were operating or under design or construction prior to 7/1/76.

Assessment

Although the emphasis of both these documents is the preservation of the safe shutdown capability during and after a fire, the guidance recognizes other sources of risk that are not related to reactor shutdown or in vessel decay heat removal. Appendix A to BTP APCSB 9.5-1 requires:

- The fire protection program for new fuel storage areas (and adjacent fire zones that could affect the fuel storage zone) be fully operational before fuel is received at the site.
- Fire protection and automatic detection for the spent fuel pool area.
- Radwaste building detection and protection.
- Materials that contain radioactivity must be stored in closed metal tanks or containers, away from ignition sources of combustibles.

Each licensee has a fire protection program that, in addition to safe shutdown requirements, has fire brigade training requirements, administrative procedures and controls, and detection and suppression requirements for plant areas that contain radioactive inventories with potential offsite consequences. For Configurations 1, 2, 3, (plant only) and 4, we recommend eliminating those requirements directly related to safe shutdown capability. Additional reductions in the scope of the 50.48 fire protection program can be examined on a plant-specific basis.

ISFSIs, under spent fuel storage Configuration 3, are subject to the fire protection requirements of Section 72.122.

Environmental Qualification

Background

The Environmental Qualification (EQ) of Electric Equipment Important to Safety for Nuclear Power Plants (10CFR50.49) was published as a final rule in the January 21, 1983 Federal Register (48FR2729). The supplementary information provided with the rule states:

> *The scope of the final rule covers that portion of equipment important to safety commonly referred to as "safety related".... Safety-related structures, systems, and components are those that are relied upon to remain functional during and following design basis events to ensure (i) the integrity of the reactor coolant pressure boundary, (ii) the capability to shut down the reactor and maintain it in a safe shutdown condition, and (iii) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines of 10CFR Part 100. Design basis events are defined as conditions of normal operation, including anticipated operational occurrences; design basis accidents; external events; and natural phenomena for which the plant must be designed to ensure functions (i) through (iii) above.*

Assessment

The EQ rule is clearly limited to electrical equipment that must function during design basis events. In response to comments on the final rule, the Commission stated that the EQ rule does not cover the electrical equipment located in a mild environment. With the permanent cessation of operations, the design basis accidents of the FSAR are limited to Section 15.7, Radioactive Release from a Subsystem or Component. The harsh environment associated with loss of coolant accidents is no longer applicable. Therefore, 10CFR50.49 can be deleted for the permanently shutdown plant.

Quality Assurance (QA) Program

Background

The plant-specific QA program that implements the Part 50 Appendix B QA requirements is described or referenced in the Safety Analysis Report per 10CFR50.34(b)(6)(ii). Under paragraph (a) of "Condition of Licenses (50.54)," the licensee is required to implement the QA program described (or referenced) in the SAR. Furthermore, paragraph (a)(3) requires NRC submittal and approval of any proposed changes that reduce the commitments in the previously accepted QA program.

Assessment

The permanently defueled plant can make selected changes to its operating based QA program without impacting public health and safety. As previously discussed in the technical specification section, each plant should evaluate the scope of their QA program and submit the revisions that are appropriate to their facility and mode of spent fuel storage for NRC approval. Perhaps R.G. 1.33 can be revised (or another RG issued) to address the QA program for the PSD plants.

Operator Requalification Program

Background

Section 54(i) of 10CFR Part 50 requires an operator requalification program that meets the requirements of 10CFR55.59(c). The licensee may not decrease the scope of the program, except as authorized by the Commission.

Assessment

Part 55 states the requirements for granting and maintaining operator's licenses and is oriented toward operating nuclear power reactors. As a consequence, portions of this section are not applicable to a permanently defueled facility. The following sections should be revised to eliminate those regulatory requirements that solely pertain to operating nuclear power reactors:

55.41, 55.43, 55.45(a), 55.59(c) - Written examinations, operating tests, and requalification program requirements should reflect the permanently defueled plant configuration and the accidents that are applicable to the permanently shutdown facility.

55.45(b) - The operating tests for a permanently defueled plant should be administered in a plant walk-through. Simulation facilities are designed for operating power reactors, have limited usefulness for the defueled configuration, and should not be required for the administration of operating tests. In addition, Section 55.53(k) should be revised to reflect any modifications to the fitness for duty program that may be adopted for the permanently shutdown nuclear power reactor.

When all fuel is removed from the plant, either to an ISFSI (Configuration 3) or offsite (Configuration 4) there is no longer any need for operators licensed under Part 55, and the requalification program can be terminated.*

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*As discussed in Section D.1, this regulatory assessment assumes an operating onsite spent fuel pool is not necessary for fuel retrievability. Therefore, licensed fuel handlers are not necessary for Configuration 3.

Operator Staffing

Background

The licensed operator staffing requirements for nuclear power reactors are delineated in Sections 50.54(k) and (m).

Paragraph (k) requires a licensed operator to be present at the controls at all times during the operation of the facility. A nuclear power unit is considered to be operating when it is in a mode other than cold shutdown or refueling. By extension, the permanently defueled condition does not require a licensed operator to be continuously present at the controls.

Paragraphs (m)(2)(i) presents onsite licensed operator staffing requirements for nuclear power reactors. The requirements are based on the number of units operating (i.e., not in cold shutdown or refueling) at a site and the number of control rooms. However, onsite staffing is required for non-operating units.

Assessment

The onsite staffing requirements of Section 50.54(m) (2)(i) should remain in effect for Configuration 1. Our concern is the continued ability to: recover from off-normal events (such as the loss of fuel pool cooling) and activate the unit(s) emergency plan. The lower decay heat of the fuel assemblies in Configuration 2 subject to the same concern as Configuration 1. There is a long time for recovery from most off normal events.* Therefore, it is not necessary to require continuous operator staffing onsite unless spent fuel or other objects are being moved within or above the spent fuel pool, or other work is in process that poses a potential near term challenge to fuel cladding integrity. Since Configurations 3 and 4 do not require licensed operators, other personnel would have to be charged with the emergency plan responsibilities.

Containment Leakage Testing

Background

Conditions of Licenses, 10CFR50.54, Paragraph (o) states that primary reactor containments for water cooled power reactors are subject to the requirements of Part 50, Appendix J. This appendix requires periodic testing to verify the leaktight integrity of the primary containment and those systems and components which penetrate the containment.

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*The representative accident sequence, a fuel assembly drop assumes an operator is present.

Assessment

The primary containment of an operating plant is one of several fission product barriers designed to protect the public's health and safety in the event of an accident. In contrast to an operating plant, a permanently defueled facility stores all of its fuel outside containment. The defueled containment is not a source of public risk; previous decommissioning studies[2-3] have determined that there are not significant offsite consequences associated with accidents that do not involve spent fuel. Therefore, the continued maintenance of containment leakage integrity does not enhance public health and safety and it is recommended that these testing requirements be eliminated for the permanently shutdown plant.

Security Plan

Background

As part of the "content of applications" of Section 50.34, applicants for a Part 50 license are required to submit a physical security plan and a safeguard contingency plan. The physical security plan addresses vital equipment, vital areas, and isolation zones and also demonstrates the applicant's compliance with the requirements of Part 73.

The safeguards contingency plan includes plans for dealing with threats, thefts, and radiological sabotage of special nuclear material in accordance with the criteria of Part 73, Appendix C, Section 50.54(p) "Conditions of Licenses", requires prior Commission approval of any changes that would decrease the effectiveness of the security plan,* the guard training and qualification plan, and the submitted portion of the safeguards contingency plan Part 73 and the associated Appendices B and C provide physical protection requirements, access authorization requirements, general criteria for security personnel and safeguards contingency plan criteria for Part 50 licensees.

Independent Spent Fuel Storage Installations also have similar requirements for the ISFSI physical security, guard training and safeguards contingency plans under Section 72.44(e), Part 72 Subpart H, Part 73, and Part 73 Appendix C.

Assessment

The intent of the physical security, guard qualification and training, and the safeguards contingency plan is to protect the facility against radiological sabotage and to prevent the theft of special nuclear material. In comparison to operating units, permanently shutdown plants have a limited number of vital areas that are necessary for the protection of those systems required to support spent fuel cooling and storage.

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*Changes that do not decrease the safeguards effectiveness of the aforementioned plans may be made without prior Commission approval.

For permanently shutdown nuclear power plants with fuel storage in the spent fuel pool (Configurations 1 and 2), the use of license amendment requests is recommended to reduce the scope of the security plan with regard to the number and extent of vital areas and equipment.* When the fuel is moved to an ISFSI or offsite (Configurations 3 and 4, respectively) there is no longer any need for the physical security, safeguards contingency or guard qualification and training plans for the permanently shutdown facility.** Please note that the ISFSI has physical security requirements under Part 72 Section 72.44(e), and Subpart H which are independent of the plant status. Under Configuration 4, all spent fuel will be shipped offsite and will become the responsibility of the DOE.

Onsite Property Damage Insurance

Background

The onsite property damage requirements for nuclear power plants are found in 10CFR 50.54(w). Each licensee is required to have a minimum coverage limit of $1.06 billion or whatever amount is generally available from private sources, whichever is less. This insurance must be dedicated to the expenses associated with returning and maintaining the reactor in a safe and stable condition in the event of an accident and, removing or controlling onsite radioactive contamination such that personnel exposure limits are consistent with the occupational exposure limits of 10CFR Part 20. In the event of an accident with estimated cleanup costs above a threshold of $100 million, paragraph 50.54(w)(4) provides for an automatic prioritization of stabilization activities.

The onsite property damage insurance requirement was instituted in March, 1982 (47FR 13750) and became effective on June 29, 1982. This regulation has been amended several times over the years. During the amendment processes, the Commission provided its views in several areas that are germane to the permanently shutdown plant. These are:

- the purpose of the regulation,
- the required amount of insurance and the updating mechanism, and
- the $100 million threshold for automatically determining stabilization priorities.

Each of these areas is discussed below. The regulatory intent is illustrated with cites from the appropriate Federal Register Notices. The Commission's philosophy is then summarized and applied to the PSD plant.

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*This reduction in the scope of the program could also conceivably reduce the size of the security force and procedures.

**References 2 and 3 and the consequence analysis for Configuration 4 (Section 4.4 of this report) indicate that once all fuel is removed the predicted offsite releases of accidents that could occur during the decommissioning process are much less than the 10CFR Part 100 limits.

The Purpose of the Regulation

The onsite property damage insurance requirement of 10CFR 50.54(w) was adopted as a final rule in 1982 (47FR 13750, March 31, 1982). As part of this Federal Register Notice, the public comments on the proposed rule were discussed. Several commenters suggested that the rule apply only to insurance covering decontamination of a facility suffering an accident and not to "all risk" property damage insurance. The Commission agreed, stating:

> *"Because decontamination insurance is the Commission's only concern from the point of view of protecting public health and safety, coverage to replace the existing facility on an "all risk" basis is beyond the scope of the Commission's authority."*

This position has been reaffirmed in two subsequent amendments to the regulation (52FR 28963 8/5/87, 55FR 12163 4/20/90. The 1987 amendment also introduced a decontamination priority which established a priority for stabilizing the reactor after an accident to prevent any significant risk to the public health and safety.

The Required Amount of Property Damage Insurance and the Updating Mechanism

When the onsite property insurance requirement, 10CFR 50.54(w), was originally instituted (47FR 13750, 3/31/82), the Commission required licensees to "take reasonable steps to obtain onsite property damage insurance available at reasonable costs and on reasonable terms from private sources".* The minimum coverage limit was specified as both:

1. the maximum amount of property insurance offered as primary coverage by either American Nuclear Insurers/Mutual Atomic Energy Reinsurance Pool (ANI/MAERP) or Nuclear Mutual Limited (NML) - $500 million, and

2. any excess coverage in amount no less than that offered by either ANI/MAERP - $85 million or Nuclear Electric Insurance Limited (NEIL) - $435 million.

Thus, the minimum required was originally $500 million primary coverage and $85 million excess coverage. By buying both excess layers, many licensees purchased a total of $1.02 billion in onsite property damage insurance (49FR 44646, 11/8/84). The Commission did not quantify a required insurance value at that time. The minimum requirement was viewed as a reasonable amount of insurance, pending the completion of a study evaluating the cleanup costs of accidents of varying severity. That study was issued as NUREG/CR-2601, "Technology Safety and Costs of Decommissioning Reference Light Water Reactors Following Postulated Accidents".[9]

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*Or to demonstrate an equivalent amount of protection

NUREG/CR-2601 evaluated cleanup costs following three full power accidents of varying severity at two reference light water reactors. The scenario 1 accident is postulated to result in 10% fuel cladding failure, no fuel melting, moderate contamination of the containment structure, but no significant physical damage to buildings and equipment. The scenario 2 accident is postulated to result in 50% fuel cladding failure, a small amount of fuel melting, extensive radioactive contamination of supporting buildings, and minor physical damage to buildings and equipment. The scenario 3 accident is postulated to result in 100% fuel cladding failure, significant fuel melting and core damage, severe radioactive contamination of the containment structure, moderate radioactive contamination of supporting buildings, and major physical damage to structures and equipment. A TMI-2 type accident was assumed in the study to be of intermediate severity (scenario 2).

The cleanup costs established in the report ranged from $105.2 million to $404.5 million for the reference PWR and from $128.5 million to $420.9 million for reference BWR. Although these costs are considerably lower than the roughly $1 billion estimated to be required to cleanup TMI-2, the NRC noted (52FR28963 8/5/87) that the estimates do not include several TMI cost components such as, inflation during the cleanup, additional decontamination of the containment building, and the cost of facility stabilization. These additional cost considerations cause the NUREG/CR-2601 cost estimates to increase to $1.06 billion for the most severe accidents studied and somewhat less for a TMI-2 type accident.

One conclusion the NRC drew from this study was that the minimum insurance requirement of $585 million would be insufficient for some accidents. Accordingly, the NRC amended 10CFR 50.54(w) (52FR 28963, 8/5/87) to require power reactor licensees to maintain at least $1.06 billion of onsite property damage insurance. The NRC noted that previous exemptions from the full amount required by 10CFR 50.54(w) were still valid. These exemptions were granted to four licensees of small reactors based on plant specific analyses of accident costs. The NRC stated:

> *"Increasing the required amount of insurance based on general technical studies in no way negates the continued validity of the specific studies upon which the existing exemptions were based."*

The August 5, 1987 Federal Register Notice also presents a summary of comments on the method of future adjustment of the insurance requirement. The NRC agreed with many commenters that an adjustment formula tied to a measure of inflation (e.g., the Consumer Price Index or the Handy-Whitman Construction Index) would not accurately reflect decontamination cost changes. Although it is expected that nuclear power reactor licensees will purchase the maximum amount of insurance that is reasonably available, the NRC reserves the right to perform periodic analyses to determine changes in accident recovery costs and to conduct rulemaking based on these analyses.

The Threshold for Automatically Determining Stabilization Priorities

In response to the 1987 final rule on changes in property insurance requirements, several petitions for rulemaking (noticed in 53FR 36335, 9/19/80) were received that requested clarification of the decontamination and stabilization priorities. As part of that rulemaking (55FR 12163, 4/2/90), the NRC amended 50.54(w)(4) to require dedication of insurance proceeds to decontamination and stabilization activities only if the estimated costs exceeded $100 million. This cutoff was viewed as a relatively minor accident where the availability of funds for stabilization decontamination activities is not considered to be an issue.

However, the Commission stated in this rulemaking that if disputes over the stabilization and decontamination process arise, the Rules of Practice under 10CFR Part 2 provide adequate procedures to resolve any issues.

Summary

This background discussion establishes that the purpose of 10CFR 50.54(w) is to protect health and safety in the unlikely event of an accident at a nuclear power plant. The minimum insurance requirement to assure post-accident recovery is based on the estimated stabilization and decontamination costs developed in NUREG/CR-2601 for two reference plants. Since it is not the Commission's intent to require more insurance coverage than is necessary for these purposes, licensees of smaller reactors have been granted exemptions from the full insurance requirement based on plant specific analyses that demonstrate lower cleanup costs. Finally, the NRC retains the authority to establish accident recovery and cleanup priorities, regardless of the estimated stabilization and decontamination costs.

Clearly the development of lower onsite property damage insurance requirements for the PSD plant is consistent with the intent of the regulation.

Assessment

Section 4 of this report developed accident consequence estimates for the four spent fuel storage configurations that were assessed for this program.

Configuration 1, "Hot Fuel in the Spent Fuel Pool," postulated rapid zircaloy oxidation of the spent fuel rods after the loss of the pool water inventory. The safety hazard analysis (Section 4) has estimated consequences that are approximately equal to a severe core damage accident. Given the potential magnitude of the consequences, it is appropriate that the onsite property damage insurance requirements of 10CFR 50.54(w) remain fully applicable for Configuration 1.

Configuration 2, "Cold Fuel in the Spent Fuel Pool," has sufficiently low decay heat loads such that the cladding will remain intact even if all spent fuel pool water is lost. Configuration 2 considers the consequences of a dropped assembly. The Configuration 2 onsite cleanup costs has been estimated at $24 million.

In lieu of long term storage in the spent fuel pool, a permanently shutdown nuclear power plant may store its spent fuel in an Independent Spent Fuel Storage Installation (ISFSI), before, during, and after, the plant itself has been decommissioned. As such, Configuration 3 must examine the regulatory requirements for the plant without fuel (similar to Configuration 4) and the ISFSI. The postulated accident for Configuration 3 is a non-mechanistic breach of the ISFSI which damages a singleBWR or PWR fuel assembly.* The Configuration 3 onsite cleanup cost is estimated at $12 million.

Configuration 4, "All Fuel Removed from the Site," assumes that all spent fuel has been shipped offsite, including any that might have been stored in an ISFSI. As discussed in Section 4, the postulated accidental radioactive releases to the atmosphere during decommissioning do not pose a significant threat to the onsite workers or the public. For the purpose of estimating onsite accident cleanup costs, the postulated scenario for Configuration 4 is the rupture of the borated water storage tank. Approximately 450,000 gallons of slightly radioactive water is released causing soil contamination. The estimated cleanup cost is $110 million.

Inservice Inspection and Testing ISI and IST Requirements

Background

10CFR50.55a, Codes and Standards, require that ASME Code Class 1, 2, and 3 pumps, valves, vessels, piping, and supports meet the testing and examination requirements set forth in Section XI of the ASME Boiler and Pressure Vessel Code. Each licensee is required to update and submit their ISI and IST Programs every ten years to the edition and addenda referenced in 10CFR50.55a(b), 12 months prior to the start of the 10 year interval. The initial interval begins at the issuance of the operating license. Section XI provides testing requirements to verify the operational readiness of pumps and valves and the structural integrity of pressure retaining components and their supports.

The ISI and IST Programs contain a plant-specific list of the applicable components, code classification, code category, examinations or tests to be performed, and the frequency and schedule of examination or testing. When the code requirements are impractical, for instance due to plant design, or would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety, the regulations permit alternatives to be used when authorized by the Commission.

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*This consequence estimate may not envelope sabotage scenarios which could conceivably involve a greater radionuclide release. These scenarios are safeguard information. The information on radio-nuclide release (if any) is not available to BNL.

Assessment

Each licensee is required to determine the ASME Code Class 1, 2, and 3 components and prepare and ISI and IST program for these components. Each program is plant specific depending on the design of the plant and the classification of components. The classification may be determined based on Regulatory Guide 1.26, NUREG-0800, or the ANSI/ANS Standards N52.1 and 51.1, depending on the age of the lant and the agreements made with the NRC. The systems important to the permanently defueled plant are radiation monitoring, fuel building, HVAC, and spent fuel pool cooling cleanup. The ASME Boiler and Pressure Vessel Codes do not address instruments and controls such as radiation monitoring. Fuel building HVAC, and spent fuel pool cooling systems may be included in the IST programs, depending on whether they perform a design basis safety-related function. Non-safety related components are not required to be examined or tested in accordance with the Code. Additionally, some plants may not include HVAC systems in the ISI/IST programs because they do not contain water, steam, or radioactive waste.

It is recommended that licensees of permanently shutdown plants reduce the scope of the ISI and IST programs to eliminate those systems that do not support spent fuel storage and handling (including cooling and cleanup) and HVAC. Although the revised program should be submitted to the NRC, approval is not necessary, unless relief requests are revised or added.

Fracture Prevention Measures

Background

Sections 50.60, 50.61, and Appendices G and H to Part 50 specify fracture toughness requirements and material surveillance programs for the reactor coolant pressure boundary of light water reactors. The intent of these regulations is to maintain reactor coolant pressure boundary integrity by assuring adequate margins of safety during any condition of normal operation (including anticipated operational occurrences).

Assessment

Once the permanently shutdown plant has been completely defueled, the measures required by these regulations are no longer necessary. These requirements can be eliminated for all spent fuel storage configurations without impacting the health and safety of the public.

ATWS Requirements

Background

The purpose of 10CFR50.62 is to require improvements in the design and operation of light water cooled nuclear power plants to reduce the likelihood of RPS failure following anticipated operational occurrences. This regulation also requires improvements in the capability to mitigate the consequences of an ATWS event.

Assessment

Although ATWS can be a significant contributor to operating plant risk, it is not applicable to permanently shutdown plants where fuel is stored in subcritical arrays. This regulation can be eliminated for all spent fuel storage configurations of the permanently defueled plants without impacting public health and safety.

Loss of All AC Power Requirements

Background

The loss of all AC power requirements Station Blackout Rule is found in 10CFR50.63. The regulation requires that all light water cooled nuclear power plants be capable of withstanding a complete loss of AC power for a specified duration and maintain reactor core cooling during that period. The NRC intent is to provide further assurance that a loss of both the offsite and onsite emergency AC power systems will not adversely affect public health and safety.

The Station Blackout (SBO) rule was published in the June 21, 1988 issue of the Federal Register (53FR23203). The supplementary information provided with the rule indicates that the purpose of this regulation is to explicitly require that nuclear power plants be designed to insure that core cooling can be maintained for a specific duration (coping period) without onsite or offsite AC power. The coping period can range from two to sixteen hours depending on the plant-specific design and the site characteristics.

Assessment

The objective of the rule is to reduce the risk of severe accidents resulting from SBO by maintaining highly reliable AC electric power systems and, as an additional defense in depth, assuring that plants can cope with a loss of all AC power for some period of time. The goal is to maintain the core damage frequency contribution of SBO to about 10^{-5}/reactor year.

Although the rule is oriented toward core damage, the objective of reducing severe accident risk due to SBO can be applied to a permanently defueled plant.

Based on the analysis in NUREG/CR-1353,[6] a total loss of spent fuel cooling would allow over 40 hours of boiloff before any spent fuel would be exposed. This time is well in excess of the maximum coping period required by the rule. The long period before fuel damage occurs allows ample time for offsite power recovery or fuel pool makeup.[*] BNL has estimated a fuel damage frequency of 5E-7 (with credit for one emergency diesel generator (EDG)) and 4E-5 (no EDGs credited) for an extended loss of all AC power.

BNL believes that permanently shutdown nuclear power plants meet the intent of 10CFR50.63. For consistency with Reg. Guide 1.155, we recommend that the existing (operating based) SBO plant procedures and training be revised to reflect the storage of all fuel in the spent fuel pool (Configurations 1 and 2).

The ISFSI of Configuration 3 should fully conform to the requirements of Section 72.122(k), however since all fuel has been removed from the plant (Configurations 3 and 4) the requirements of 10CFR50.63 are not applicable.

Maintenance Effectiveness

Background

The NRC amended its regulations under 10CFR50.65 to require commercial nuclear power plant licensees to monitor the effectiveness of maintenance activities on safety significant plant equipment. The intent is to minimize the likelihood of failures and events caused by the lack of effective maintenance. The rule will require that licenses:

- Perform annual evaluation of the effectiveness of the maintenance program.
- Assess the overall impact of monitoring and maintenance activities (which require taking equipment out of service) on the performance of safety functions.

The rule will become effective on July 10, 1996.

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[*]Reference 6 has estimated a 24 hour recovery period for actions that require access to the spent fuel pool. These could include the use of the fire protection system to provide pool makeup. Remote recovery actions, such as offsite power recovery, are not limited by the auxiliary building radiation levels and must be accomplished before boiloff exposes the fuel.

Assessment

Section 50.65, paragraph b (scope of the monitoring program) includes safety-related structures, systems, and components that are relied upon to remain functional during and after design basis events to prevent or mitigate the consequences of accidents that could result in potential offsite exposure comparable to 10CFR Part 100 guidelines. Also included within the scope of the maintenance effectiveness program are non-safety related structures, systems, and components (SSCs) that are relied upon to mitigate accidents. Furthermore, draft regulatory guide DG-1001 [DG-1001, 8/1/89] clarifies the scope of the rule as including "SSCs in the balance-of-plant that would significantly impact safety or security."

Using the draft regulatory guide and other industry guidance each licensee will develop a prescriptive maintenance effectiveness program to meet the intent of the rule.

Plants that have formally ceased operations prior to July 10, 1996 (the effective date of the rule) are not expected to have implemented a maintenance effectiveness program. It is recommended that these facilities be exempted from the requirements of the rule.

Plants that operate after July 10, 1996 should have a maintenance effectiveness program in place. The scope of the program will vary from plant-to-plant based on plant-specific design and operating attributes. When a plant is permanently shutdown many of these structures, systems, and components can be removed from the maintenance effectiveness program. For these plants, the scope of the maintenance effectiveness program can be reduced to reflect the permanently shutdown plant configuration, i.e., it would only apply to the structures, systems, and components necessary to support safe fuel storage in the spent fuel pool (Configurations 1&2).

The requirements of Section 50.65 are not applicable to spent fuel storage Configurations 3 and 4.

Periodic FSAR Update Requirements

Background

10CFR50.71(e) requires NPP licensees to file FSAR revisions annually or six months after each refueling outage (provided the interval between successive updates to the FSAR does not exceed 24 months). The updated FSAR shall "include the effect of all changes made in the facility or procedures described in the FSAR all safety evaluations performed by the licensee either in support of requested license amendments or in support of conclusions that changes did not involve an unreviewed safety question all analyses of new safety issues performed by or on behalf of the licensee at Commission request."

The NRC position on the continued applicability of 50.71(e) to permanently shutdown plants appears to be evolving. Schedular exemptions from 50.71(e) have been issued to PSD licensees in the past.[7]

However, more recently, the Yankee Nuclear Power Station received an exemption from the FSAR update requirements.[8]

Assessment

After a decision to permanently shutdown a facility has been formalized with the NRC, a licensee may begin making extensive changes to plant structures, systems, and components that are no longer necessary. Each of these changes will require a 50.59 safety evaluation which in turn requires a FSAR review. The continuance of the FSAR update requirement will provide a somewhat current plant reference source for future safety evaluations and will also continue to serve as a licensing document. In the supplemental information provided as part of the Final Rule [45FR30614, May 9, 1980] the scope of the rule was specifically extended to include older plants without FSARs including the Indian Point 1 and Humboldt Bay plants that were permanently shutdown at the time. In addition, wenote the periodic[*] FSAR update requirements for ISFSIs, a passive storage system, without the support systems required for fuel storage in the spent fuel pool. It is recommended that the FSAR update requirements of 50.71(e) be maintained for all spent fuel storage configurations, with schedular exemptions as necessary to encourage a timely submittal that documents the plant at major decommissioning milestones. However, the scope of the document is expected to be reduced to reflect the decommissioning process, i.e., the removal of plant systems, structures, components, and procedures, that are no longer necessary from a health and safety perspective. The ISFSI update requirements of 70.72 remain, although for consistency, a biennial update period should be considered.

Training and Qualification of Nuclear Power Plant Personnel

Background

In 1993 the NRC amended its regulations [58 FR 21904, 4/26/93] to require that each applicant and each holder of a license to operate a nuclear power plant establish, implement, and maintain a training program. The new requirement, 10CFR 50.120, uses a systems approach to training to ensure nuclear power plant personnel will be qualified to operate and maintain the facility in a safe manner for all modes of operation.

The rule requires training and qualification of the following nuclear power plant personnel:

- Non-licensed operator
- Shift supervisor
- Shift technical advisor

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[*]10CFR72.70 currently requires an annual FSAR update for ISFSI licensees. The similar requirement for Part 50 licensees was revised from an annual to a refueling outage basis not to exceed 24 months. (57FR39353, 8/31/92).

- Instrument and control technician
- Electrical maintenance personnel
- Mechanical maintenance personnel
- Radiological protection technician
- Chemistry technician
- Engineering support personnel

Licensed operators, such as control room operators and senior control room operators, are not covered by this rule and will continue to be covered by 10CFR Part 55. Because some senior control room operators may also be shift supervisors, only those aspects of training related to their shift supervisor function are covered by this rule.

As part of the public comments to the proposed Rule, several commenters recommended that facilities undergoing decommissioning, where all fuel has been permanently removed from the reactor vessel, or those with a possession only licensee, not be subject to this Rule. The Commission disagreed, stating that the provisions of the Rule are applicable to all Part 50 licensees. The Commission maintained that the systems approach to training embodied in the Rule will ensure that training programs are revised to reflect changing plant conditions. Permanent changes to the plant (i.e., decommissioning) that make some or all of the existing training programs unnecessary can be addressed by the exemption process. Since the public risk associated with the permanently shutdown nuclear power plant is associated with the spent fuel, it is recommended that the requirements of 50.120 continue for Configurations 1 and 2 for only those personnel that are responsible for fuel handling and the continued safe storage of the spent fuel.

As shown in the safety hazard analyses of Section 4, after the spent fuel has been moved to an ISFSI or offsite, the risk to the public is negligible. The training and qualification requirements of 50.120 can therefore be removed for Configurations 3 and 4.

Material Control/Accounting of Special Nuclear Material (including US-IAEA Agreement

Background

Part 70, Sections 51 and 63 provide general material balance, inventory, recordkeeping, and status report requirements that are applicable to nuclear power reactors. Section 53 refers to 10CFR74.13(a) and 75.35 which provide additional detailed material status report requirements including reporting form numbers and submittal dates.

Independent spent fuel storage installations have similar requirements as specified in 10CFR72.72, 72.76, and 75.35.

Assessment

The material control and accounting requirements of Parts 70, 74, and 75 remain fully applicable for permanently shutdown plants in spent fuel storage Configurations 1 and 2. Licensees in Configurations 3 should be exempted from the Section 70.51 and .53 and, as applicable, Part 74. Material accounting requirements will remain for the ISFSI under Parts 72 and, as applicable, Part 75. If all fuel is removed from the site, the material control and accounting requirements of Part 70, and all of Parts 74 and 75 are not longer applicable.

Financial Protection Requirements

Background

The financial protection requirements for large nuclear power plants* are found in Part 140 of 10CFR. At the present time, paragraph 140.11(a)(4) requires a primary layer of financial protection of $200 million. A secondary layer of financial protection is also mandated. This is an industry retrospective rating plan providing for deferred premium charges equal to the pro rata share of the public liability claims and costs. Under this plan, the current maximum deferred premium charges for each nuclear reactor which is licensed to operate is $75.5 million with respect to any nuclear incident.** No more than $10 million per incident is required in a calendar year. The total financial protection for any incident would equal the primary layer of $200 million plus the secondary layer of $75.5 million times the number of reactors covered, or in excess of $8 billion.

This liability insurance covers claims resulting from a nuclear incident or a precautionary evacuation. In addition to accidents involving offsite releases, public evacuation and land contamination, the insurance covers liability arising from power plant effluents, storage and transportation of spent fuel,*** and radioactive waste materials. Included in the insurance coverage are defense costs for claims settlement.

10 CFR Part 140 was established in 1957 pursuant to Section 170 of the Atomic Energy Act of 1954, commonly called the Price-Anderson Act. One of the purposes of the Act was to protect the public by assuring the availability of funds for the payment of claims arising from a catastrophic nuclear incident. The Act required the AEC's reactor licensees to furnish financial protection (in the form of nuclear

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* i.e., a nuclear reactor facility that is designed for producing 100,000 electrical kilowatts or more.

** plus any surcharge assessed under subsection 170o (1)(E) of the Atomic Energy Act of 1954, as amended.

*** The liabilities and indemnification requirements associated with the transfer of spent fuel from the licensee to the Department of Energy will be evaluated on a case by case basis at a future time when spent fuel is shipped to a repository.

liability insurance or the equivalent) to cover public liability claims against the licensee and all others who might be liable for a nuclear incident. A second major provision required the AEC to indemnify the licensee and all others who might be liable in the amount of $500 million over and above the financial protection required. The Act also limited the liability from a nuclear incident to the sum of the financial protection required plus the AEC's indemnity. For large reactor licensees this resulted in a statutory liability limit of $560 million. The Act had similar provisions for certain licensees not operating reactors and to certain AEC contractors.

The financial protection requirement for large nuclear power plants was (and remains) the maximum amount of liability insurance available at a reasonable cost and on reasonable terms from private sources. The amount was originally $60 million. The required amount has been increased in step with increases in the amount of privately available nuclear energy liability insurance. The current requirement for this primary layer of insurance is $200 million. Other licensees generally have lesser financial protection requirements which consider type, size, and location of the licensed activity and "other factors pertaining to the hazard."

In 1975, the Price-Anderson Act was modified and extended until 1987 (Public Law 94-197). This amendment established a secondary layer of insurance by requiring that a retrospective premium of $2 to $5 million be established for large nuclear power plants. Part 140 was revised (42FR 46 1/3/77) to establish a retrospective premium of $5 million per facility per incident. The NRC chose the $5 million level because such a premium would not present an undue burden on any size utility. Moreover, since the $5 million requirement was the highest allowed by Public Law 94-197, it would result in the maximum financial protection available to pay public liability claims.

In 1988, Public Law 100-406 modified and extended the Price-Anderson Act to the year 2002. The retrospective premium was increased to $63 million per reactor per incident. This limit was subsequently increased to $75.5 million (58FR 42851 8/12/93) by Section t of the Act, based on the consumer price index change since 1988.

This discussion of the offsite liability insurance requirement has established that one intent of the Price-Anderson legislation is to protect the public by ensuring that timely compensation is available in the event of claims arising from a catastrophic nuclear incident. Unlike the onsite property damage insurance requirement, the offsite liability levels as mandated by Congress do not appear to have an explicit technical basis.

The primary insurance requirement, presently at $200 million, is based on the maximum amount of liability insurance available from private sources. Similarly, there does not appear to be an explicit technical basis for the secondary layer retrospective premium of $75.5 million per reactor.

Although the permanently shutdown nuclear power plant has a lower public risk, many activities that have the potential for public liability claims will continue until all radioactive materials are removed and the site is released for unrestricted access. This implies that the offsite liability insurance requirement should continue although, for most configurations a lower requirement should suffice.

Assessment

There are three major considerations that are germane to this offsite liability assessment. Each is discussed below:

- **The Relationship of Accident Probability to the Liability Insurance Requirements**

One purpose of the Price-Anderson Act was to protect the public by assuring the availability of funds for the payment of claims arising from a catastrophic nuclear incident. Probabilistic Risk Assessments (PRAs) provide a mechanism to examine the relationship of accident frequency and accident consequence for a given enterprise. Full power PRAs of nuclear power plants show increasing consequences with decreasing accident frequencies. The accident consequences can be used to determine liability insurance levels.

Although Congress did not explicitly state its intent when specifying or amending the Price-Anderson Act, some inferences can be drawn from a review of the hearing transcripts.

On March 3, 1976, shortly after the Price-Anderson Amendments Act of 1976 (Public Law 94-197) was adopted, the Joint Committee on Atomic Energy held a hearing to consider whether the financial risk to utilities under the Price-Anderson system should be increased.[10] The hearing transcript provided the following insights:

From the prepared statement of Larry Hobart, Assistant General Manager, American Public Power Association (p. 34)

> *Public Law 94-197 was the result of extensive committee hearings and vigorous Congressional debate extending over a two-year period. During Congressional consideration of the legislation, the level of financial risk to be imposed on electric utilities was the major focus of attention. Testimony was taken on a variety of approaches to the question. The range of retrospective premiums provided under current law is the end-product of that very detailed examination.*
>
> *The decision by Congress took into account the conclusions of this committee relative to risk to the public, including evaluation of the findings of the study "An Assessment Accident Risks in U.S. Commercial Nuclear Power Plants" prepared under the direction of Dr. Norman C. Rasmussen of the Massachusetts Institute of Technology. The committee stated in its report of November 13, 1975, on this legislation that: "Insofar as the amount of financial protection for the public is concerned, both Dr. Rasmussen testimony before the joint Committee last year and the final report*

affirm that the total of public and private indemnity provided for by this bill is adequate to cover any credible accident which might occur."

As part of the general discussion, committee member representative John B. Anderson of Illinois stated (P. 11):

One further comment on the question of the $560 million limit on liability. We did have some testimony before Joint Committee when we considered the extension of Price-Anderson to the effect that this would afford protection for about 96 percent of all the accidents that might occur.

In other words, that 96 percent of the probable accidents that could occur would be below the extent of the limits imposed on liability under this statute and the kind of accident that would exceed that amount would be one that would probably occur once in every 5,000 years and that as the pool floats upward, as it will do under the legislation, as I know the Senator is aware, to about $1 billion by 1985 this would include 99 percent of all accidents that might occur. In other words, accidents that would exceed that $1 billion would likely occur once in 10,000 years.

The witness, Senator Charles H. Percy from Illinois responded in part,

The committee was very wise to establish through the Rasmussen report the fact that the risks are relatively low. We needed some means of bringing it down from a 10,000-year span to what we can really comprehend in relation to our own insurance policies. We don't have to be concerned about 10,000 years so much as the probability of an accident occurring once in 10,000 chances in 1 year or once in a thousand chances in 10 years. The Rasmussen study shows that when 100 reactors are on line, the probability over a 10-year period of an accident with $900 million in property damages, a 2,000 square mile decontamination area, a 130 square mile relocation area, 300 early illnesses and total health effects over a 30 years of 5,100 latent cancer deaths, 42,000 thyroid nodules and several hundred genetic defects, is one in a thousand.

On the basis of this testimony we can extrapolate that the frequency (F) of a release resulting in the stated consequences is:

F/ reactor year x 100 reactors x 10 years = 1.0E-3, therefore:
F = 1.0E-6/reactor year

These statements (and the intent of the Joint Committee) can be interpreted two ways:

1. The intent of the committee was to ensure that the primary and secondary layers of financial indemnity will afford protection for about 96 to 99 percent of the accidents that might occur.

2. The intent of the committee was to determine a credible accident frequency, and establish indemnity levels based on the estimate consequences of that credible accident.

For the purposes of operating power reactors these two interpretations have the same outcome. However, for the PSD plant they can produce disparate results, when the release frequency distribution is different from the full power operation of a nuclear plant. For example, if a release frequency ranged between 1E-7 and 1E-10, with 1E-9 and greater comprising 99 percent of the total frequency, interpretation number 1, would require the financial protection levels based on a 1E-9 accident. However, interpretation number 2 would not require any liability insurance.

It is likely that Congress implicitly assumed a credible accident frequency (interpretation number 2). We believe that the intent of Congress in establishing a retrospective premium in the range of $2 to $5 million was to ensure that adequate funds were available to cover any credible accident that might occur. That level of funds appears to be $1 billion. The associated "credible" accident frequency is about 1E-6 per reactor year.

The release frequency estimates for the spent fuel storage configuration representative accident sequences are provided in Section 3. The release frequency for the Configuration 1 accident is in the E-6 range for both BWRs and PWRs. The spent fuel assembly drop (Configuration 2) is 3E-4 events per year. The ISFSI release frequency (Configuration 3) of 6E-6 events per year is from an EPRI study. However, as discussed in Section 3, it is our judgement that this frequency is overstated by at least two orders of magnitude. The estimated release frequency is approximately 3E-7 events per year. The Configuration 4 seismically induced borated water storage tank (BWST) rupture has been estimated at 2E-7 events per year.

Table B.2 A Comparison of Consequence Estimates

	Early Fatalities	Latent Fatalities	Societal Dose (Person-Rem)	Boundary Dose (Rem)	Condemned Land (Sq. Miles)	Total Offsite Costs $
Configuration 2	0	2	4000	0.009	0.0	neg
Configuration 3	0	0.22	690	0.472	0.0002	neg
TMI 2[1]	0	0.4	~2000	0.100	0.00	neg[2]

1. TMI 2 accident information is from the Rogovin Report (Ref. 12)
2. Established based on milk and vegation sampling results reported in Reference 12. All samples were well under EPA protective action levels.

- **The Relationship that Accident Consequence Calculations Have to Actual Liability Expenses**

Consequence codes such as the MELCOR Accident Consequence Code System (MAACS) are used to estimate the outcomes of radiological accidents in terms of health effects, population dose, and economic cost. It appears that one bases of the offsite liability requirement for large power reactors is an estimate of accident consequences. However, these calculations are not necessarily representative of actual experience.

For example, Table B.2 presents the consequence estimates for Configurations 2 and 3 using the MAACS Code. The Three Mile Island Unit 2 accident data is also provided for comparison. The table shows that the TMI 2 offsite health and economic consequences are similar to the estimates for Configuration 2 and 3. Yet, as of 1993, $60 million has been awarded settlement of claims arising from the TMI 2 accident. A significant number of claims were still unsettled as of 1993.[11]

There clearly is a disparity between the expected consequences and the public's perception of an accident. The Rogovin Report[12] recognized this stating:

> *In our view, the fact that there will be no adverse radiation health effects, or very minimal effects, from the Three Mile Island accident has not been clearly understood by the public. It is clear to us that the public misconception about the risks associated with the actual releases measured during the accident, as well as about the risks associated with nuclear power plants generally, has been due to a failure to convey credible information regarding the actual risks in an understandable fashion to the public.*

Despite significant education efforts, the majority of the public is not comfortable with nuclear power. In all likelihood, the public mistrust of all things nuclear will continue for the foreseeable future. In this environment the public reaction to relatively minor incidents will be exacerbated, (e.g., precautionary evacuation) and result in economic consequences that are far in excess of code predictions.

- **The PriceAnderson Requirements for Non Operating Reactors and ISFSIs**

Section 170 of the Atomic Energy Act,* Part a requires that:

> *"Each licensee issued under Section 103 or 104 and each construction permit issued under Section 185 shall, and each licensee issued under Section 53, 63, or 81* **may**, *for the public purposes cited in Section 2 I. have as a condition of the license a requirement that the licensee have and maintain financial protection of such type and in such amounts as the Nuclear Regulatory Commission (in this*

*Commonly known as the Price-Anderson Act.

section referred to as the "Commission") in the exercise of its licensing and regulatory authority and responsibility shall require..." *(emphasis added)*

The NRC must require financial protection for licensees issued under Section 103 (commercial licenses), Section 104 (medical therapy and research and development) and for construction permits and operating licenses under Section 185. Section 170b gives the Commission the authority to require less than the maximum amount of primary financial protection, in consideration of other factors including, the type, size, and locations of the licensed activity. However, the Act specifies primary and secondary insurance amounts for facilities designed for producing substantial amount of electricity. Financial protection is not mandated for Sections 53, 63, and 81 which addresses the domestic distribution of: special nuclear material, source material, and byproduct material, respectively.

There has been significant debate regarding the applicability of Section 170 to permanently shutdown facilities. After a sufficient cooling period such that there is no longer the threat of rapid zircaloy oxidation, the accidents that could be associated with the PSD facility have significantly reduced consequences. Cases can be made for removing the offsite liability insurance requirement or continuing it with less than the maximum amount required for the permanently shutdown facility.

Section 4 of this report developed accident consequence estimates for the four spent fuel storage configurations that were assessed for this program.

Configuration 1, "Hot Fuel in the Spent Fuel Pool," postulated rapid zircaloy oxidation of the spent fuel rods after the loss of the pool water inventory. The safety hazard analysis has estimated consequences that are approximately equal to a severe core damage accident.

Configuration 2, "Cold Fuel in the Spent Fuel Pool," has sufficiently low decay heat loads such that the cladding will remain intact even if all spent fuel pool water is lost. Configuration 2 considers the consequences of a dropped assembly. The safety hazard analysis, as discussed in Section 4 of shows negligible offsite costs.

In lieu of long term storage in the spent fuel pool, a permanently shutdown nuclear power plant may store its spent fuel in an Independent Spent Fuel Storage Installation (ISFSI), before, during, and after, the plant itself has been decommissioned. As such, Configuration 3 must examine the regulatory requirements for the plant without fuel (similar to Configuration 4) and the ISFSI. The postulated accident for Configuration 3 is a breach of the ISFSI which damages a single BWR or PWR fuel assembly.* The estimated offsite cost is negligible

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*This consequence estimate may not envelope sabotage scenarios which could conceivably involve a greater radionuclide release. These scenarios are safeguard information. The information on radionuclide release (if any) is not available to BNL.

Configuration 4, "All Fuel Removed from the Site," assumes that all spent fuel has been shipped offsite, including any that might have been stored in an ISFSI. As discussed in Section 4, the postulated accidental radioactive releases to the atmosphere during decommissioning do not pose a significant threat to the onsite workers or the public. For the purpose of estimating onsite accident cleanup costs, the postulated scenario for Configuration 4 is the rupture of the borated water storage tank. Approximately 450,000 gallons of slightly radioactive water is released causing onsite soil contamination and potential contamination of the water table. BNL has performed calculations that indicate tritium levels will be below the maximum concentration limit for drinking water at the site boundary. Offsite remediation has not been considered and again offsite costs are considered to be negligible.

Given the potential magnitude of the consequences, it is appropriate that the offsite liability insurance requirements of 10CFR Part 140, both the primary and secondary levels, remain in place for Configuration 1.

The insurance recommendations for the remaining configurations are not as straightforward. Qualitative justifications can be made for anywhere from $0 to $200 million.

Since the analyses show minimal offsite consequences, a case can be made for eliminating the offsite liability requirements for Configurations 2, 3, and 4. Any liability awards *should* be minimal and the licensee should be able to pay those awards in a timely manner, thereby satisfying the intent of the Price-Anderson Act.

Conversely, the $200 million figure recognizes the possibility of a large suit for alleged damages due to routine, low level radioactive effluents from the plant during decommissioning.

All things considered, a $100 million offsite liability insurance requirement is a reasonable compromise for the permanently shutdown plant. The TMI 2 experience has shown that significant judgements can be awarded, despite negligible offsite consequences. It is also recommended that these plants be allowed to withdraw from the secondary level of protection. In addition, the exemption process could be used to justify lower plant specific requirements, as deemed appropriate.

For the independent spent fuel storage installations (ISFSIs) that are not covered under an existing site policy, it is acknowledged that a lower liability limit could be justified. The passive nature of the installation, and the expected lack of radioactive effluents, routine or otherwise, conceivably results in less liability exposure.

Annual Fees for Licensees

Background

Part 171 of 10CFR, "Annual Fees for Nuclear Power Reactor Operating Licenses," was published on September 18, 1986 [51FR33224] as a final rule. The rule assessed an annual fee for FY1987 for every power reactor licensed to operate. The annual fee was instituted to comply with the statutory mandate of the Consolidated Omnibus Budget Reconciliation Act of 1985. The scope of this section was expanded [56FR31472, 7/10/91] to include other entities including nonpower reactors, materials licensees, part 72 ISFSI licensees, fuel facilities, etc., in response to the congressional mandate requiring the NRC to recover approximately 100% of its budget authority in FY1991 and the four succeeding years. In the Responses to Comments, Section D, Specific Fee Issues of the Final Rule, the NRC responded to the issue of annual fees for shutdown plants. Two commenters had indicated that charging the full annual power reactor fee was not fair because certain costs allocated to all power reactors were not applicable to permanently shutdown plants. The Commission responded that the proposed rule excluded power reactors with a POL* from the FY 1991 fee base. This waiver was extended and remains in effect for FY95.

Assessment

The NRC is required to recover approximately 100% of its budget authority. The licensing and inspection fees assessed under Part 170 recover the costs of providing individually identifiable services to specific applicants for, and holders of, NRC licenses and approvals. Part 171 provides for the recovery of NRC budgeted costs for generic regulatory activities for each class of licensee. For example, the generic activities associated with power reactor licensees include: reactor decommissioning, license renewal, construction permit, and operating license reviews. Also included are generic costs such as the Incident Response Center and certain other NRC efforts that can support other licensees, but are primarily established for the power reactor licensee. Costs attributable to types of licenses other than power reactors (i.e., part 72 licensees) consists of generic regulatory costs and other costs not recoverable under Part 170, including rulemaking, upgrading safeguards requirements, modifying the standard review plans and developing inspection programs.

Permanently shutdown power reactor licensees continue to require NRC services, although not to the extent of a full power licensee. It is recommended that the Part 50 licensees, authorized to possess but not operate a nuclear power reactor be assessed as a group for the NRC services that are to be provided. If the appropriate fees cannot be accurately assessed at this time, perhaps a fee that is equivalent to the annual ISFSI fee can be instituted.

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*or with a formal NRC order prohibiting placing fuel back in the reactor vessel.

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11. ABSTRACT (200 words or less)

An evaluation of the nuclear power plant regulatory basis is performed, as it pertains to those plants that are permanently shutdown (PSD) and awaiting or undergoing decommissioning. Four spent fuel storage configurations are examined. Recommendations are provided for those operationally based regulations that could be partially or totally removed for PSD plants without impacting public health and safety.

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